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# Advanced Nuclear Reactor Safety Issues and Research Needs

**Workshop Proceedings**
**Paris, France**
**18-20 February 2002**

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Nuclear Safety

# Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs

**Paris, France**
**18-20 February 2002**

**Co-sponsored by the**
**International Atomic Energy Agency**
**and**
**Organised in collaboration with the**
**European Communities**

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

## ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14 December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

## NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

## FOREWORD

On 18-20 February 2002, the OECD Nuclear Energy Agency (NEA) organised, with the co-sponsorship of the International Atomic Energy Agency (IAEA) and in collaboration with the European Commission (EC), a Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs. It was attended by a broad range of parties with a potential stake in the development and deployment of advanced nuclear power plants, including designers, utility executives, regulators and researchers.

Currently, advanced nuclear reactor projects range from the development of evolutionary and advanced light water reactor (LWR) designs to initial work to develop even further advanced designs which go beyond LWR technology (e.g. high-temperature gas-cooled reactors and liquid-metal-cooled reactors). These advanced designs include a greater use of advanced technology and safety features than those employed in currently operating plants or approved designs. The objectives of the workshop were to:

- facilitate early identification and resolution of safety issues by developing a consensus among participating countries on the identification of safety issues, the scope of research needed to address these issues and a potential approach to their resolution;
- promote the preservation of knowledge and expertise on advanced reactor technology;
- provide input to the Generation IV International Forum Technology Roadmap.

In addition, the workshop tried to link advancement of knowledge and understanding of advanced designs to the regulatory process, with emphasis on building public confidence. It also helped to document current views on advanced reactor safety and technology, thereby contributing to preserving knowledge and expertise before it is lost.

The meeting was structured into four main sessions:

- Session 1 focused on basic safety principles, and requirements, and on the implementation of defence-in-depth. It also provided an introduction to major novel designs and their safety cases.
- Session 2 was devoted to the identification of issues important for nuclear safety and their assessment.
- Session 3 investigated how to deal with these safety issues and identified questions and concerns, while stressing those that could be solved through research and identifying research needs.
- Session 4 developed conclusions and recommendations regarding safety issues and research needs.

The role of international co-operation in the field of safety research was emphasised, as was its role in preserving knowledge and competence.

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## SUMMARY AND RECOMMENDATIONS*

### Introduction

An OECD NEA Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs was organised from 18 to 20 February 2002. It was co-sponsored by the IAEA and organised in collaboration with the European Commission. It was attended by more than eighty participants, representing eighteen countries and four international organisations. Twenty-six papers were presented.

Currently, advanced nuclear reactor designs range from the development of evolutionary and advanced LWR designs to initial work to develop even further advanced designs which go beyond LWR technology (e.g., high-temperature gas-cooled reactors and liquid metal reactors). These advanced designs include a greater use of advanced technology and safety features besides those employed in currently operating plants or approved designs. The purpose of the Workshop was to promote early consensus among Member countries on identification of advanced nuclear reactor safety and research issues, including possible paths for their resolution. Its objective was to bring together a broad cross-section of parties – designers, utilities, regulators and researchers – with a potential stake in the development and deployment of advanced nuclear power plants, to:

- Facilitate early identification and resolution of safety issues by developing a consensus among participating countries on the identification of safety issues, the scope of research needed to address these issues and a potential approach to their resolution.
- Promote the preservation of knowledge and expertise on advanced reactor technology.
- Provide input to the US DOE Generation IV Technology Roadmap development.

The Workshop also tried to link advancement of knowledge and understanding of advanced designs to the regulatory process, with emphasis on building public confidence. And it helped to document current views on advanced reactor safety and technology, thereby contributing to preserving knowledge and expertise before it is lost.

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- Session 1 focused on basic safety principles, and requirements, and on the implementation of defence-in-depth. It also provided an introduction to major novel designs and their safety cases.
- Session 2 was devoted to the identification of issues important for nuclear safety and their assessment.

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* The following summary and conclusions was drafted by C.E. Ader, G. Cognet, W. Frisch, M. Gasparini, K.W. Hesketh, J. Hyvärinen, J.E. Lyons, H. Niwa, A. Porracchia, M. Vidard, H. Wider and J. Royen.

- Session 3 discussed how to deal with these safety issues and identified questions and concerns, stressing those that can be solved through research and identifying research needs.
- Session 4 developed conclusions and recommendations regarding safety issues and research needs.

The results of the Workshop will be documented as proceedings, to be published by the NEA. The following document, prepared by the Organising Committee of the meeting, is a summary of the conclusions reached by the Workshop and a summary of the sessions.

**Conclusions and Recommendations**

The outcomes of Workshop sessions can be summarised in the following conclusions:

The basic principle of nuclear safety, defence-in-depth, continues to be employed also in advanced reactors. However, it was recognised that future and advanced reactors pose several questions and challenges to the implementation of defence-in-depth. In the past, this has been achieved primarily through deterministic implementation of provisions and multiple physical barriers against the release of fission products, and by measures to prevent accidents and mitigate their consequences The emphasis put on prevention and/or mitigation differs among the various advanced concepts. The approach to the safety of future reactors will need to be derived from a more advanced interpretation of defence-in-depth fully integrated with PSA insights. How the best integration of the deterministic and probabilistic concepts will be achieved is still a major open question.

The advanced reactor concepts discussed in the workshop were mostly limited to Advanced Light Water Reactors (ALWRs), High-Temperature Gas-Cooled Reactors (HTGR) and Liquid-Metal Cooled Reactors (LMR). The concepts discussed can be divided roughly into two categories: mature ones, (more or less) ready for market, such as Framatome-ANP's SWR-1000 or Westinghouse AP-600, and preliminary ones, such as IRIS (an ALWR), and most LMRs and HTGRs. A common feature to all advanced reactor types is that they promise safety enhancement over the current generation of plants; likewise, the safety significance and provisions to be made against external hazards are common questions that pertain to all future designs.

Mature ALWR concepts are characterised by increased simplicity and streamlining in their safety system design, significant amount of passive (system) features, and explicit consideration of severe accidents as a part of their design basis. Regarding severe accidents, the ambitions of their technical and regulatory treatment varies between Europe and United States. European vendors and regulators specifically require qualification of the dependability of their severe accident capabilities (which does not mean that all related technical issues would already have been categorically resolved; design features are selected also on the basis of PSA insights to effectively eliminate severe accident sequences that would be overly complex to manage) while in the US, PSAs are relied upon more extensively to identify severe accident vulnerabilities and appropriate measures to reduce the risk from severe accidents.

As to LMRs, considerable experience base from operating sodium-cooled reactors exists, and convergence seems to be occurring in treatment of certain major issues such as Core Disruptive Accidents and sodium related issues. As far as lead/bismuth cooled reactors are concerned, significant remaining questions relate (among others) to materials and thermal-hydraulics issues (integrity, corrosion, thermal loads and heat transfer, irradiation effects, etc.). It should be noted, though, that

considerable operating experience (about 80 reactor-years) has been gained with Russian submarines using the same type of coolant. Several research institutions in OECD Member countries are building research facilities to intensify experimental and analytical HLM investigations.

As to HTGRs, some amount of actual operating experience exists, and regarding future HTGRs, the picture of main issues is clear: HTGR safety cases, as presented so far, rely very heavily on the fuel as the main (if not sole) fission product barrier, and hence all fuel issues become prominent. These include fuel (concept) qualification with very high confidence level, manufacturing issues, fuel handling during operation, and improved understanding of fuel failure mechanisms and modes. HTGR designs include promising features against both criticality and decay heat removal related functional concerns, but ultimately their success will depend on quality of the fuel. Also certain well-known systemic safety issues such as air-water ingress into the reactor and reactor vessel integrity with respect to thermal shock remain to be addressed to an extent that convinces the whole reactor safety community. Most of the fundamental research into currently fashionable HTGR design features seems to date from 20-30 years back and there does not appear to be much of recent experimental effort to satisfy this need, by either confirming earlier results or closing existing gaps. Proponents of HTGR maintain that the plant needs no leak-tight containment in the conventional sense against its internal threats due to greatly enhanced safety; however, recent attention to external hazards and the fact that the relative importance of external hazards increases with enhanced safety against internal hazards may raise the question of the need for a containment or other adequate protection against external hazards.

Identification of specific research needs for any reactor type can only follow after a *consistent overall safety case* has been established for it. This consistent overall safety case (i.e., identification of needs) helps identify where (remaining) research needs are and what level on uncertainty reduction (confidence) is necessary. Ideally, the safety case should render manageable all confidence requirements of each individual safety question and safety factor; only then can research problems be formulated properly (problem definition so that it will have a definite solution knowable to adequate accuracy and obtainable at reasonable cost). Research supporting the passive systems development of mature ALWR concepts seems to fulfil this objective (or comes close).

After due consideration of Workshop discussions summarised above, the Organising Committee recommends that the CSNI consider the following possibilities of future actions:

- workshops to discuss the safety cases and their supporting evidence could be arranged, with more definite technical focus (e.g., fewer different technologies in one workshop) and priority on near-term deployable design concepts;
- the preparation of such workshops should be led by countries that have immediate interest in these options and the meetings should be open (on an invitation basis) also to capable and interested non-OECD participants, because much actual work towards future reactors is being done also outside the OECD;
- explore and identify possibilities of international co-operative research projects;
- take actions to compile and preserve the extant knowledge bases, especially for technologies developed 20+ years ago (key HTGR fuel testing). A good example of how to do this exists in the form of Computer Code Validation Matrices, developed for (mainly) LWR applications by former CSNI Principal Working Groups; the most important part of this effort is storage of the chosen experiments in the NEA Data Bank (preferably in digital form). *Internationally recognised and guaranteed storage is the only way of guaranteeing that essential data, knowledge and understanding are not lost.*

Irretrievable loss of some LWR test data has already occurred; in order to avoid loss of essential test data related to other reactor types, it would be advisable to start setting up corresponding international data bases as a first priority (as a minimum, as soon as any new data start to become available);

- more generally, in order to maintain competence, take action to preserve acquired knowledge and experience in areas where R&D is at a standstill at the moment, such as LMFBR development. Such action would cover aspects of information storage and retrieval, and information transfer, in industry, universities and regulatory bodies.

The Organising Committee notes that definite identification of specific research needs for any individual reactor design depends on its associated safety case (among other things, national / international requirements to be fulfilled). Current variations in implementation of basic safety principles and differing ambition levels preclude more definite listing than the one that has been given above; however, a longer list of issues is provided in the summary of Session 3. Those issues are likely to need to be addressed to an extent that depends on the requirements specified by the overall safety case.

## SESSION SUMMARIES*

### Session 1: Safety Principles, Safety Requirements and Implementation of Defence-in-depth – Introduction to Major Novel Designs and their Safety Cases

Session 1 included five papers.

The first paper presented a method for the development of safety requirements for advanced reactors. The approach is based on the general concept of defence-in-depth (DID) as defined and described in IAEA publications, and integrated by probabilistic considerations. The approach is general and can be applied to any kind of reactors and it is being tested at the IAEA to assess the safety of LWRs and to create the technical basis for the preparation of safety requirements for HTGR.

The second paper made the case for the need to risk-inform the traditional approach of DID. The paper discussed two possible approaches that have been proposed. The first maintains the traditional DID approach, but PSA insights are used to evaluate and confirm the effectiveness of DID. The second approach retained the DID philosophy at the high level, but utilizes the insights from PSAs directly in the implementation of DID decisions, with PSA as the primary decision-making tool. DID is a result of decisions made to compensate for uncertainties.

The third paper emphasized the need for early interactions between the designers and regulatory authorities to:

- identify safety issues early in the development of new concepts and to identify possible paths for their resolution;
- identify needed technology development programmes;
- identify needed new or different regulatory guidance.

Regarding this latter point, some of the challenges to implementing a risk-informed framework were discussed, including lack of operating experience for new designs.

Design characteristics and safety issues in High Temperature Reactor were described in the fourth paper. The requirement of "catastrophe-free nuclear technology" in Germany was explained. In order to fulfil it, the fission product release is restricted to less than $10^{-5}$ of the inventory. Current evaluation shows that the fuel temperature does not exceed 1 600°C, which satisfies the requirement. Further research needs include fuel performance, improvement of fuel, air/water ingress to primary circuit, integrity of pressure boundary and underground siting. Discussions include: cost increase in underground siting, reactivity change due to fuel motion in earthquakes, maximum burnup, containment material concerning failure mode (burst or not).

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* Drafted by the session chairpersons and the organising committee of the workshop.

In the fifth paper, after addressing the usefulness of FBR fuel-cycle as a sustainable energy supply system, the historical perspective was reviewed to discuss the safety approaches taken in the previous sodium-cooled LMRs. Conventional DID approach, which has been taken in LWRs, was also applied in LMRs. Regarding CDA, its prevention was stressed with high priority, while the necessity of mitigation measures was stated for reinforcement of elimination of the recriticality concern and achievement of in-vessel cooling/retention. Further discussions included sodium-air/water interaction, cladding material to achieve high burnup, and the necessity of fuel cycle for realizing a FBR system.

Topics from Session 1 that warrant further discussion are:

- role of PSA and DID concept;
- balance between prevention and mitigation for new concepts which shift achievement of safety to prevention;
- decision process for deciding on the need for containment vs. confinement.

**Session 2: Issues Important for Safety and their Assessment**

Session 2 was devoted to the identification of issues important for nuclear safety and their assessment. Sixteen papers were presented during this session. The safety issues were mainly addressed in two different ways:

- on the basis of particular design concepts (10 papers);
- in the frame of regulatory activities (4 papers).

Moreover, two papers, from CEA and on the Generation IV initiative, presented overall development strategies.

The various reactor concepts presented in the first group of presentations mentioned above, cover a wide range of reactor types including evolutionary water cooled reactors, more advanced water reactors such as the integral concept IRIS and a reduced moderation concept in Japan, gas cooled reactors and liquid metal cooled reactors. As a consequence of this broad spectrum of reactor types, the presented safety issues also were numerous and of different nature. The degree of knowledge on these safety issues is also very different, because the presented concepts had a very different degree of design advancement. While some papers referred to plants with already existing operating experience (e.g. Superphenix, the Japanese HTTR, the Chinese HTR 10), other designs are still in an early conceptual phase. Though the size of a reactor concept was not discussed as a particular safety issue, it is interesting to note that the presented design concepts varied from 10 $MW_{th}$ (HTR 10) up to 1700 $MW_{el}$ (ABWR II).

The majority of the presented water-cooled design concepts are advanced water-cooled reactors between 600 $MW_{el}$ and 1700 $MW_{el}$ (PWR, BWR and CANDU-type). In terms of safety they have some common characteristics. In all design concepts severe accidents are considered and design features are provided to cope with them, especially by strengthening the ultimate barrier function, the containment. The amount of effort realized on this $4^{th}$ level of defence-in-depth is different in the various designs and depends on the safety requirements imposed on them. With respect to necessary R&D, it has been stated in general that much has already been done, especially in the area of passive safety system behaviour and severe accident phenomena, and that additional effort might be necessary for final confirmation.

The gas cooled reactor concepts have been presented as “severe accident free” in the sense that accidents with severe core damage have an extremely low estimated probability. For this reason the presented gas-cooled reactor concepts do not foresee a containment structure as a last tight barrier against radioactive releases to the environment. The necessary R&D effort is higher than for LWRs (for more details see below).

Safety issues for liquid metal cooled reactors have been presented on the basis of three very different design concepts (Superphenix, the lead-bismuth-cooled SVBR-75/100, and the Japanese LMFR). Important safety issues have been raised in the discussion (see below).

Concerning the presentation of safety approaches and licensing issues for ALWR (Finland, France/Germany, US) a general convergence can be seen with respect to important issues such as aim for safety improvement, implementation of defence-in-depth, risk informed elements in basically deterministic requirements and treatment of severe accidents. For the latter aspect, the required means and the necessary validation to prove the effectiveness of design measures against severe accidents differ among the various regulatory systems.

The paper on licensing of PBMR gives a list of still open items, which is longer than that for ALWRs because of the lack of experience and knowledge from predecessor plants, and indicates the necessary future effort, especially in the area of fuel behaviour and safety analysis tools (codes, code validation). The role of the containment function is still under discussion.

During the discussions following the individual papers of Session 2 many safety issues have been addressed and/or emphasized by questions and comments. Some general issues have been touched several times such as pros and cons of underground siting, the involvement of the public during the development of safety approaches and within licensing processes, and the question of a “balanced design” (e.g. with respect to prevention and mitigation of severe accidents, or design against internal events and external hazards).

After the presentations on the liquid metal cooled reactors, questions have been raised on:

- possibility of steam generator leaks reaching the core of a lead-bismuth reactor;
- corrosion problems of steel in lead-bismuth contact;
- sodium leaks and fires;
- sodium-water reactions;
- core disruptive accidents.

The presentations on water cooled reactors in general gave the impression that the designs are already rather mature and that most of the safety issues have been resolved by design features, analysis and/or experimental validation. Nevertheless numerous questions have been raised. Examples of issues discussed are:

- The essential design changes during the extrapolation from AP 600 to AP 1000 (mainly increased power density according to M. Carelli).
- The validity of experimental verification of containment outer shell cooling when extrapolating from AP 600 to AP 1000.
- The effectiveness of in-vessel retention after core melt accidents in large PWRs such as the Korean APR 1400 and the Japanese ABWR II.

- The appropriateness of providing both in-vessel retention and ex-vessel cooling as severe accident mitigation means (e.g. possibility of steam explosion if core debris enter a flooded cavity).
- The extended use of passive systems (e.g. in the German SWR 1000 or the passive heat removal system in the Japanese ABWR II).
- The significance of negative void coefficients (M. Bonechi states that the NG CANDU has a negative void coefficient).
- R&D on fuel and fuel channel behaviour for NG CANDU.
- The question of "proliferation proof" fuel cycles (M. Bonechi states that the NG CANDU fuel cycle fulfils the IAEA specifications).
- The problem of excluding extremely rare events from the design by probabilistic cut-off values and/or deterministic exclusion criteria (e.g. definition of "practical elimination" in the French/German approach).
- The non-consideration of severe accident mitigation measures when core melt frequency is supposed to be extremely low (e.g. as stated for the IRIS concept as a principle; however, the vessel cavity which is in IRIS to guarantee core coverage could in principle also double as ex-vessel retention measure.).

For gas-cooled reactor concepts the situation with respect to open safety issues is different from the ALWR situation. Operating experience and experimental validation is much less and stems from a more remote period of time, raising the additional problem of retrieval of relevant information. A comprehensive list of open issues can be found in the various papers. Some of the items which received special attention during the discussion, are:

- Air and/or water ingress.
- Fuel behaviour.
- Fuel handling risks.
- Safety system classification.
- Demo plant versus tests in real plant, e.g. during commission tests (is instrumentation adequate to provide sufficient information for code validation?).
- Significance of external hazards, especially when frequency of internal events is significantly reduced.
- Need for a containment function (or not?).
- Clear identification of open issues (paper of Y.L. Sun on the HTR 10 presents a good balance of what is known and what is not yet known).

In Session 2, both the paper presentations and the contributions during the discussion have led to a comprehensive identification of safety issues, thus representing a good basis for the approach of Session 3, "How to Deal with Safety Issues, Questions and Concerns – Identification of Research Needs".

## Session 3: How to Deal with Safety Issues, Questions and Concerns? – Identification of Research Needs

This session discussed how to deal with the safety issues and identified questions and concerns. The papers stressed those issues that can be solved through research and identified research needs.

The session began with a paper by T. Okkonen (STUK) that described the role of research when assessing, and finally demonstrating, the safety of future reactor concepts. The paper provided a top-down planning perspective by addressing a general set of safety factors related to new reactor projects. The research needs then can be based on the safety factors and related challenges. Research was presented as a three-phase learning process of exploration, consolidation, and verification. Knowledge is increased slowly at first through initial exploration of the problem. Then as basic principles and concepts are refined, knowledge is rapidly consolidated and accumulated as real understanding is gained. Finally, as the field matures, uncertainties are reduced through verification, but there is less new information to be gained.

The remaining papers in the session provided utility and international views on research needed in water-cooled, gas-cooled, and liquid-metal reactor technologies. In previous sessions, other authors also presented on-going or proposed research. G. Fiorini (CEA) presented the following tentative list of safety and reliability crosscut research and development from the Generation IV forum:

- Design basis transients and accidents – static and transient analyses.
- Severe plant conditions.
- Environmental impact.
- Safety-related architecture: economical evaluation.
- Licensing approach: risk-informed and risk-based regulations.

The research needs presented during the conference are summarized below by reactor type.

***Crosscut issues for all reactor types: (G. Serviere, EDF)***

- Operators should continue to have a role to play, with appropriate interface systems.
- Designed to be easily inspected and quickly maintained or repaired if necessary.
- Identify materials that will limit the amount of fission products released from the core and the amount of activated products in the reactor coolant system.
- Materials should be analysed for compatibility with the environment and fully characterised for mechanical properties over the entire spectrum of contemplated temperatures and pressures.
- Creep behaviour should be fully investigated for structures operating at high or very high temperatures, including creep-fatigue interaction.
- Manufacturing and construction procedures (e.g., welding techniques) need to be developed and validated.

- Eliminate materials that could prove to be extremely difficult to deal with during decommissioning.

***Advanced light water reactors***

EDF View on Next Generation Reactors Safety and Operability Issues: (G. Serviere, EDF).

Integrated type reactors:

- Fuel enrichment.
- Integrated component reliability.
- Dynamic behaviour of the reactor in case of transients and accidents.

Supercritical LWRs:

- Fluid behaviour under supercritical conditions in case of an accident.
- Critical flow in case of break.
- Flow regimes during transition between sub-critical and critical flow conditions.
- Fuel cladding dry out in case of high heat flux.
- Reassessing scaling conditions for test facilities, including phenomena importance.
- Numerical modelling of the transition regime (oscillations due to rapidly changing fluid properties).
- Mechanical behaviour of overpressure protection components when discharging super-critical fluid.
- Component and structure reliability.
- Re-qualify fuel for anticipated operating conditions.

Current and Future Research on Passive Safety Devices (W. Brettschuh, Framatome-ANP).

- Completed:
  - Emergency condenser.
  - Containment cooling condenser.
  - Passive pressure pulse transmitter.
  - Passive flooding line with valve.
  - Containment behaviour in event of core melt with large gaseous and aerosol releases.
  - Reactor pressure vessel exterior cooling.
  - Reactivity control – steam driven scram tanks.
- Ongoing or planned:
  - Reactivity control – fast acting boron injection system (FABIS).

- Passive internal reactor pressure vessel flooding – spring supported check valve.
- Vent pipes and quenchers for safety relieve valves.
- Control rod drives.

Fuel Development for Advanced Light Water Reactors (W.D. Krebs, Framatome-ANP).

- Materials behaviour.
- Fuel behaviour.
- Breeder reactor fuel (long term).

***Sodium-cooled reactors***

*General Research Needs: (S. Kondo, JNC).*

- Passive Safety Features (passive shutdown and natural circulation heat removal).
- Elimination of recriticality under core disruptive accidents.
- In-Vessel Retention without challenging the containment.
- Advanced fuels for fast reactors (R.J.M. Konings, Institute for Transuranium Elements).

*More Specific Research Needs:*

For conventional LMRs: (Natta, IPSN)

Avoiding local accidents due to extensive control rod withdrawal:

- Introduce computer logic that each control rod can only be moved by a maximum amount (in Superphenix by 15 mm)

Avoid air ingress into primary system:

- Investigate possible reasons for air ingress (it happened in SPX)

For future LMRs: (JLMR)

For mitigation core disruptive accident:

- Develop special fuel assembly with inner duct structure for fuel discharge into retention plates in the lower core support structure.
- Design lower core support structure for long-term coolability to ensure IVR.

Control of interaction with air:

- Guard pipe is proposed to suppress sodium fire both for primary and secondary piping. All penetrations should be covered to ensure double boundary system.

Control interactions with water:

- Development of new steam generator (probably using Pb/Bi as an intermediate fluid) in order to avoid a possible sodium/water interaction. This would also make a secondary sodium system unnecessary.

Lead/bismuth-cooled reactors: (J.U. Knebel, FZK)

Although Russian scientists seem to have mastered all the critical aspects of Pb/Bi cooled systems, OECD countries should perform considerable confirmatory research. In the important field of materials/corrosion and thermal-hydraulics of this coolant extensive research has begun in several OECD countries (France, Germany, Italy, Japan, Sweden, Switzerland and USA). The specific areas in which research is needed are:

- Corrosion mechanisms of steels in contact with liquid lead and lead/bismuth.
- Protection mechanisms and corrosion resistance enhancement.
- Mechanical behaviour of steels being in contact with lead or lead/bismuth.
- Irradiation effects on mechanical properties.
- Irradiation effects on liquid metal corrosion.
- Purification and oxygen control systems.

Thermal-hydraulics:

- Improvement and assessment of numerical codes with physical models which are validated for reactor relevant geometries and conditions.
- Single-effect and integral effects experiments of fundamental flow configurations and reactor relevant geometries, in order to benchmark the numerical codes.
- Measurement techniques.

Flow configurations and geometries:

- Laminar/turbulent heat transfer along thermally highly-loaded surfaces.
- Flow mixing under forced flow, transition and buoyant flow conditions.
- Bubbly two-phase flow with heat transfer.
- Flow field and heat transfer in a heat exchanger.

***High-temperature, gas-cooled reactors (consolidated list)***

*General research needs result from:*

- establishment of list of events to be studied and determination of the associated limits.
- identification of the needed computer codes and qualification needs.
- identification of necessary physical data.
- assessment of uncertainties.

*Specific research areas identified include:*

- Fuel performance – testing of fuel under normal and accident conditions, including reactivity insertion accidents; testing beyond predicted conditions to identify "cliff edge".
- Fuel qualification – verifying quality of fuel in production facilities.
- Graphite behaviour – long term behaviour, measurement of physical and thermal characteristics.
- High temperature material behaviour – creep-fatigue data and environmental characteristics.
- High temperature components – behaviour of components (e.g., control rods, gas turbines) for high temperatures applications.
- Verification and validation of computer codes – includes assessment or development of data for code validation.
- Air and water ingress accidents – behaviour of fuel and graphite, potential for graphite fire, and methods for intervention.
- Fission product behaviour in primary circuit – plate out, lift off and transport of fission products and graphite dust during normal operation and accident conditions, system and component contamination and impact on personnel exposure during maintenance.
- Normal operation and accident scenarios – modelling of passive decay heat, core geometry changes, reactivity insertion, air and water ingress, containment vs. confinement.
- Protection of reactor pressure vessel against hot spot formation
- Fuel handling and storage – non-proliferation aspects of fuel cycle, and behaviour of fuel during long term storage.

# SESSION 1

## Safety Principles, Safety Requirements and Implementation of Defense-in-depth
## Introduction to Major Novel Designs and their Safety Cases

***Chairman: M. Bonaca (USNRC ACRS)***
***Co-chairmen: C.E Ader (USNRC) and H. Niwa (JNC, Japan)***

# DEFENCE-IN-DEPTH AND DEVELOPMENT OF SAFETY REQUIREMENTS FOR ADVANCED NUCLEAR REACTORS

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Division of Nuclear Installation Safety

## Abstract

The paper addresses a general approach for the preparation of the design safety requirements using the IAEA Safety Objectives and the strategy of defence-in-depth. It proposes a general method (top-down approach) to prepare safety requirements for a given kind of reactor using the IAEA requirements for nuclear power plants as a starting point through a critical interpretation and application of the strategy of defence-in-depth. The IAEA has recently developed a general methodology for screening the defence-in-depth of nuclear power plants starting from the fundamental safety objectives as proposed in the IAEA Safety Fundamentals. This methodology may provide a useful tool for the preparation of safety requirements for the design and operation of any kind of reactor. Currently the IAEA is preparing the technical basis for the development of safety requirements for Modular High Temperature Gas Reactors, with the aim of showing the viability of the method. A draft TECDOC has been prepared and circulated among several experts for comments. This paper is largely based on the content of the draft TECDOC.

## Introduction

There is a large number of proposed advanced nuclear reactors and there are also different definitions of the term "advanced". In 1997 the IAEA published a TECDOC on Terms for describing new, advanced nuclear power plants (NPPs) [1]. According to the definitions proposed in this TECDOC, an advanced plant design is a design of current interest for which improvement over its predecessors and/or existing designs are expected. Advanced designs consist of evolutionary designs and designs requiring substantial development efforts. The latter can range from moderate modifications of existing designs to entirely new design concepts. They differ from evolutionary designs in that a prototype or a demonstration plant is required, or that work is still needed to establish whether such a plant is required.

The different design approaches, technologies and safety features of advanced concepts indicate that the full application of existing safety requirements, mostly developed for large water cooled reactors may need, in some cases, interpretation or adaptation. For some innovative concepts there is a need to develop a tailored set of safety requirements derived from the general consolidated principles of nuclear safety, which better incorporates the specific characteristic of a given concept. The IAEA Safety Standards and the ongoing work on implementation of defence-in-depth for different type of reactors provide a useful starting point and a suitable framework for this purpose.

## General safety objectives

The IAEA publication The Safety of Nuclear Installations [2] sets out basic objectives, concepts and principles for ensuring the safety of nuclear installations in which the stored energy or the energy developed in certain situations could potentially result in the release of radioactive material from its designated location with the consequent risk of radiation exposure of people. The principles are derived from the following three fundamental safety objectives (the following five paragraphs are reproduced from Ref. [2].

**General Nuclear Safety Objective:** To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.

This General Nuclear Safety Objective is supported by two complementary Safety Objectives dealing with radiation protection and technical aspects. They are interdependent: the technical aspects, in conjunction with administrative and procedural measures, ensure defence against hazards due to ionising radiation.

**Radiation Protection Objective:** To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.

**Technical Safety Objective:** To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

In order to achieve these three safety objectives and to demonstrate that they are achieved in the design of a nuclear power plant, comprehensive safety analyses are carried out to identify all sources of exposure and to evaluate radiation doses that could be received by the public and by workers at the installation, as well as potential effects of radiation on the environment. The safety analysis examines: (1) all planned normal operational modes of the plant; (2) plant performance in anticipated operational occurrences; (3) design basis accidents; and (4) any event sequences that may lead to severe plant conditions. The design for safety of a nuclear power plant applies the principle that plant states that could result in high radiation doses or radionuclide releases are of very low probability of occurrence, and plant states with significant probability of occurrence have only minor or no potential radiological consequences. For advanced NPPs it is expected that the technical need for external intervention measures may be very much limited or even eliminated.

The three safety objectives as described above are expressed in general terms. However, they clearly delineate the safety approach to all nuclear installations and to developing safety principles and safety requirements. There is no evidence, for the time being, of any need for substantial changes to these objectives – they still represent the starting point for the preparation of safety requirements for any advanced or future reactor.

**The defence-in-depth strategy**

The general strategy to meet the safety objectives with large confidence and in a "measurable" way is the implementation of defence-in-depth.

The actual level of safety of an NPP is determined by the compliance of the design with detailed requirements and criteria (deterministic and probabilistic). In other words, the level of safety depends on the way defence-in-depth is implemented in the design, taking into account the implications of the specific features and technology.

The strategy for defence-in-depth [3] is twofold: first, to prevent accidents and, second, if prevention fails, to limit their potential consequences and prevent any evolution to more serious conditions. Accident prevention is the first priority. The rationale for the priority is that provisions to prevent deviations of the plant state from well-known operating conditions are generally more effective and more predictable than measures aimed at mitigation of such departure, because the plant's performance generally deteriorates when the status of the plant or a component departs from normal conditions. Thus preventing the degradation of plant status and performance generally will provide the most effective protection of the public and the environment.

The implementation of the concept of defence-in-depth in the design of a plant provides a series of levels of defence (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails. This strategy has proved to be effective in compensating for human and equipment failures, both potential and actual.

There is more than one way to implement defence-in-depth, since there are different designs, different safety requirements in different countries, different technical solutions and varying management or cultural approaches. Nevertheless, the strategy represents the best general framework to achieve safety for any type of NPP.

Generally, several successive physical barriers for the confinement of radioactive material are put in place. Their specific design may vary depending on the activity of the material and on the possible deviations from normal operation that could result in the failure of some barriers. Therefore,

the number and type of barriers confining the fission products is dependent on the reactor technology adopted.

Defence-in-depth is generally structured in five levels. Should one level fail, the subsequent level comes into play. Table 1. summarizes the objectives of each of the five levels and the primary means of achieving them. The correct implementation of defence-in-depth ensures that a failure, whether mechanical or human, at one level of defence, and even combinations of failures at more than one level of defence, will not propagate to jeopardize defence-in-depth at subsequent levels. This requires the independence of the different levels of defence.

The general concept of defence-in-depth as articulated by the IAEA is now widely known and adopted, even though, in some cases, defence-in-depth is still solely interpreted as the availability of multiple physical barriers to the release of fission products.

Table 1. **Levels of defence-in-depth (from INSAG-10)**

| **Levels of defence** | **Objective** | **Essential means** |
|---|---|---|
| Level 1 | Prevention of abnormal operation and failures | Conservative design and high quality in construction and operation |
| Level 2 | Control of abnormal operation and detection of failures | Control, limiting and protection systems and other surveillance features |
| Level 3 | Control of accidents within the design basis | Engineered safety features and accident procedures |
| Level 4 | Control of severe plant conditions including prevention of accident progression and mitigation of the consequences of severe accidents | Complementary measures and accident management |
| Level 5 | Mitigation of radiological consequences of significant releases of radioactive materials | Off-site emergency response |

**The fundamental safety functions**

To ensure safety (i.e. to meet allowable radiological consequences during all foreseeable plant conditions), the following fundamental safety functions shall be performed in operational states, in and following a design basis accident and in and after the occurrence of severe plant conditions:

- control of the reactivity;
- removal of heat from the core; and
- confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases.

The possible challenges to the safety functions are dealt with by the provisions (inherent characteristics, safety margins, systems, procedures) of a given level of defence. All mechanisms that can challenge the successful achievement of the safety functions are identified for each level of defence. These mechanisms are used to determine the set of initiating events that encompass the possible initiations of sequences. According to the philosophy of defence-in-depth, if the evolution of a sequence is not controlled by the provisions of a level of defence, it will be by the subsequent level, and so on.

Figure 1 shows the logic flow diagram of defence-in-depth. The objective is always to maintain the plant in a state where the fundamental safety functions (confinement of radioactive products, control of reactivity and heat removal) are successfully fulfilled. Success criteria are defined for each level of defence-in-depth.

As the objective of the first level of protection is the prevention of abnormal operation and system failures, if it fails, an Initiating Event comes into play. Then the second level of protection will detect the failures and control the abnormal operation. Should the second level fail, the third level ensures that the safety functions are further performed by activating specific safety systems and other safety features. Should the third level fail, the fourth level limits accident progression through accident management, so as to prevent or mitigate severe accident conditions with external releases of radioactive materials. The last objective (fifth level of protection) is the mitigation of the radiological consequences of significant external releases through the off-site emergency response. Some off-site measures should be taken preventively and independently from the success of Level 4 provisions.

Some challenges/mechanisms may compromise the effectiveness of the considered level of defence by affecting either the performance of the safety function directly or the reliability of a safety provision. The effectiveness of a level of defence is determined by the ability of the provisions to cope with mechanisms that challenge the performance of safety function. The probability associated with challenges/mechanisms, the reliability of the safety provisions called for and the associated potential radiological consequences will define the risk corresponding to the accident sequence considered.

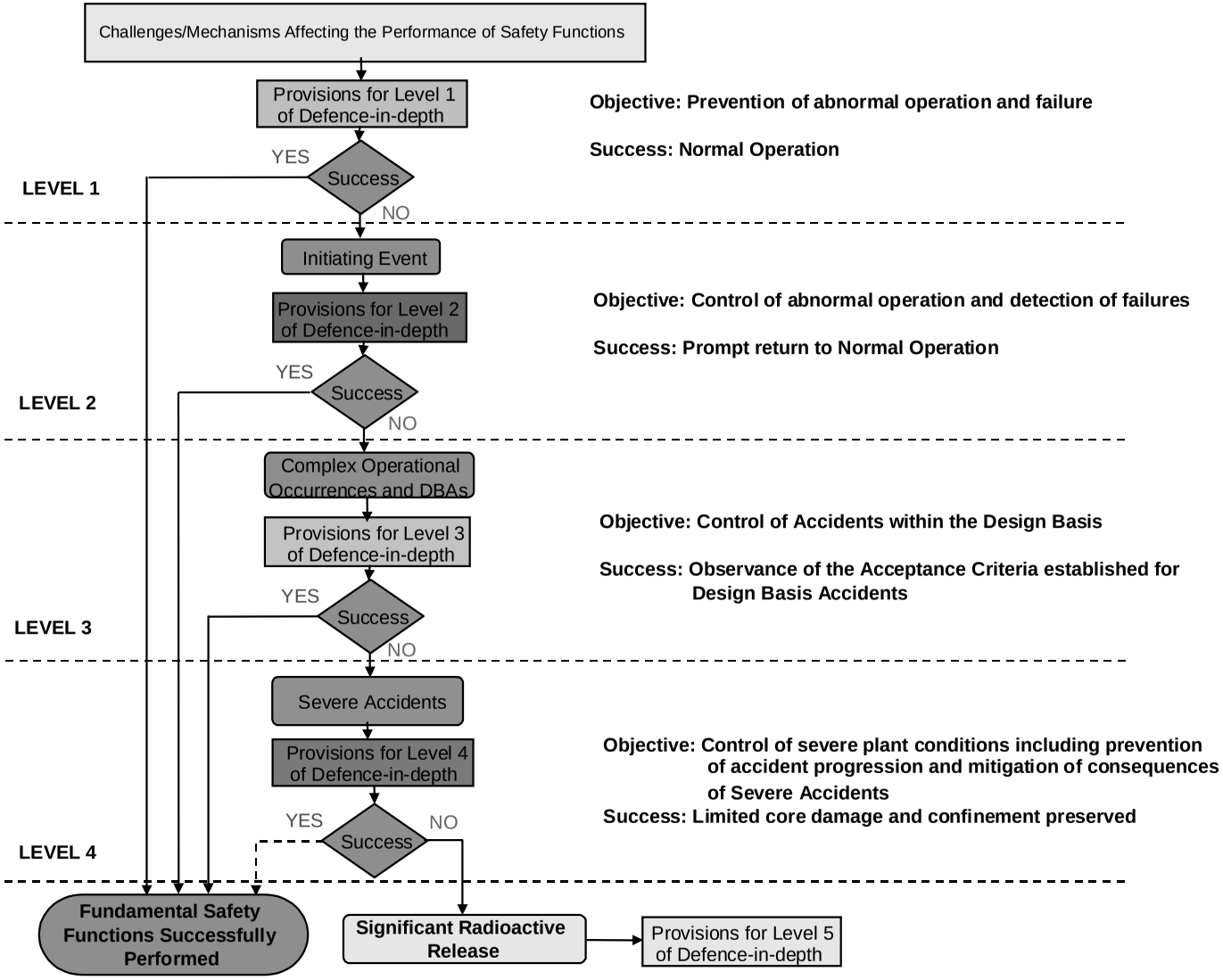
**Safety approach for current reactors**

Operating NPPs are largely designed following a safety architecture dictated by the implementation of the defence-in-depth strategy. This means that the plant is deterministically designed against a set of normal and accident situations according to well-established design criteria in order to meet the radiological targets. The current design approach has been shown to be a sound foundation for the safety and protection of public health, in particular because of its broad scope of accident sequence considerations, and because of its many conservative assumptions which have the effect of introducing highly conservative margins into the design that, in reality, give the plant the capability of dealing with a large variety of sequences, even beyond those included in the design basis.

The deterministic approach is complemented by probabilistic evaluations with the main purpose of verifying that the design is well balanced and there are no weak areas or systems that would allow the possibility of risky sequences. Probabilistic safety assessment is recognized as a very efficient tool for identifying those sequences and plant vulnerabilities that require specific complementary preventive or mitigative design features.

This safety approach is reflected in the existing IAEA Safety Requirements for the design of NPPs [4].

Figure 1. **Logic flow diagram of defence-in-depth**



**The IAEA safety standards series**

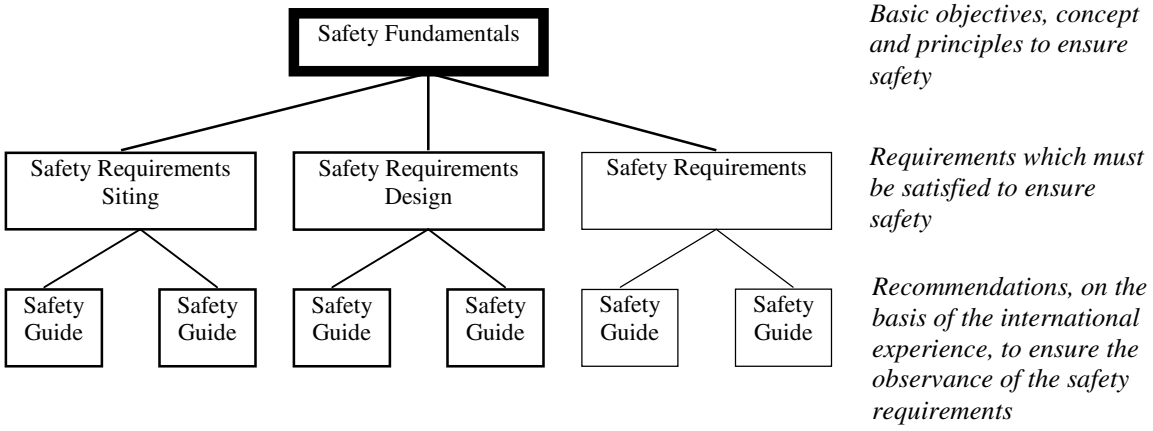
Under the terms of its Statute, the Agency is authorized to establish standards of safety for protection against ionizing radiation. The IAEA publications of a regulatory nature are issued in the IAEA Safety Standards Series, covering nuclear safety, radiation safety, transport safety and waste safety. There are three categories within the Safety Standards Series, schematically depicted in Figure 2, with the following aims:

**Safety Fundamentals:** present basic objectives, concepts and principles of safety and protection in the development and application of nuclear energy for peaceful purposes.

**Safety Requirements:** establish the requirements that must be met to ensure safety. These requirements, which are expressed as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals.

**Safety Guides:** recommend actions, conditions or procedures for meeting safety requirements. Recommendations in Safety Guides are expressed as "should" statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements.

Figure 2. **Hierarchy of the IAEA Safety Standards Series**




An extensive process was established some years ago to review all NUSS publications to produce a better-organized and consistent set of documents. Several of these new publications have been released and are also available on the Internet site of the IAEA.

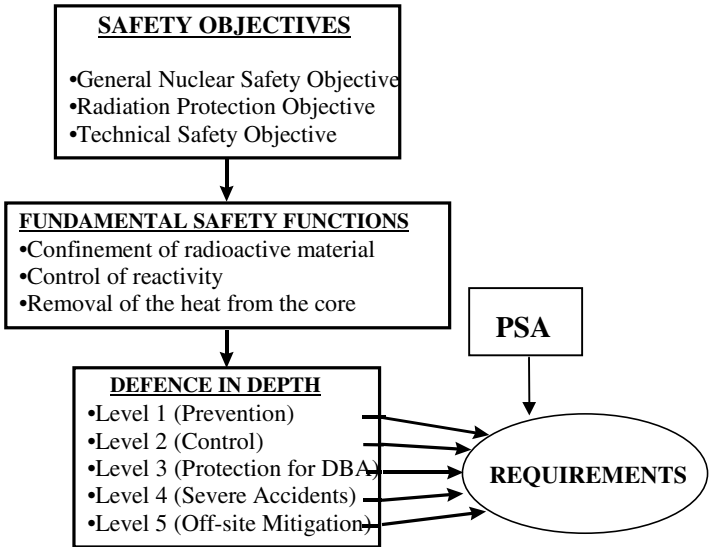
Because the process for the preparation of these publications involves several experts inside and outside the IAEA and the review and approval mechanism involves all the Member States of the Agency, the safety Standards are consensus documents that reflect established and well accepted safety rules.

**Main tenets observed in the development of the safety requirements for design**

The Requirements for the design of NPPs play an important role in establishing the safety level of the product. They also have a great impact on the cost of the plant and operating procedures.

The logical process behind the drafting of the Safety Requirements for the Design is represented in Figure 3 and briefly described below.

Figure 3. **Logical process underpinning the safety requirements**




The Safety Requirements derive from a set of limited safety principles, which directly descend from three well-established safety objectives. The safety objectives define the general targets that are to be achieved by a nuclear installation to protect the operators and the population. They are the same for all nuclear installations, including nuclear reactors, and are independent of the kind or size of any given installation.

For nuclear reactors, compliance with the safety objectives is achieved when the three fundamental safety functions *Confinement of radioactive material*, *Control of the reactivity* and *Removal of the heat from the core* are fulfilled for all the plant operational, accidental and post accidental conditions in accordance with radiological targets.

The correct implementation of the strategy of defence-in-depth ensures that the fundamental safety functions are reliably achieved and with sufficient margins to compensate for equipment failure and human errors.

Defence-in-depth has been proved to be generally applicable and very effective in assuring safety in NPPs. It can be used as primary guidance for the preparation of safety requirements. As a matter of fact, as it has been shown by INSAG [5] that there is correspondence between the five levels of defence-in-depth and the requirements.

Preparing the requirements means to establish a set of rules that indicate what shall be done to implement each level of defence-in-depth. The basically deterministic concept of defence-in-depth is integrated with probabilistic considerations (e.g. system reliability, probabilistic targets, etc.) that also provide input for additional requirements and ensure a well-balanced design to cope with all Postulated Initiating Events (PIEs).

**The top-down approach**

The safety requirements for NPPs have reached the current status through a long development process, which has incorporated the results of extensive plant operating experience and the experience gained from the lessons of the past. The current safety requirements define the safety approach developed and refined over the course of many years. Although they have mostly been developed for large water cooled reactors, it is reasonable to assume that they are a very good starting point for the preparation of the design requirements for Advanced Reactors, including non-water cooled reactors such as high temperature gas cooled reactors (HTGRs) and liquid metal cooled reactors (LMCRs).

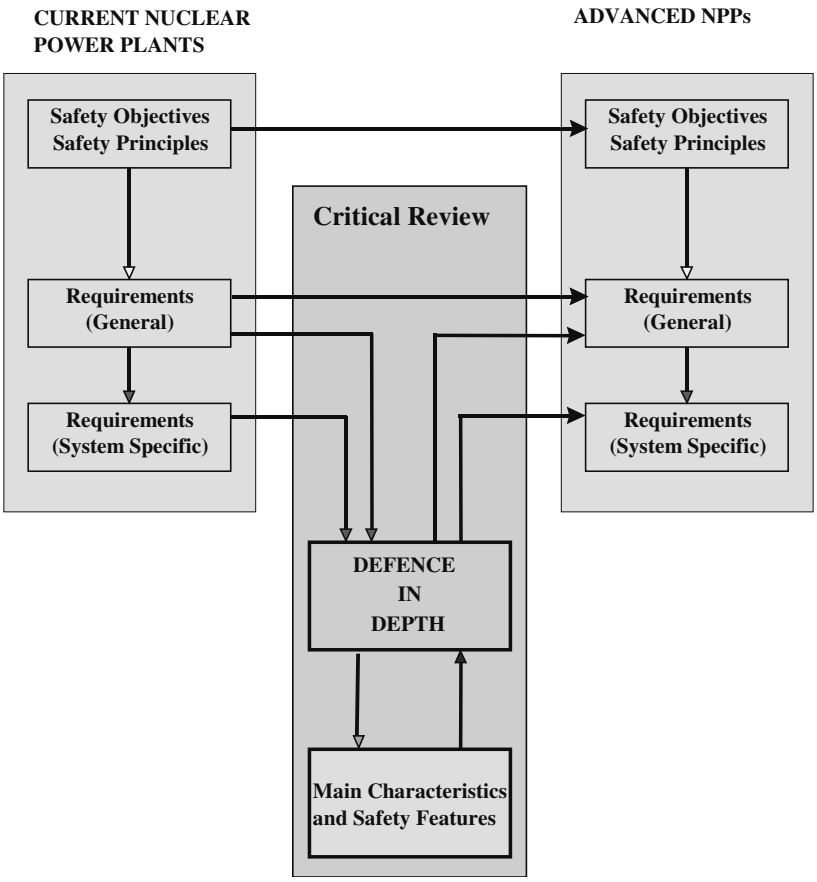
The proposed top-down approach consists of a systematic review of the existing requirements for the design of nuclear power plants [4] starting from the most general (applicable to all NPPs) and down to the most specific and more technology dependent. This process is schematically presented in Figure 4.

The Requirements for a specific type of reactor are generated through a critical interpretation of the "objectives" and "essential means" associated with each level of defence-in-depth (see Table 1), and the full understanding of the safety features of the specific reactor.

The criterion for judging the applicability or adequacy of a requirement for an existing NPP to a different reactor should be based on the full understanding of the contribution of the requirement to defence-in-depth. The "transfer function" (central box in Figure 4) that establishes the requirements for a generic nuclear reactor plant from the requirements for an existing NPP should not simply be interpreted as a filter to accept or not a requirement, but as a mechanism to generate new requirements if they are necessary because of the features of the specific nuclear reactor plant. For example, an inherent feature that fulfils a safety function in a very reliable way could allow for a relaxation of the requirements for a safety system or even to the possible elimination of the safety system that normally performs that function in water reactors. On the other hand, specific features or materials could possibly introduce failures that could initiate events for which adequate preventive or mitigative measures could be necessary.

This process will lead to the compilation of a consistent set of requirements organized in a hierarchical way with the general requirements at the top and the more specific at the bottom. Moving from the top down, the requirements will become more specific and more dependent on the particular technology.

Figure 4. **Generation of requirements for advanced NPPs**




## Main characteristics and safety features of advanced reactors

Because of the large variety of proposed designs for Advanced Reactors and their different degrees of innovation, it is quite arduous to make safety and licensing considerations that are valid for all reactors without being too general. Specific considerations can only be formulated for specific reactors. However, there are common aspects that deserve to be mentioned.

Advanced Reactors will certainly incorporate design solutions to enhance defence-in-depth and in particular the first level (prevention) and the second level (detection and control of failures).

A major effort directed at enhancing accident prevention through the improvement of the strength of the first level of defence-in-depth (higher reliability of systems, better protection against external events, …) will lead, in general, to a smaller number of significant initiating events, and consequently, to less stringent requirements on mitigative systems. This can be achieved through technical solutions that eliminate some accident sequences by design, using the intrinsic capability of the systems in a more effective way, reducing the power density, increasing the design margins, increasing the time constants for the overall reactor system in order to slow down the transients response of the system. These measures will provide tolerance to failures (Level 2 of defence-in-

depth) allowing more time for automatic control and operator actions and avoiding these failures to develop into accidents. In addition, they simplify the design of control systems and the actions required from the operators, and they decrease the number and severity of challenges to structures and safety systems (Level 3 of defence-in-depth).

The limited size of some reactors will allow, for example, to achieve the decay heat removal function with simple and reliable systems.

***Severe accidents***

For new reactors an important goal is to further reduce the potential radiological consequences of accidents. Severe accidents have to be considered systematically from the early phase of the designing process. The likelihood of core damage is expected to be very low and this should be demonstrated in a clear and convincing manner, for example through integral reactor testing.

The target proposed by INSAG [5] (CDF $<10^{-5}$ together with the practical elimination of sequences that could lead to large early radioactive release) can be used as reference. Severe accidents that could imply late containment failure would be considered in the design process with realistic assumptions and best estimate analysis so that their consequences would necessitate only protective measures limited in area and time (elimination of the need for any prompt off-site response).

***Risk informed decision making***

The challenge for the future is to develop more confidence in the PSA tools and to demonstrate that sufficient defence-in-depth can be achieved through simpler and cheaper technological solutions. Risk informed decision making plays an important role in the development and optimization of future reactors to achieve high levels of safety and to reduce the cost in particular through simplification of safety systems. Risk informed decision making will also contribute to a more rational safety classification of structures, systems and components, which is based on a sound ranking of their importance to safety.

***Simplification and use of passive features***

The simplification of plant systems with extensive use of inherent and passive features is a well-established trend for Advanced Reactors, especially with regard to safety systems. This will contribute to reduce the need of active engineering features.

The goal of greater design simplification goes hand in hand with the goal of increased design margins, robustness and response time. A simplified system is one that is more easily operated and maintained, which has reduced the number of components to the minimum necessary to provide all safety and performance functions (thereby reducing the number of failure points and modes), and which will be resilient to human errors in operation.

Passive systems offer the opportunity to eliminate complex active systems that rely on a large number of safety grade support systems by applying the advantages of simple gravity driven or thermal gradient driven safety systems. The challenge is to demonstrate the capability and the reliability of these passive systems and to deal with their longer time response.

The extensive use of passive components poses some problems for performing reliable PSAs. The safety of these reactors is determined by Initiating Events of very low probability. The consequences of these events that can be very serious, are determined by the direct phenomenological response of the plant to these events, rather than by a sequence of failures of systems, which individually have higher probabilities and which can be analysed and modelled with much less uncertainty.

***Digital instrumentation***

Systems based on digital technology have demonstrated very high reliability in many industrial applications, including NPPs, allowing for a very good supervision and control of the plant. New generation designs offer the opportunity to fully implement this technology and benefit from the associated advantages.

**The objective-provisions tree**

The method of the objective-provisions tree described below represents a preliminary attempt to systematically address the 'critical review' of the implementation of the defence-in-depth as indicated in Figure 4.

The logical framework of the objective-provisions method is graphically depicted in terms of a tree such as shown in Figure 5. At the top of this tree is the level of defence-in-depth that is of interest, followed by both the objectives to be achieved and the barriers to be protected.

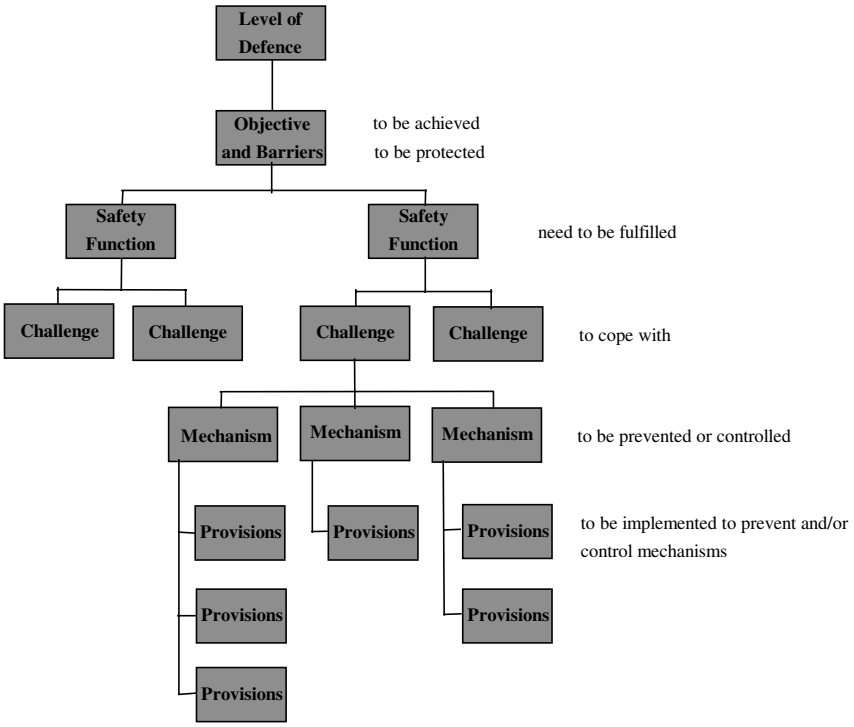
The objectives can be directly derived from those of Table 1. For example the main objective for Level 3 is to achieve the control of accidents within the design basis. This main objective can be developed and expressed in terms of more specific objectives such as: (a) limit damage to the fuel, (b) avoid any consequential damage to RCS and (c) maintain the confinement of radioactive products.

For each level of defence, the three main safety functions can be detailed into a consistent set of subfunctions that should be assured (e.g. control of reactivity: shut down the reactor, maintain the reactor in safe shutdown conditions, …).

For each subfunction the challenges to their fulfilment can be identified. These challenges are general processes or situations that can prevent adequate performance of the safety functions (e.g. reactivity excursions that could damage the fuel before shutdown). The challenges arise from a variety of mechanisms that have also to be identified. The identification of the mechanisms that can challenge the achievement of a safety function is an essential task in the development of the logical framework for inventorying the defence-in-depth capabilities of an NPP. Once the mechanisms are known, it is possible to determine the provisions necessary to prevent and/or control them.

This methodology is being currently applied at the Division of Nuclear Installation Safety of the IAEA to prepare the guidelines for the development of safety requirements for modular high temperature gas cooled reactors. If completed successfully, the same approach will be extended to other reactors, including research reactors.

Figure 5**. Defence-in-depth objective-provisions tree**




## Concluding remarks

Advanced Reactors will adopt design solutions that will enhance defence-in-depth features and will have a level of safety even higher than the best large NPPs currently in operation or being designed. The level of safety will be assessed in a more transparent way.

The safety requirements for the design of Advanced Reactors can be generated through a review process of the requirements established for NPPs currently in operation. This process will be based on a critical application of the strategy of defence-in-depth. The adoption of technical solutions allowed by the size of the reactor and the specific technology will probably lead to the enhancement of prevention by improving the strength of the first level of defence and the consequent relaxation of the requirements for expensive mitigative systems.

The use of small independent modules rather than a single large plant can also represent a viable solution to achieve simplification and cost reduction (e.g. heat removal function with simple passive systems).

The challenge for the future is to develop more confidence in the PSA tools and to demonstrate that sufficient defence-in-depth can be achieved through simpler and cheaper technological solutions. Risk informed decision making will play an important role in the development of future reactors of any kind. It will help to achieve high levels of safety and reduce costs, in

particular through simplification of safety systems and a sound and well balanced safety classification of structures, systems and components.

The standardisation, prefabrication and modularity of the facilities and the simplification of the licensing procedures through a certification process are suitable means to reduce the costs.

Additional benefits can be obtained through harmonising licensing criteria procedures used by the nuclear community to the greatest possible extent, based on worldwide scientific resolution of technical issues and accepted standards of safety adequacy.

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# DEFENSE-IN-DEPTH AND NEW REACTORS

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Defense-in-Depth (DID) is the structured approach to nuclear reactor safety that is at the basis of the safety features of the current generation of operating plants. This approach developed as a means of compensating for uncertainties in equipment and human performance, and it has evolved since the 1950s from its early use as a reactor safety guiding principle to its current broad, systematic application as an overall safety philosophy incorporating lessons learned from the current generation of operating reactors. The NRC white paper on risk-informed and performance based regulation [1] defines DID as "…an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. This philosophy ensures that safety will not be wholly dependent on any single element…The net effect of incorporating defense-in-depth …is that the facility…tends to be more tolerant of failures and external challenges."

In practical terms, DID results from the implementation of multiple measures to prevent and mitigate accidents, to contain their consequences, and to establish an acceptable balance between prevention and mitigation. Its pervasive application in reactor safety design and regulation is translated into many precepts and technical requirements of the current body of regulation (i.e., redundancy, diversity, separation…).

It is recognized that this approach to reactor safety has proven generally effective. For example, the implementation of containments and the consideration of surrogate bounding accidents in the design of reactor safeguards have provided us with protection in events that were once considered remote if not impossible, but have actually occurred or have become possible due to unexpected equipment malfunctions and operator errors. On the other hand, the absence of explicit consideration of probabilities associated with assumed scenarios and equipment failures has resulted in instances of inadequacies in the application of DID measures. Examples include the application of single failure in sequences where multiple failures are probable, and the narrow focus on safety-related systems and components.

Thus, lessons learned from the current generation of reactors suggest that for the next generation of reactor designs the DID philosophy should be retained, but its implementation should be guided by the probabilistic insights provided by a comprehensive probabilistic safety analysis (PSA). Since PSA may suggest strategies and tactics that differ from and occasionally conflict with prescriptive DID requirements, there is a need to clarify how DID requirements and risk insights will be integrated.

NRC's Option 3 approach to risk-inform the current body of light water reactor (LWR) regulation [2] provides a framework that could be adapted to new reactor designs. In this approach, quantitative safety objectives are established, and risk components are allocated to prevention and mitigation in order to establish a "reasonable" balance between these two strategies. DID tactics (i.e., redundancy, diversity, independence) are then used to implement these strategies. Where the expected balance between strategies cannot be achieved, or in the presence of large uncertainties regarding the effectiveness of one of the strategies, more stringent requirements may be imposed on the remaining strategy. In this approach, a comprehensive PSA is used to implement DID strategies and tactics and to demonstrate that the intended objectives are met, but risk information remains subsidiary to DID precepts.

Because of PSA's inherent ability to analyze, suggest effective design strategies to meet acceptance criteria, and quantify risk and uncertainties, a new risk-informed design and regulatory process has been proposed whereby PSA is the primary decision-making tool and DID is the aggregate of provisions made to compensate for uncertainty and incompleteness in our knowledge of accident initiation and progression [3,4]. This approach depends on the establishment of quantitative safety goals and on carrying out probabilistic safety analyses, including analyses of uncertainties. DID measures are selected to compensate for residual uncertainties and incompleteness after the capability of the analyses are exhausted. This approach is flexible and can be adapted to accommodate a high level DID approach and a low level risk-informed approach where DID strategies and tactics are used as desirable design attributes, but PSA guides how DID measures are ultimately defined and implemented.

Licensing of new reactor designs is likely to challenge some of the tenets of DID that have become institutions of the current generation of LWRs. A risk-informed approach where PSA is not subsidiary to DID precepts would provide a more flexible regulatory approach, better capable of accommodating new designs and concepts. But this approach would place expectations on PSA quality and scope that may be unachievable for designs that lack the extensive experience base of current LWRs.

As we proceed to review new proposed reactor designs, basic questions regarding the acceptability of proposed safety features will need to be answered, and the answers may depend on how DID requirements and risk insights are integrated, i.e., on whether PSA remains subsidiary to DID or vice versa:

- Will containment be required, or will confinement be acceptable for certain designs? How will we reach this decision? What role will risk information play in this decision?
- Can imbalance between prevention and mitigation be justified based on risk insights?
- If frequency-consequence curves are used as acceptance criteria, how will the expected balance between prevention and mitigation be assessed?
- How will design basis accidents be identified? Using the traditional approach of surrogate events bounding effects of concern with the design? Or will events be excluded based on probability considerations?
- How will the single failure criterion be modified to include consideration of multiple probable failures?

For each new proposed design, the answers to these questions will depend on how DID requirements and PSA insights are integrated. In defining an acceptable approach we need to recognize that the DID philosophy has served us well and elements of it must be retained for new

reactor designs. However, DID requirements should provide enough flexibility to allow new designs to capitalize on PSA insights and to introduce innovation in design and safety concepts. The main limit to the use of PSA should depend on the inherent limitations in the quality of the PSA due to the limited experience base of the new reactor designs.

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# BUILDING A SAFETY CASE FOR ADVANCED REACTOR DESIGNS

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US Nuclear Regulatory Commission, USA

## Abstract

The U.S. Nuclear Regulatory Commission (USNRC) is currently engaged in pre-application reviews for two advanced reactor designs: the AP1000, an advanced light water reactor; and the Pebble Bed Modular Reactor, a high temperature gas-cooled reactor. Two other pre-application reviews are planned for the GT-MHR, a high temperature gas-cooled reactor, and the IRIS, advanced light water reactor. These pre-application reviews provide for early interaction between the USNRC and the reactor designers to identify key safety and policy issues, propose paths for their resolution and establish a regulatory framework providing guidance on applicable requirements that are different from current requirements. The USNRC is also developing an advanced reactor research plan to identify research necessary to provide the data and analytical tools to support an independent assessment of the safety of these designs.

## Discussion

The U.S. Nuclear Regulatory Commission (USNRC) is currently conducting two pre-application reviews for advanced reactor designs: the AP-1000, an advanced light water reactor; and the Pebble Bed Modular Reactor (PBMR), a high temperature gas-cooled reactor (HTGR). Two other pre-applications reviews are planned for the Gas Turbine-Modular Helium Reactor (GT-MHR), also an HTGR, and the International Reactor Innovative and Secure (IRIS), an advanced light water reactor. These reviews are being conducted under the framework of the USNRC's Advanced Reactor Policy Statement [1], which was first issued in 1986 and reissued in 1994. This statement sets forth the Commission's policy regarding the review of, and desired characteristics associated with, advanced reactors. For purposes of this policy statement, advanced reactors are considered those reactors that are significantly different from current generation light water reactors and includes reactors that provide enhanced margins of safety or use simplified or other innovative means to accomplish their safety function. The Commission expects that as a minimum, these advanced reactors will provide the same degree of protection of the public and the environment required for the current generation light water reactors, but also expects that they "will provide enhanced margins of safety and/or utilize simplified, inherent, passive or other innovative means to accomplish their safety function." Among the specific attributes that the Commission believes should be considered in advanced designs are: highly reliable and less complex shutdown and decay heat removal systems; longer time constants to allow more time before reaching adverse conditions; simplified safety systems; designs that incorporate defense-in-depth; and designs based on existing technology or a suitable technology development program. It is believed that incorporation of some or all of these attributes may assist in establishing acceptability or licensability of a proposed design with minimum regulatory burden and help in the understanding by the public.

Because advanced reactors are likely to have characteristics and features that are different from the existing generation of light water reactors, it is recognized that new or modified regulatory guidance may be needed and that new design features may require a commitment to a suitable technology development program to support their safety case. Accordingly, the Advanced Reactor Policy Statement encourages early interactions between the regulator and the applicant and/or designers to facilitate the early identification of safety and regulatory issues and to identify possible paths for their resolution. It is this early interaction that the USNRC is currently engaged in for the AP1000 and PBMR advanced reactor designs.

The Westinghouse AP-1000 design is based on the AP-600 advanced light water design that the USNRC previously reviewed and certified under 10 CFR Part 52, "Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants" [2]. As part of the pre-application review of the AP1000, the USNRC is assessing the applicability of the AP600 test program and analysis codes to the AP1000 design. This early identification of technical and safety issues, along with any necessary technology development programs, will support a decision by Westinghouse regarding the feasibility of seeking a design certification for the AP1000. Because it is estimated that about 80% of the AP1000 design is similar to the AP600 design, the safety review efforts are expected to be less challenging than for other advanced reactors such as the PBMR.

The PBMR is a modular HTGR under development in the Republic of South Africa and is being considered for licensing in the United States by Exelon Generation, USA. The proposed design includes certain innovative aspects of design, technology, and operating characteristics that are unique to the PBMR. As such, because many of the current USNRC reactor regulations are specific to light water reactors, they may not be applicable to the PBMR. Likewise, due to the different technology and approach to safety employed by the PBMR, new requirements will likely be necessary in some areas.

Because of the more limited operating experience with HTGR technology, the pre-application activities for the PBMR include a preliminary assessment of both HTGR technology and PBMR specific technology to identify key safety and policy issues. HTGRs, such as the PBMR, involve characteristics that make their approach to protecting public health and safety very different from reactor designs currently licensed in the United States. For example, when considering the traditional layers of defense-in-depth, modular HTGRs typically shift the emphasis from mitigation features to highly reliable prevention features. Specifically, the PBMR proposes to shift much of the containment function to fuel capable of withstanding high temperatures and to rely on simpler and more passive decay heat removal processes that also rely on high temperature material behavior (e.g, graphite). These and other differences from current LWRs are expected to lead to a number of safety, technology and policy issues. Fuel performance, high temperature material performance and containment vs. confinement are examples of issues will need to be addressed.

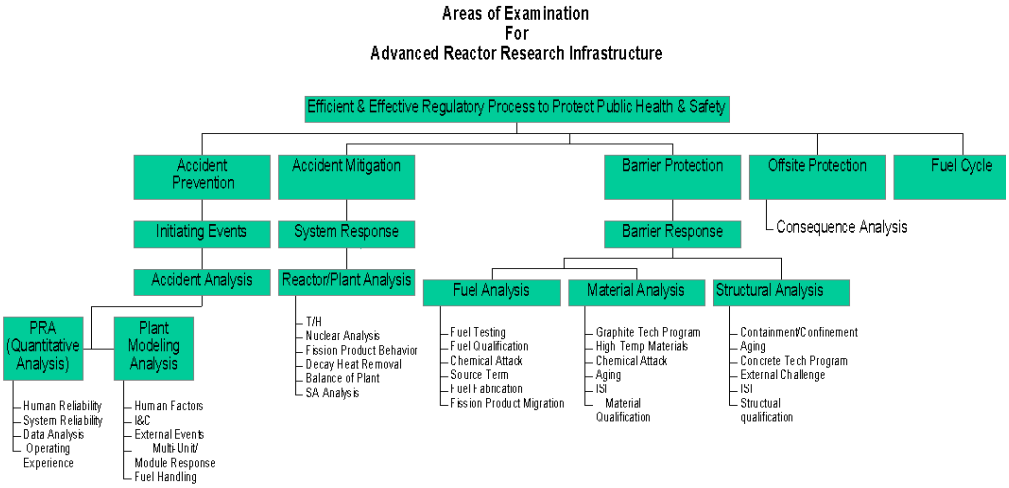
The eventual resolution of these safety and policy issues is only possible if there is a sufficient understanding of the basic technologies involved in the designs. Therefore, a key part of building a safety case for advanced reactor designs is to identify the necessary technology infrastructure needed to review an actual application. While it is the designer/applicants responsibility to conduct the research necessary to support its application, the USNRC also conducts safety research necessary to support its regulatory decisions and to provide an independent confirmation of the key elements of an applicant's safety case. It is important that the USNRC have an independent capability to verify the plant response to accidents, particularly those related to loss of coolant, decay heat removal, and reactivity insertion. Such independent capability is valuable in providing a deeper understanding of plant behavior under a wide range of off-normal conditions, which can result in insights that contribute to the quality and thoroughness of the safety review. It is through this independent research and analysis that there can be greater public confidence in the ultimate safety of these advanced designs.

In an effort to assess the adequacy of the technology supporting these advanced designs and to guide its research program, the USNRC is developing an advanced reactor research plan. This plan will assess key research areas and identify specific research topics and layout a road map for the USNRC's research program over the next several years. A systematic and structured (top-down) approach is being used to identify research needs for the safety review of advanced reactors (Figure 1). This effort also is taking into consideration ongoing research initiatives in the international arena, as well as opportunities for future cooperation. In support of this effort, in October 2001, a two-and-a-half day workshop was held at the USNRC that focused on HTGRs. National and international experts discussed a wide range of topical areas and identified research topics that were considered to be high priority. These research topics include: high-temperature material performance; nuclear-grade graphite behavior; fuel performance; containment performance; adequacy of data and analytical tools; and accident scenarios [3]. The priorities given to various research areas will reflect research needs for those designs that the USNRC is currently reviewing or expects to review in the near term. Thus, our current focus is on three general topics, namely, high-temperature gas cooled reactors, advanced light water reactors and regulatory framework developmental activities to assure the needed predictability and versatility in the longer term.

As discussed above, for advanced reactors different from current light water reactors, such as the PBMR, certain reactor regulations may not be applicable to design features of the advanced reactors. Likewise, new requirements may also be needed to address the different approach to safety proposed by these new designs. In the case of the PBMR, Exelon has proposed to use risk-informed approaches to support its safety case at the NRC [4]. They are proposing to use existing regulations, where existing regulations may apply or partially apply, and look to the USNRC to develop new regulatory criteria where existing regulations do not cover the safety review needs. The screening

process proposes to use PSA technology to identify such new regulatory criteria. Their proposal is currently under review, but a number of issues will need to be considered to successfully implement this approach, including the role of the defense-in-depth to address uncertainties (including the issue of containment vs. confinement), limited or no operating experience for use in a PSA, identification of or need for appropriate risk metrics (CDF and LERF) and the selection of licensing and design basis events.

**Draft**




In summary, through the pre-application reviews of the AP1000 and PBMR, the USNRC is actively engaged with the reactor designer/applicants to achieve an early identification and resolution of safety issues. At the same time, the USNRC is developing the necessary technical infrastructure to support its regulatory decisions.

## References


1. U.S. Federal Register Notice, "Regulation of Advanced Nuclear Power Plants; Statement of Policy," July 8, 1986, 51 FR 24643 and revised July 12, 1994, 59 FR 35461.
2. U.S. Code of Federal Regulations (CFR), Title 10, Part 52, "Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants," April 18, 1989.
3. Summary Report, Workshop on High-Temperature Gas-Cooled Reactor Safety and Research Issues, October 10-12, 2001, USNRC, Washington, DC (Adams Accession ML013650004).
4. Letter from K. Borton to US Nuclear Regulatory Commission, "Exelon Generation Company's Proposed Licensing Approach for the Pebble Bed Modular Reactor (PBMR) in the United States," dated August 31, 2001.

# SAFETY OF HTR – STATE OF KNOWLEDGE AND NECESSARY RESEARCH

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## 1. New requirements on safety of future nuclear power plants

World-wide there are many projects to design nuclear reactors for future applications, which offer more safety compared to today's systems. In Germany as an example, the modified atomic power act from 1994 requires a very high safety standard for future plants, the total radioactivity must be retained inside the power plant in all cases of accidents, independently from any probabilities.

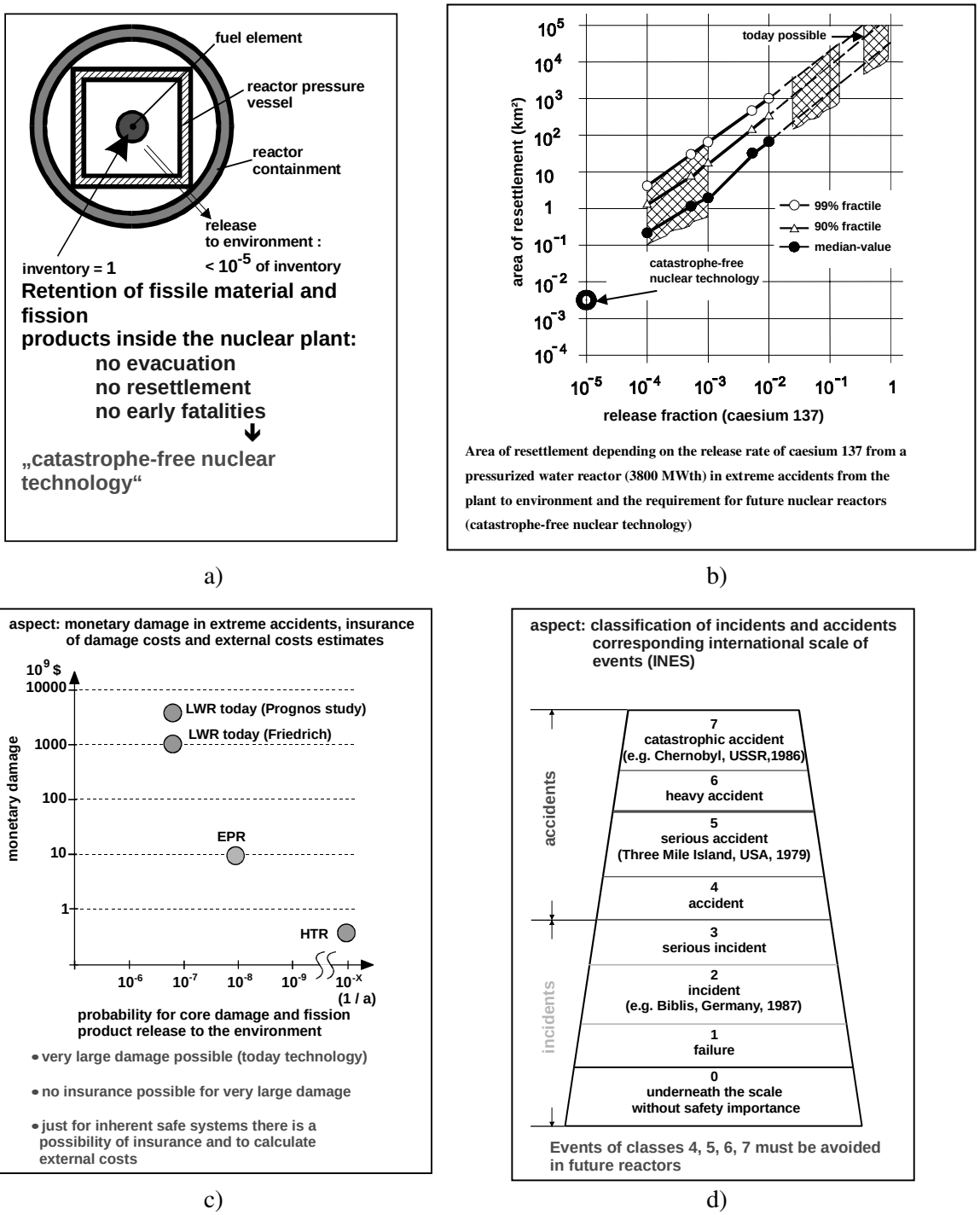
Evacuation and relocation of people must not be necessary after accidents, no intolerable radiological consequences are allowed to occur outside the nuclear power plants. This type of technology, which fulfils these requirements, can be named „catastrophe-free nuclear technology". For the case of a reactor with a thermal power of 300 MW which is explained later on, as an example, the amount of release should be less than of $10^{-5}$ of the fission product inventory inside the fuel elements (Figure 1a).

The safety requirements of this future nuclear reactor technology and the importance to realize "catastrophe-free" nuclear technology shall be explained additionally by Figure 1b. Dependent on the total release of radioactive isotopes with long half-life time ($^{137}$Cs with 30 years as an example) the contamination and loss of land could be large in case of today's nuclear reactors. This contamination must be avoided.

This loss of land and people causes a very large monetary damage too (Figure 1c). In these numbers the results of the German risk study on light water reactors where used to calculate numbers including cost figures for loss of land, infrastructure, and loss of people (death and cancer). The monetary damage for severe melting accident with this extreme radiological consequences would be very large. Certainly there are some differences in the height of monetary damage caused by the chosen assumptions. If the EPR concept with the retention of a molten core inside the reactor containment could be realized the damage would be reduced very much in future, but still would be very high because of loss of plant and the necessity to remove a molten core. Additionally the probability of this accidents is considered not to be smaller than $10^{-8}$ year$^{-1}$.

Figure 1. **Safety requirements for future nuclear reactors**

a) New requirement for retention of fission products inside the nuclear power plants (following the German Atomic law from 1994)
b) Area of resettlement caused by release of caesium 137 to the environment and contaminated area (following results of German risk study, PWR 3800 $MW_{th}$)
c) Monetary damage caused by severe nuclear accidents dependent on probability of occurrence
d) The INES-scale for nuclear incidents and accidents (IAEO, Vienna, 1990)



Just if it is possible to realize “catastrophe-free” nuclear technology and to retain the fission products inside the fuel elements and if the plant has a small power, the monetary damage would be relative small. Just in this case the calculation of external costs of nuclear energy becomes possible and an insurance for the damage caused by accidents of nuclear power plants will be a realistic possibility. Insurance of nuclear power plants and really small external costs are preconditions for the future acceptance of nuclear energy.

Catastrophe-free nuclear technology is achieved if the radioactive substances remain contained inside the reactor plant in all possible cases of accidents so that no significant radiological consequences will result for the environment, i.e., no immediate fatalities, no late fatalities, no evacuation, no resettlement, and no changes in eating and drinking habits. This corresponds to the requirement of the IAEO (Figure 1d) too, and means that the categories 5, 6 and 7 must not occur.

The principle of catastrophe-free nuclear technology as defined above is applied world-wide already today, when spent fuel elements and vitrified wastes undergo dry interim storage in cast-iron or steel containers which are cooled by free convection of air and radiation of heat.

## 2. HTR-concept with inherent safety

The principle of inherent safe reactors shall now be explained on behalf of a special design for the HTR. This conceptual design of a reactor completely excludes the melting of the core and the release of fission products from the fuel elements is almost impossible so far as the fuel temperature stays below 1600°C. The key element for safety behaviour is the spherical fuel element with excellent fission product retention in normal operation and in all accident condition.

TRISO-Coated fuel particles (Figure 2a) are embedded in spherical graphite fuel elements, they will practically completely retain the fission products during accidents up to a fuel temperature of 1600°C, as confirmed by detailed experiments (see Figure 2b). The fuel elements furthermore, can be designed in future corrosion resistant to air and water vapour due to additional silicon carbide coatings of 100 to 200 µm thickness on the surface, these are under development.

Figure 3 shows the concept of such a high temperature reactor with a thermal power of 300 MW which is characterized by the following special features: an annular core, coated particle fuel (TRISO) which is embeded into spherical graphite fuel elements which entirely retain fission products up to a temperature of 1600°C in all accidents over a very long time. The primary reactor system is arranged in prestressed pressure vessels which cannot burst. Destructive mechanical influences on the core are therefore impossible. The decay-heat removal from the reactor plant functions in a self-reliant way only by help of heat conduction, heat radiation and natural convection of air. The primary system is contained in an inner concrete cell, which is connected to a filter and a stack. By means of a flap and a self-acting closure system, consisting of a granulat silo for instance, which can flow downwards after depressurization of the primary system, the amount of air in this inner concrete cell is very limited. Analysis carried out until now show, that a concept like this obviously meets with the demands of a catastrophe-free nuclear technology, as far as four principles of stability are fulfilled: nuclear, thermal, chemical and mechanical stability. As the following chapter show, this is indead the case in this design.

Nuclear stability is reached by means of a self-reliant restriction of the nuclear power production as well as of the fuel element temperatures in severe reactivity accidents by an appropriate lay-out and dimensioning of the core. Permanently strong negative feed-back coefficients of

temperature and a minimal excess reactivity insice the core are preconditions. This is possible because of the continous loading of fuel elements which avoids excess reactivity for burn-up compensation.

Thermal stability is gained by means of realizing a self-reliant decay heat removal. This is realized without overheating the fuel elements caused by a low power density of the core, large heat capacity of the core, short ways of heat transportation in the core, application of temperature-resistant ceramic materials, high values of heat conduction in the core area and by permanent outer heat sinks.

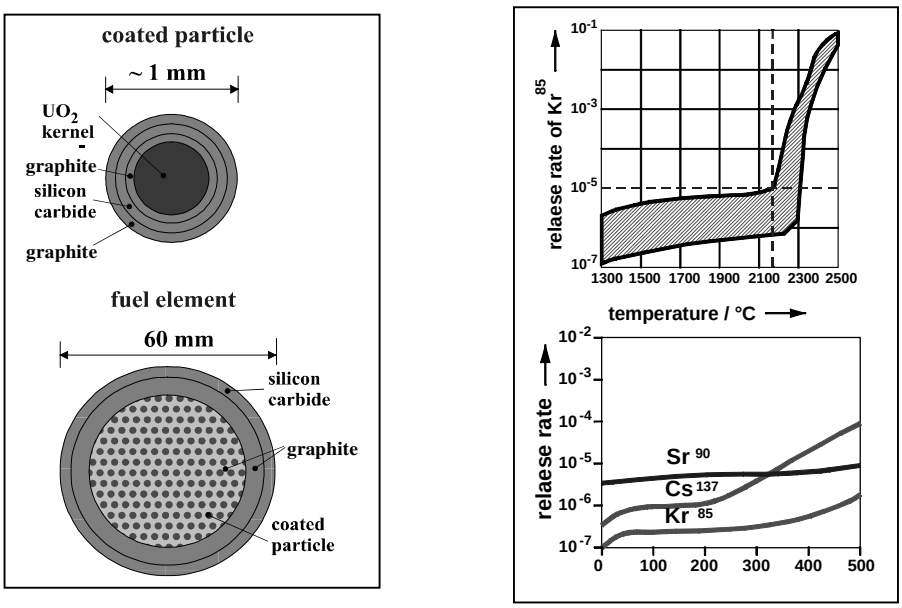
Chemical stability is realized by the very effective limitation of air ingress into the primary circuit in accident. This is gained by a prestressed design of the primary system and by the limitation of air, which is available to corrosion processes, to the content of the inner concrete cell inside the reactor building. This is realized by self-reliant mechanismen.

The principle of mechanical stability is guaranteed by the application of a prestressed primary enclosure which cannot burst. A lot of axial and radial tendons cause compressive stresses in the walls of the vessel system. Therefore, cracks never can grow. Overpressure is reduced self-actingly by the design of the vessel structure, i.e. the opening of welding lips.

The reactor should be arranged underground to exclude damages form extereme impacts from the outside (Figure 4). The thickness of earth and other materials above the reactor building depends on the assumptions on the accidents.

Figure 2. **Future high-temperature reactors with enhanced safety**

a) Coated particle (TRISO) and spherical fuel element.
b) Fission product release from spherical fuel elements in severe accidents: dependence on temperature (Kr 85) and time (Sr 90, Cs 137, Kr 85) at 1 600°C.




a) b)

Figure 3. **Future high-temperature reactors with enhanced safety**

Concept of an inherently safe HTR: ISR 300 (Inherently Safe Reactor with 300 $MW_{th}$), (1) annular core, (2) prestressed reactor pressure vessel, (3) prestressed vessel for steam generator, (4) prestressed connecting vessel, (5) control- and shut off system, (6) helium circulator, (7) hot gas duct, (8) steam generator, (9) cell cooler, (10) inner concrete cell, (11) reactor building, (12) fuel loading device, (13) fuel discharge device, (14) flap, (15) concrete closure, (16) storage vessel for spent fuel elements

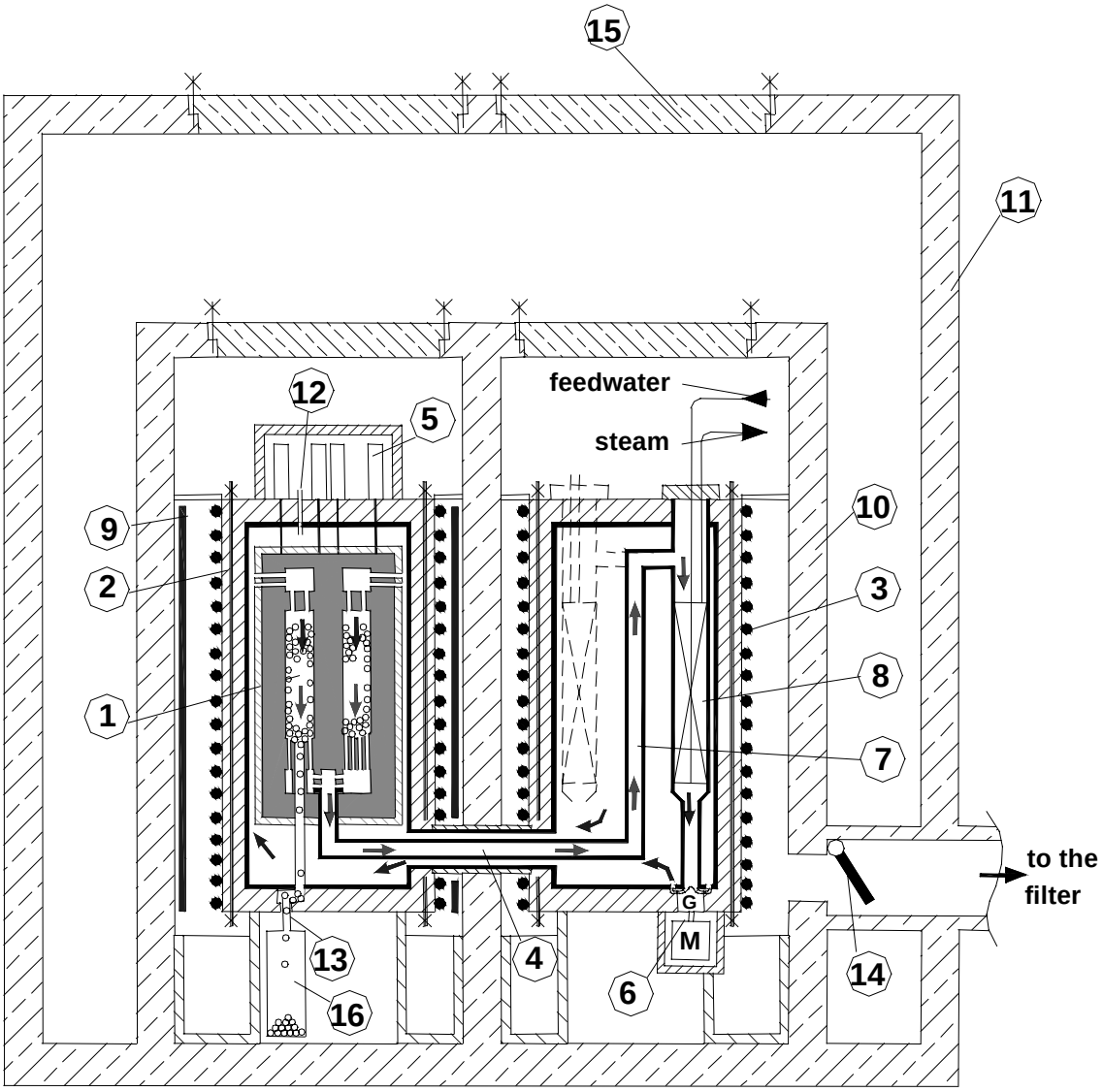
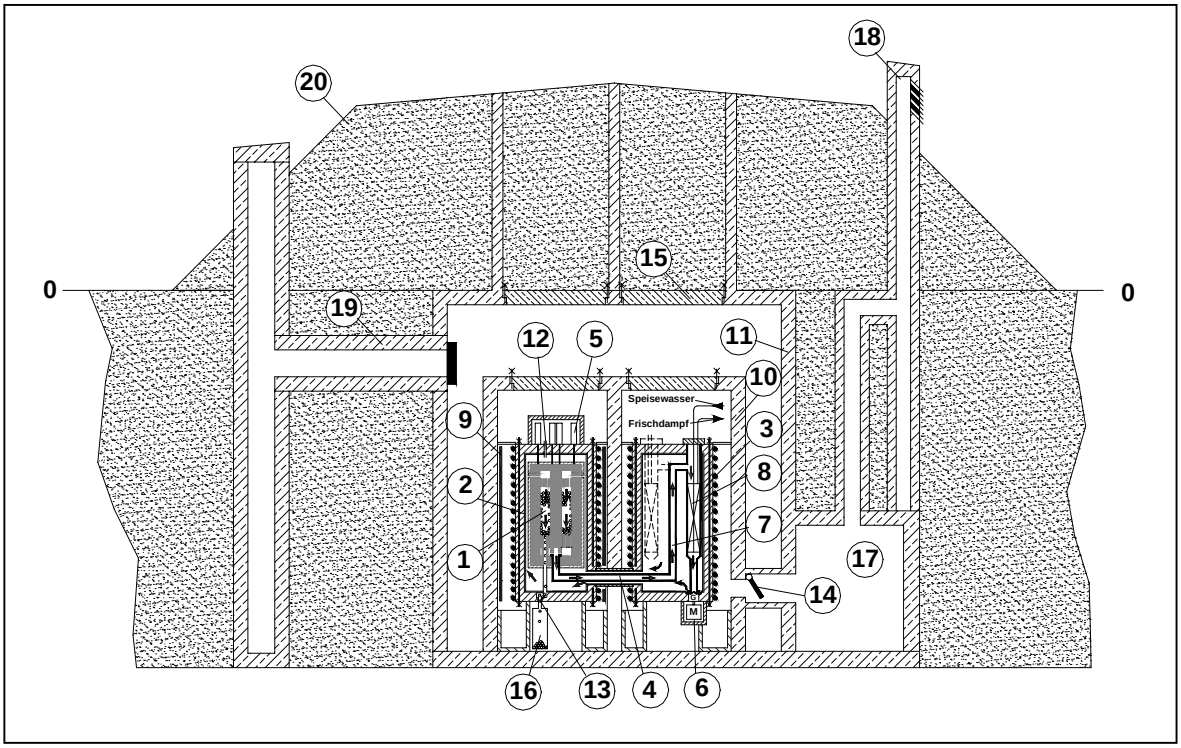


Figure 4. **Future high-temperature reactor with enhanced safety (ISR 300): underground and protected siting of reactor building**

((1) to (16) see Figure 2c, (17) sedimentation area + filter, (18) stack with filters, (19) transport channel, (20) cover of reactor building




## 3. Safety aspects of modular HTR

The range of catastrophe-free nuclear technology must be defined. Following Figure 5a there are accidents from internal and from external reasons which can be foreseen and which are covered by the licensing process. All these accidents must not cause unallowed release of radioactivity from the plant. Category 3 contains some accidents which are far beyond today's licensing procedure, they require partly new measures of design, e.g. underground siting against terroristic attack.

Modular HTR use the principle of self-acting decay heat removal and the limitation of fuel temperatures below allowed values, which are so low that nearly no fission product release occurs (1600°C). In this reactor the decay heat is transported from the core, through the core structures and from there to an outer heat sink (water cooled surface cooler or finally concrete structures) just by radiation, conduction and natural convection or air (see Figure 5b).

The figure shows maximum fuel temperature in the core dependent on time, if the usual assumption of loss of coolant and loss of active decay heat removal is made. This corresponds to the case of the 300 $MW_{th}$ core in Figure 3. Just few fuel elements are at the high temperatures for a short time. Therefore the fission products release is very small, limited to values lesser than $10^{-5}$ of the

inventory. A reactor, which fulfils the principle of self-acting decay heat removal is called a thermally stable reactor too.

The principle of self-acting decay heat removal and the limitation of fuel temperatures to acceptable values (< 1600°C) is even fulfilled in very extreme situations as Figure 5c indicates. If the reactor vessel totally is covered with rubble – as a result of a very extreme earthquake or of a terrorist attack – even then the principle of self-acting decay heat removal works and the maximum fuel temperature is limited. Certainly it takes more time, before the temperatures drop again, but finally the decay heat is released to the environment. This means that even very extreme accidents do not cause catastrophic releases of fission products. The temperature of the vessel system stays below 500°C in this accident, there is no internal helium pressure in the system. Detailed analysis show that the release of fission products from the fuel elements during such an accident stays below an integral value of less than $10^{-5}$ of the inventory. Additionally there is deposition of fission product inside the primary system and in the reactor building.

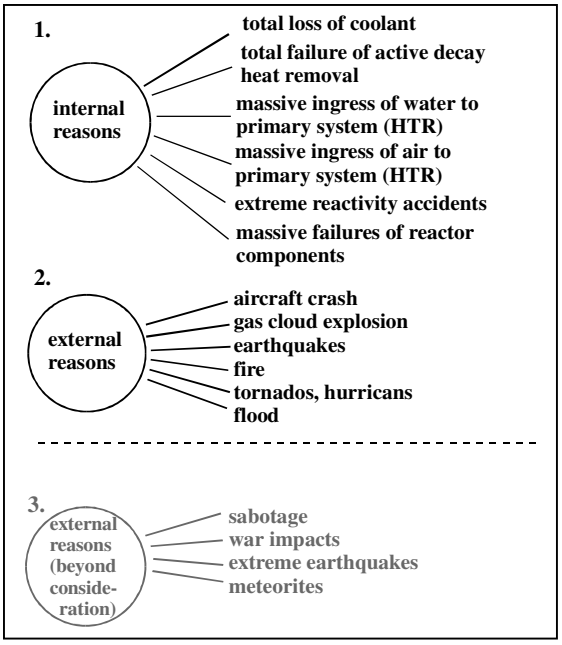
Very extreme accidents of reactivity have been considered for modular HTR. Results in Figure 5d are valid for a 200 $MW_{th}$ plant, but the results will be similar for the 300 $MW_{th}$-core mentioned here. Self-acting limitation of power and fuel temperatures in nuclear transients is guaranteed by nuclear stability: Because of the continuous loading and disloading of the reactor there is practically no excess reactivity for burn-up compensation in the core. The prompt temperature coefficient of the system is strongly negative. The behaviour of TRISO coated particles embedded into graphite is very well suited related to extreme reactivity accidents too. Although there is no physical reason for this assumption, it was assumed that a sudden loss of all control rods of the first control system in a very short time would happen. The result is that the maximal fuel temperature stays below 1600°C even for this very extreme assumptions. If the second shut down system would be lost totally from a cold state of core in a short time the maximum fuel temperatures would stay below 1600°C too.

An overpressure in the primary cooling circuit is reduced by safety valves, rupture discs and finally by the concept of the vessel itself. The welding lips would open and reduce the pressure. This safety behaviour of the vessel has been tested and verified.

Figure 6a shows considerations on the accident “air ingress”. The ingress of large amounts of air into the primary system is impossible because of use of prestressed primary vessel system. Only very small amounts of air can enter the core because possible openings in the primary system are limited (diameter of openings < 65 mm, few openings have 250 mm diameter). Burn-up of matrix graphite by air and release of fission products by the accident “ingress of air into the primary system” is negligible small, because the amount of air available for corrosion processes is very limited. This is the content of the inner concrete cell of around 5000 $m^3$ of air, which allows the corrosion of maximum 500 kg graphite. After a depressurisation accident the concrete cell is closed by self-acting measures (flap, granulat silo). Simple interventions can be carried out to avoid even this relatively small corrosion damage of graphite structures, because the reactor building is free of contamination and therefore access to the cell is possible. The openings in the primary system can be closed by sand, foam or other suited materials. Therefore, a protection of investment is possible. Air ingress accidents in totally are not important for the modular HTR following the concept shown here.

Figure 5. **Safety aspects of modular HTR**

a) Overview on accidents.
b) Behaviour of reactor in case of loss of total cooling.
c) Behaviour of reactor in case of loss of total cooling and total destruction of reactorbuilding.
d) Behaviour of reactor in extreme reactivity accident: fast loss of total first shut down system.

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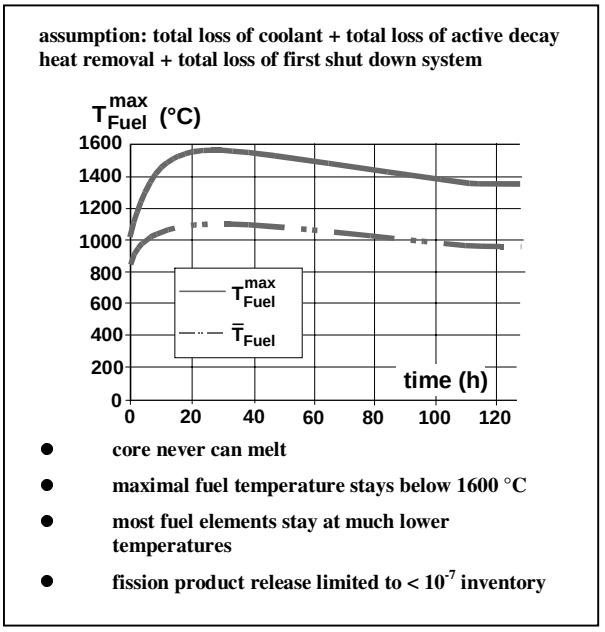


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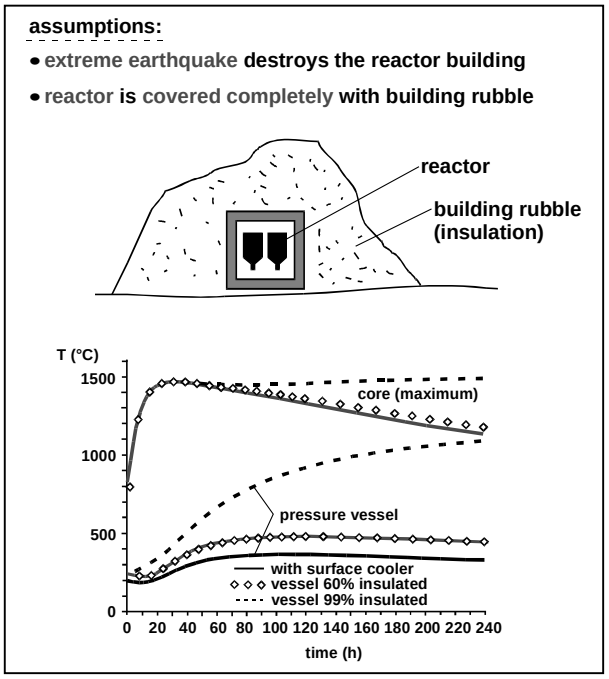


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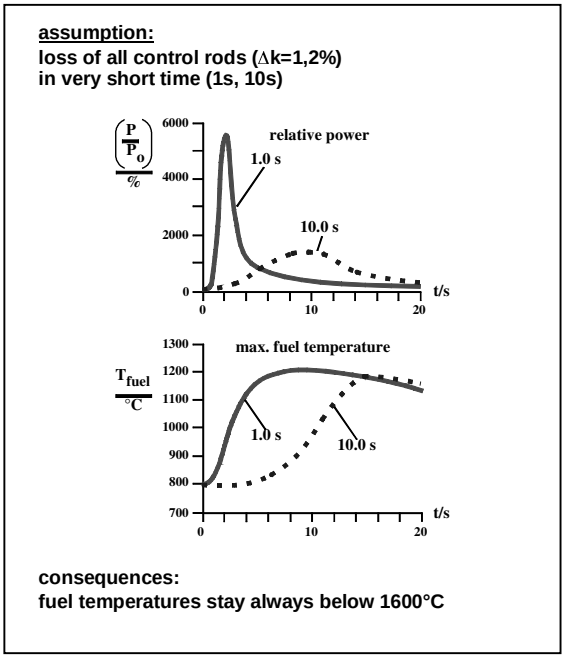


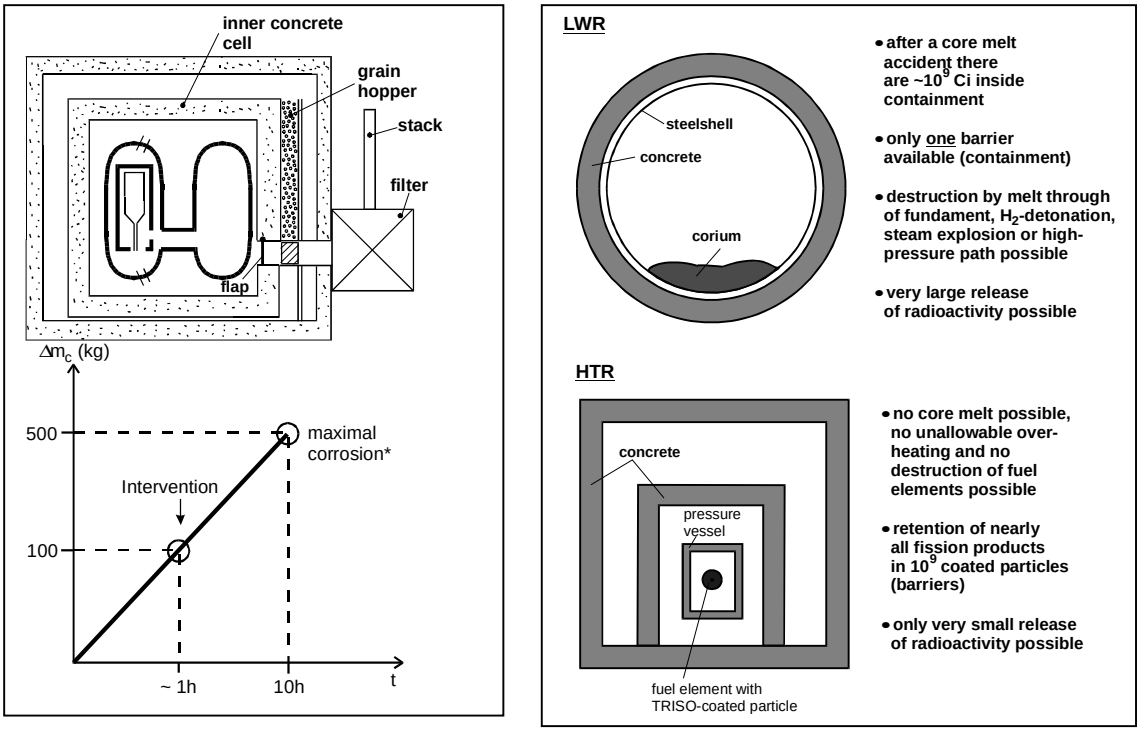

d)

The HTR containment must not be dense as that of LWR (see Figure 6b). The real enclosures for the fission products in all accidents are the $10^9$ coated particles inside the fuel elements. They cannot be destroyed by accidents and have a very good retention capability for the fission products and fissile material. The HTR containment however, has to be designed against outer impacts as the LWR containment too, that means a wall thickness of concrete of 2 m as used for the last nuclear power plants in Germany, as an example.

Overall the release rates of fission products are very small. The filter can retain additionally nearly all the solid and aerosol type fission products, which have been released from the fuel elements during heat up accidents. Following the experience from HTR plants and the heating-up experiments for spherical fuel elements with TRISO particles, the final release to environment is very small as required in Chapter 1. The inventory of $^{137}Cs$ e.g., is 2.5 x $10^7$ GBq, from this less than 10 GBq would be released into the environment including the action of the filter.

Figure 6. **Safety aspects of modular HTR**

a) Behaviour of the reactor in case of air ingress.
b) Aspects of containment of HTR and LWR.




a) b)

Overall the following conclusions regarding safety of this modulare HTR are valid:

- the fuel never will melt; no super-heating above 1600°C possible;
- the self-acting decay heat removal cannot fail;
- the reactor would withstand even large reactivity transiens; temperatures stay below 1600°C;
- the reactor vessel cannot burst: no large air ingress possible; no deformation of core or change of composition possible;
- large water ingress into the primary circuit causes no problems;
- the reactor building withstands standard outer impacts;
- even against extreme impacts from outside there are large safety margins.

The consequence of this behaviour is, that no non-allowable large fission product release is possilbe in case of accidents; requirements of "catastrophe-free" nuclear technology are fulfilled.

## 4. Knowledge and necessary research on safety relevant topics of HTR

The main aspects of safety behaviour of modular HTR have been investigated very well. The following topics described here, are clear today. Some safety questions which need further deepening are explained at the end of this chapter.

By the operation of HTR plants and by accompaning broad safety research programmes of the last decades the state of knowledge is as following:

- spherical fuel elements with TRISO coated particles retain the fission products inside the particles very effectively till accident temperature of around 1600°C for a long time, for several hundred hours;
- the self-acting decay heat removal in case of total loss of cooling from the core through the reactor structures and from there via the surface of reactor pressure vessel to an permanent outer heat sink is proven by large experiments and by tests in the AVR. The main parameters governing the heat transport in this chain – all transport steps just by conduction, radiation and free convection of air – are well known;
- all well designed HTR have a strong negative temperature coefficient of reactivity, this is proven by experiments in all HTR plants. The very remarkable experiment in the AVR (blockade of all shut down rods) should be mentioned here;
- the very robust behaviour of fuel elements and core in dynamic events has been proven;
- the effects of corrosion of fuel elements by air or water in case of ingress of these media into the primary circuit is known; the conditions of these accidents have been tested in large test facilities;
- the integral behaviour of fuel elements has been tested by operation of the AVR and THTR – in all one million fuel elements – and the overall experience is very good. The fuel quality has reached a very high standard. Especially the quality of the coatings was very good, the free uranium contamination of the graphite matrix was very small for the spherical fuel elements in the last step of development;

- burst protected prestressed reactor pressure vessels have been developed (concrete, steel or sphero-cast iron) and partly qualified in operation (concrete);
- the technology of very effective filters for solid fission products and aerosols is available and can be applied to HTR systems with low depressurization rates in loss of coolant accidents;
- underground siting for small HTR plants has been already studied extensively in 1970 and was considered feasible already at that time . Important advantages for the safety concept of modular HTR were recognised at that time.

Of course there are some topics which should be part of further developments and research and should be deepened:

- The quality of fuel and the qualification of fuel before and during operation need further attention; a further progress could be possible by introduction of new layers in coated particles to get even higher retention of fission products.
- Plutonium-coated particles are necessary for reactors to burn the weapon grade plutonium or plutonium from reprocessing effectively.
- The development of corrosion resistant fuel elements using thin SiC-layers is promising, it needs a broad development- and test-programme.
- Further experiments to deepen the knowledge on air ingress to primary circuit and the consequences for fuel elements and graphite structures can be done are necessary for all HTR concepts which use “normal” forged steel vessels for the primary enclosure.
- Detailed work on prestressed burst protected vessels is necessary.
- Intervention methods to minimize the consequences of air and water ingress into the primary system needs further effort; they help to protect the investment of plants.
- Tests on safety relevant components at high temperatures (shutdown rods, KLAK-systems) should be carried out at real conditions.
- Work related to the HTR containment and its characteristic specifications regarding all accidents is necessary.
- Specific questions of underground siting (mechanical impacts, retention of fission products in soil structures) should be deepened.
- Non-proliferation aspects of HTR fuel cycle are very important topics of further research.
- Specific questions of safety in connection with the gas turbine application (dynamic behaviour, specific accidents) are important to be solved.
- Final direct storage of spent HTR fuel elements requires additional work; especially continuation of leaching experiments and development of additional ceramic protection layers for the spent fuel elements.

## 5. Conclusion

Modular HTR plants can be realized with inherent safety. The reactor system can be designed to fulfil the requirements of catastrophe-free nuclear technology: the fission product release to the environment in all accidents is restricted to less than $10^{-5}$ of the inventory. The core never can melt or overheat to unallowed temperatures above 1600°C. By the use of an annular core a thermal power rating of 300 MW or even more can be realized. This safety behaviour can be demonstrated in an integral full scale-experiment. This experiment can be carried out with a real reactor too, because the core will not be destroyed by these experiments and thus the consequences of the accident demonstration are insignificant.

Additionally to the application for steam generation, the reactor system can be used for coupling with gas turbine cycle, combined cycle, process heat applications. Especially for co-generation processes this reactor is very attractive, because of the medium power. The power plant is economically attractive compared to other future options, including world market coal, especially if a series production is realized and a modular arrangement is used for plants. Furthermore, world-wide fabrication of components will reduce the cost remarkable. The world market to this type of reactor with the broad spectrum of application is large. Nevertheless, international co-operation is necessary to establish this type of reactor technology world-wide.

# SAFETY ISSUES AND APPROACH FOR LIQUID-METAL REACTORS

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**Abstract**

In the present paper, the generic safety characteristics of LMRs are discussed first. Sodium-cooled LMRs are mainly reviewed because of their long history and experience. Secondly historical perspective is reviewed briefly to discuss the safety approaches taken in the previous LMRs, designed, licensed or actually built. Finally recommendations are given for a safety approach for future advanced LMRs. We believe a fast-neutron LMR concept is regarded as one of the most-promising candidates for next-generation advanced nuclear reactors for which we should further investigate in the future.

## 1. Introduction

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

In the present paper, the generic safety characteristics of LMRs are reviewed in Section 2. Sodium-cooled LMRs are mainly discussed in this paper. Secondly historical perspective is reviewed in Section 3 briefly to discuss the safety approaches taken in the previous LMRs, designed, licensed or actually built. Finally recommendations are given for a safety approach for future advanced LMRs.

## 2. Key safety characteristics of LMRs

A basic safety approach in designing LMRs is essentially the same as one taken in light water reactors (LWRs). Namely a reactor system shall be designed based on a defense-in-depth concept, with a primary emphasis on preventing and detecting abnormal occurrences. Then safety design measures shall provide, under postulated abnormal conditions, appropriate means to shut the reactor down, cool the residual heat in the reactor core and contain radioactive materials within the

reactor facility. Different levels of logical defense lines are considered in the defense-in-depth concept. If we take a definition of International Atomic Energy Agency, this concept for the current-generation nuclear power plants consists of the following five levels [1,2]:

- Level-1: Prevent abnormal occurrences,
- Level-2: Control abnormal operation,
- Level-3: Control accident and contain radioactive materials,
- Level-4: Manage severe accidents, and
- Level-5: Mitigate off-site consequences.

The thorough discussions on the defense-in-depth concept should be prepared in a different paper of the Workshop, and hence are not repeated here. We should at least note that the design basis events (DBEs) are considered normally up to Level-3. The safety design should be assessed in a sufficiently conservative way. Level-4 is to provide an additional safety margin to the plant to prevent the accident development to severe accidents and to mitigate their consequences. Level-5 obviously includes administrative actions for on- and off-site emergency procedures, as well as the plant safety design consideration to facilitate such actions.

Even though the philosophy involved in the defense-in-depth concept has been universally accepted, what are technically included in it should not be regarded as solid static entities. An innovative reactor design with effective passive features to prevent and mitigate severe accidents may eliminate the need for off-site emergency procedure in Level-5. This final point will be discussed in Section 4.

The DBEs, commonly consisting of anticipated operational occurrences and (design basis) accidents, are postulated for safety analyses to confirm the validity of plant safety features and their functions in a conservative way. For LMRs, with a low pressure system and single-phase coolant system, the sequences of DBEs are rather benign. Even an incident of severe coolant leakage is unlike to lead to rapid core uncovery. Hence there is normally no need to provide an emergency core cooling system that is commonly equipped in high-pressure LWR systems. The specific safety characteristics of coolant, however, must be carefully taken into account in LMRs. For example, chemical reactions of sodium are usually treated in safety analyses, since their consequences can be energetic when it comes into contact with air or water upon coolant boundary failure.

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

## 3. Historical perspective of safety approach

### *3.1 Safety approaches in prototype LMRs*

The development of sodium-cooled fast reactors has a long history. The purpose of this paper is not to provide an entire review of the history but to discuss the historical perspective and the present status on the safety approach generally taken in the past. Thus we focus more on the recent LMR plants, actually designed, built and or operated during the 1970s and 1980s. In this section, the safety approaches are briefly reviewed for the four typical sodium-cooled fast reactors: Superphenix in France, SNR-300 in Germany, CRBRP in the United States and Monju in Japan.

*Superphenix* [3, 4]

Superphenix is a large pool-type commercial-size fast reactor of a 1 240 MWe output. The design of the plant was made feasible from French experience in the former two fast reactors, Rapsodie and Phenix. The basic safety approach is to provide safety design measures based on a defense-in-depth concept. A special characteristic is that large hydraulic and thermal inertia of the plant provides an inherently stable plant dynamics, as well as the negative reactivity feedback due to radial expansion of fuel and blanket subassemblies. For instance, a long grace time is available for more than 10 minutes before coolant boiling even in the event of unprotected (without scram) accidents.

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

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

*SNR-300* [5-7]

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

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

*CRBRP* [8-10]

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

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

*Monju* [11, 12]

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

The Monju plant was licensed and plant construction was completed. The operation of the plant was stopped since 1995 when the sodium leakage accident occurred in the secondary cooling system. A licensing procedure was initiated in 2001 to improve the plant against sodium leakage events.

*Summary*

In a word, a rather coherent safety approach was taken in those LMRs developed in the 1970s and 1980s. Namely, these plants were designed based on the defense-in-depth principles with appropriate consideration on sodium reactions. Even though their early designs consider CDAs directly in the safety design, the treatment in safety evaluation is different from DBEs with best-estimate method and assumptions being commonly used. More recently they tend to be regarded clearly as an event category of beyond DBE. The purpose of CDA analysis is therefore to provide or confirm an additional safety margin of the plant strictly designed for DBEs.

### *3.2 Safety Approaches in Advanced LMRs*

In the next-generation LMR plants designed during the 1990s, a more advanced innovative design approaches were taken for improved safety especially to cope with CDAs. In this section, we provide a concise review of the four fast reactor development programs: European Fast Rector, ALMR in the United States, BN-800 in Russia and D-FBR in Japan.

*European Fast Reactor* [13, 14]

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

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

*ALMR* [15-17]

The U. S. Advanced Liquid-metal Reactor (ALMR) is a small modular reactor with a metal-fueled core. One power block consisting of three reactor modules, has a net electric output of 465 MWe. The reactor concept is the power reactor innovative small module (PRISM). Metal fuel is operated at relatively low power and a large thermal expansion introduces negative reactivity feedback. Other passive safety features are provided by core radial expansion, axial expansions of control-rod drive lines and reactor vessel. The gas-expansion modules (GEMs) are added to mitigate unprotected loss-of-flow CDAs. In addition, the active shutdown system has a diverse system called the ultimate shutdown system, in which B4C absorber spheres are dropped manually into the central channel of the core. These active and passive features make the reactor shutdown capability of PRISM very reliable.

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

*BN-800* [18]

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

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

*D-FBR* [19]

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

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

*Summary*

The LMRs designed during the 1990s have incorporated many innovative design ideas. It was shown that the safety objectives can be met in both the large-scale plants (EFR, BN-800 and D-FBR) and intermediate-scale modular plant (ALMR). Even though none of these plants but BN-800 in Russia were actually licensed or constructed, it is hopeful that these plants are licensable. Moreover, many of the advanced and innovative design concepts developed are extremely useful for designing future LMRs.

## 4. Recommendation for future safety approach

Based on our experience as reviewed in the previous section, we can conclude that the safety technology and actually licensable approach exist in the world for sodium-cooled LMRs from small to large scale. The safety technology or achievement of safety research programs is documented and available in open conference proceedings. This does not mean, however, that our future efforts are unnecessary. There remain those safety issues, although not many, that we should resolve or improve. In this section, recommendations are given for the key safety issues and safety approach.

***General Objectives***

The future advanced LMRs must be developed much further to make then economically competitive with the LWRs in future generation, such that they can be commercialised. An inherent capability of fast-neutron reactors for either breeding or burning plutonium and minor actinides also has to be studied further in such a way that the requirements for nuclear non-proliferation are met at the same time. Another important objective is to design a reactor system that is best compatible with the related fuel cycle technologies (fuel fabrication and spent fuel reprocessing). This last objective is necessary to develop an integral system of reactors and fuel-cycle facilities to sustain to supply a continuous energy resource in a long time scale.

The safety objectives must be met at the same time in a well-balanced way with the above objectives. For the safety of any future advanced LMRs, a higher level of safety standard may be necessary, if we believe they can eventually replace the LWRs that must rely upon uranium resource.

An approach to be taken in future-generation LMRs may call for a "step-wise" increase in a safety level, such that the risk level associated with the reactor should be kept sufficiently low even without off-site emergency procedure [20]. This objective may provide a challenging safety goal to be explored and is also useful for enhancing public acceptance of the reactor. Questions remain as to how a quantitative acceptance criterion is defined with respect to risk and how one can demonstrate the reactor meets the safety objective.

***Key Safety Issues***

Some of the safety issues to be resolved are discussed. First the chemical and material-compatibility problems associated with liquid-metal coolant must be controlled. Second activation of coolant nuclides should be carefully considered in the design, especially when a future LMR tries to eliminate the secondary heat-transport system.

A preventive part of the defense-in-depth approach can be strengthened by provision of passive safety features for more reliable CDA prevention. A number of design ideas have been studied thus far, but their function and effectiveness should be demonstrated.

Even though the occurrence frequency can be made sufficiently low, consideration of CDAs should be required assuming their occurrences. As discussed in Section 2, there is a recriticality potential in a fast-neutron, plutonium-fueled core. The safety approach to eliminate the recriticality concern will be highly useful, because with this approach, severe accidents in LMRs can be simply regarded as similar to LWRs. In addition, it is mush easier to achieve in-vessel cooling and retention of post-accident core debris in LMRs because of excellent heat-transport characteristics of liquid-metal coolant. In other word, we can better emphasize the salient features of LMRs.

The role of the containment must be also discussed. In contrast to the high-pressure LWR system, there is no significant containment loading mechanism in LMRs. With sufficient measures available for preventing and mitigating CDAs, a question may arise as to whether we still need a containment. Because of crucial safety features of the containment as a final physical barrier to confine radioactive materials, and understanding uncertainty and limitation in human perception, the requirement for the containment should be carefully determined.

***Recommendations for Safety Design Approach and Goal***

The role and effectiveness of a probabilistic safety assessment (PSA) must be addressed. A risk concept has been already introduced in the safety approach for EFR, for example. A risk-informed approach in an early design stage is recommended for attaining well-balanced safety design. Although design details are not available yet in a conceptual design stage for performing a detailed PSA, preliminary probabilistic assessment should be extremely beneficial for systematically comprehending the safety (risk) characteristics of a plant with respect to a risk potential. Design improvement can be effectively made in such a way of appropriately controlling and minimizing the risk.

Early contact with regulatory authorities is highly and always recommended for advanced reactor concepts with innovative design features where experiences and practices are limited in licensing procedures. The procedure taken by the U. S. Nuclear Regulatory Commission to perform safety review during a pre-application stage is desired for efficient licensing activities.

Assuming the future advanced LMR design takes advantage of innovative and passive safety features to prevent and mitigate severe accidents, we can come back to an improved concept of defense in depth (see Section 2). If the future safety approach can be harmonized with the concept of not requiring an off-site emergency procedure, we can argue to modify the definition of the Level-5 defense line. It should be still noted that off-site emergency procedures have been already prepared by means of national laws and regulations in many countries.

## 5. Concluding remarks

In this paper, we have reviewed the safety characteristics of LMRs and the safety approaches taken in those LMRs which were actually designed, built and operated in the 1970s and 1980s and planned 1990s. These experiences clear show that the LMR technology has been matured well to a level that such a reactor concept is licensable in any country. The next-step development and commercialization of LMRs in many countries has not been promoted as originally intended, though, not because of safety reasons but rather of economical or political reasons.

In the future advanced and innovative LMRs, a safety level can be further improved especially enhancing prevention and mitigation features with more emphasis on passive safety features. Based on the future development, we can then construct new safety logic to eliminate major mechanisms for containment loading and hence large-scale radiological release. Thus we will be able to argue that a future safety goal may be defined such that the risk level associated with the reactor should be kept sufficiently low even without off-site emergency procedure.

We believe a fast-neutron LMR concept is regarded as one of the most-promising candidates for next-generation advanced nuclear reactors for which we should further investigate in the future. The concept can be technologically matured in a short-time scale to become economically-competitive to future-generation LWRs, publicly acceptable, environmentally advantageous, and potentially providing sustainable energy resources over many centuries to come.

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# SESSION 2

## Issues Important for Safety and their Assessment

*Chairman: J. Hyvärinen (STUK, Finland)*
*Co-chairmen: K.W. Hesketh (BNFL, UK), W. Frisch (GRS, Germany),*
*G. Cognet (CEA, France) and A. Porracchia (CEA, France)*

# SAFETY ASPECTS OF SVBR-75/100 REACTOR

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## SVBR-75/100: design features

- Monoblock (integral) design, neither valves nor pipelines with lead-bismuth coolant.
- Two-circuit system of heat removal.
- Use of natural circulation in all circuits for decay heat removal.
- Guard vessel.
- Fuel subassemblies without shrouds.
- Steam generators with multiple natural circulation with saturated steam generation.
- Low-speed gas-tight motor of less than 500 kW power for main circulating pump.
- Reparability and possibility for reactor equipment replacement.
- Subassembly by subassembly refuelling of the whole core at a time.
- Multi-fuel capability ($UO_2$, MOX with MA, nitride fuel) in the same reactor design met all the safety requirements and standards.

## SVBR-75/100: background

| | |
|---|---|
| 1995 | Conceptual design of 2nd unit renovation of NovoVoronezhskaya NPP with SVBR-75 reactors. |
| 1998 | RosEnergoAtom (NPP operation organisation) approved the project. |
| 1999 | Stagnation of works: BREST lead cooled reactor was preferred by Minatom RF. |

1999-2000 Development of conceptual design of 1200 MW NPP with SVBR-75/100 reactors for JNC, Japan.

2001 Conceptual design of new NPP (2 units x 1600 MW) with SVBR-75/100 reactors for Russia.

07.12.2001 Minatom RF approved the project.

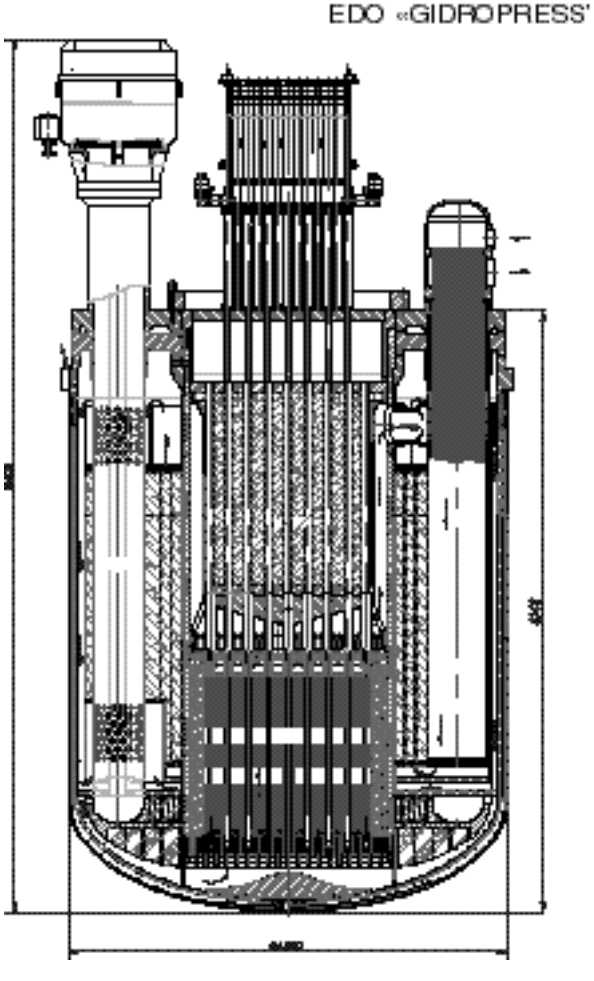
Alexander Roumyantsev, RF Minister for Nuclear Energy, (“Nezavisimaya” newspaper):

… There is no single-valued answer yet, that very BREST is a future of the atomic energy …

… There are alternative reactor designs with lead-bismuth coolant. Research works are completed for them. Development works should be done to build …

… I cannot say, that there are concrete results which make possible to build NPP on basis of new reactors, we are at a stage of investigations yet …

**SVBR-75/100: main performances**



| Designation | Value |
|---|---|
| Thermal power (nominal), MW | 280 |
| Steam capacity, t/h | 580 |
| Steam pressure (saturated), MPa | 9.5 |
| Feed water temperature, °C | 240.9 |
| Primary coolant flowrate, kg/s | 11 760 |
| Primary coolant temperature, outlet/inlet, °C | 482/320 |
| Core dimensions: diameter x height, m | 1.645 x 0.9 |
| Number of fuel elements | 12 114 |
| Number of control rods | 37 |
| Mean power density in the core, $MW/m^3$ | 140 |
| Mean linear load of the fuel element, kW/m | ~24.3 |
| Time between refuelling, year | ~8 |
| Core charge, ($UO_2$) with uranium: mass, kg / enrichment, % | 9 144/16.1 |
| Number of steam generator modules | 2 ı 6 |
| Number of main circulating pumps | 2 |
| Power and pressure of main circulating pumps, kW / MPa | 450/0.55 |
| Core lifetime, effective hours | 53 000 |
| Lead-bismuth coolant volume in the reactor, $m^3$ | 18 |
| Reactor monoblock dimensions: diameter x height, m | 4.53 x 7.55 |

**SVBR-75/100: reactor arrangement**

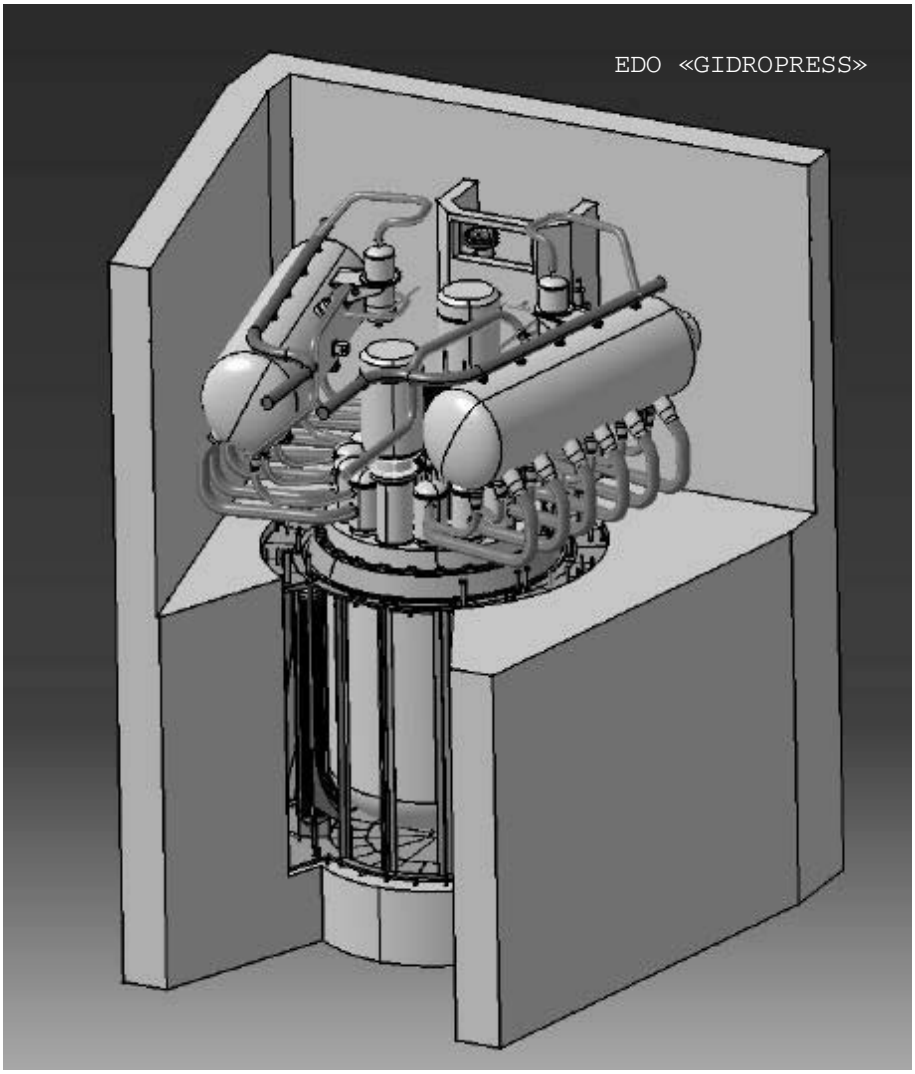
Reactor is arranged in a water tank.

Tank design rigidity can withstand earthquake up to number 9 of MSK.

As an element of radiation shielding, the tank preserves structural elements of reactor compartment against activation.

As an element of passive heat removal system, the tank provides the reactor with passive autonomous heat sink; reactor parameters remain within the safe operation limits.

As an element of SG leak localising system, the tank serves as a bubbler to condense steam while big SG leak occurs.



**SVBR-75/100: refuelling**

Refuelling is necessary once in ~8 years.

Before refuelling the reactor should be stopped for a month for heat removal.

All the core is reloaded at a time, subassembly by subassembly.

Each subassembly is placed into its hermetic container filled with liquid lead.

Containers with spent subassemblies can be stored in a dry repository cooled with naturally-circulated air.

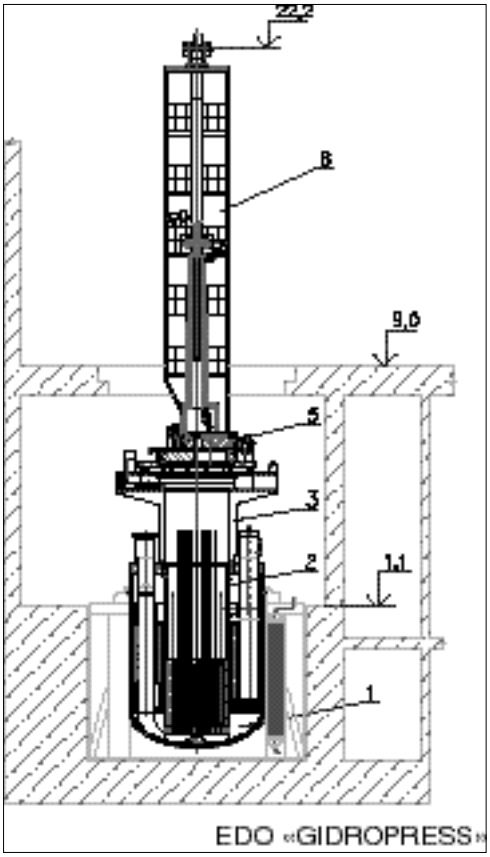
One set of refuelling equipment can be used for several reactors within the same NPP unit.

While delivered to developing countries, both refuelling and other nuclear-hazardous works are not provided.

After core lifetime exhaustion reactor should be cooled for a year till the coolant solidifying and transported to the manufacturer for refuelling and necessary preventive repair as a one unit.

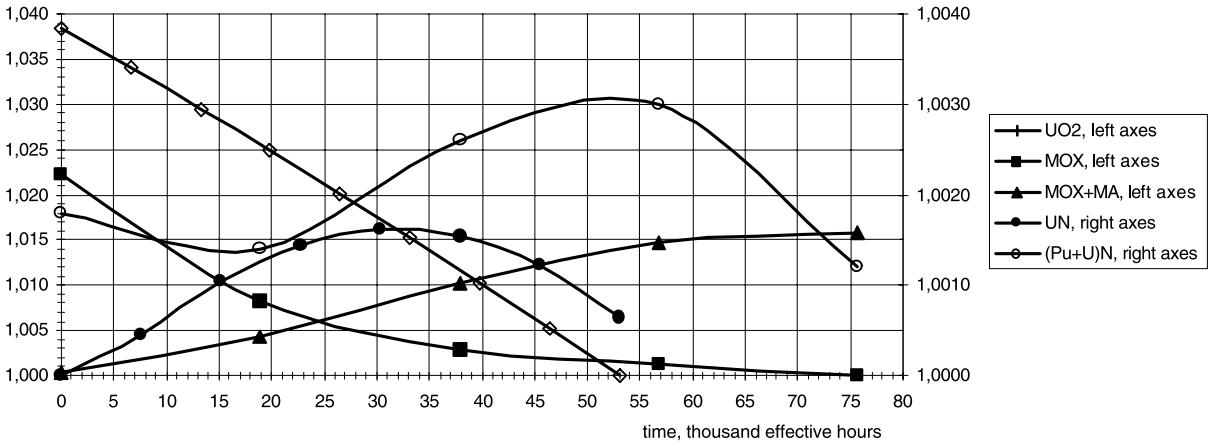
Another reactor is delivered to the customer.

After refuelling and preventive repair the reactor unit is ready to be delivered to a NPP site (with solid coolant while transporting).



**SVBR-75/100: multifuel capacity**

N.N.Novikova, IPPE




Oxide fuel: reactivity change during the core lifetime $\Delta\rho(T)$ is more than 1$, safe reactivity control algorithm should be used.

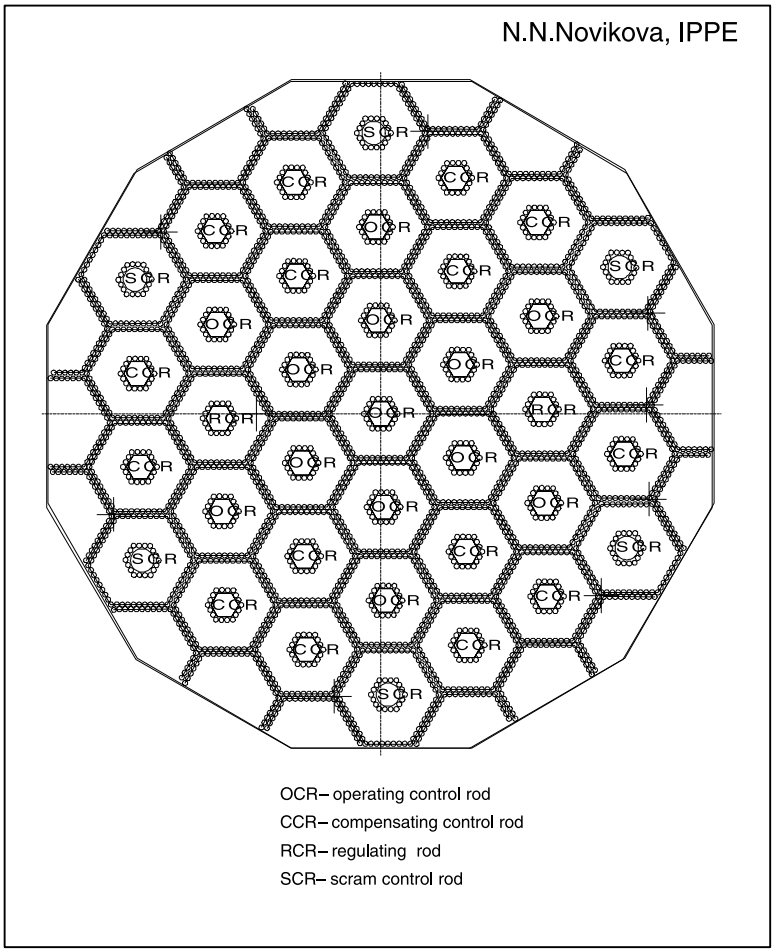
Nitride fuel (4,5): reactivity change during the core lifetime $\Delta\rho(T)$ is less than 1$.

UN fuel makes possible to prolong the core lifetime to 100 000-120 000 effective hours using safe reactivity control algorithm.

**SVBR-75/100: Safe Reactivity Control Algorithm**

- Core lifetime T is divided to micro-campaigns $\Delta t$, which has operation reactivity margin $\Delta\rho(\Delta t)$ less than 1$.
- Maximum value of operation reactivity margin $\Delta\rho(\Delta t)$ during a micro-campaign $\Delta t$ is ~0.65$; micro-campaign duration is ~6500 effective hours (~1 year).
- To limit the operation reactivity margin, there are two groups of compensating rods:
  - operating compensating rods (OCR, 13 units), available for an operator;
  - compensating control rods (CCR, 16 units) which disconnected from reactivity control system in reactor operation.
- Safety and control rods are always available for an operator.
- At the beginning of core lifetime all the CCR completely inserted into the core and disconnected from reactivity control system.
- When core micro-campaign is exhausted, reactor should be stopped, all available control rods should be completely inserted into the core, 2 CCR rods should be connected to the system, completely withdrawn from the core and disconnected again.
- Sequence of withdrawing CCR is optimised to reduce azimuth irregularity of power density distribution in the core.
- While starting operation reactivity margin is more that 1$ due to negative power reactivity effect; safety under this condition is provided by limited velocity of OCR withdrawal.

**SVBR-75/100: basic neutronics**



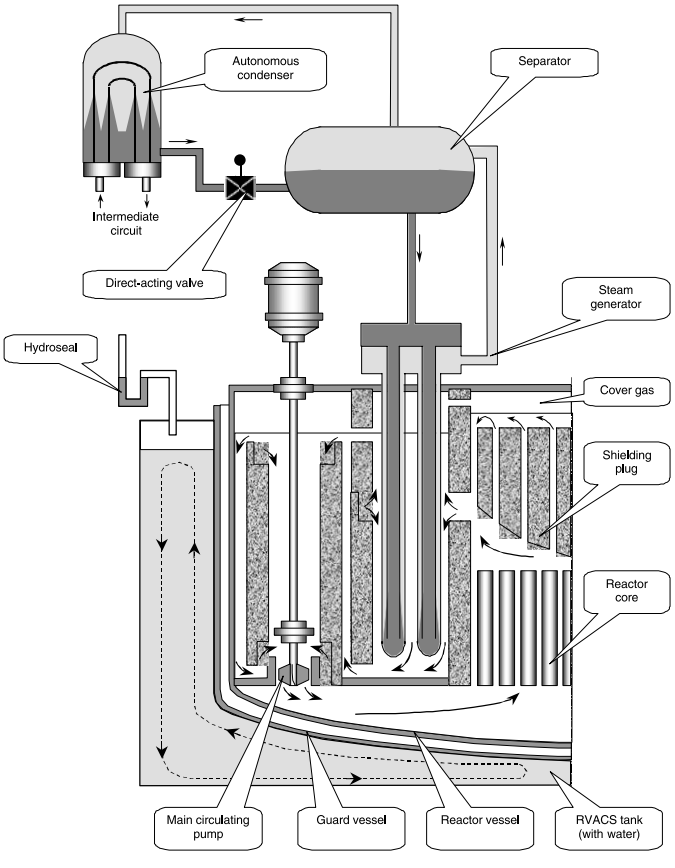

| Designation | Value |
|---|---|
| Fuel type | $UO_2$ |
| Core charge with $^{235}U$, kg | 1 470 |
| Core lifetime, effective hours | 53 000 |
| $\beta_{eff}$ at the beginning of core lifetime<br>at the end of core lifetime | 0.722<br>0.585 |
| Reactivity change during core lifetime, $\beta_{eff}$ | 5.72 |
| Power reactivity factor, (warming up from 200° to rated power), $\beta_{eff}$ | – 1.2 |
| Reactivity worth of 29 CCR + 2 CR, $\beta_{eff}$ | 8.95 |
| Maximum reactivity worth of one CCR rod, $\beta_{eff}$ | 0,33 |
| Core micro-campaign duration, effective hours | ~ 6 500 |

**SVBR-75/100: autonomous heat removal**

Autonomous heat removal system provides decay heat removal to an intermediate circuit.

The system capacity is 6% of rated reactor power.

The system provides reactor operation in starting and other technological modes without the turbine unit.



**SVBR-75/100: SG leak localising system**

SG leak localising system can condense steam in case of one SG tube rupture.

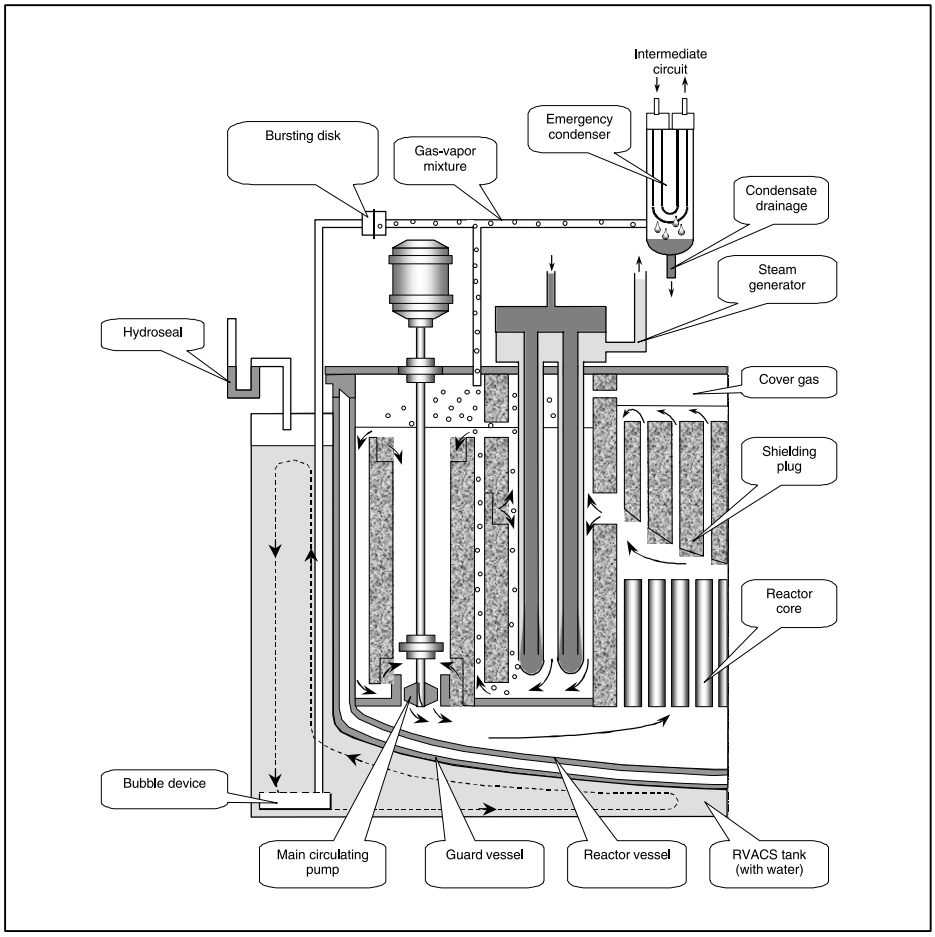
In this case steam pressure in reactor vessel does not exceed 0.5 MPa.

Condensate (of water volume in the separator is 5 ton) is drained in a tank for liquid radioactive waste storage.

In case of a big SG leak the bursting disk will snap into action under steam pressure above 1.0 MPa.

The steam will be forwarded to RVACS tank, which serves as a bubbler.

Non-condensing gases will be passed into filter-ventilation system.



**SVBR-75/100: passive heat sink**

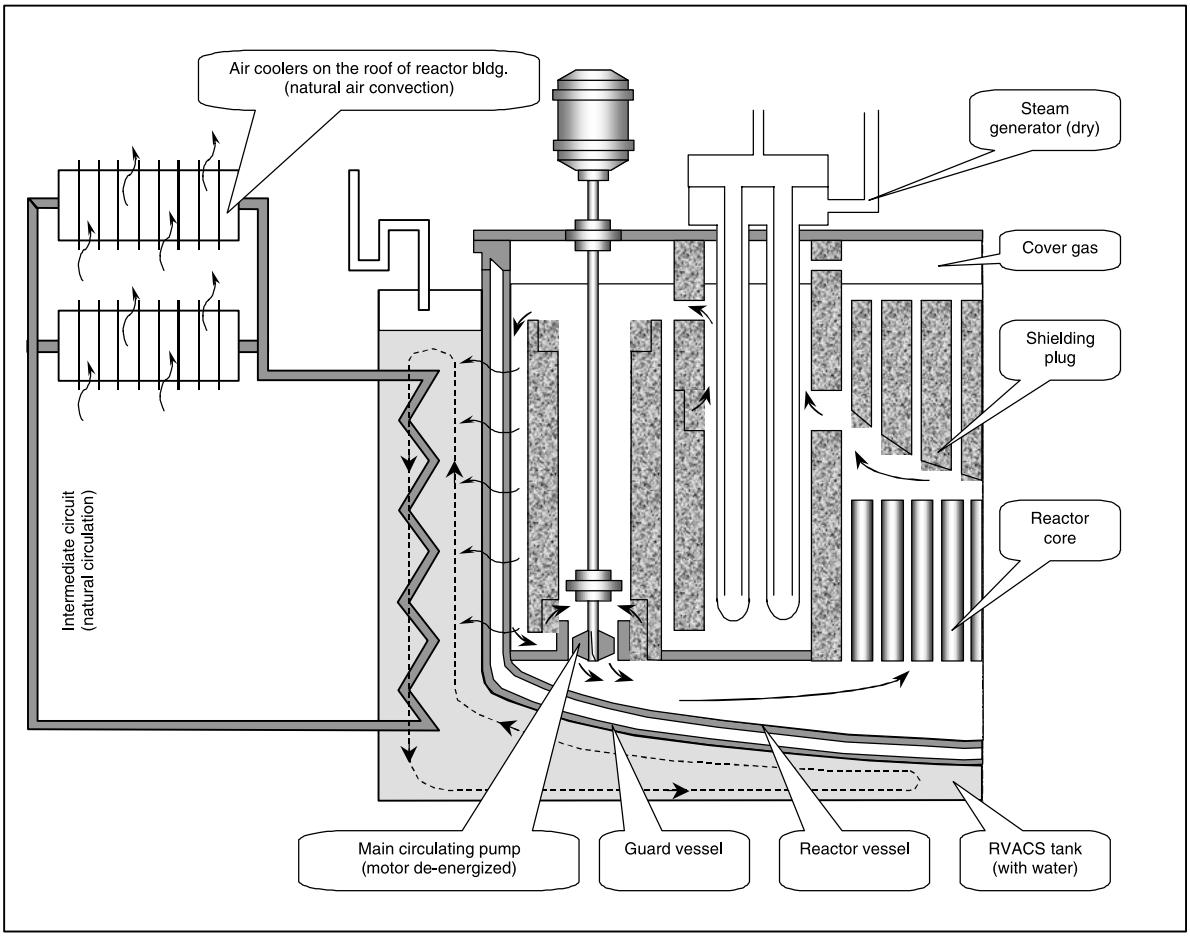
Reactor vessel auxiliary cooling system (RVACS) is always in operation.

Under normal condition removes up to 0.2% of rated reactor power via the walls of reactor and guard vessels to 110 $m^3$ RVACS water tank.

The tank is supplied with built-in heat exchangers cooled with naturally circulated water of the intermediate circuit.

Intermediate circuit removes heat to air coolers on the roof of reactor building.

Heat dissipated in atmosphere due to natural air convection.



**SVBR-75/100: passive heat sink**

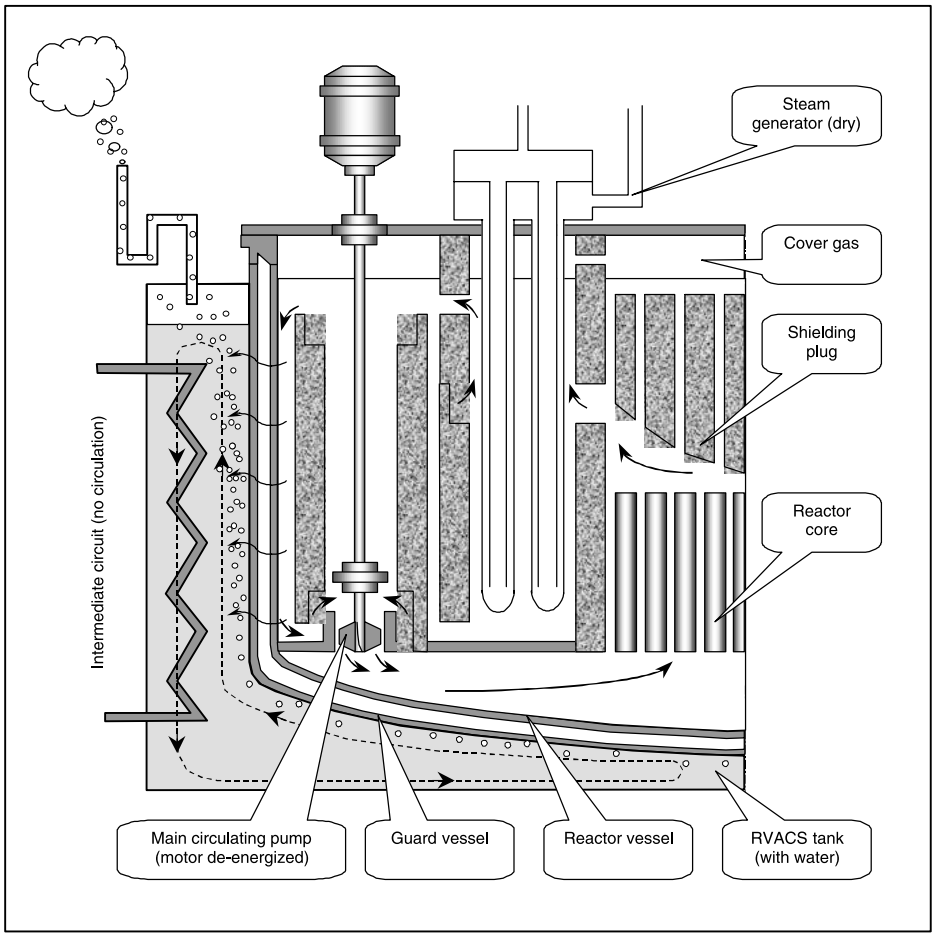
In an abnormal mode, while regular heat removal systems are fault and coolant temperature increases up to 700°C), up to 0.4% of rated reactor power can be removed to the RVACS water tank.

To reduce capital cost, it is possible to design intermediate circuit with forced water circulation.

In this case, NPP blackout results in reactor cooling down due to water evaporation from the RVACS tank.

Water evaporation makes possible to keep the reactor parameters within operation limits during 48 hours.

If necessary, this let-along period can be prolonged using further design improvement.

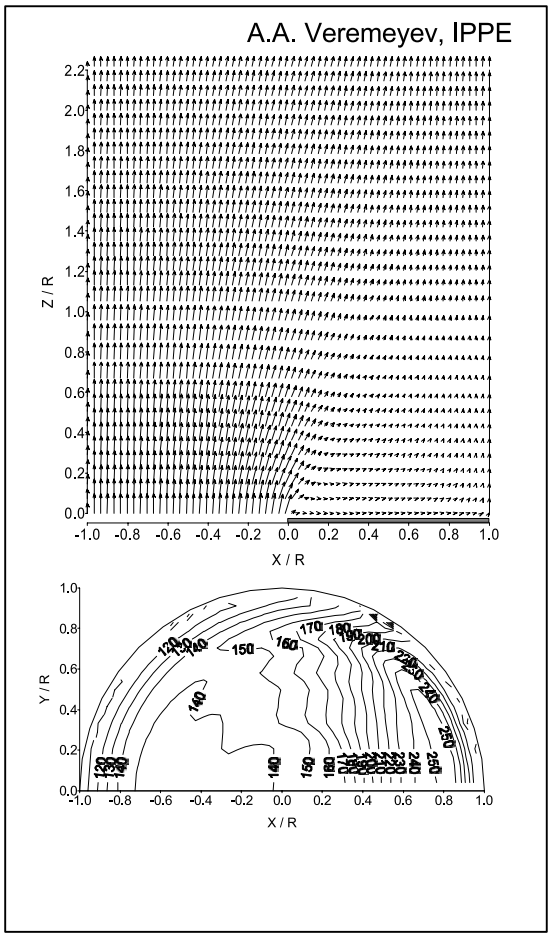


**SVBR-75/100: core blockades**

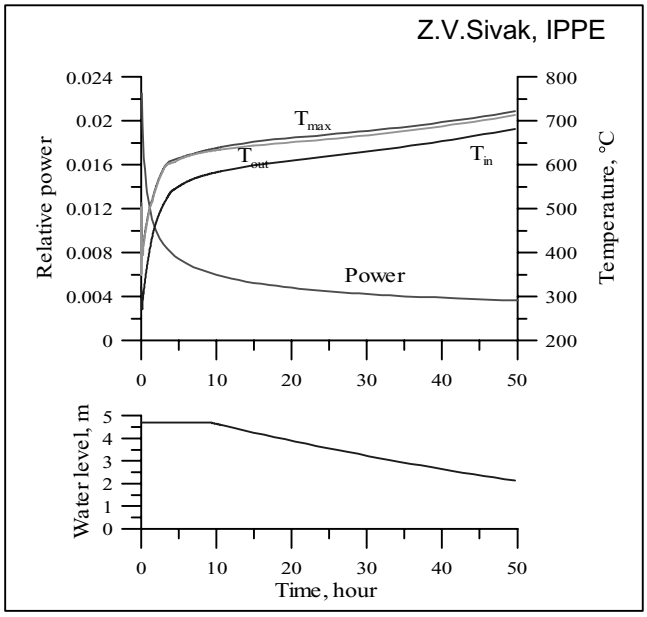
Three types of core blockade were investigated:

- Blockade of the central sub-assembly at the inlet.
- Blockade of ½ core open flow area at the inlet (the worst case).
- Blockade of the central sub-assembly at the middle of fuel element height (at the beginning of the fuel part).

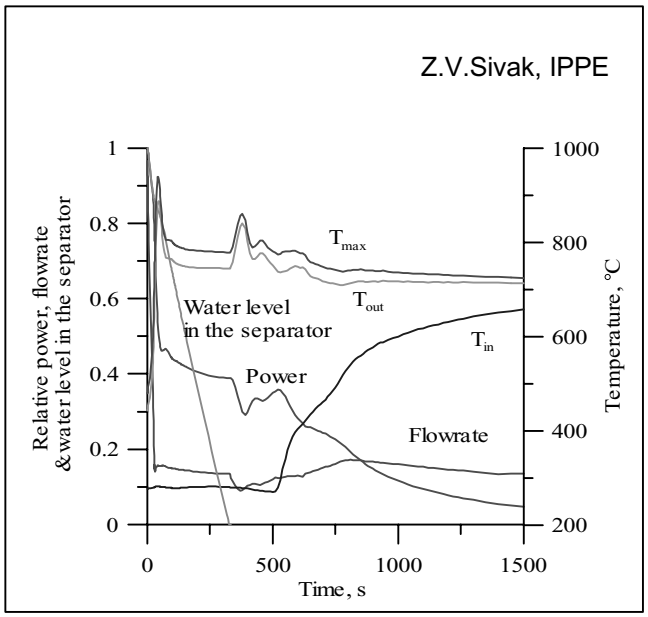
The temperature values obtained for the condition simulated do not exceed safe operation limits.



**SVBR-75/100: NPP blackout**



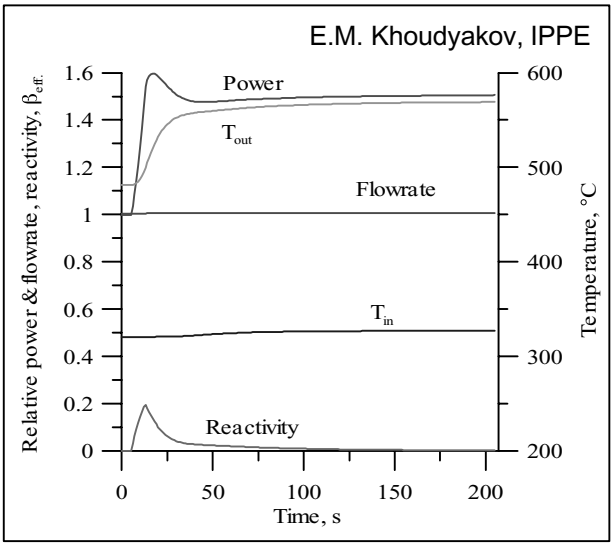

Long term reactor cooling down at the end of core lifetime under NPP blackout conditions **with scram**. Reactor decay heat is removed due to water evaporating from the RVACS tank.



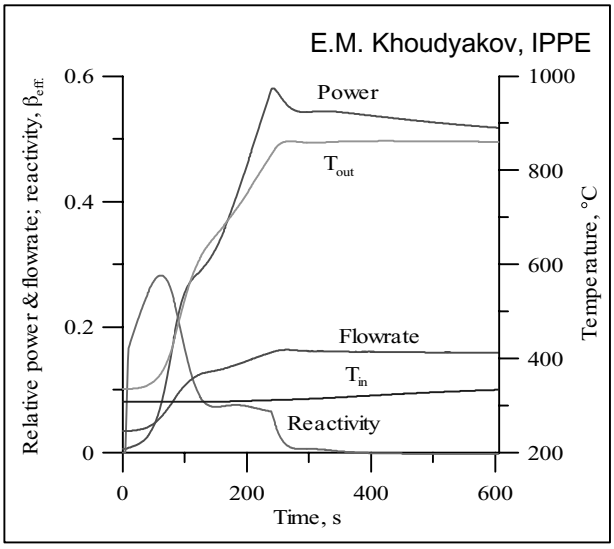

Reactor cooling down at the end of core lifetime under NPP blackout conditions **without scram** (first period, when reactor power is higher than decay heat). Reactor decay heat is removed due to negative reactivity feedback.

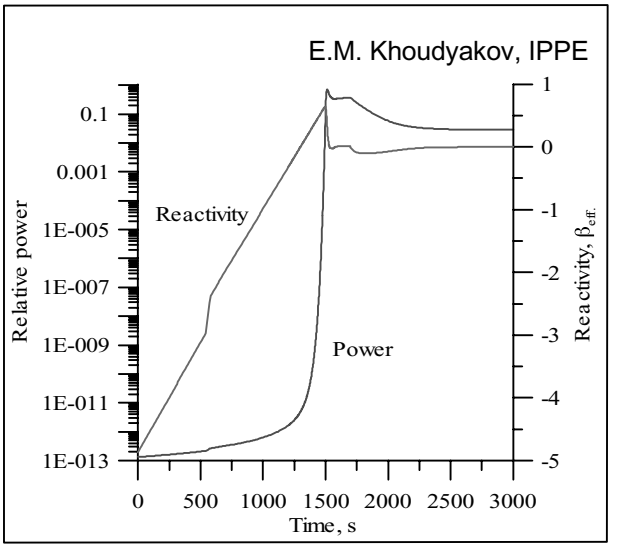
**SVBR-75/100: reactivity incidents**

**Withdrawal of all available control rods at rated reactor power without scram.**

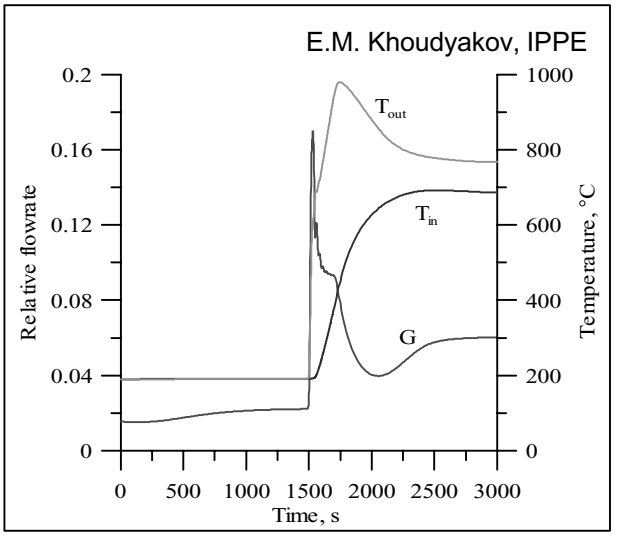



**Withdrawal of all available control rods at zero reactor power without scram**






**Withdrawal of all available control rods from sub-critical reactor without scram**



**SVBR-75/100: Prospects**

04.2002 Examination of the Conceptual Design of a new NPP (2 units by 1 600 MW) with SVBR-75/100 reactors by Scientific and Technical Committee of RosEnergoAtom.

05.2002 Examination of the Conceptual Design of a new NPP (2 units by 1 600 MW) with SVBR-75/100 reactors by Scientific and Technical Committee of Minatom RF.

2003-2010 Development works on:

- Renovation of the 2nd unit of NovoVoronezhskaya NPP.
- 200-400 MW nuclear co-generation plant for Voronezh.
- 200-400 MW nuclear co-generation plant for Arkhangelsk.

5-6.03.2003 Info: Minatom, Russian Federation and CEA, France organise a seminar "Future nuclear power systems and technology to provide a regime of nuclear weapon non-proliferation" in Saclay Research Center. Russian delegation with representatives of several research institutes to be headed by Vladimir Emeliyanov, Deputy Head of Department of Nuclear Power, Minatom, Russian Federation.

# THE R&D ISSUES NECESSARY TO ACHIEVE THE SAFETY DESIGN OF COMMERCIALISED LIQUID-METAL COOLED FAST REACTORS

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**Abstract**

Within the framework of the feasibility study on commercialised fast reactor cycle systems (hereafter described as F/S), the safety design principle is investigated and several kinds of design studies are now in progress. Among the designs for liquid-metal cooled fast reactor (LMR), the advanced loop type sodium cooled fast reactor (FR) is one of the promising candidate as future commercialized LMR. In this paper, the safety related research and development (R&D) issues necessary to achieve the safety design are described along the defence-in-depth principle, taking account of not only the system characteristics of the advanced loop concepts but also design studies and R&D experiences so far. Safety issues related to the hypothetical core disruptive accidents (CDA) are emphasized both from the prevention and mitigation. A re-criticality free core concept with a special fuel assembly is pursued by performing both analytical and experimental efforts, in order to realize the rational design and to establish easy-to-understand safety logic. Sodium related issues are also given to ensure plant availability and to enhance the acceptability to the public.

## 1. Introduction

Fast reactor (FR) cycle systems supply sustainable energy both to allow long-term use of nuclear power by recycling TRU fuels and to have a potential to be harmonized with the environments by burning the minor actinides (MA) and transmuting long-life fission products (LLFP). It is therefore expected that FR cycle systems will realize both the effective use of natural uranium resource and the mitigation of environmental burden.

Among various kinds of coolants, the sodium cooled FR system has the greatest potential in core performance, compact plant design and technical feasibility. It is therefore that the sodium cooled FR has been developed in the world so far. The safety requirements for the next generation reactors based on the defence-in-depth principle will be well reflected into the design with confidence, since the Japanese Demonstration Fast Breeder Reactor (DFBR) design study[1] had already taken into

1. DFBR design study had been performed from 1986 to 1999 by Japanese nine electric power companies, Electric Power Development Company (EPDC) and Japan Atomic Power Company (JAPC). The MOX fuelled 660 MWe sodium cooled fast breeder reactor concept with top-entry

account those requirements and developed some key issues such as passive shutdown system, natural circulation capability and mitigation of consequences due to the CDA [1,2,3,4,5].

There are, however, several issues to be resolved in order to be the future commercialized reactors: Firstly, the construction cost of LMR has not yet been achieved to be equivalent to that of future LWRs. The innovative system design is required to reduce the plant materials less than the conventional designs by incorporating the advanced technologies and materials. Secondary, the countermeasures against the CDA consequences should be well accommodated to ensure the rational plant design with easy-to-understand mechanism, even though the reactor core size may become larger and passive safety features would be provided to enhance the prevention capability. Thirdly, anticipated incidents related to boundary failures of sodium or structure failures in sodium should be earlier settled to ensure less impact on the plant systems for higher plant availability. Related to this point, In-service Inspection and Repair (ISI&R) technology should be more improved and the system design should be accommodated with the ISI&R requirements, which will be able to assure the preventive maintenance like those of LWRs. These three points should be carefully taken into account to both system design and safety design.

This paper describes the safety requirements for future LMR, and the innovative plant concepts which are now under study in the F/S and the R&D issues necessary to achieve the safety design along the defence-in-depth principle [6,7].

## 2. Safety design requirements

General requirements set in the F/S are consisted of the followings:

- Safety design must be achieved by defence-in-depth principle.
- Core damage frequency (CDF) should be less than $10^{-6}$/ry.
- Passive safety features should be introduced to further reduce the CDF.
- No evacuation will be required even under Core Disruptive Accidents (CDA).

Crucial safety design points will be described along the defence-in-depth principle late after a brief explanation of the innovative LMR design concepts. The CDF less than $10^{-6}$/ry is a probabilistic design target for FR systems studied in the F/S. This target value would be achieved by taking account of safety characteristics of the system and also the DFBR design experiences. Two independent reactor shut down system (RSS) and redundant decay heat removal system with natural circulation capability are effective to ensure the high reliability. The level-1 PSA would be performed from the early stage of the design study in order to confirm the validity of the safety design.

The last two requirements are so-called beyond design basis requirements. The first is to enhance the passive safety features against crucial initiators for CDA, i.e., anticipated transients without scram (ATWS), since the time period into the whole core damage is too short to manage by operators. The complete passive safety features would be achieved by enhancing the natural circulation capability even under the passive shutdown conditions. The second is the ultimate design

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three loops system was studied aiming at the introduction of DFBR in the first decade of the 21st century. The related R&D issues had been performed both in Japanese utilities and also in PNC (reformed to JNC from 1998), in Central Research Institute of Electric Power Industry (CRIEPI), and in Japan Atomic Energy Research Institute (JAERI).

target to prevent the radioactive release even under CDA conditions. To achieve this target, the in-vessel retention (IVR) against CDA consequences and no impact on the containment integrity are design requirements. The IVR had already been pursued by ensuring structural design margin against certain energetic re-criticality in DFBR, which resulted in fission products (FP) release and sodium spillage due to energetics in the containment. However there exists several uncertainties for the evaluation of energetics and less experimental data especially for a large scale core. Therefore, the elimination of re-criticality events under the CDA sequences has been strongly required. The restriction of core design parameters such as sodium void worth and the introduction of special fuel assemblies would be necessary to satisfy this requirement. As a result, the IVR and no impact on the containment integrity under CDA conditions would be achieved in an easy-to-understand way. Eventually the release of radioactive materials would be prevented with the assurance of the containment integrity and thus no evacuation would be required from the plant.

The significant point in the passive shutdown and the elimination of re-criticality should be realized in simple and its effectiveness should be demonstrated for better acceptability to the utilities and the public.

The safety design concept to satisfy these requirements is illustrated in Figure 1.

## 3. Design concepts for sodium cooled FR in Japan

Within the framework of the F/S, the advanced loop type reactors are one of the promising candidate as future commercialized sodium cooled FR [8]. There are two different reactor sizes, i.e., a large scale-reactor of 1 500 MWe and a medium sized modular reactor of 500 MWe with the similar plant configuration. In both designs, several new technologies have adopted to achieve the economical target, i.e., introduction of new system and materials such as three-dimensional (3D) seismic isolation systems, 12Cr-steel piping, and oxide dispersion strengthened ferrite steel (ODS) cladding, novel plant configuration such as two-loop system with compact reactor vessel, new sub-system such as the integrated IHX with pump, and new fuel assemblies for re-criticality free core.

The reactor core design aims at the high burn up, where the core average is 150 GWd/t and the pin peak will be 200 GWd/t. Both MOX and Metallic fuels has been selected as the candidate as the core fuel. The first candidate of the cladding material is ODS, which is expected to ensure the irradiation integrity more than 200 GWd/t, and now the material irradiation is in progress and the fuel pin irradiation is under preparation.

The design concept is shown in Figure 2 together with indicating four crucial innovative technologies, which are adopted to reduce significantly the plant materials. To realise this design, it is necessary to carry out several R&D issues until 2015, at longest. Here, the safety related R&D issues have been described in the next section.

## 4. Safety related R&Ds along the defence-in-depth principle

### *1. First level: prevention of occurring incidents and failures*

The system design with new material, subsystems and structures should have rational design margin difficult to be failed or not to result in the incidents. For this purpose, the new material,

subsystems and structures will be carefully designed and fabricated based on the characteristics understanding and mock-up experiments.

As the basis of rational system design, the seismic isolation technology is one of the crucial issues, which would reduce plant materials and standardize the system design especially in Japan. The two-dimensional seismic isolation technology has been already developed and the related design guideline for FR system has been published in Japan. However, the 3D seismic isolation technology should be developed in order to improve rationality, e.g., much smaller relative displacement among fuel assemblies, control rods, and the core support plate against vertical response acceleration. Also buckling of reactor vessel would be more easily avoided. The 3D seismic isolation system is now under investigation by producing several concepts as shown in Figure 3 [9].

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

The two loop system and compact reactor vessel design has raised several R&Ds to ensure the plant integrity related to thermal hydraulics, such as thermal stress, cover-gas entrainment, and flow induced vibration issues which would be facilitated by higher flow velocity of coolant in main piping. The integrated components with IHX and mechanical pump may enhance the fretting wear of IHX tubes. The evaluation methods for the fretting wear in 12Cr steel and related experimental data are necessary to be investigated to ensure its integrity. Figure 4 summarises the R&D items to ensure the technical feasibility for the advanced loop system concept.

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

## *2. Protection of incidents or transients into accidents*

The early detection capability of safety protection system is significant to minimize the influences to plant systems, such as temperature increases in fuel, cladding and structures. From the safety point of view, the reliability of RSS and DHRS is essential and the safety criteria for fuel pins and 12Cr steel are necessary to be developed.

The CDF less than $10^{-6}$/ry can be achieved by providing redundant and independent safety function with diversity. The two independent reactor shutdown systems (RSS) have been designed within the DFBR Project in Japan, where the diversity of crucial parts of RSS are enhanced by adopting independent multi-signals for de-latching, different rod insertion mechanisms, different de-latch mechanisms, and rod structures such as solid and articulated. The preliminary PSA result showed the unavailability of RSS is sufficiently less than $10^{-6}$/demand, even under the assumption of common cause failures [10]. As for the decay heat removal, the natural circulation capability is crucial to

achieve higher reliability for a longer mission period, even though the redundant forced circulation systems were adopted and the safety function is ensured without coolant injection. Based on these results, the CDF less than $10^{-6}$/ry would be satisfied.

In the advanced loop concept, two loop system without check valves is employed and thus the pump stick would become the severest accident in the design basis event (DBE) comparing with the conventional three or four loops design. However some design adjustments, e.g., a delay time of the primary pump trip in the intact loop and a halving time of the primary flow rate within the reasonable range, make it possible to restrict the maximum cladding temperature less than the safety criteria. Thereby the safety criteria for ODS fuel pins would be prepared through the safety experiments.

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

### *3. Mitigation of accident consequences*

The consequences of DBE would lead to no radioactive release due to neither fuel pin failures nor penetrations from primary coolant to the containment. The pipe break of the cover gas piping would result in also no release of radioactive materials, since the double boundary system would be adopted in order to avoid sodium fire in the containment that is caused by accompanied sodium with gas leaking. There are small possibilities in the accidents during fuel handling system such as drop of fuel assembly, and accidents in the radioactive material storage tanks. The confinement function would be adopted to mitigate the consequences in fuel handling system and similar design measure would be adopted in the radioactive material storage tanks. Nevertheless the containment would be provided as the final barrier to the environments.

### *4. Prevention and mitigation of severe accidents*

The probability of causing the severe core damage would be less than $10^{-6}$/ry, but both the passive safety features and mitigation measures against CDA would be provided to enhance safety. As the selection of these features, the great importance has been put on simple and reliable mechanisms together with demonstration capability.

- Passive Safety Features

  A restraint core concept with core barrel structure has been adopted in Japan. The cause of rod stuck has been studied by analyzing the contact modes between absorber rods and guide tubes through experiments. As a result, an articulated structure has been adopted in the upper most position of absorber rods to enhance insertion capability. The absorber rods are always inserted inside the guide tubes and locate just above the active core during operation requiring only 1 m stroke for scram. By taking account of these design features, the absorber rods will not stuck before whole fuel assemblies will be compacted by terrible deformation. A certain design margin of insertion function is

required against beyond maximum design earthquake conditions. Therefore cause for scram failure has been identified as multiple failures of safety protection systems and of de-latch mechanism. Sodium coolant, on the other hand, has superior characteristics as regard to thermal conductivity and heat transportation with sufficient sub-cool margin to boiling. Indeed there is ca 350°C. of sub-cool margin at the maximum cladding temperature point in the best estimate sense. And the absorber rod has a large negative reactivity worth and its motion is one-dimensional within a few seconds by gravity. The increase of coolant temperature can be expected all types of ATWS, i.e. unprotected loss of flow (ULOF), unprotected transient over power (UTOP) and unprotected loss of heat sink (ULOHS).

The above is the background to select the self actuated shutdown system (SASS) as the most promising passive shut down system developed from DFBR design study, where the Curie point magnet is adopted in the magnetic circuit to de-latch absorber rods. The temperature sensitive alloy has a large design freedom to apply any reactor core concept not only MOX fuel but also Metallic fuel cores. The preliminary evaluation of ULOF indicates its effectiveness as shown in Figure 5. A comprehensive development work has been carried out both out-of-pile and in-pile experiments concerning the thermal ageing, thermal fatigue and creep, thermal transients, and irradiated influences. The effectiveness of SASS has been confirmed by the transient tests using full mock-up SASS in an out-of-pile facility. The verification of integrity against thermal transients and of preventing spurious de-latch remain up to now. The thermal transient test is now under continuation and design effectiveness against spurious de-latch is scheduled to be verified through in-pile operations in the experimental FR “JOYO”.

In the advanced loop concept, the natural circulation capability would be enhanced by reducing the pressure drop of core and adopting intermediate reactor auxiliary cooling system (IRACS) and primary reactor auxiliary cooling system (PRACS), that would increase the flow rate under natural circulation. And thus the passive cooling even under passive shutdown condition would be attained. The remaining point is natural circulation capability of direct reactor auxiliary cooling system (DRACS) that will be used under loop maintenance conditions. Design and analytical efforts accumulated in DFBR design study will be reflected into future study and there seem less experimental needs.

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

- Mitigation of CDA

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

against energetic re-criticality during the transition phase (T/P). The thermal design margin has been also investigated to ensure the IVR [11].

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

In the course of ULOF accident, power excursion due to coherent sodium boiling has been in major concern, because large sodium cooled core tends to have larger positive sodium void worth. In order to avoid severe energetics during I/P, the sodium void worth should be limited under certain value. In our study the reference value has been set less than six dollars (6$) for MOX core and 8$ for Metallic core based on a theoretical consideration. The data-base for MOX fuel has been greatly accumulated through in-pile experiments in TREAT and CABRI and theoretical analyses for various types of core design have been carried out [12]. Those in-pile experiments give clear evidences for boiling propagation velocity and fuel dispersion velocity as the function of power level under ULOF conditions. By applying these data to reactor conditions, the void limitation has been obtained.

Figure 6 shows typical ULOF results for 150 MWe MOX core with 7.4$ (case a) and 5.4$ (case b) analyzed by SAS4A, where the super prompt excursion is not reached in (case b) but is exceeded in [case a]. All the reactivity change is greatly accelerated in (case a) but the certain reactivity balance is established in (case b). Indeed the negative feedback effects from Doppler and fuel dispersal has dominated the transients and the net reactivity will never reach the super prompt criticality in (case b).

As for the fuel discharge in T/P, special fuel assemblies have been proposed to enhance the molten fuel discharge as shown in Figure 7. The Fuel Assembly with Inner DUct Structure (FAIDUS) has been selected, because the fuel discharge process would be nearly one-dimensional and contains less uncertainty. The scale of fuel discharge is own fuel assembly and it is possible to be demonstrated by the experiment.

Figure 8 is the T/P analytical results for FAIDUS and the conventional cores using SIMMER-II. As shown in this figure, the fuel discharge would be fairly enhanced in FAIDUS without resulting in energetics, whereas the conventional core may cause several power excursions due to the molten pool movements. The simplicity and demonstration capability of FAIDUS would contribute to establish easy-to understand safety logic. The experimental program named EAGLE had been launched from 1998 as the collaboration work between Japan and the republic of Kazakhstan [13]. Preparatory experiments both in-pile and out-of-pile are now in progress.

The FAIDUS assembly remains several R&Ds for fabrication and irradiation integrity. On the other hand, another concept ABLE (Axial BLanket partially Eliminated) is expected to improve core performance than that of FAIDUS and less R&Ds are required for assembling. However the fuel discharge process of ABLE contains more uncertainty and needs much time to complete the fuel discharge. To ensure the elimination of re-criticality, analytical efforts are now undergoing. After the detailed understanding for fuel discharge process in ABLE, the experimental requirements would be clarified.

The design of the lower core support structure is crucial to cope with the post accident material relocation and heat removal for the debris. In FAIDUS core, relative high temperature melt would be transported directly to the lower core support structure and thus careful design measures should be provided to ensure the in-vessel retention. Several measures to ensure the in-vessel retention have been found as shown in Figure 9.

With respect to Metallic fuel, further in-pile experiments, especially for ULOF conditions, would be required and analytical code would be modified for reactor analyses. The fuel discharge process would be also verified in the experiments. The advantage of Metallic fuel is low melting temperature and thus out-of-pile test may simulate the reactor conditions.

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

## 5. Specific issues for sodium coolant

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

- Sodium leak

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

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

- Sodium water reactions in SG

  As for the sodium water reactions, we are now investigating three different design approaches. The first one is a conventional single tube SG and addressed as reference

design. The point is to enhance the reliability of early detection system for water leak. The earlier detection system especially against small leaks would be adopted to prevent the propagation of tube ruptures and early restarting of plant operation, although the detection sensitivity would be less in the larger SGs in the advanced loop concept. For safety concerns, the wastage data and high temperature creep strength for 12Cr steel is necessary to be obtained. A comprehensive evaluation method for tube rupture propagation behaviors would be developed with taking account of superimpose of wastage and overheating rupture modes, where the hydrodynamics of sodium and water-steam and the local coolability of water in tubes would be precisely evaluated. For this purpose the experimental efforts would be required.

The multiple failures of mitigation system would be also significant, even though failures of the blow down system and isolation of feed water would be assumed as the beyond DBE. In such a case, large-scale tube ruptures may occur and higher pressure may challenge to the integrity of reactor coolant boundary. An experimental data to demonstrate self-limiting behaviors of tube ruptures due to sodium movement in the reacted zone is desirable to clarify a rational design margin for the primary boundary and also secondary systems. Providing such a design margin for the boundaries, it is easy to explain that the chemical potential of sodium will never connect to the reactor safety.

The second design approach is introducing the double-wall-tube SGs with secondary system. This concept will minimize the consequences for sodium-water reactions and simultaneous penetration break would be addressed as beyond DBE by providing the detection system both for water and sodium. A porous metal filled type and a mechanically bonded type had been tested in ETEC within the framework of DFBR design study. The applicability to the advanced loop type is now under investigation.

The third one is developing a new type SGs without Secondary system, where the possibility of sodium water reaction would be ruled out. The lead bismuth alloy is used as intermediate heat transfer medium, and there are two different design concepts: triple tube concepts, where the lead bismuth exists annulus zone between inner water tube and outer sodium tube(tube type concept), and water tubes and sodium tubes are separately installed in the lead bismuth tank (pool type concept) [14]. The fundamental feasibility is now under investigation with seeking for another concepts.

The above three approaches will be progressed during phase-II study of F/S (2001-2005). We will select the most promising SG concept from the views of technical feasibility, construction cost and anxiety for the consequences of sodium water reaction. These efforts for more reliable and robust SG design can be significant to conquer the drawback of sodium cooled FR.

## 6. Concluding remarks

The sodium cooled FR system is the one of the promising concept investigating within the framework of the F/S. The safety designs are now in progress for the advanced loop system. There are several kinds of R&D needs concerning safety, but crucial issues such as passive safety and elimination of re-criticality have been already launched and these are expected to complete within several years. As the result of these efforts, the sodium cooled FR system would establish easy-to-understand safety principle that would satisfy the complete passive safety features and rule out the re-

criticality issues during the CDA sequences. The IVR and no challenge to the containment under CDA conditions would be expected with confidence.

Long-term R&D issues are mainly related to the developments of new material and new subsystems, those are necessary to reduce the plant material for the improvement of the construction cost. Sodium related R&D issues have of great importance and need medium period, where ISI&R and SG issues should be highly stressed.

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Figure 1. **Safety design concept for LMR**

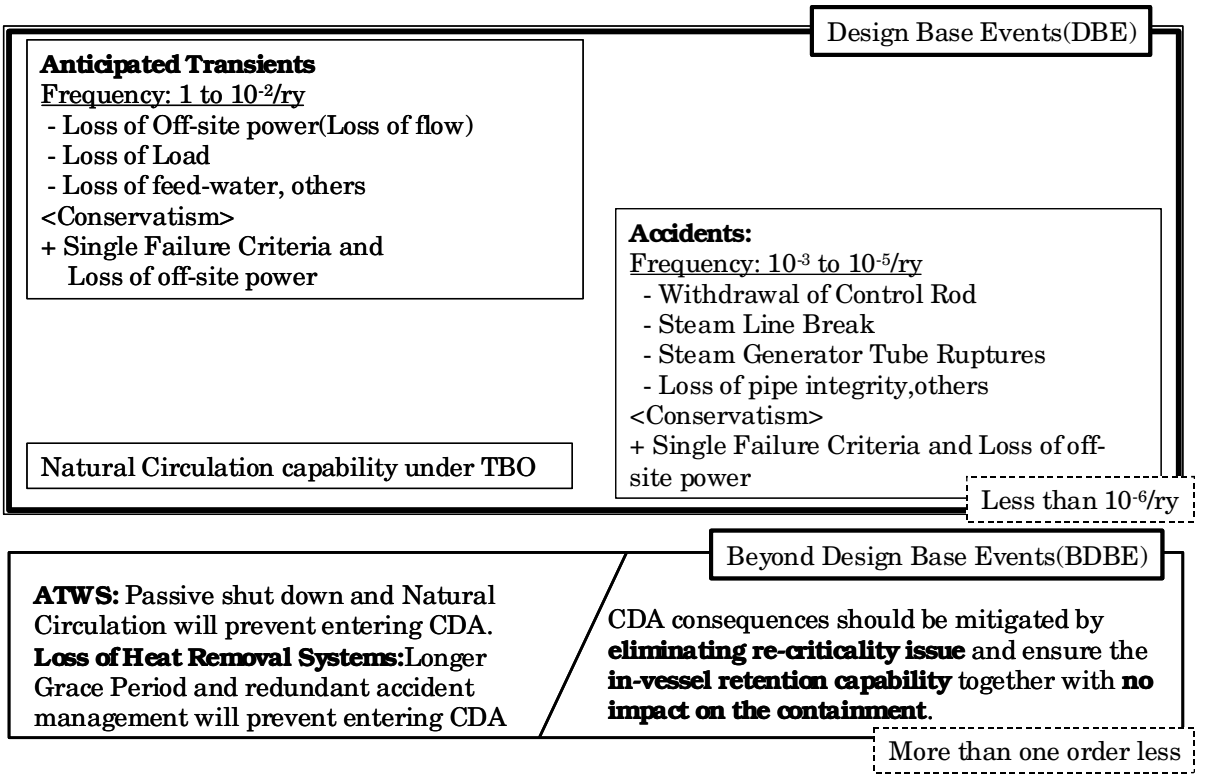



Figure 2. **Advanced loop concept and its innovative technologies**

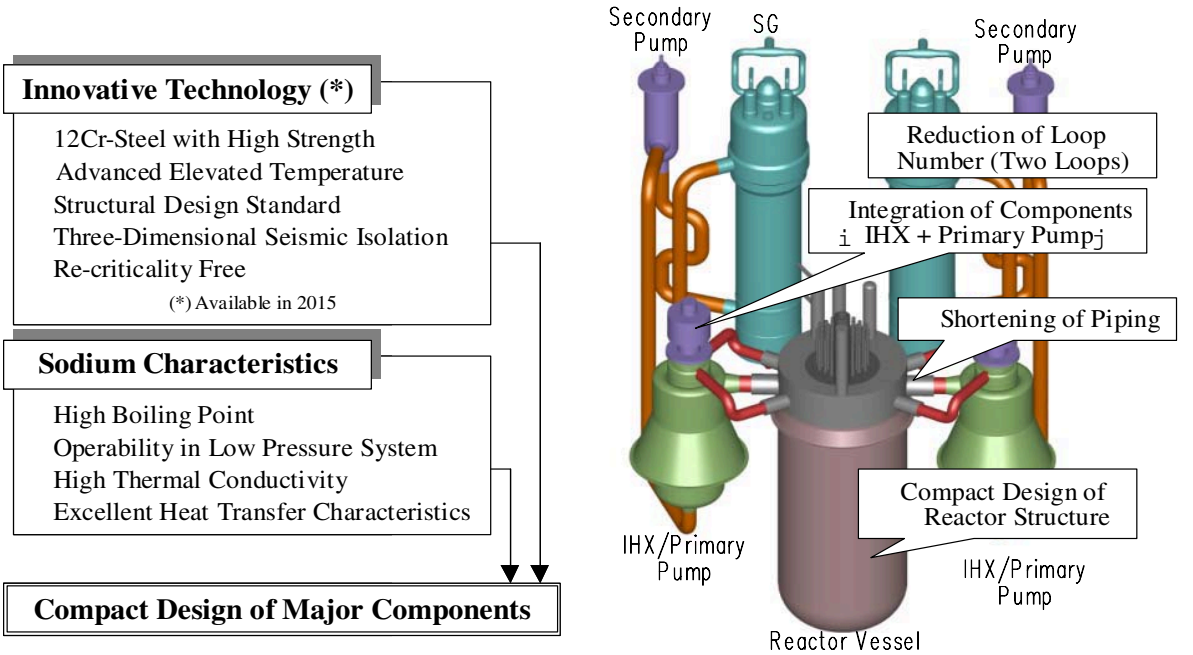


Figure 3. **Approach to the development of 3D seismic isolation system**

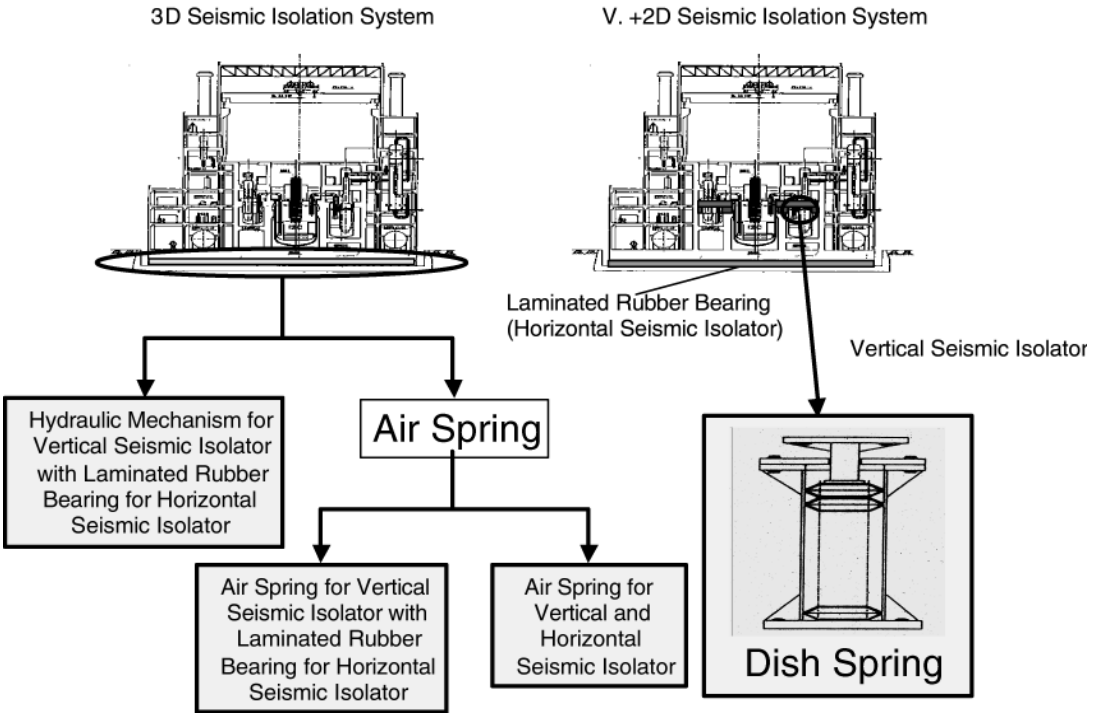



Figure 4. **Advanced loop systems and related R&D issues**

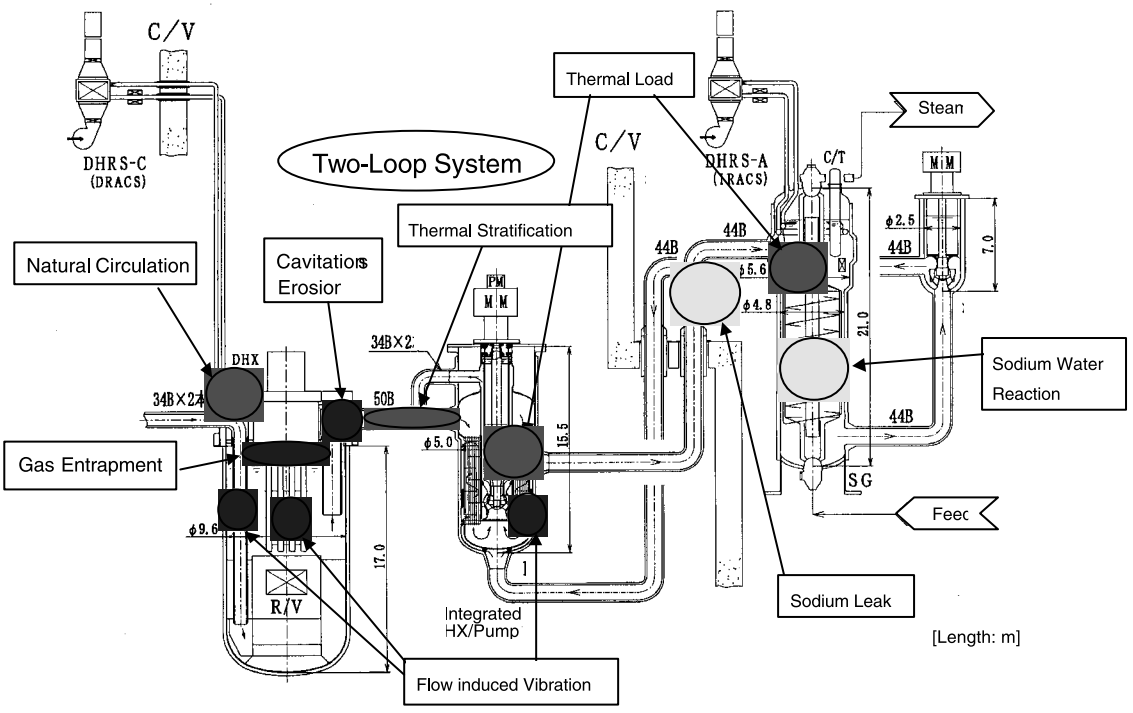


Figure 5. **Passive shutdown capability under ULOF in 1 500 MWe MOX LMR**

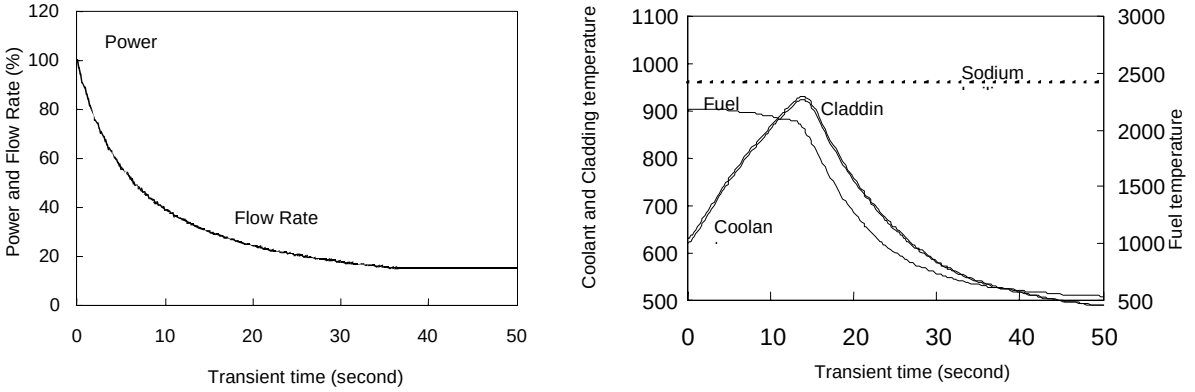
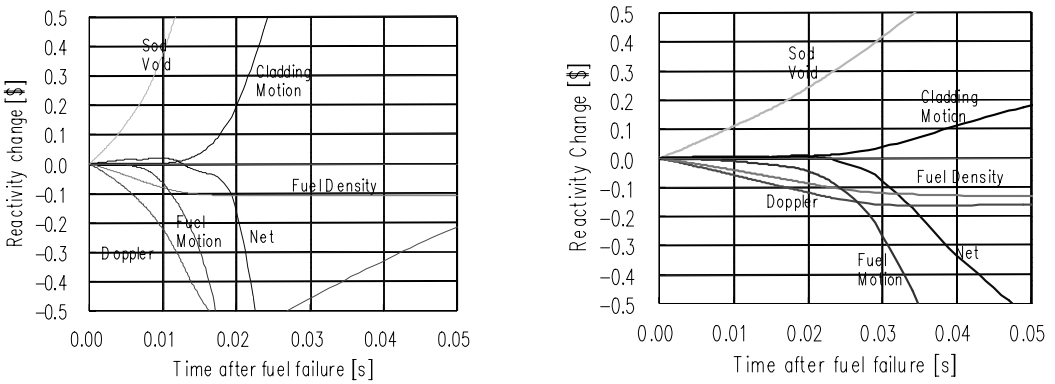



Figure 6. **Effect of sodium void worth during the initiating phase of ULOF**

| Case | Void worth (positive sum) ($) | Peak net reactivity ($) | Peak normalised total power (P/Po) | Peak average fuel temperature (K) |
|---|---|---|---|---|
| a. Void 7.4 $ | 7.4 | 1.015 | 1 200 | 3 210 |
| b. Void 5.4 $ | 5.2 | 0.964 | 110 | 2 750 |




Case a: Void 7.4 $ Case b: Void 5.4 $

Figure 7. **Fuel assembly candidates for re-criticality free core**

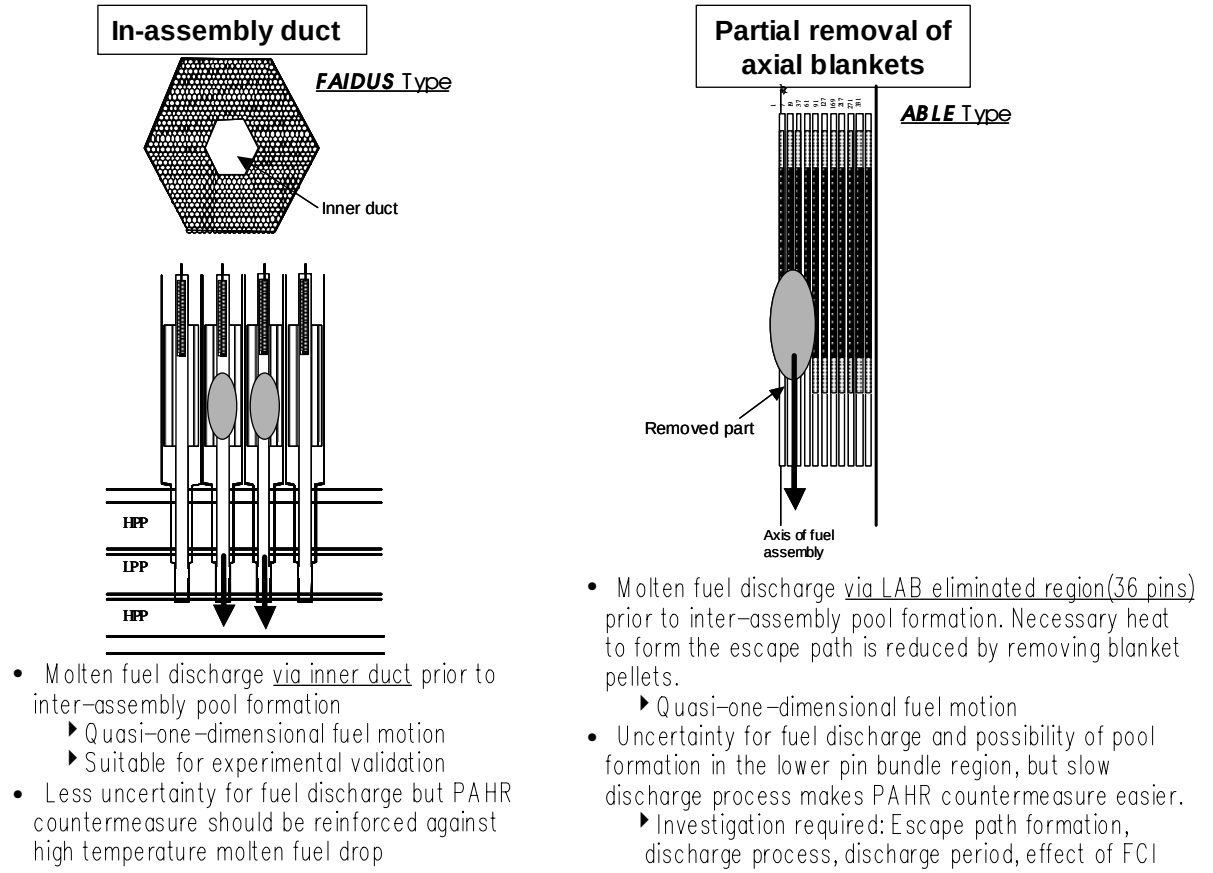



Figure 8. **Typical power profile of CDA in re-critically free core with FAIDUS**

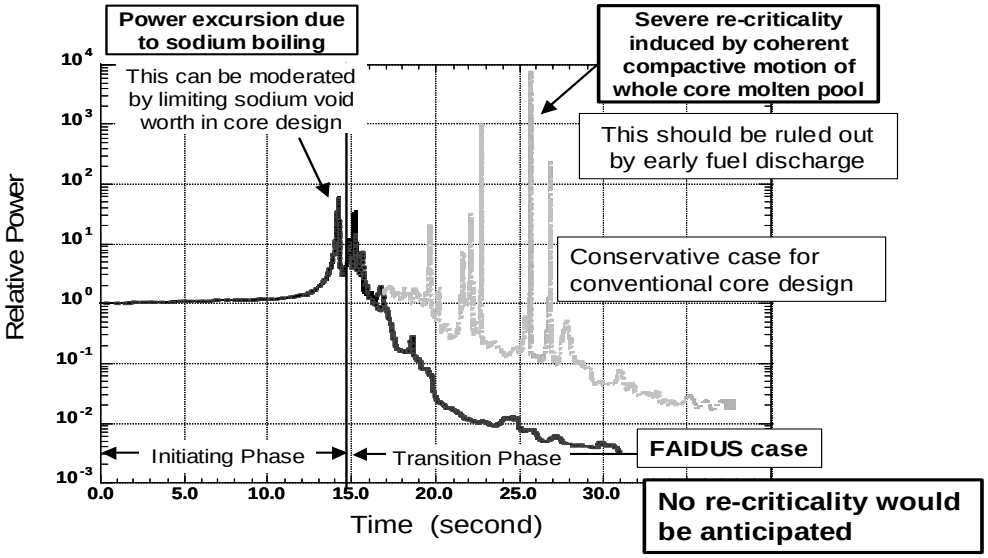
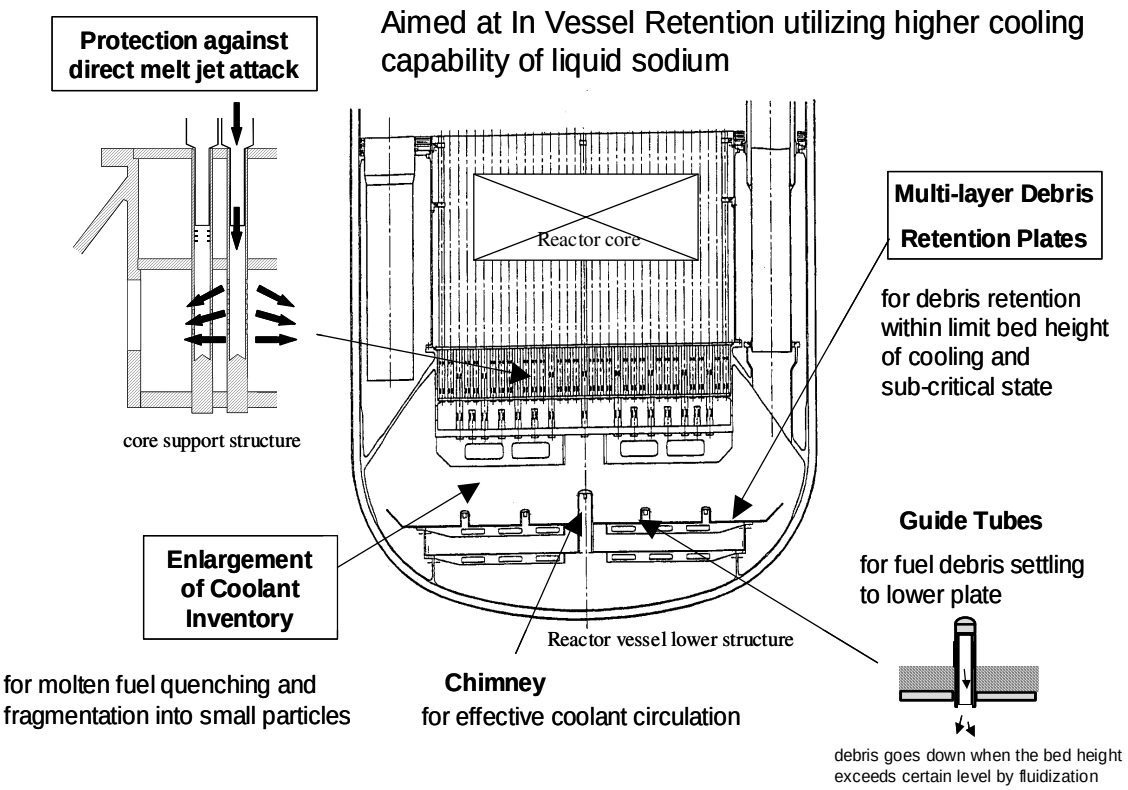


Figure 9. **Design measures for post accident heat removal**



# SAFETY ISSUES FOR LMFBRs:
# IMPORTANT FEATURES DRAWN FROM THE ASSESSMENTS OF SUPERPHÉNIX

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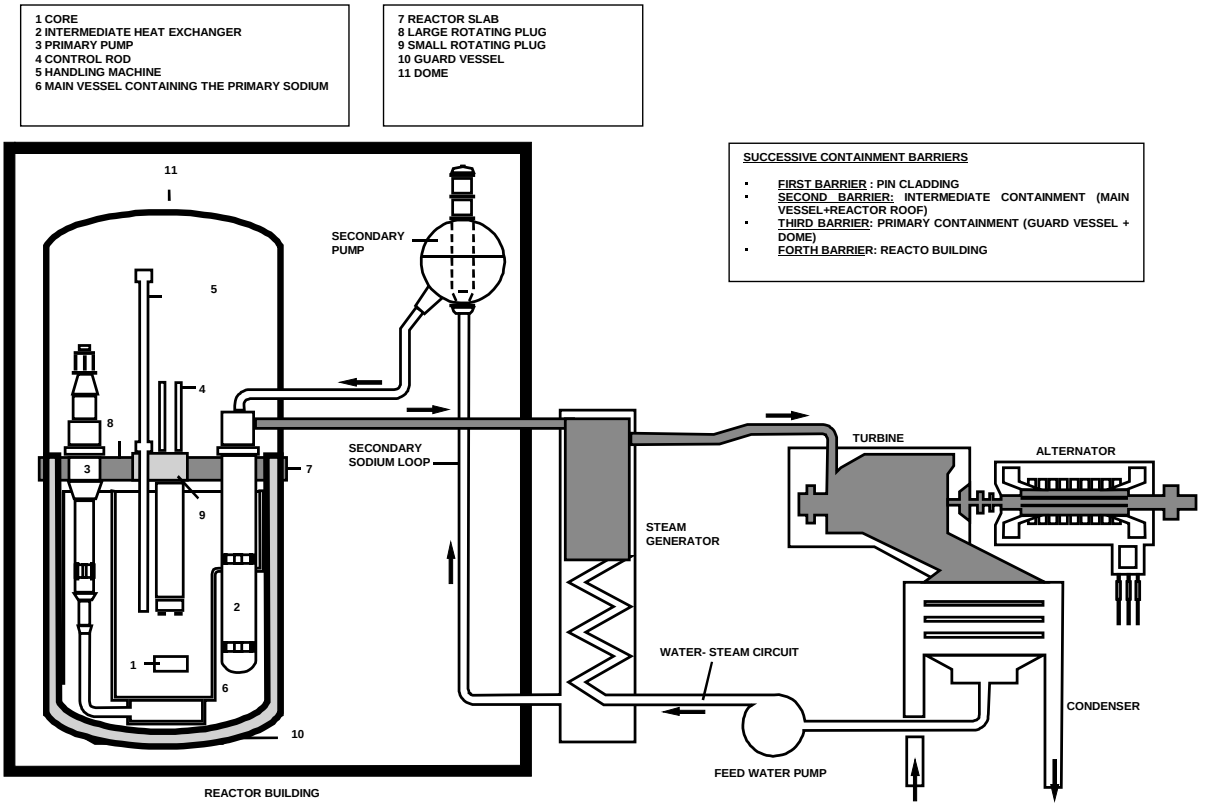
Superphénix, which is built on the site of Creys-Malville, is still the biggest LMFBR plant that has been in operation. It is a pool type reactor, as Phenix and the RNR 1 500 and EFR projects. Figure 1 schematizes the general arrangement of the normal circuits and of the containment system of Superphénix. The main vessel contains 3 000 t of sodium; there are four primary pumps and four secondary loops, each of them having two intermediate heat exchangers, one steam generator and one secondary pump on the cold leg.

Figure 1. **General arrangement of the normal circuits and of the containment of Superphénix**



After the analysis of the preliminary safety (1974-1975), the construction was authorised by decree of the Prime Minister in 1977, the authorisation for fuel loading and star-up to 3% was given by the minister of industry in July 1985 and full power was achieved in December 1986. The plant was operated until the end of December 1996, producing the equivalent of 320 EFPD, corresponding to half of the maximum burn-up of the first core. The plant was definitively stopped on the 20th of April 1998 by a decision of the French government.

During this period of 25 years of licensing, construction and operation of Superphénix, others discussions and preliminary licensing procedures were started for new projects, mainly the RNR 1500 French project and the EFR European project. The operation of Superphénix was also marked by several incidents, which led to additional licensing procedures and important modifications. This period was also marked by an important work of research and development in the safety field, mostly related to the issues concerning hypothetical core disruptive accidents (HCDA) and sodium fires; further, this period was marked by the Three Mile Island accident in 1979 and the Chernobyl accident in 1986.

Table 1 gives some milestones of the licensing, construction and operation of Superphénix together with some indications about the assessments concerning the RNR 1500 project.

Table 1. **Some milestones on the licensing, construction and operation of Superphénix and on the RNR 1500 project**

| | |
|---|---|
| 1973 | Recommendations on safety criteria applying to the Superphénix 1200 MWe Fast Reactor Power Station, issued by the head of SCSIN |
| 1974-1975 | Assessment by the “Groupe Permanent Réacteurs” |
| 1977, May 12 | Authorization of construction decree signed by the prime minister |
| 1985, July 19 | Authorization for starting loading up to 337 fissile assemblies, signed by the Minister of Industry |
| 1985, September 4 | Authorization for achieving the loading for criticality and stat-up to 3% nominal power |
| 1986, December 9 | Reactor at full power |
| 1987, March | Sodium leakage from the main vessel of the fuel storage |
| 1987 to 1991 | Replacement of the sodium storage tank by a gas handling device. Ultrasonic inspection of the main vessel. Procedure to enhance the defence-in-depth in case of leakage of the main vessel. |
| 1989, January 12 | Authorization to restart the plant |
| 1990, July 3 | Important pollution of the primary sodium by air ingress |
| 1990 to 1992 | Reassessments of the operation procedures and of the sodium fire issues |
| 1992, July 29 | Additional conditions required, by the Prime Minister, with respect to the sodium fire issue for the restarting of the plant |
| 1992-1994 | Important works in the secondary galleries and the steam generator buildings related to the sodium fire issues |
| 1994, July 11 | New decree of authorization of creation |
| 1994, August | Restart of the plant |
| 1995 | Leakage of the cover gas of an intermediate heat exchanger |
| Dec. 1995 to Dec.1996 | After the repair of the heat exchanger operation of the plant at full power |
| 1996 December 30 | Normal shut down of the plant at 320 EFPD for pressure test of the steam generator and fuel shuffling |
| 1997 February 28 | Annulment of the decree of the 11th of July 1994 by the Conseil d’État |
| 1998, April 20 | Confirmation by the ministers of finance, industry and environment that the plant must be definitively shut down. |

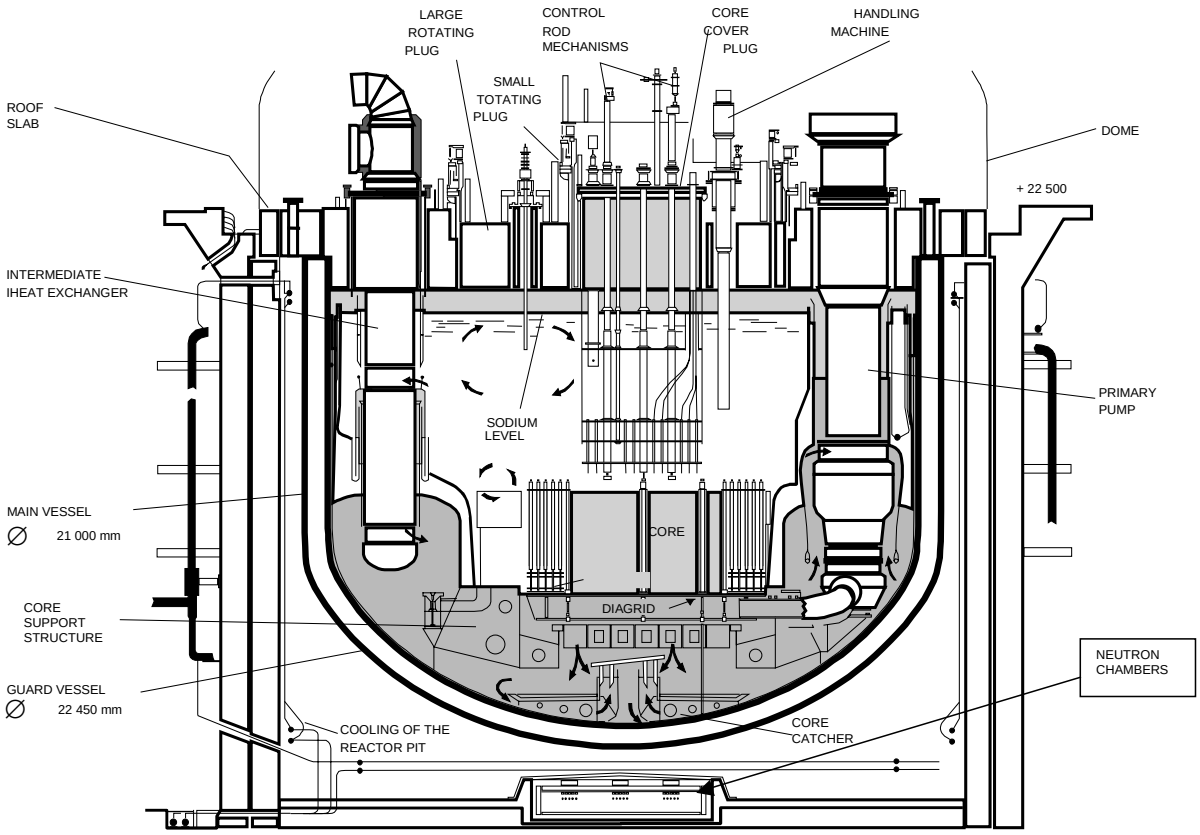
The purpose of this paper is to present some items which were discussed during this period of 25 years and which should be of interest for future LMFBRs. Several presentations on the safety issues of Superphénix, of the RNR 1 500 and the EFR projects have been made in the past (see for instance the references [1,2,3]). In this presentation, we shall discuss the key issues concerning the safety criteria and options taken with respect to severe accidents, i.e. core melt accidents, giving details on some specific which are less known since they were assessed only lately for Superphénix, sometimes in connection with the on-going safety researches.

To discuss the safety criteria and options concerning the severe accidents, one can make a distinction between the so-called Hypothetical Core Disruptive Accidents (HCDA) and the two other types of core melt accidents that are the loss of the primary coolant inventory and the loss of decay heat removal

## 1. Hypothetical core disruptive accidents

Figure 2 shows a cross section of the primary circuit and of its primary containment that includes the guard vessel and the dome, surrounding the intermediate containment that includes the main vessel and its roof (reactor slab and rotating plugs).

Figure 2. **Primary circuit, intermediate and primary containments of Superphénix**



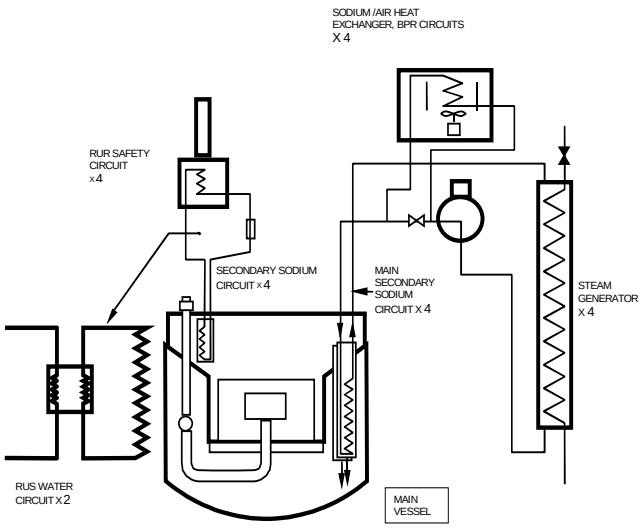
Accordingly to the decree of creation authorization, the primary containment must withstand the consequences of an accident releasing abruptly a mechanical energy of 800 MJ and leading to the ejection of 1 metric ton of sodium in the dome, burning as a spray fire corresponding to an overpressure of 3 bar relative. These requirements were made to cope with the consequences of a molten core accident, which can lead to a mechanical energy release either by a thermodynamic fuel coolant interaction (FCI) between the molten fuel and the sodium or by a vaporization of the fuel due to the recompaction of the damage core after a first melting. This last scenario corresponds to the called "Bethe Tait accident".

This type of conditions was generally applied to the containments of the LMFBR projects such as Rapsodie, Phenix, SNR300, Clinch River or Monju. It is linked to the fact that the core of LMFBRs are not in their most reactive configuration and so may be subject to a prompt critical excursion either by sodium voiding (in the case of big cores) or by recompaction after a first melting of the core (which is the so called Bethe Tait accident).

The studies made for Superphénix have shown that:

1. the intermediate containment withstands the 800 MJ work energy release (this was shown by calculations qualified on mock-up tests and a representative final 1/20 mock-up test); locally, the plastic deformation of the main vessel reaches 5%; the maximum ejected quantity of sodium through the roof is of the order of 250 kg; this is well below 1t, which is the quantity corresponding to the design overpressure of the dome in D (faulted) conditions;
2. the core catcher installed under the core support structure (see Figure 2) could maintain nearly the whole fissile material in a coolable and sub critical geometry [4] and the upper internals could withstand at least 70 % of the fissile fuel when ejected in the upper hot plenum;
3. after the mechanical energy release (i.e. the explosion), the reactor could be cooled by the internal (and external) emergency cooling circuits (RUR and RUS, see Figure 3), as these circuits would withstand the sodium dynamic loadings (which was a design condition for the RUR circuits which were added in a late stage of the project).

Figure 3. **Decay heat removal systems of Superphénix**



These last two points concern the long term cooling of the core, which is a necessary condition for maintaining the reactor in safe state. This question could only clarified lately in 1979, when correlations on the cooling of debris bed in sodium became available and it was then possible to design an internal core catcher with a cooling capacity of the entire molten core [3]. Further, it was demonstrated that this internal core catcher was protected by core support structure from the effects of the explosion and so should keep its geometry during the accident. Another point was to check that the main intermediate heat exchangers would not loose their integrity during the explosion since this could lead to an increase of the level of the primary sodium not tolerable for the mechanical behavior of the roof.

## 2. Classification and initiators of HCDAs

During the course of the licensing of Superphénix, it was clearly recognized that the HCDAs could not be considered as design basis accident, since firstly they could not be assessed taking into account all the possible conservative assumptions and margins, as this must be the case for design basis accidents, and secondly some points of the demonstration could be considered as weak points. For instance, one cannot be "100% sure" that the core catcher will keep its horizontality after the explosion, or that no internal missile could not damage the heat exchanger or that the ejected fuel in the upper plenum would not accumulate in some part of the upper structures forming a critical mass which could by-pass the core catcher and fall on the main vessel.

Since the containment system could only be considered as a medium line of defence (let us say "with 90% confidence") for the mitigation of the consequences of HCDAs, the accidents of this type were considered as Beyond Design Basis Accidents. The counterpart of this position was that it was clearly necessary to make the maximum possible efforts to prevent this type of accidents which means firstly to look very carefully to all the possible causes, which will be presented shortly hereafter.

### *Gas entrainment in the core. Core movements*

Voiding of a significant part of the core could result in a prompt critical excursion; such a voiding could result either from massive introduction from the diagrid through the subassembly inlets or from simultaneous clad ruptures in a large part of the core.

The risk of accumulation of argon in the diagrid and consequent sudden entrainment in the core was thoroughly studied theoretically and experimentally, with experiments on mock-up and on the reactor. An experiment of depressurization of the argon cover gas of an intermediate heat exchanger was also performed.

Since the relative movements of the subassemblies are not strongly constrained, the various possibilities of concentric compaction of the core have been assessed, looking to different possible initiators

From these various analysis and experiments, it was concluded that there was no possibility of gas entrainment in the core or core compaction leading to a significant positive reactivity insertion. Concerning the possibilities of simultaneous clad rupture only the earthquake, which would initiate an early reactor scram, appears to be a possible cause.

***Subassembly and advertent control rod withdrawal accidents***

Neutronic studies have shown that, even if the control rod are totally inserted in the core, the core can come back to a local criticality if 3 to 7 adjacent subassemblies are compacted, depending either this occurs in the outer or the center zone of the core. In this situation, in particular if the cladding steel is ejected above the fissile material, it appears that the surrounding hex-cans could melt so that the melting of the subassemblies would propagate to the whole core. Two accidents that could lead to this situation have been identified: the so-called subassembly accident and the inadvertent control rod withdrawal accident.

Concerning the subassembly accidents, it appears that, with the instrumentation of the core (the reactor scram is activated by either by the detection of delay neutron in the sodium or by the measurements of the temperature of each subassembly outlet), only a total instantaneous blockage could lead to the propagation of the melting of one subassembly to its neighbors. With the conception of the inlet of the subassembly this event seemed quite impossible.

Concerning the inadvertent rod withdrawal accident, the analysis has shown that this type of accident was badly protected since the protection by the neutron chambers (working with a 2/3 logic was rather late due to the flux deformation. The treatment of the outlet subassembly temperatures had to be modified in order to improve the protection. However, it appeared that it was difficult to stop the reactor before some melting in the central part of the pins of the subassemblies close to the withdrawn rod, this depending of the initial peak power of the pins (see Figure 4). In such a situation the question was either one could tolerate or not a certain amount X of molten fuel in the fuel pins of several subassemblies.

This question has been discussed from a general point view in a paper given at the 1985 Knoxville conference [2]; one has to fulfill two conditions: 1) a non-rupture condition (condition I), 2) a non-propagation condition if one rupture nevertheless occurs. This second condition called "condition II" has to be imposed since to satisfy condition I cannot give enough confidence that there would be no rupture during the transient since several thousands of pins are concerned by the loadings of the transient; so condition II can be defined as "there must be no-propagation toward a core melting if some adventitious clad failures occur".

In the case of the inadvertent rod withdrawal accident, the analysis has shown that, for the type of pins of Super-Phénix, with hollow pellets, the condition I was fulfilled even if there was an important amount of melting in the central volume of the pins.

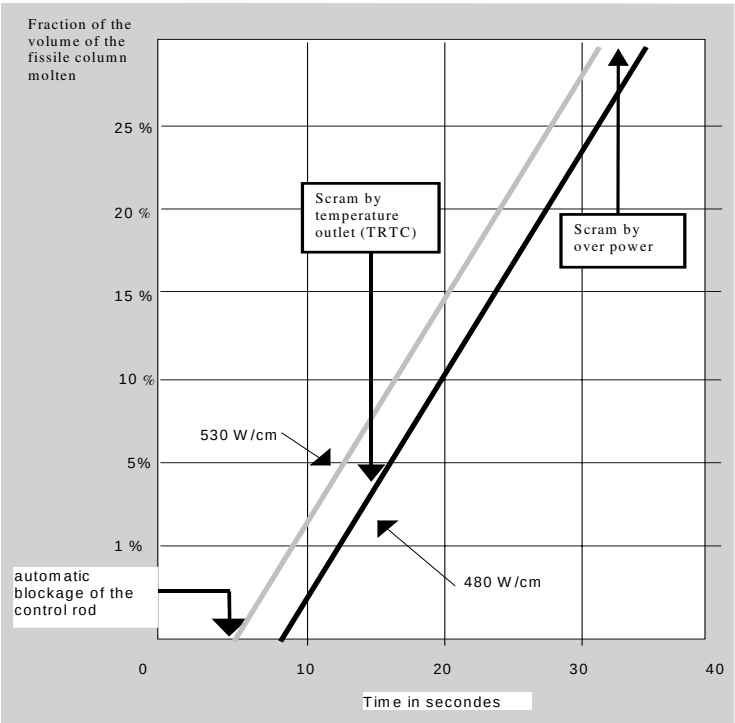
Concerning condition II, it seems clear that if there is a fuel ejection during the transient, this ejection would cause a rapid propagation of the failures to the adjacent pins by over heating of their cladding, with again ejection of the fuel. So that this was clearly a process for a subassembly melting, which could affect several subassemblies near the withdrawn rod.

To determine until what fraction X of molten condition II could be fulfilled, two special slow overpower experiments, in which a clad rupture could be induced during the transient, have been performed in CABRI, one leading to a fuel ejection and the second to no fuel ejection. The results, which have not yet been published and analysed thoroughly, have shown that an amount of the order of 10% of the section of the fuel pellet at the peak power of the fuel seems to be tolerable.

For Super-Phénix, it was decided to limit the consecutive movement of each rod: 1) first by a procedure asking to the operator not to rise each rod consecutively more than 10 mm, 2) secondly with an electronic device which stop the rod if it is risen consecutively more than 15 mm. There were some

further limitations on the maximum peak power and the total reactivity of the core during its operation until 320 EFPD, so that in practice the fraction X could be kept equal to zero until the action of the first scram signal, which is the one delivered by the measurements of the sodium temperatures at the subassembly outlets (see Figure 4).

Figure 4. **Fraction of molten fuel as a function of time in the inadvertent control rod withdrawal accident**




***Unprotected loss of flow accidents***

Without going into the details of all the possible behaviors following a loss of flow accidents without scram (ULOF), it seems useful to point out here that, in the case of Superphénix, the probability of such an accident was very low. This is due to the redundancy of the shut down system.

Further in the cases of loss of the power supply with failure of the protection system, the operator had at least 9 minutes to scram the reactor manually. This feature was linked to inertia wheels that were installed on the power supplies of the primary pumps, which allowed avoiding a rapid decrease of the primary sodium flow in case of loss of power, in order to make possible the cooling by natural convection in such a situation.

The only ULOF sequence with an estimated probability of the order of $10^{-8}$/year was an unprotected simultaneous slow down of the four pumps due an error of operator on the common command of the four pumps. In this situation, according to best estimate calculations the sodium boiling would nevertheless not occur; with a conservative set of parameters, it appeared that boiling could occur, but that the operator had at least 3 minutes to stop the reactor.

Concerning, the envelope character of the figure of 800 MJ:

- there have been extensive works on the possible primary excursions following a ULOF, with the SAS3D [4] and the PHYSURAC codes; these calculation have shown that the figure of 800 MJ was a good envelope[1];
- for the re-compaction which could follow a core melting, at the time of the start-up only simplified calculations were available; it was only recently that it was possible to check with SIMMER2 calculations that the figure of 800MJ is also correct with respect to the transition phase issue.

## 3. General approach for future projects

To include or not mitigation features of the consequences of HCDAs was a matter of numerous discussions for the future projects.

In 1982, there was some agreement between the applicant and the safety authority [6] to use criteria that, at least in principle, could avoid including any mitigation features against HCDAs in the conception of the containment. These criteria were expressed in terms of lines of defence, a notion that was first used by the DOE [7] for the research programs on LMFBRs and also promoted by P. Tanguy [8].

In the line of defence method that was used in the discussions between the applicant and the safety authority, one further uses a qualitative quantification of each line of defence by making a distinction between **strong lines of defence** and **medium line of defence**.

A **strong line of defence** corresponds either to:

- the components or systems which are needed in normal or anticipated transients and whose failure would lead to an accident classified in 3rd or 4th category; straightforward examples of such a first line of defence are: the main vessel, the normal decay heat system;
- the safeguard components or systems, which are designed with respect to the conditions imposed by hypothetical accidents; examples of these lines of defence are the guard vessel and the emergency decay heat removal system.

In the usual design approach, the two first lines of defence belong the design basis of the reactor, the objective being to limit the radiological consequences below the" limits of the 4th category, which may be qualitatively defined as releases for which no significant out of side restriction should be necessary; these lines of defence corresponds to components or systems designed and fabricated with the highest standards, taking into account conservative margins. Redundancy, diversity and geographical separation must be introduced in the design of all the concerned systems or components, in order to achieve a high degree of reliability coherent with the general probabilistic objectives and to cope with common failure modes that may impede the required independence of the line of defence.

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1. One must notice that the 800MJ figure corresponds to the work energy produced by the expansion of the vapor bubble down to 625 $m^3$, which is roughly the volume of the cover gas + the deformation volume of the main vessel during the explosion. For an expansion down to one atmosphere the energy work would be 2000MJ.

A **medium line of defence** essentially corresponds to what is called an additional beyond design basis feature, the objective being an additional reduction of the risk by a factor of 10 to 100, either by a further prevention of severe accidents or by mitigation of their consequences. These lines of defence are not necessarily designed, realized and qualified with the same conservatism than the strong lines of defence.

Typically, as this was stated before, the containment of Super-Phénix can be considered as a medium line of defence.

The preceding schema must be adapted to each particular risk, in particular when dealing with low probability abnormal transients that can be associated to the failure of a medium line of defence, or design basis external events that can be virtually identified as the failure of a strong line of defence. In fact the approach is essentially a simplified probabilistic approach which can be used when the systems are not to much redundant; in particular, its application to the decay heat removal systems is difficult and in any case must be completed by a probabilistic assessment.

By weighting for 1 each strong line of defence and ½ each medium line of defence, the general criteria which was settled down [5] for the RNR 1 500 project was to identify with respect to each particular risk a total number of lines of defence with a weight of at least 2 ½.

An important point, which must be emphasized, is that when assessing a risk reduction feature, no qualitative difference is made between reducing the probability of occurrence of an accident and mitigating its consequences. This had of course the important output that the licensing authority accepted not to impose the mitigation of the consequences of core melts accident for the containment design. The consequences of such an approach were the need for very strict and extensive requirements for the prevention of core melt accidents.

Nevertheless after additional discussions, in 1983, it was agreed [8] to keep some mitigations features HCDAs in the containment design of the RNR 1 500 project: an internal core catcher with a capacity of the full molten core and some resistance of the vessel closure were kept. This decision was more or less based on a “cost-benefit” analysis.

Afterward the line of defence approach was widely used by the utilities and engineering companies of the different countries involved in the EFR project.

After the Chernobyl accident, although the “2 ½ criterion” was kept for the prevention of core melt accidents, their was a clear proposal of the European utilities and engineering companies for keeping all the necessary features to mitigated the consequences of HCDAs up a certain level of work energy release, lower than 800 MJ. With respect to Superphénix, an important feature of the RNR 1 500 and EFR projects was the suppression of the dome.

## 4. Loss of the primary sodium inventory

The loss of the primary sodium inventory due to a leakage of the main and the guard vessel would lead to severe consequences to the environment, in particular if it occurs when there is still a significant residual heat in the fuel.

From the beginning of the assessment of the Super-Phénix project, it was clear that in case of a leakage of the main vessel, the reactor would have to be rapidly discharged outside the main vessel

since one could have to face a further leak in the guard vessel. Further, the reactor would have to be definitively stopped.

This fact was recognized for the RNR 1 500 project [2]: as a direct application of the line of defence criteria, the reactor cavity was designed in order to cope with simultaneous leakages in the main and in the guard vessel. This feature was kept for the EFR project [9]. The advantage of this design feature was that it was also possible to replace the external fuel storage tank by a storage of the fuel internal to the main vessel.

Concerning Superphénix, after the leakage of the main vessel of the storage tank, this vessel had to be removed from its cavity and replaced by a transfer facility, which could not be used to store the fuel. For that reason, it was necessary to install a special procedure (U4), with several additional devices, in order to be able to face an eventual leakage of the main vessel during the operation of replacement of the storage tank and also after the replacement. With these procedures and the associated additional devices, it was possible to face a leakage of the guard vessel occurring 3 months after the leakage of the main vessel.

## 5. Decay heat removal. Risks of common failures

The loss of the decay heat removal would lead to severe consequences.

For Super-Phénix, this function was studied by a probabilistic assessment for which special method was developed in order to deal with the long-term failures [10].

In the RNR 1 500 project, the heat exchangers on the secondary loops were suppressed. Correlatively the performances of the water steam systems were improved and the cooling capacities of the four emergency cooling loops, called RRA, were enhanced. In order to deal with such a decay heat removal system, it was necessary to look to the possibility of failure by common mode due to the design or to the fabrication. A method to deal with such failures that do not occur simultaneously was developed [13] and was applied to the RNR 1 500 project, using the data observed on the components of Phénix, where such common mode failures have been observed in particular on the intermediate heat exchanger. This method clearly shows the advantages of diversifying the four independent loops in two different design and fabrication.

This method is also applicable to the decay removal system of EFR that has six independent emergency cooling (RRA) loops.

## 6. The sodium fire issue

Another causes of common mode failure that can impede the decay heat removal function are the sodium fires. This issue has led to numerous various analysis and finally to important modifications in the secondary galleries inside the reactor, as well in the intermediate galleries and in the steam generator buildings.

Although one of the design basis options for Superphénix was to take the sodium fire corresponding to the guillotine rupture of the biggest secondary loop, which is the cold leg of 1 m diameter, the actual original design was done taking into account the pool fire corresponding to this rupture, ignoring what could happen between the rupture and the flour of the galleries. This “pool fire

option corresponded to a burning rate of sodium of about 2 kg/s, which led to a design pressure of the galleries of 40 mbar relative, with exhaust valves equivalent to 0.8 $m^2$ in each gallery.

During the discussion of the safety options of the RNR 1 500 project, it was clear that the hypothesis of a guillotine rupture was incredible and could be neglected for the future reactor as it was possible to use "leak before break" arguments to demonstrate that any crack would give a detectable leak before becoming unstable under the most severe loadings, which are the safe shut down earthquake and the sodium-water reaction in a steam generator; by unstable, one must understand that the crack propagation during the transient loadings must not lead to a significant rupture of the pipe. It was also clear that to apply such "leak before break" argument, it was necessary to improve the leak detection since experiments had shown that some small leaks could not be detected before the pipe was severely corroded.

Since, it was clear that the Superphénix galleries were not designed with respect to all the consequences of the guillotine rupture of the biggest pipe, it was proposed to apply the RNR 1 500 approach to Superphénix. At the time of the reactor start-up [1], a leak section of 10 $cm^2$ corresponding to a sodium spray fire of 15 kg/s was then retained for the design of the galleries. Nevertheless, this option already needed some modifications on some openings leading the reactor hall in order to increase the exhaust surface for these 15 kg spray fires.

In April 1990, there was a small sodium leak on an auxiliary circuit of 65 mm diameter; the analysis then showed that there was a necessity to assess the possible leaks on all the auxiliary circuit, which in fact were not equipped with individual leak detection systems. This led to conclude that to take into account a spray fire of 15 kg per second was insufficient and that further modifications were needed.

The sodium fire issue was completely reassessed in 1990 and 1991 after the pollution of the primary sodium together with the reassessment of some important issues. It was then asked to the applicant to fulfill all the conditions corresponding to the initial option of the guillotine rupture of the biggest sodium pipe, and in particular to take into account the spray part of the fire. After several discussions, the applicant was not authorized by the prime minister to restart the reactor in July 1992, without having made all the necessary modifications.

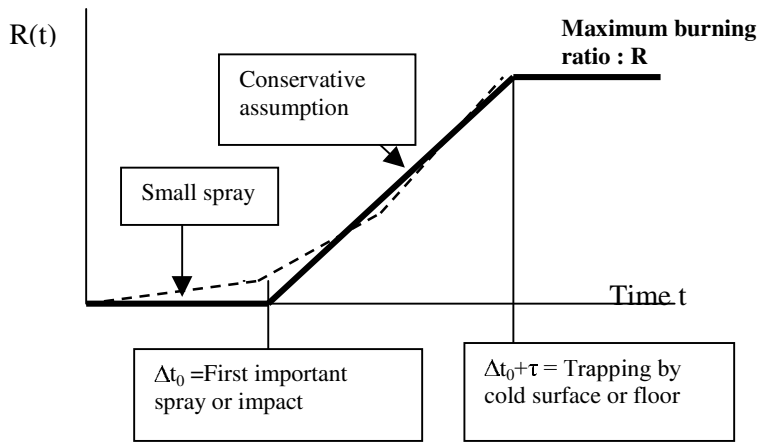
The modifications made in the secondary galleries were very important. They included:

- the separation of each gallery into 11 cells connected together by large openings; the idea being that this partition limit the quantity of oxygen which can react with the sodium, without limiting the overall capacity of the exhaust valves;
- the opening of additional exhaust valves in the wall of the reactor building (24 m2 of additional exhaust surface to be compared to 0.8 m2 initially); with this additional capacity, it was possible to cope with 70 kg/s spray fires without aerosols release within the reactor hall; additional openings toward the reactor hall were also installed in order to cope with the guillotine rupture of the biggest pipe, which corresponds to an initial sodium flow at 350°C of 3 300 kg/s increasing to 12 000 kg/s in 2 seconds;
- the reinforcement of the structure of the secondary galleries so that they could support an overpressure of 300 mbar, to be compared with 40 mbar initially.
- the installation of insulation on the concrete walls in order to avoid any water release during the fire and so sodium water reactions with a production of hydrogen.

To take into account all the consequences of the doubled ended rupture of the biggest sodium pipe, it was necessary to clarify what could happen in the first second of the fire since the maximum of pressure was obtained before 1 second in the case of a significant spray fire.

An important step in the solution of the sodium fire issue was to recognize that the burning rate of the spray sodium reaches its maximum, not immediately, but after a delay τ of the order of 1 to 3 seconds, depending on the size of the building cell. More precisely, ignoring the variation of the burning rate with the oxygen concentration, one can write as a function of time the burn mass Δm(t) of a mass of sodium $\Delta m_0$ ejected at the time t=0, as: $\Delta m(t) = \Delta m_0 R(t)$, with $R(t)$ varying from 0 to its maximum value R in a time $\Delta t = \Delta t_0 + \tau$, where $\Delta t_0$ represents the time of impact[2] of the ejected mass and $\tau$ represents the burning time of the spray mass. The variation of the function $R(t)$ is schematised on the Figure 5.

Figure 5. **Schema of the burning ratio R(t) (see formula 1)**




Taking now into account the variation of the rate of burning with the oxygen concentration, one can write the burning sodium flow as:

$$q_{burning}(t) = C(t)_{O2} / C(0)_{O2} \int_0^t q_{ejected}(t') \frac{dR(t-t')}{dt} dt' \qquad (1)$$

In practice, in the calculations, one can neglect the burning before the start of the actual spray ; further one can suppose a linear variation of $R(t)$ between zero and its maximum value $R$ (see figure 5). For the safety analysis, this procedure is conservative since the depletion of oxygen during the beginning of the sodium spray (before $\Delta t_0$. on Figure 5) is neglected, which enhance the calculated pressure.

2. $\Delta t_0$ can be equal to zero if the spray is immediate or if there is no impact.

For a constant ejected flow rate $q_0$ starting at time $t = 0$, the integration of the formula (1) gives the following burning sodium flow:

$$q_{burning}(t) = C(t)_{O2} / C(0)_{O2} q_0 R(t) \qquad (2)$$

When looking to the beginning of the pressure increase in the sodium fire experiments, it appears indeed that the form is initially parabolic and then linear after a time of at least one second corresponding to τ. The reason for this delay τ for the establishment of the full burning rate R is that the spray fire does not immediately occupy its full space. Further it can be shown that, for a constant ejected flow rate, the maximum burning ratio R (supposing that the oxygen concentration is constant) is more or less proportional to the time constant τ, which can be understood by the fact that the ratio R is related to the mean time of flight of the sodium drops until they are caught by cold surfaces, which is in fact the delay time τ.

In 1990, although there was quite a number of available experiments, all these experiments were done in close vessels and the theoretical effort was mainly focused on fitting the overall pressure variation [12], with no strong interest in the first second of the fire, which is in fact the main problem for the case of big sodium spray fires, when one has to cope with the inertia of the opening of the exhaust valves, so that the question of kinetics and transfers of masses are very important.

Table 2 shows the values of the parameters R and τ which could be obtained [13] on the available experimental results, with a linear variation of $R(t)$, after implementing formulas[3] of type (1) in the version V0 of the FEUMIX code [12].

Table 2. **Best fits of the experiments available in 1992**

| Facility | Test | Hole Ø [mm] | Ejected sodium* flow rate [kg/s] | Initial speed [m/s] | Distance to Impact [m] | Time τ [s] | Maxim. spray burning flow [kg/s] | Maxim. R of R(t) [%] | Comments |
|---|---|---|---|---|---|---|---|---|---|
| 22 $m^3$ facilty of Cadarache | IGNA2.2 | 10 | 1.5 | 23.3 | 2.25 | 1.2 | 0.28 | 19.5 | Impact on the walls |
| | IGNA3.2 | 26 | 5.4 | 12.4 | 2.25 | 2.0 | 0.87 | 18.2 | Impact on the walls |
| | IGNA+2 | 26 | 8.53 | 19.6 | 2.25 | 1.25 | 0.95 | 12 | Impact on the walls |
| 220 $m^3$ FAUNA facilty of KfK | FCA 2.2 | 10 | 1.76 | 28.0 | 4.25 | 1.2 | 1.06 | 60 | No impact |
| | FCA 3.2 | 26 | 5.05 | 11.8 | 4.25 | 2.8 | 4.12 | 87 | No impact |
| | FCA3.2 | 26 | 5041 | 12.8 | 6.50 | 2.1 | 3.82 | 74 | Impact on the walls |
| | FCA+2 | 26 | 7.85 | 18.2 | 4.25 | 3.5 | 5.11 | 74 | Impact on the walls |
| | FS 6 | 4 mm x 271 | 56.0 | 20.0 | 4.50 | 2.7 | 13.37 | 63 | No impact. The ejection lasted 1 s, with a constant flow |

* In these tests, the temperature of the ejected sodium was between 500 and 550°C

3. For the reactor calculations, it was also necessary to modify the modeling of the exhaust valves in order to treat correctly the inertia of their opening for big sodium spray fires.

In 1993, three additional tests were performed with high sodium flow rates in the KfK FAUNA facility (FCA 150) and in the Cadarache JUPITER facility of 3600 $m^3$ (IGNA 3602 and IGNA 3604).

Table 3 gives the main characteristics of these tests and of the parameters R and $\tau$ coming from the fits made with the version 3B of the code FEUMIX.

Table 3. **Results obtained in the last tests, which were performed in 1993**

| Facility | Test | Temperature of the ejected sodium [°C] | Ejected sodium* flow rate [kg/s] | Time $\tau$ [s] | Maximum. R of R(t) [%] | Comments |
|---|---|---|---|---|---|---|
| FAUNA 220 $m^3$ | FCA 150 | 550 | 150.0 | 1.7 | 15 | Upward flow |
| JUPITER 3600 $m^3$ | IGNA 3602 | 549 | 90.0 | 1.2 | 27.2 | Upward flow after impact on a plate |
| | IGNA 3604 | 530 | 225.0 | 1 | 13 | |

The results of table 3 show clearly that the maximum ratio decreases with the flow rate.

Another test (IGNA 2002) with 2 cells was also performed in 1993 with a flow rate of 136 kg/s, with the objective of estimating the entrainment of sodium in the cell adjacent to the cell where the fire first occurs. This test, which validated the concept of the modifications made in the galleries, will not be detailed here.

Concerning the modifications made in secondary sodium galleries, the applicant has validated the solution by taking into account two maximum initial flow rates of 1650 kg/s at 545°C on the hot leg and 3 300 kg/s at 350°C on the cold leg with respective maximum burning ratio R of 40 and 20% reached linearly in 1 second. The envelope character of these numbers was established by 2D calculations with the code PULSAR, which was adjusted on the available experiments in particular to fix the size of the droplets which has to be introduce in the data. For the propagation of the fire in the adjacent cells, due to non-burned droplets, it was suppose that the burning of the droplets was complete and instantaneous in the adjacent cells. Regarding the results obtained with the interpretation of the IGNA 2002 test, these hypothesis were conservative.

In all the interpretation work which was performed, a very important point was to be sure that there was no delay in the self inflammation of the droplets once they are created or introduced in an adjacent cell. There has been no evidence of any significant delay, which could of course lead to an explosive mixture. An important point was also to deal with the risk of air re-entrance in the cell before the end of the leakage.

It seems useful to mention that the solution which was adopted for Superphénix seems to be optimum since the secondary galleries were not inerted with nitrogen. With this respect, one must notice that inerting with nitrogen is not a solution to deal with big sodium leaks unless one designs the civil work for an overpressure of the order of 2 bar relative; one can also install exhaust valves on the galleries, but this solution has the same disadvantages as the Superphénix solution without its benefits, but with the additional difficulties of using nitrogen.

## General conclusion

A lot of progresses on the safety of LMFBRs were made during the licensing and the first year of the operation of Superphénix. This paper has been focused on the details of some safety questions which may have a generic character for future reactors of this type.

Some other questions such as the leakage of the storage tank in 1987, the in-service inspection of the main tank, the leakage of the cover gas in an intermediate heat exchanger in 1995, as well as some issues as the protection of a steam generator in the case of a leakage or a break of a water tube in the sodium, have not been discussed although they were very important.

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# SAFETY FEATURES AND RESEARCH NEEDS OF WESTINGHOUSE ADVANCED REACTORS

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## Abstract

The three Westinghouse advanced reactors – AP600, AP1000 and IRIS – are at different levels of readiness. A600 has received a Design Certification, its larger size version AP1000 is currently in the design certification process and IRIS has just completed its conceptual design and will initiate soon a licensing pre-application. The safety features of the passive designs AP600/AP1000 are presented, followed by the features of the more revolutionary IRIS, a small size modular integral reactor. A discussion of the IRIS safety by design approach is given. The AP600/AP1000 design certification is backed by completed testing and development which is summarized, together with a research program currently in progress which will extend AP600 severe accident test data to AP1000 conditions. While IRIS will of course rely on applicable AP600/1000 data, a very extensive testing campaign is being planned to address all the unique aspects of its design. Finally, IRIS plans to use a risk-informed approach in its licensing process.

## Introduction

In the late 80s Westinghouse initiated the next step in the evolution of light water reactors, introducing the passive design where passive safety features (core emergency cooling, containment cooling), which rely on laws of nature like gravity and natural circulation, replaced all active safety features of "traditional" PWRs. The embodiment was the AP600 reactor design that obtained a Design Certification in 1999. It is very important to note that improved safety came together with plant simplification. Compared to a conventional 600 MWe plant, AP600 has 50% fewer valves, 80% less safety related piping, 45% less seismic building volume, and no pumps or fans required for safety operation.

In spite of this, and surpassing the economic goals defined by the utilities at the time, the capital cost of AP600 is still not competitive with today's low priced combined cycle plants. Therefore, Westinghouse upgraded the 600 MWe AP600 design to the almost 1 100 MWe AP1000 version which will have economics competitive with the most advanced fossil units. AP1000 has of course the same safety features of AP600 and is backed by the results of the extensive AP600 test campaign. Its design certification process, started in 2001, is targeted for completion in 2004.

At the beginning of 2000, Westinghouse took a further major step in the progress of light water reactors technology, by initiating the design of the integral configuration IRIS (International

Reactor Innovative and Secure). While AP600/1000 replaced active with passive safety systems and simplified the PWR design, IRIS adopted a "safety by design" approach to eliminate the root cause of some accidents and thus eliminate as well the safety systems designed to cope with their consequences. Consequently, the design was further simplified with respect to AP600/1000. In terms of economics, IRIS is a small-to-medium power modular design and pursues the economy of multiples standardization, fabrication and installation. IRIS has completed the conceptual design and is currently in the preliminary design stage; it plans to initiate NRC licensing preapplication in mid-2002, with a goal to reach design certification by 2008.

## AP600/AP1000 Safety Features

For small break LOCAs, the relevant safety features of the AP600 and AP1000 designs are the core makeup tank, and the automatic depressurization system, which depressurizes the primary system in a controlled way to near containment pressure. The core makeup tank provides injection flow by gravity to the reactor vessel at any reactor coolant system pressure. The automatic depressurization system provides controlled venting of the reactor coolant system to reduce pressure and allow transition to injection by accumulators and then gravity driven injection from the In-containment Refueling Water Storage Tank (IRWST). For large LOCAs these features are augmented by flow coastdown from the high inertia centrifugal canned motor reactor coolant pumps (RCPs).

Core heat is rejected into the containment vessel atmosphere through the natural circulation passive residual heat removal heat exchanger during non-LOCA events or by direct release during LOCA events. Heat is then transferred through the containment shell and removed by ambient air on the outside surface of the containment. For these events the core remains covered and heat is transferred to the atmosphere without the use of any pumps or fans.

## IRIS Safety Features

IRIS adopts a "safety by design" approach, that is, to exploit to the fullest what is offered by its design characteristics (chiefly, integral configuration – see Figure 1 – and long-life core) to:

- Physically eliminate the possibility for some accident(s) to occur;
- Decrease the possibility of occurrence of most of the remaining scenarios; and
- Lessen the consequences if an accident actually occurs.

The approach is not new. It is nothing more than good engineering and it is the goal of all reactor designers. However, the case for IRIS is quite different because the integral configuration, if properly exploited, allows implementation of the safety-by-design approach to an extent which is not possible in traditional, loop LWRs. The most obvious implication is that large LOCAs cannot occur since there are no primary system components outside the reactor vessel. Large LOCAs are, however, only the most obvious example. Other accident scenarios such as LOFAs, steam and feed line breaks, steam generator tube rupture, and loss of off-site power can be eliminated or downgraded. Table I reports how the safety-by-design is implemented in IRIS (for a more detailed discussion see Reverences. 1-3). The result, as shown in Table II, is that out of the 8 class IV accidents typically considered for a PWR like AP600, 7 can be eliminated or reclassified to a lower Class and the only one remaining (refueling accident) has a significantly lower probability than for a typical PWR, because of more infrequent refuelings and no fuel assembly shuffling.

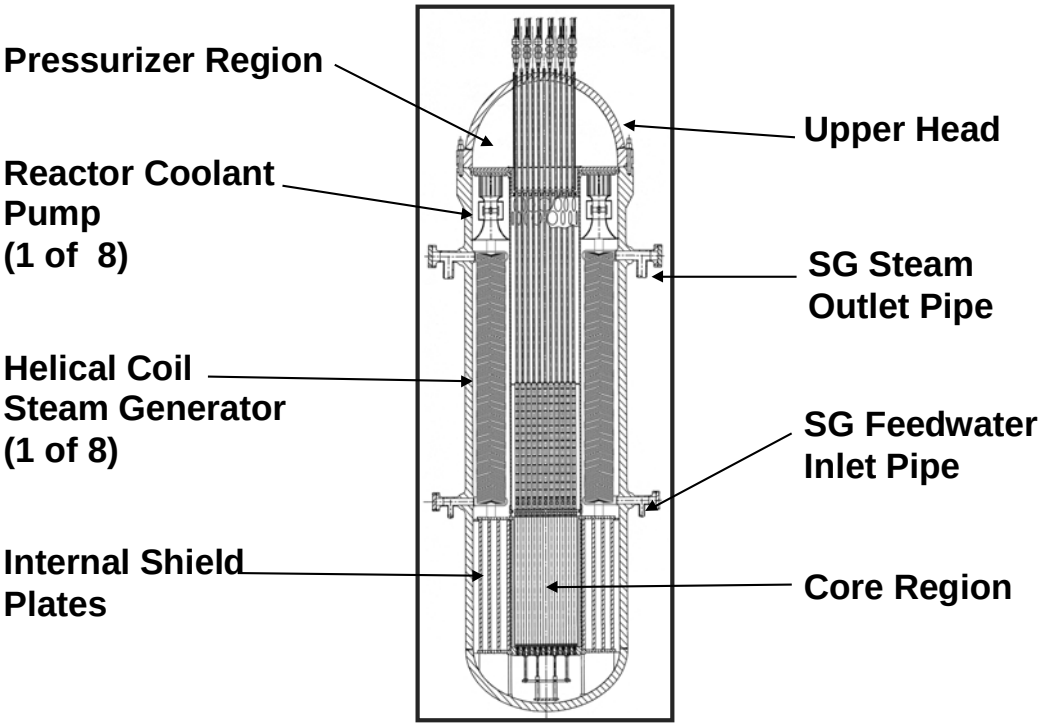
Table 1. **Implementation of IRIS safety by design**

| Design Characteristic | Safety Implication | Related Accident | Accident Disposition compared to current PWR |
|---|---|---|---|
| Integral reactor configuration | No external loop piping | Large LOCAs | Eliminated |
| Tall vessel with elevated steam generators | Can accommodate internal control rod drives | Reactivity insertion due to control rod ejection | Can be eliminated |
| | High degree of natural circulation | Loss-of-all-flow events (e.g. loss of offsite power) | High partial natural circulation mitigates consequences/reduces required pump inertia |
| Low pressure drop flow path and multiple RCPs | Core flow remains above DNB limit with sudden loss of one pump | LOFAs (e.g., pump seizure or shaft break) | Reduced consequences - no core damage occurs |
| Large water inventory inside vessel | Slows transient evolution helps to keep core covered | Small-medium LOCAs | Core remains covered even with no safety injection assumed |
| Reduced size, higher pressure containment | Reduced driving force through primary opening | | |
| Inside the vessel heat removal | | | |
| High pressure steam generator system | Primary system cannot over-pressure secondary system | SGTR | Reduced consequences – accident terminated quickly by simple automatic isolation |
| | No SG safety valves required | Steam and feed line breaks | Reduced probability and reduced consequences |
| Once through SG design | Low water inventory | | |
| Long life core | No partial refueling | Refueling accidents | Reduced probability |

Table 2. **Typical PWR Class IV accidents and their resolution in IRIS design**

| | Accident | IRIS safety by design | IRIS resolution |
|---|---|---|---|
| 1. | Steam system piping failure (major) | Reduced probability Reduced consequences | Can be downgraded to lower classification |
| 2. | Feedwater system pipe break | | |
| 3. | Reactor coolant pump shaft seizure or locked rotor | Minimal consequences | Eliminated as safety concern |
| 4. | Reactor coolant pump shaft break | | |
| 5. | Spectrum of RCCA ejection accidents | Can be eliminated | Not applicable (with internal CRDMs) |
| 6. | Steam generator tube rupture | Reduced consequences | Can be downgraded to lower classification |
| 7. | Large LOCAs | Eliminated | Not applicable |
| 8. | Design basis fuel handling accidents | Reduced probability | Still Class IV, but with significantly lower probability than for current PWRs |

Figure 1. **IRIS integral primary system**

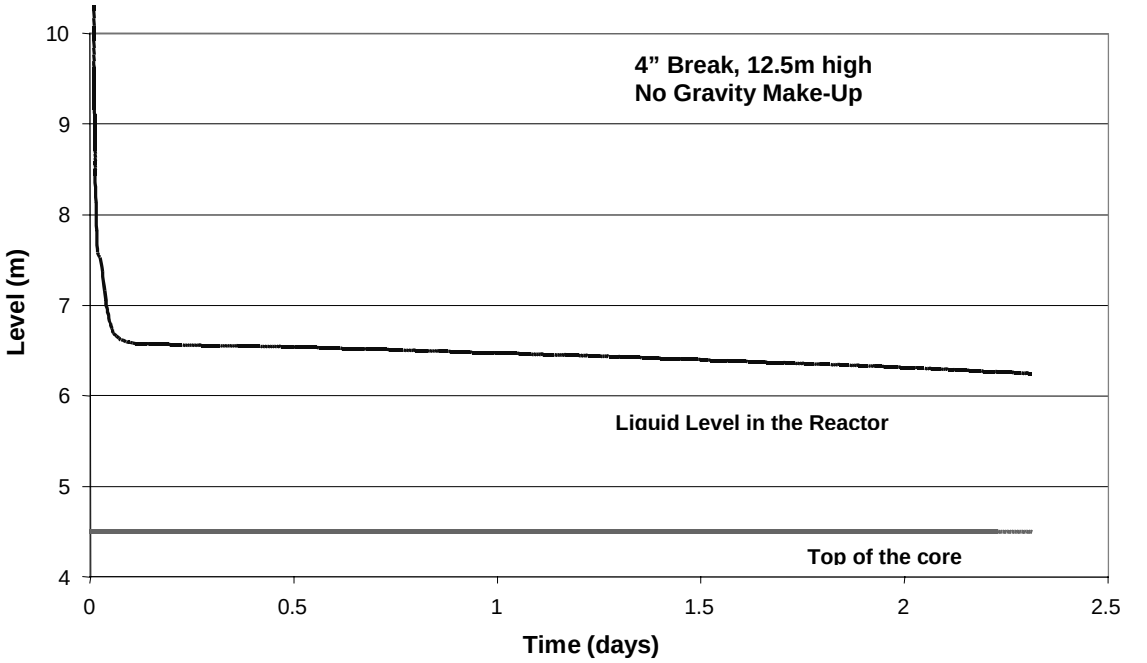



A unique IRIS feature is the thermohydraulic coupling of the vessel with a small spherical containment. Besides fulfilling the usual containment functions, the IRIS containment, in concert with the integral vessel, practically eliminates small and medium LOCAs as a safety concern. The underlying principle is quite simple: during an accident, the pressure differential across the break decreases quickly and becomes very close to zero, thus choking the egress flow. This is possible because the vessel pressure decreases due to the internal heat removal by the steam generators, while the containment pressure is higher than in traditional LWR containments. In fact, the IRIS containment can, other parameters being equal, sustain a peak pressure 4 times the value of LWR containments (factor of 2 due to spherical vs. cylindrical configuration and an additional factor of 2 because the diameter of the IRIS containment is half that of AP600).

Consequently, the core remains covered for an extended period of time (several days and possibly weeks, depending on the heat removal rate on the containment surface) without any emergency water injection or core makeup (see Figure 2). Therefore, IRIS requires no Emergency Core Cooling System (ECCS). There is, however, a suppression pool for limiting the containment peak pressure, which, similarly to the AP600 core makeup tanks, can also double as core makeup in a unforeseen emergency. IRIS has four (three independent) decay heat removal systems: steam generators, natural circulation driven heat exchangers located outside the containment, air and water heat removal from the containment surface.

The IRIS Vessel is located in a cavity open to the containment at an elevation above the top of the core. The cavity collects the containment condensate and is connected to the gravity makeup of the suppression pool. The IRIS core thus remains always covered over the whole range of accident sequences.

Figure 2. **The IRIS core still remains under 2 meters of water after 2 days in the worst small-to-medium LOCA Accident**




Other IRIS safety features are its high degree of redundancy (e.g., IRIS has eight steam generators and eight fully internal pumps), and the high degree of natural circulation, thanks to the significant thermal centers distance and the low pressure drop circulation path in the integral configuration and the open lattice core. The locked rotor/pump seizure accident, a Class IV in AP600, has so minimal consequences that no reactor trip is necessary (a conservative steady state calculation indicated that almost a 100% margin to DNB still exists at full power operation with seven pumps).

## AP600/1000 Safety Testing Program

Design certification of AP600 was backed by a very extensive testing campaign; AP1000 relies on the results of these tests, properly scaled to the higher power configuration, for its design certification process.

Tests conducted during the AP600 Conceptual Design Program in the late 80s provided input for plant design and to demonstrate the feasibility of unique design features. Tests, conducted in the 90s for the AP600 Design Certification program, were devised to provide input for the final safety analyses, to verify the safety analysis models (computer codes), and to provide data for final design and verification of plant components. An AP1000 specific Phenomena Identification and Ranking Table (PIRT) and scaling analysis and a review of preliminary safety analysis for AP1000 show that AP600 and AP1000 exhibit a similar range of conditions for the events analyzed. This provides justification that the AP600 data are sufficient to meet the requirements of 10 CFR Part 52 for AP1000. Table III is a list of the AP600 tests and AP1000 evaluations. Note that the AP1000 project reviewed each of the AP600 tests described and assessed their applicability to AP1000. These evaluations showed that the AP600 tests are sufficient to support AP1000 safety analysis.

Table 3. **AP600 design tests and AP1000 evaluation**

| Purpose | Test |
|---|---|
| LOCA Mitigation | Core Makeup Tank Performance<br>Passive Safety Injection System Check Valve<br>Automatic Depressurization System Hydraulic |
| | |
| Containment Cooling | Integral Containment Cooling<br>Passive Containment Cooling System Heat Transfer<br>Passive Containment Cooling System Water Distribution<br>Passive Containment Cooling System Wind Tunnel |
| | |
| Non-LOCA Transients | Passive Residual Heat Removal Heat Exchanger Performance<br>Departure from Nucleate Boiling |
| | |
| Integral Systems Tests | Low Pressure Integral Systems<br>Full Height Full Pressure Integral Systems<br>NRC Low Pressure Integral Systems |
| | |
| Component Design Tests | Incore Instrumentation System<br>Reactor Coolant Pump/Steam Generator Airflow<br>Reactor Coolant Pump High Inertia Rotor/Bearing |
| | |
| AP1000 Evaluation | AP1000 PIRT and Scaling Assessment<br>AP1000 Plant Description and Analysis |

The AP600 tests related to the plant safety functions were selected based on the plant features that are different from current PWRs and where directly applicable experimental data were not available. The tests simulate plant features as required to demonstrate the phenomena being examined. To validate the computer models, these experiments were modeled using the same computer codes used for plant analyses. Testing of some plant component designs was required to verify their reliability and manufacturability. Other component tests provided data for design optimization. A very brief discussion of the AP600 tests follows.

***Core Makeup Tank Performance Test***

The core makeup tank is unique to the AP600 and AP1000 design. This experiment verified the natural circulation and draining behavior of the core make-up tank over a full range of flowrates, pressures and temperatures. It also provided data to support the design and operation of the tank level indication, which acts as a control for the automatic depressurization system (ADS).

A one-eighth diameter and one-half height scale core makeup tank was constructed and instrumented to obtain condensation rates within the tank adequate to verify computer models.

***Passive Safety Injection System Check Valve Tests***

Both the AP600 and the AP1000 use check valves to isolate passive systems from the reactor coolant system. Tests were conducted to measure check valve pressure drop from very low flow to full flow conditions. Detailed data on initial valve opening, valve disk behavior and flow versus differential pressure were obtained for individual check valves as well as for valves installed in series.

***Automatic Depressurization System Hydraulic Tests***

The purpose of these tests was to simulate the automatic depressurization system, to confirm the capacity of the automatic depressurization system valves and spargers, and to determine the dynamic effects on the IRWST structure. A pressurized, heated water/steam source was used to simulate the water/steam flow rate from the AP600 reactor coolant system during various stages of plant blowdown through the automatic depressurization system.

***Integral Containment Cooling Test***

This test examined the combined effect of natural convection and condensation on the interior of the containment with film evaporation and air flow cooling on the exterior. This test demonstrated the operation of the passive containment cooling system over a range of operating conditions, including operation at low environmental temperatures. The cylindrical vessel used for this integral test was 3 feet wide and 24 feet high.

***Passive Containment Cooling System Heat Transfer Test***

A one-eighth scale steel containment structure with external water film and natural circulation air cooling and modeled containment internal compartments was constructed and used for this test. This test accurately models both the containment dome and side wall heat transfer areas. It complements the integral containment experiment that simulates the side wall condensation and evaporating film heat transfer. Measured parameters were the condensation heat flux distribution, the resulting heat transfer coefficients, the air/steam mass ratios, and the resulting liquid film evaporation rates. Both the integral containment cooling test and this larger scale containment test were modeled to verify the safety analysis computer codes and to demonstrate the scalability of the results.

***Passive Containment Cooling System Water Distribution Test***

This test was performed to examine and finalize the AP600 containment water distribution characteristics. The results provide input into the containment safety analysis computer codes for water coverage of the containment shell. The test was performed on a full-scale 1/8th sector of the containment dome. Measurements of water film velocities and film thickness variation as a function of flow rate and radial distance on the dome were obtained.

***Passive Containment Cooling System Wind Tunnel Tests***

Containment cooling relies on natural circulation of air to enhance evaporative cooling of the containment shell during a design basis event. Wind tunnel tests were performed to demonstrate that

wind does not adversely affect natural circulation air cooling through the shield building and around the containment shell.

An approximately 1/100-scale model of the AP600 plant, including the adjacent buildings and cooling tower structure, was constructed and instrumented with pressure taps. The model was placed in a boundary layer wind tunnel and tested at different wind directions. The results were used to design the shield building air inlet and exhaust arrangement and to determine the loads on the air baffle. Variations in site layout and topography were addressed using an approximately 1/800-scale model of the site buildings and local topography.

Tests were also conducted in a larger, higher speed wind tunnel on an approximately 1/30-scale model. These tests were conducted to confirm that the early test results conservatively represented those expected at full scale Reynolds numbers and to obtain better estimates of the baffle loads in the presence of a cooling tower.

***Passive Residual Heat Removal Heat Exchanger Performance Test***

The PRHR heat exchanger is located in the IRWST. This heat exchanger, which is connected directly to the reactor coolant system, transfers core decay heat and sensible heat energy to the IRWST water and depends only on natural circulation driving forces. This test determined the heat transfer characteristics of the PRHR heat exchanger and the mixing characteristics in the IRWST. The results confirmed the heat exchanger size and configuration.

***Departure from Nucleate Boiling Test***

Due to the shorter coastdown of the AP600 canned motor reactor coolant pumps, the flow rates at the time of minimum DNBR are somewhat below previously correlated flow rates. DNB testing was performed to extend the DNB correlation to lower flows. These critical heat flux tests were conducted using a 5x5, full length heated rod bundle with non-uniform radial and axial heating distributions.

***Low-Pressure Integral Systems Test***

The primary purpose of this experiment was to examine the operation of the long-term makeup path from the in-containment refueling water storage tank. In addition, analyses of this experiment demonstrated that the water flow through the core limits the long-term concentration of boric acid. The facility was capable of simulating high-pressure system responses.

The test modeled the reactor vessel, steam generators, reactor coolant pumps, in-containment refueling water storage tank, the automatic depressurization system vent paths, the lower containment and the connecting piping, hot legs, cold legs, core makeup tanks, PRHR heat exchangers, accumulators, and pressurizer. Water is the working fluid and the core is simulated with electric heater rods scaled to match the core power levels consistent with the test scaling approach. Tests were performed to simulate various small-break LOCAs with different break locations, break sizes, with and without non-safety systems operating. The Westinghouse safety analysis methods were compared to the test.

### *Full-Height, Full-Pressure Integral Systems Test*

This test facility was configured as a full-height, full-pressure integral test with AP600 features including two loops with one hot leg and two cold legs per loop, two core makeup tanks, two accumulators, a PRHR heat exchanger and an automatic depressurization system. The facility included a scaled reactor vessel, steam generators, pressurizer and reactor coolant pumps. Water was the working fluid and the core was simulated with electric heater rods. Tests were performed simulating small break LOCAs, steam generator tube ruptures and a steam line break transient.

### *Incore Instrumentation System Test*

Systems similar to the AP600 and AP1000 top mounted fixed incore detector instrumentation have been demonstrated in operating plants. A test was performed to demonstrate that the system will not be susceptible to electromagnetic interference from the nearby control rod drive mechanisms.

### *Reactor Coolant Pump/Steam Generator Airflow Test*

The airflow test was performed to identify effects on the pump performance due to non-uniform channel head flow distribution, pressure losses of the channel head nozzle dams and pump suction nozzle, and possible vortices in the channel head induced by the pump impeller rotation. The air test facility was constructed as an approximate one-half scale mockup of the outlet half of the channel head, the two pump suction nozzles, and two pump impellers and diffusers. The channel head tube sheet was constructed from clear plastic to allow smoke flow stream patterns to be seen. The test showed that no flow anomalies or vortices in the channel head were induced by the dual impellers.

### *Reactor Coolant Pump High Inertia Rotor/Bearing Tests*

A rotor, manufactured of depleted uranium clad with stainless steel, was incorporated into the hermetically sealed, high inertia centrifugal canned motor reactor coolant pump to provide the required flow coastdown performance for loss of flow transients. Tests were performed to determine friction and drag losses, to verify the operating performance of the pivoted-pad bearings, and to develop a detailed quantitative knowledge of the factors influencing bearing design and performance. Approximately 1000 cycles of starts and stops were also performed as a life test to demonstrate that the rotor will maintain its dimensional stability.

### *Beyond Design Basis Tests to Support Severe Accident Investigations*

A number of tests were performed to establish correlations and coefficients for the prediction of AP600 in vessel retention capability. Containment becomes flooded during design basis events to a level above the reactor vessel nozzles. This provides the mechanism for direct heat removal from the reactor vessel to the water surrounding it. Tests were performed to support the analysis showing that direct vessel cooling for AP600 maintains reactor vessel integrity for postulated severe accident scenarios. Similar testing was performed to extend the range of the correlations so that they also apply to AP1000.

**IRIS Research Needs**

Many traditional LWR accident sequences, starting with LOCAs, are not credible in the IRIS design. Thus, existing regulatory requirements written for loop-type PWRs need to be reviewed for their applicability to IRIS. A substantial streamlining is expected, once it is accepted by the US NRC that IRIS has indeed a lower number of possible accident sequences. On the other hand, because of its novel engineering features, extensive testing needs to be performed. This will be the most immediate research need.

IRIS will take advantage of the very extensive AP600/1000 testing campaign where applicable. For example IRIS adopts the same Passive Containment Cooling System, thus the scalability of related tests to IRIS conditions will be examined. The IRIS containment is much smaller than AP600 and thus closer in size to the test facility. Also, IRIS natural circulation decay heat removal heat exchangers are quite similar to the AP600 PRHR. On the other hand, all the significant differences from loop PWRs will be tested; this will include both the individual components as well as their integral behavior. These tests will be conducted on properly scaled models, and therefore the first activity will be a rigorous similitude analysis. Past experience has amply proved that time and money spent on achieving proper similitude and test planning have a many-fold return on hardware, testing and redesign savings.

Key individual components to be tested are the internal helical steam generators and the internal "spool" pumps. Both individual performance and interactive effects (e.g. flow effects due to positioning of the pumps on top of the steam generators) will be investigated. Scaled mockups in thermal hydraulic similitude will be used. Since water is the coolant in both model and prototype (same Prandtl number), thermal similitude is not a major concern and modeling can be focused on hydraulic and structural (vibrational) similitude. Testing of steam generator characteristics has already been performed in the late 90s by IRIS team member Ansaldo on a 20MWt scaled mockup (Ansaldo had developed the helical steam generator for their ISIS reactor design, which is in many aspects a precursor of IRIS). Similarly, the Washington Group has already conducted extensive testing of the primary pumps.

The other major class of tests will be focused on confirming the safety analyses results. In this context, interaction effects, such as the thermalhydraulic coupling of vessel and containment, are of primary importance. This will require demonstrating temporal as well as thermalhydraulic similitude. It will be impossible to simulate the entire transient behavior in any scaled model, short of a full scale mock-up. A possible approach is to "segment" the transient and simulate, with proper boundary conditions, only selected portions of the overall transient. Different models will be needed for the various portions, in order to properly reproduce boundary conditions and transient behavior. But first, analyses must confirm the adequacy of this approach in satisfactorily reproducing the transient behavior and also its economic advantage over a full-scale mock-up in the event that several different models are needed.

The availability of suitable testing facilities is obviously critical to the success of the IRIS pre-certification application project. Not by fortuity, many of the IRIS team members have specific test facilities that could be utilized for separate effect and integral effect testing. For example, prospective team member OKBM was the main designer of integral PWRs for the Russian submarine Navy and at the end of the 80s had developed a land version for commercial operation. Not only is OKBM the only organization in the world with experience in designing and building IRIS type reactors, but it also has an experimental facility where integral type effects can be examined. Also testing is planned to be conducted at facilities previously used for AP600, like Oregon State University in the US and SIET in Italy.

The second major research need for IRIS is how to implement in its licensing a new risk informed design and regulatory process which has been developed by Westinghouse over the last few years. Its implementation will require that new PRA models be developed to reflect unique IRIS design and safety features such as the interaction between the reactor vessel and containment during a small-to-medium LOCA. As in the current PRA method, the success paths are established based on deterministic analyses. The event sequences controlling the design are, then, based on the PRA results, not on "arbitrary" past experience.

The extensive testing campaign will provide the IRIS specific data base necessary to quantitatively support the risk-informed methodology. Additionally, where applicable, IRIS will draw upon the extensive data base from existing LWRs. The IRIS project will have to reflect the "safety-by-design" features, performance testing, and uncertainties in the PRA and make corresponding design decisions. This means that design features may have to be added or modified to meet the safety and operational criteria. The use of the PRA as an over-arching evaluation and decision-making tool also means that design features may also be deleted, if they do not produce a significant benefit to plant safety or performance. The three main areas of needed research are: (1) development of methods for handling all types of uncertainties that would be incurred in a new design, especially those previously addressed by the inclusion of deterministic design margin and subjective judgments; (2) development of probabilistic design criteria and the means for selecting probabilistic regulatory criteria; and (3) development of procedures for conducting the regulatory process.

Expectations are that implementation of risk-informed methods will provide an integrated method for evaluating the design and resolving regulatory issues, without reverting to arbitrary judgments which have a significant impact on the viability of the design. For example, a very detailed evaluation of offsite release sequences (both their frequency and magnitude) is expected to provide an impartial evaluation of the necessity for offsite emergency planning. Consistently with one of DOE's objectives for Generation IV reactors, our early judgment is that it may be demonstrated that offsite emergency planning in not required for IRIS.

This judgment is based on the fact that to start with, the safety-by-design approach that exploits the integral configuration characteristics provides a very strong, deterministic, starting basis. As previously shown, almost all sequences leading to Class IV accident scenarios in loop PWRs no longer do so in IRIS. Preliminary PRA analyses confirm that the Core Damage Frequency of events such as small LOCAs, line break and loss of off-site power is order of magnitudes lower than in AP600. On the other hand, application of risk-informed methodology to the System 80+ design indicated (Ref. 4) that the probability of the considered accident sequence can be substantially reduced, even by a decade with respect to "traditional" evaluations. Thus, when risk-informed regulation is applied to the significantly more benign IRIS accident scenarios it might be expected to demonstrate a reduction in the overall release probability sufficient to justify no need for off-site emergency response.

The newness of the IRIS design also provides a good opportunity to develop generic methods for other new nuclear plant designs (Generation IV) based on a reasonable combination of experience (proven LWR) and new challenges (e.g., development of probabilistic safety criteria, addressing the issue of no emergency planning). It is believed that the implementation of risk-informed research for IRIS and the interactions with the NRC on the development and implementation of new methods for regulatory positions during the IRIS licensing process will allow establishment of new risk-informed design and regulatory methods and procedures for Generation IV reactor designs in general.

Pre-licensing IRIS activities are planned to start in mid-2002 with a review of its approach to similitude analysis, test planning, and facilities selection. This will be followed by a review of preliminary PRA analyses and their incorporation in the overall risk-informed regulation framework.

## Conclusions

Westinghouse advanced water reactors portfolio covers a wide range in terms of readiness, size, design approach, safety features and related research needs. AP600/1000 represent the Generation III (passive designs)/III+ (with improved economics), are designed certified or close to be, have been extensively tested and thus require very little, if any, additional research. The highly innovative IRIS design represents Generation IV, has greatly simplified safety systems and requires substantial research to experimentally support the results of the safety analyses and to implement risk informed design and regulation.

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# APR1400 DESIGN: ITS SATETY FEATURES AND ASSOCIATED TEST PROGRAM

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## Abstract

APR1400 is an evolutionary PWR design being developed in Korea. With the completion of the design, the licensing review of the standard plant (similar to the design certification in 10CFR52) is in the final stage. The new safety features in APR1400 are the safety injection system with direct vessel injection, POSRV with spargers, in-containment refueling water storage tank, and severe accident mitigation system. To validate the new features, associated test programs were conducted. These features have been reviewed in detail during the licensing review and the integrated approach of testing and analytical supporting work has been effective in the issue resolution. The site for the APR1400 is chosen near Kori NPP site and the construction project for the twin units, Shin-Kori Unit 3&4 is in progress. The commercial operation of the Unit 3 is planned for September 2010. The implementation experience of the certified standard design into an operating plant will be useful for other countries interested in the standard design licensing approach.

## 1. Introduction

As a long-term advanced nuclear reactor development program, Korea Electric Power Corporation (KEPCO) has been developing an evolutionary PWR plant called Korean Next Generation Reactor (KNGR) since 1992. It has been renamed as APR1400 with its construction plan approval. APR1400 is a 1400 MWe evolutionary PWR design. APR1400 has been developed in three phases. The first phase is called the conceptual design phase. After surveying the candidate reactor types, we chose to develop an evolutionary PWR and set the top-tier requirements. The second phase is called the basic design phase, started from March 1995 till February 1999. The basic design of APR1400 has been completed. Standard Safety Analysis Report (SSAR) and specifications for major NSSS equipment have been completed. Detailed description of the design can be found in [1]. The third phase has started from March 1999. In this phase, considering the cost pressure of alternate fuel sources as well as global energy market, a major design optimization has been performed. The key challenge was how to reduce the cost while maintaining the overall safety. Detailed description of the optimization process and result is presented in [2]. The third phase has ended in December 2001.

The development team of the APR1400 consists of all major nuclear companies in Korea. Korea Electric Power Co (Korea Hydro & Nuclear Power Co. was a part of KEPCO before

restructuring) led the development project. KOPEC and KNFC designed Nuclear Steam Supply System (NSSS) and fuel. KHIC (now DHICO) reviewed the manufacturability of the NSSS design. KHIC also produced detailed designs of the reactor vessel and the steam generator. KAERI has performed several test programs to support the APR1400 development. In parallel with the APR1400 design development, licensing approach similar to the design certification in U.S. has been developed and became effective in 2000.

Currently, KINS (Korea Institute of Nuclear Safety) review for the design certification is on the way. Also, the construction program of the first two APR1400 units is in place and in progress as planned; first concrete in June 2005 and the commercial operation in September 2010. The site is near Kori nuclear power plants. Hence the two units are named as Shin-Kori Unit 3 and 4.

In this paper, we introduce the APR1400 design and its safety features. Then we review the test programs that have been important to support the licensing of APR1400. We, then, end the paper with the review of the licensing status.

## 2. APR1400 General Description

### *2.1 NSSS Design*

The major components of the primary loop are the reactor vessel and two coolant loops. Each loop consists of one hot leg, a steam generator, two reactor coolant pumps and two cold legs. A pressurizer is connected to one of the hot leg. A schematic diagram of APR1400 NSSS is shown in Figure 1. The core consists of 241 fuel assemblies. Each fuel assembly consists of 236 fuel rods (16×16 array) and 5 guide tubes. The core is designed for an operating cycle of 18 or more months with a discharge burnup up to 60,000MWD/MTU. The thermal margin is around 13%.

The steam generators are vertical U-tube heat exchangers with peerless type steam dryers, moisture separators. Inconel 690 is chosen as the steam generator tube material. Tube plugging margin of 10% has been reserved in the design. The hot leg operation temperature is set at 615ºF to reduce the tube corrosion probability.

The capacities of the PZR and the SGs (especially secondary side) are increased from that of current designs. The increased capacity of the pressurizer accommodated the plant transients without power operated relief valves. Conventional spring loaded safety valves mounted to the top of the PZR are replaced by the pilot operated safety relief valves (POSRVs). POSRVs will perform RCS overpressure protection and safety depressurization functions.

### *2.2 General Arrangement: 4 Quadrant Concept*

APR1400 is arranged on the twin-unit concept and slide-along arrangement with common facilities such as the radwaste building. The general arrangement is shown in Figure 2. The APR1400 plant consists of the nuclear island and the turbine island. Nuclear island includes containment building, auxiliary building, and diesel generator area. Turbine island has turbine building and annex building.

The auxiliary building, which accommodates the safety systems and components, surrounds the containment building. It is divided into four quadrants from the safety perspective. The four

quadrant arrangement ensures the physical separation of the safety systems including pumps. The safety injection pumps are located in the auxiliary building and each pump is located in each of four quadrants surrounding the containment. This will ensure the redundancy against external events such as fire, and internal flooding. The emergency diesel generator rooms are also separated and located at the symmetrically opposite side.

The containment is a post-tensioned concrete cylinder with a hemispherical dome. The steel containment liner is provided to ensure the leak-tightness. The containment building is designed to provide biological shielding, external missile protection, and to sustain all internal and external loading conditions which may reasonably be expected to occur during the life of the plant.

### *2.3 I&C and MMIS Design*

APR1400 is equipped with digitialized Man-Machine Interface System(MMIS) which encompasses the Control Room Systems and Instrumentation and Control(I&C) systems. The APR1400 MMIS concept is schematically depicted in Figure 3. The APR1400 MCR design is characterized by 1) compact workstations for operators, 2) Large Display Panel for overall process monitoring of the plant to be shared among operating crew 3) multi-functional soft controls for discrete and modulation control, 4) computerized procedure system and 5) safety console for dedicated conventional miniature button type controls provided to control essential safety functions.

I&C system is based on microprocessor-based multi-loop controllers for the safety and non-safety control systems. Engineering workstation computers and industrial personal computers are used for the two diverse data processing systems. To keep the plant safety against common mode failures in software due to the use of digital systems, controllers of diverse types and manufacturers will be employed in the control and protection systems. For data communication, a high speed fiber optic network is used. Since the S/W is heavily relied on in full digital MMIS, stringent S/W qualification process is established and the quality assurance program will be followed for the life cycle of APR1400.

The human factor engineering is an essential element of the Control Room facility design and Man-Machine Interface(MMI) design and its principles are systematically employed to ensure safe and convenient operation. Operating experience review analysis, function analysis, and task analysis is performed to provide systematic input to the MMI design. Partial dynamic mockup has been constructed based on the simulator of predecessor plant (KSNP) system models. This facility was used to perform verification of suitability of the MMI design.

## 3. APR1400 Safety Systems and Features

### *3.1 Safety System Design Philosophy*

The review of operating experience and the insights from the PSA, the strict adherence to design basis has its limitation in improving the safety of the plant. To further improve the safety of the plant, it is important to focus on more likely initiators such as transients, small-break LOCA and SGTR. Also, with the TMI-2 incident, design features against the severe accident are also necessary. Recognizing this, EPRI URD [3] proposes that the new ALWR will be designed in accordance with the licensing design basis to meet the licensing criteria and at the same time be designed with an

additional safety margin. The additional safety margin is to improve the investment protection as well as the public health.

APR1400 design followed the same philosophy in designing the safety system. The emphasis is on the response to more frequent initiating events such as transients and small-break LOCA. The severe accident mitigation features are also added on the safety margin basis. For example, the safety system is designed so that the reactor fuel will not uncover following the event of small break LOCA up to 15 cm pipe break. Another important design philosophy for safety is the increased design margins. Examples are the requirement of 10 ~15% core thermal margin, sufficient system capacity for operator recovery action time of more than 30 minutes, and 8 hours station blackout coping time.

### *3.2 Safety Systems*

The Safety systems consist of the safety injection system, safety depressurization system, in-containment refueling water storage system, auxiliary feedwater system, and containment spray system. A schematic diagram of arrangements and locations of safety systems are shown in Figure 4.

The safety injection system(SIS) is designed to inject to the upper downcomer directly. The safety injection lines are mechanically 4 trains and electrically 2 divisions without the tie branch between the injection lines. Each train has one safety injection pump and one safety injection tank. The common header currently in the SIS trains is eliminated. Functions for safety injection and shutdown cooling are separated. A fluidic device is located at the discharge of the safety injection tank (SIT). It is a passive system to inject the borated water into the RCS at a lower rate when the SIT level reaches a set level. The system has a capability to reduce the discharge flow to 10% of the maximum flow. This system will enhance the performance against the loss of coolant accidents by lengthening the water injection time.

The refueling water storage tank is located inside the containment. The spillover from the RCS through the break as well as containment spray would return to the IRWST. Through the IRWST the current operation modes of high pressure, low pressure, and recirculation during LOCA are merged into one operation mode (i.e., safety injection). The functions of IRWST are as follows; the storage of refueling water, a single source of water for the safety injection, shutdown cooling, and containment spray pumps, a heat sink to condensing steam discharged from the pressurizer for rapid depressurization to prevent high pressure core melt or to enable feed and bleed operation, and coolant supply to the cavity flooding system in case of severe accidents to protect core melt.

The AFWS is designed to supply feedwater to the SGs for RCS heat removal in case of loss of main/startup feedwater systems. In addition, the AFWS refill the SGs following a LOCA to minimize leakage through pre-existing tube leaks. The AFWS is a 2 divisions and 4 trains system. The reliability of the AFWS has been increased by use of two 100% motor-driven pumps, two 100% turbine-driven pumps and two independent safety-related emergency feedwater storage tanks as a water source instead of condensate storage tank.

The Containment Spray System (CSS) is a safety grade system designed to reduce containment pressure and temperature from a main steam line break or LOCA and to remove fission products from the containment atmosphere following a LOCA. The CSS uses the IRWST and has two independent trains. The CSS provides sprays of borated water to the containment atmosphere from the upper regions of the containment. The spray flow is provided by the containment spray pumps which take suction from the IRWST. The CS pumps are designed to be functionally interchangeable with the

Shutdown Cooling System (SCS) pumps. The CS pumps and CS heat exchangers can be used as a backup to the SCS pumps and heat exchangers to provide residual heat removal or to provide cooling of the IRWST.

### *3.3 Design Features against Severe Accidents*

The design approach differentiating the APR1400 from operating nuclear power plants is the consideration of severe accidents in the design. The measures to cope with severe accident are divided into two categories, prevention and mitigation. We minimized the possibility of severe accident. Severe accident prevention features can be summarized as follows:

- Increased design margin such as a lager pressurizer, larger steam generators, and increased thermal margin.
- Reliable ESF systems such as SIS, AFWS, and CSS.
- Extended ESF systems such as SDS with IRWST, AAC.
- Containment bypass prevention.
- SDS and IRWST.

The severe accident phenomena and issues addressed in the APR1400 design are hydrogen combustion, high pressure melt ejection (HPME) and direct containment heating, core debris coolability, containment performance and equipment survivability.

Safety Depressurization and Vent System (SDVS) serves an important role in severe accident mitigation. When core melts down at a high RCS pressure, the SDVS can be used to depressurize the RCS. The depressurization would ensure that HPME does not occur after vessel breach.The lower RCS pressure is necessary also for external reactor vessel cooling (ERVC) described below.

To control hydrogen that would generate from fuel rod zirconium-water reaction, passive autocatalytic recombiners (PARs) complemented by glow plug igniters are provided. The PARs are passive equipment that works without electricity and operator actions.

For core debris coolability, in-vessel retention (IVR) of corium is pursued as a key severe accident management strategy. ERVC System has been incorporated which uses a branch of shoutdown cooling recirculation loop through the IRWST and RCS to submerge the reactor vessel outer surface by flooding the entire cavity. Figure 5 schematically shows how the ERVC and IVR can be achieved. The evaluation shows that this strategy is effective for all depressurized core damage scenarios except large LOCA.

As a defense in depth purpose, Cavity Flooding System (CFS) is provided to flood reactor cavity below the reactor vessel for the purpose of cooling the core debris in the reactor cavity and scrubbing fission product releases should ERVC system fail. The water delivery from the IRWST to the reactor cavity is accomplished by gravity once operator opens isolation valves. Also, the large cavity floor area allows for spreading of the core debris enhancing its coolability within the reactor cavity region.

To prevent and mitigate direct containment heating (DCH), the APR1400 reactor cavity is configured to promote retention of, and heat removal from, the postulated core debris during a severe

accident. Core debris chamber virtually eliminates the potential for containment loading due to direct containment heating. The passage of the core debris to the upper part of the containment is designed to be convoluted so that core debris could hardly escape from the reactor cavity even under high pressure core melt ejection.

### *3.4 APR1400 PSA*

The core damage frequency(CDF) for the internal events were estimated to be 2.25E-6 per year. The LOCA categories of initiating events dominate (43%) the core damage frequency profile. Of the LOCA categories, small LOCA (17.2%) and steam generator tube rupture(10.3%) dominate the core damage frequency. In the transient categories, the loss of feedwater(20.9%), station blackout (14.9%), and anticipated transient without scram(14.9%) of events governs the core damage frequency. Figure 6 presents the CDF contributions by initiating events.

The external event analysis for APR1400 was performed using bounding site characteristics. The core damage frequency due to external events is 4.36E-7/RY considering the fire and the flood induced events. The results show an increased safety compared to the conventional plants by virtue of design characteristics such as divisional area and safety/non-safety equipment rooms separation and increased seismic capacity of 0.3g.

The core damage frequency safety goal meet the design goal of $1.0 \times 10^{-5}$ /RY. The containment failure frequency for all events is expected to be $2.84 \times 10^{-7}$ /RY and less than the design goal of $1.0 \times 10^{-6}$/RY. This assessment result was used for standard design certification process and will be updated in the detailed design stage considering site information and detailed design information.

## 4. Experimental Program in Support of APR1400 Safety System Design

### *4.1 Direct Vessel Injection Test*

One of the important characteristics of safety injection system of the APR1400 is the adoption of the direct vessel injection (DVI). There has been questions regarding the detailed T-H phenomena when the cold SI water is injected directly into the superheated steam in the upper downcomer. Specially, the ECC bypass and spillover with the direct vessel injection are of key interest.

The objective of the test is to understand major T-H phenomena in downcomer during LBLOCA reflood phase and is to provide experimental data for evaluating the predictability of safety analysis codes and for improving their models/correlation. Test facility consists of downcomer annulus, core simulator, 3 intact cold legs, broken cold leg, steam/water separator, containment simulator, steam supplier, and ECC supplier.

The separate effect tests using this test facility were very useful to understand the amount of ECC bypass and sweep-out and steam condensation effect. The test result was used to benchmark TRAC code and subsequently used in the licensing review.

### *4.2 Fluidic Device Performance Test*

This is a full-scale test for the performance evaluation of the passive flow control device, Fluidic Device(FD), which is adopted to APR1400. Tests have been carried out to evaluate the discharge characteristics of FD. The characteristics of discharge flow rate, level changes in the both SIT and stand pipe, and SIT pressure change are measured for each test case.

### *4.3 Unit Cell Sparger Test*

The APR1400 provides an IRWST and Safety Depressurization and Venting System (SDVS). SDVS consists of POSRVs (Pilot Operated Safety Relief Valve) and spargers. When POSRV discharge occurs, the steam with high pressure and temperature is discharged through the spargers attached at the end of the pipings of the Safety Depressuriation System. Before the steam is discharged, water and air existing inside the piping are discharged. The discharged air oscillates with a low frequency and produces dynamic loads, that may cause severe impact on the IRWST structures.

KAERI performed a series of blowdown tests to produce experimental data required for evaluating the performance of the SDVS, using a unit cell sparger.

This test is performed to generate pressure forcing function and time history of radius of bubble cloud which are produced based on the wall pressures measured in the unit cell test with a APR1400 prototype sparger. The bubble cloud pressures in the unit cell test could not be measured directly, since the installation of support for the sensors was extremely difficult due to the high pressure loads of discharging fluid. So, the pressure of bubble cloud was estimated using the measured wall pressures and the electrode analogy test results performed in ABB-Atom for the development of a BWR sparger. The major parameters affecting the bubble cloud pressure are the maximum steam mass flux in discharging pipe, the maximum pressure in discharging pipe during air clearing, the sparger submergence depth, the bubble cloud volume, the bubble cloud location, the pool temperature, the initial condition of discharging pipe, the subsequence actuation and the opening time of relief valve. The parameters of the unit cell test were compared with those of the APR1400 design data and the bubble cloud pressure of APR1400 was calculated by introducing appropriate correction factors reflecting the differences. The pressure forcing function of APR1400 was generated by processing the bubble cloud pressures of unit cell test. The time history of bubble radius was calculated based on the pressure forcing function.

### *4.4 Experiments on In-Vessel Retention*

For in-vessel retention in the APR1400 design, two major uncertainty issues have been raised: one is the effectiveness of heat removal capability by water flowing through the annular space between the reactor pressure vessel (RPV) and the thermal insulation. The other is the integrity of the in-core instrument (ICI) nozzles that are welded at the RPV bottom head under expected core debris thermal load of about 400 $kW/m^2$.

To resolve the former issue, a 1/16 scaled experiment was conducted by Pennsylvania State University (PSU) to measure critical heat flux (CHF) and to study the effect of the insulation and ICI nozzles. For the experimental design, a scaling analysis model was developed to simulate APR1400 design configurations and CHF was measured under both subcooled and saturated water conditions. The experimental results indicated that at high-heat-flux levels, the ICI nozzles have a strong influence on the bubble departure and act like a flow regulating device to relieve the effect of the recirculating

back-flow on the two-phase motion in the bottom region. This led to an increase in the local CHF and the CHF exhibited a minimum near the minimum gap of the annular flow channel. The measured CHF was larger than those measured in the ULPU test. This resulted in the conclusion that we can conservatively use ULPU CHF values to estimate thermal margin for IVR for APR1400 RPV configuration and thermal conditions.

For the second issue, an experiment was carried out by Korea Atomic Energy Research Institute (KAERI) demonstrating integrity of the ICI nozzles at the RPV bottom head under APR1400 thermal load conditions. A full scale bottom head slice section with one nozzle was fabricated and simulant core debris ($Al_2O_3$) was used to simulate APR1400 thermal load. The experimental result showed that integrity of the nozzle penetration is not challenged when the vessel is submerged in water.

## 5. Licensing Approach for APR1400 Standard Design

The design certification process in Korea is similar to that in U.S.. The certification rule similar to 10CFR52, Subpart B, Standard Design Certification has been finalized. We have submitted Standard Safety Analysis Report and Design Document (design description and associated ITAAC). The licensing review of these material is in progress. We expect the approval of standard design during the first half of this year. A legislation granting one-step approach similar to 10CFR52, Subpart C Combined Licenses are under discussion. Without one-step licensing in place, the first APR1400 plant, Shinkori unit 3, has to submit PSAR. However, the section referring to the Standard Safety Analysis Report will not be reviewed in PSAR submittal. The items related to CP/OL items will be reviewed in detail.

As far as technical requirements are concerned, the current licensing requirement is followed. Safety requirement against design basis accidents is the same as that of operating plants. The difference is in the area of severe accident mitigation features. To guard against the severe accident, severe accident policy statement is in place. The requirement for the safety against severe accidents for ALWR, such as APR1400, is to consider the severe accident mitigation and management in the design. In this regard, U.S.NRC SECY93-087 has been used as guidance.

During the review of SSAR, the focus has been on the new design features. The adequacy of safety injection system (with the direct vessel injection) in the double-ended guillotine break has been reviewed carefully. In this regard, the integrated approach of KAERI experimental program and the TRAC supporting calculation was very useful. Also, the performance of the spargers and IRWST has been reviewed in detail. In severe accident area, the use of PARs in hydrogen mitigation system and the usefulness of IVR strategy has been reviewed in detail. In this regard, the panel of international experts reviewed the report on IVR applicability to APR1400.

Once the construction is in progress, we need to submit CP/OL items as well as perform ITACC as stated in SSAR and Certified design material. CP/OL items are related to site-specific design information. ITAAC is related to show that the as-built design meet the criteria set in the certified design.

## 6. Summary

The standard design of APR1400 has been completed. The licensing of APR1400 standard design will be completed during the first half of this year. The focus of the licensing review is on the new design features. To answer succinctly, an integrated approach of testing and supporting analytical code calculations is very useful.

As a first country to build the plant with the licensing similar to 10CFR52 standard design, we expect some trials and errors in the process of obtaining the operating license. However, we think that the licensing and construction experience of eight KSNP plants will be very useful for us to resolve them. Also, we think that our experience will be useful to other countries that are considering the similar approach.

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[2] Oh S. J. et al., "Use of Probabilistic Safety Assessment in Decision Making of Korean Next Generation Reactor Design Optimization", PSAM-5, Proceedings of the 5th International Conference on PSAM held on November 27 – December 1, 2000, Osaka, Japan, Vol. 3, pp1847-1855.

[3] Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document", Volume 1.

[4] 10CFR52, Early Site Permits; Standard Design Certifications; And Combined Licenses for Nuclear Power Plants.

Figure 1. **Schematic diagram of APR1400 NSSS**

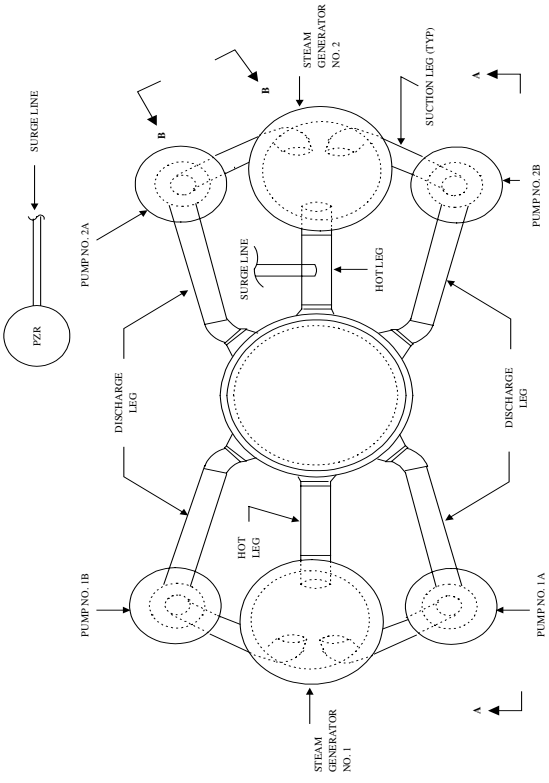



Figure 2. **The general arrangement with complex building of KNGR**

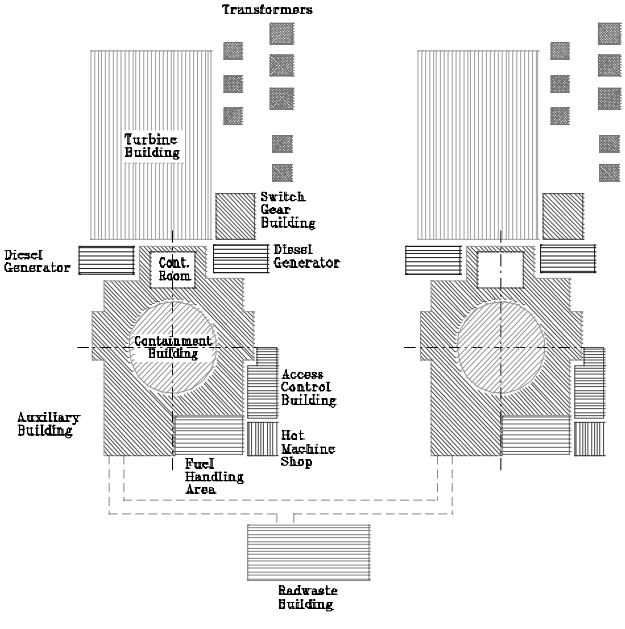


Figure 3. **APR1400 Man-machine interface system(MMIS)**

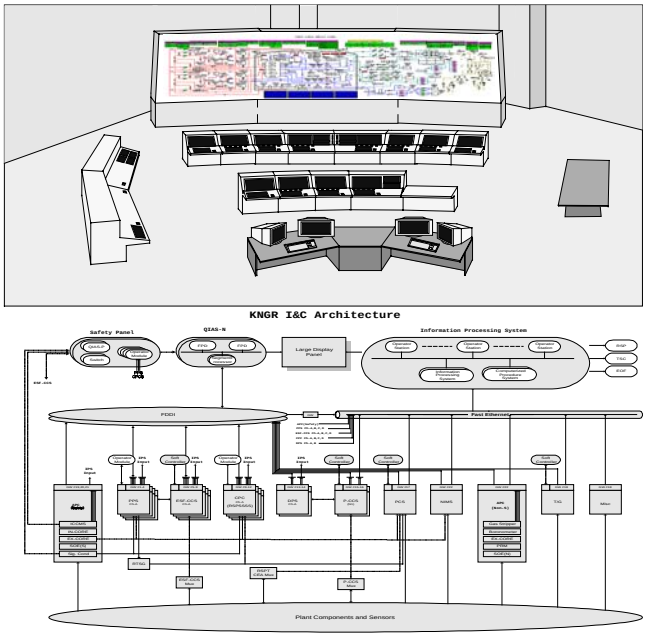



Figure 4. **Safety systems**

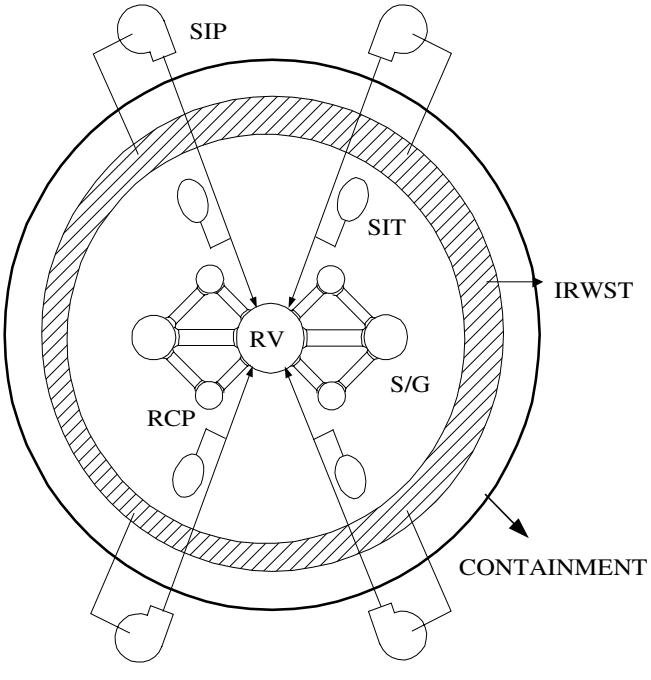


Figure 5. **Concept of the in-vessel retention and design implementation**

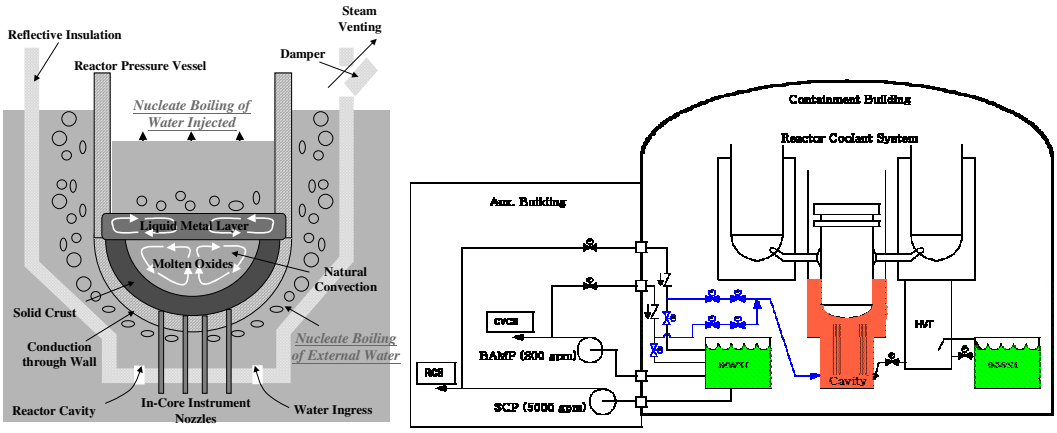
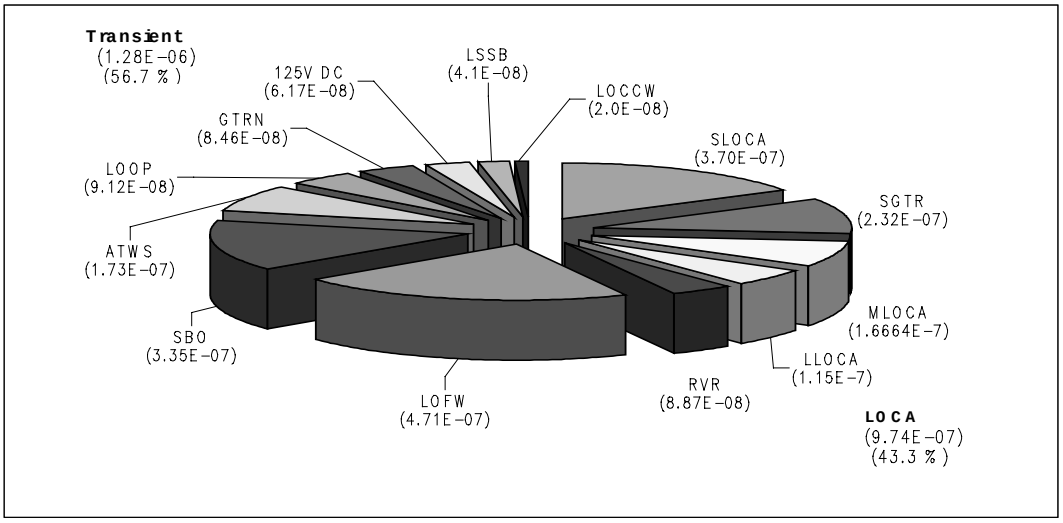



Figure 6. **Core damage frequency by each initiating event**



# SAFETY CONCEPT OF THE SWR 1000, AN ADVANCED BOILING WATER REACTOR WITH PASSIVE SAFETY FEATURES

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## Introduction

For technical products and production facilities, the prevention of equipment failures as well as of harm to the general public and the environment is a fundamental requirement. For nuclear power plants, nuclear safety is a question of paramount importance. In the development of nuclear power plants over the last few decades, the "defense-in-depth" safety philosophy has proven to be the most suitable method for preventing the release of radioactive materials.

The main prerequisite for assuring nuclear safety is a reactor core that is “inherently safe”, i.e. has physical characteristics that will reliably prevent rapid power transients. For new nuclear power plants, the defense-in-depth principle is structured as follows (Table 1):

- Prevention of operational disturbances through high-quality design, manufacture and operation of major plant systems and components.
- Provision of limitation and control equipment for preventing operational disturbances from developing into accidents.
- Provision of redundant and diverse systems for accident detection and control in order to prevent unallowable damage to the nuclear fuel and unallowable releases of radioactive materials to the environment.
- Provision of equipment that, in the postulated event of severe fuel damage, will restrict the consequences of such an event to the plant itself and thus preclude any necessity for large-scale emergency response actions in the vicinity of the plant.

Furthermore, the nuclear fuel, components containing high and intermediate activity inventories as well as plant safety systems must be protected against natural and external man-made hazards.

As regards the actual production of electricity in a nuclear power plant, the equipment provided for nuclear safety is nothing more than a cost factor. However, nuclear power has to compete with power generated from fossil and renewable energy sources for which this cost component does not apply. On the other hand, it is indisputable that operation of a nuclear power plant must not lead to any significant risks to the public or the environment. What this means for designers of advanced nuclear power plants is that they must incorporate nuclear safety requirements into their overall plant designs in such a way that safety is increased but costs are reduced.

Table 1: **The "defense-in-depth" safety philosophy of the SWR 1000 compared to previous nuclear plants**

| Safety Levels | Previous Nuclear Plants | SWR 1000 |
|---|---|---|
| 1st level | High-quality design, construction and operation to prevent offnormal operating conditions and accidents | – Proven technology and quality controls taken over |
| 2nd level | Reliable control and limitation equipment for preventing offnormal operating conditions from developing into accidents | – Proven technology taken over<br>– **Plus** • **Lower core power density, and Large water volume in RPV** |
| 3rd level | Safety systems for accident control and limitation of fuel cladding damage | – Proven technology partially taken over<br>– **Plus:** • **Diverse passive system for activation of safety systems**<br>• **Passive safety systems**<br>• **Large water reservoirs for long grace period** |
| 4th level | Residual risk | – **Equipment and provisions for control of a core melt accident so that no emergency response actions are necessary in the plant environs** |

## Safety concept of the SWR 1000

In the following, the advanced boiling water reactor "SWR 1000" is used as an example for showing how these conflicting demands can be met. Nuclear safety should not be a matter of complex system architectures and procedures, or be dependent on rapid intervention by operating personnel. Instead, it should be founded on inherent physical characteristics, simple design principles and modes of operation based on laws of nature.

To achieve these goals, the following requirements were defined for the safety concept of the SWR 1000:

- Clear and simple systems engineering through consistent use of passive safety equipment.
- Increased safety margins.
- Good accident control behavior through slower reaction to off-normal conditions.
- Increased grace periods (up to several days) after the onset of accident conditions before active intervention by operating personnel is required.
- Effect of human error on reactor safety to be minimized or avoided entirely.
- Much lower probabilities of occurrence for severe accidents involving core melt than in previous designs.
- Limitation of the effects of a core melt accident to the plant itself, i.e. eliminating the need for large-scale emergency response actions such as temporary evacuation or permanent relocation of the local population.
- Economic competitiveness.

Implementation of these requirements has led to an SWR 1000 safety concept with the following key design features:

- Reactor core with low power density.
- Large water inventory inside the reactor pressure vessel (RPV) to ensure good thermal-hydraulic behavior in the event of an accident; i.e. excellent slow-acting accident control capabilities.
- Control of transients without the need for coolant makeup supply to the RPV from an external source.
- Large heat storage capacity inside the containment thanks to large water inventories in the core flooding pools and the pressure suppression pool.
- Passive equipment for heat removal from the RPV and the containment.
- Large core flooding water inventory available inside the containment for discharge by gravity flow into the RPV following reactor depressurization.
- Passive activation of key safety functions such as reactor scram, reactor pressure relief and depressurization and isolation of main steam and feedwater lines, providing diversity with respect to activation by safety instrumentation and control (I&C) equipment.
- Passive accident control without the need for power supplies, activation by I&C systems or intervention by operating personnel in the initial days following the onset of accident conditions, and subsequent unlimited heat removal via simple active measures.
- Nitrogen-inerted containment atmosphere to preclude hydrogen combustion and hydrogen reactions inside the containment in the event of a severe core melt accident.
- Extended containment pressure load-bearing capacity to accommodate the quantity of hydrogen arising from 100% zirconium oxidation in the event of a severe accident.
- Passive cooling of the RPV exterior in the event of a core melt scenario to ensure retention of the molten core inside the RPV.

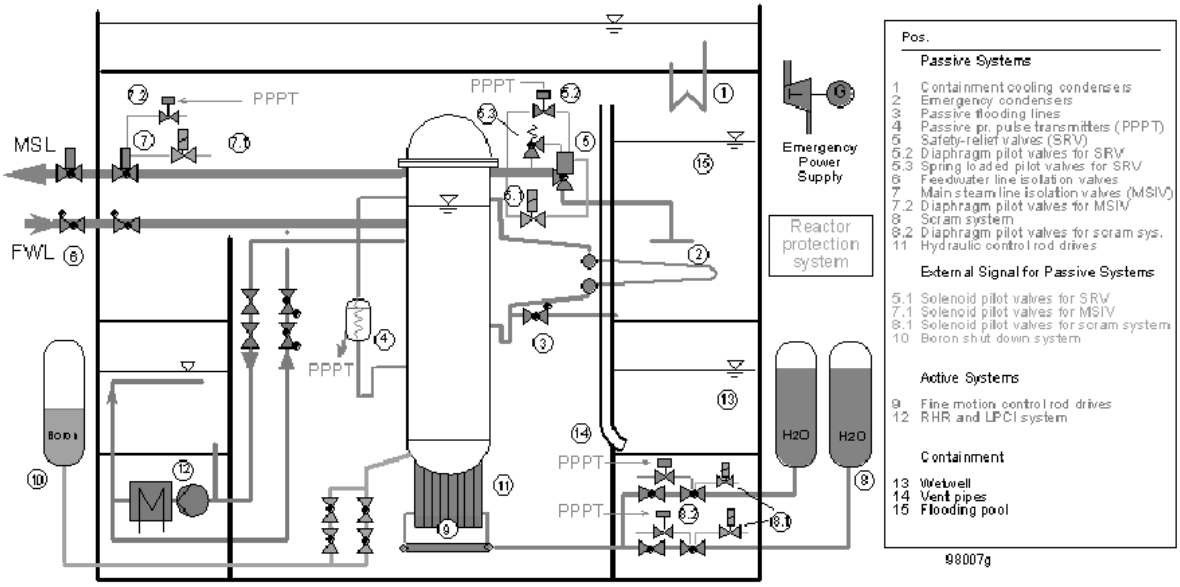
**Active and passive safety systems**

The following active and passive safety systems are provided for the main safety functions:

- Reactor protection system
  Apart from its digital, programmable safety I&C system, the safety concept features an activation system of diverse and redundant design which requires no I&C equipment for the safety functions of reactor scram, reactor pressure limitation and depressurization, containment isolation and prevention of excessive reactor coolant feed.
- Reactivity control
  Diverse means of reactor shutdown are available: control rods and boron injection. The control rods possess a diverse-design drive system consisting of a fine-motion control rod drive equipped with electric motor, gearing and threaded spindle and a hydraulic drive with accumulator-driven scram system. Fast boron injection provides diversity with respect to shutdown by the control rods.

- Reactor pressure relief and depressurization
  Diverse-design, system-fluid-operated main valves are equipped with pilot valves of diverse design. For pressure relief: solenoid valves (battery-powered and activated by I&C equipment) and spring-loaded pilot valves (passive). For depressurization: solenoid valves and diaphragm valves (activated by passive pressure pulse transmitters).

Figure 1. **Passive and active safety systems**



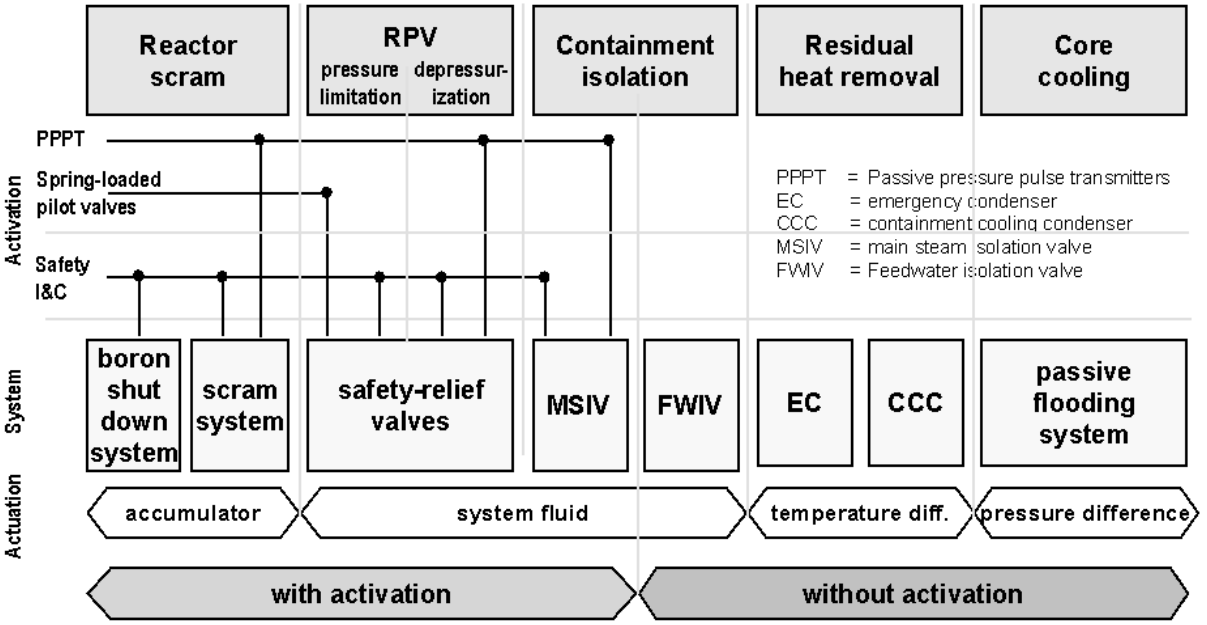

- Containment isolation
  Main steam lines: diverse-design, system-fluid-operated main steam isolation valves with diverse pilot valves (solenoid and diaphragm valves).
  Feedwater lines: two system-fluid-operated feedwater isolation valves (check valves) plus an additional valve of diverse design in each line.

- Residual heat removal from the RPV
  For residual heat removal at high reactor pressure levels, four entirely passive emergency condensers are provided which, following a drop in reactor water level due to reactor scram, are capable on their own of removing residual heat from the RPV to the water of the core flooding pools. These components are also capable of reducing reactor pressure.

- Core flooding
  In the event of a loss-of-coolant accident (LOCA), two residual heat removal (RHR) and low-pressure coolant injection (LPCI) systems are available for supplying makeup coolant from the pressure suppression pool to the RPV. In addition there are four passive core flooding lines equipped with check valves which flood the core with water from the core flooding pools after reactor pressure has been reduced.

- Emergency power supply system
  A two-train emergency power supply system with a diesel generator in each train and batteries of sufficient capacity is provided for supplying emergency power to safety-related electrical loads.

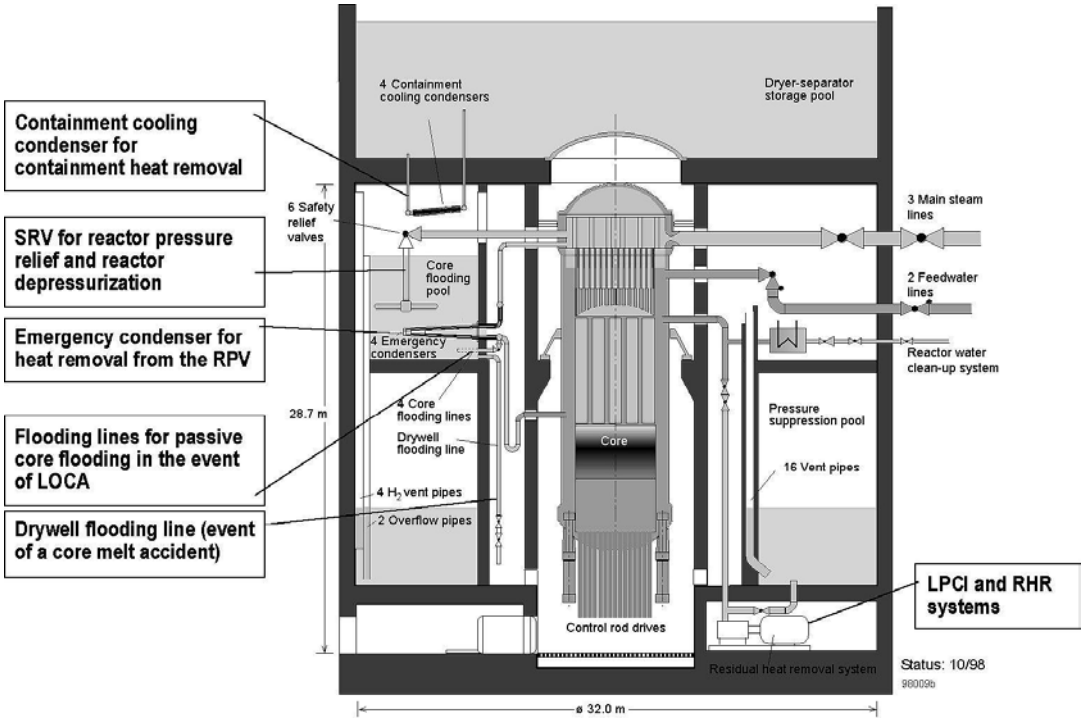
Figure 2. **The passive safety equipment for the different safety functions**




- Containment
  As in all contemporary BWR plants, a concrete containment with steel liner and pressure suppression system has been chosen. However, contrary to the previous practice of using prestressed concrete, the containment shell will be made of reinforced concrete. Inside the containment – which like all pressure-suppression-type containments is subdivided into a pressure suppression chamber and a drywell – there are additionally four large core flooding pools that technically belong to the drywell. The core flooding pools serve on the one hand as a heat sink for passive heat removal from the RPV by the emergency condensers as well as for the safety-relief valves and, on the other, as a water reservoir for gravity-driven flooding of the core at reduced reactor pressures following a LOCA. Apart from ventilation systems, the containment also houses the systems which communicate directly with the RPV, such as the high-pressure section of the reactor water cleanup system and parts of the two-train RHR system. The RHR pumps and heat exchangers are installed in separate compartments underneath the pressure suppression chamber which are isolated from the containment atmosphere. This arrangement ensures that the redundant RHR pumps are installed with physical separation and structural protection but nevertheless remain accessible after the occurrence of an accident. The containment also accommodates the new equipment provided for passive accident control. This comprises the safety-relief valves with their additional pilot valves, the emergency condensers for passive removal of heat from the RPV to the water of the core flooding pools, the containment cooling condensers for passive heat removal from the containment to the shielding/storage pool situated above

it, the passive core flooding lines and the passive pressure pulse transmitters provided for safety function activation.

Figure 3. **Containment of the SWR 1000 with passive and active safety features**




## Probabilistic Safety Assessment

The probabilistic safety analyses conducted for this safety concept with safety systems featuring a high degree of redundancy and diversity of design have revealed that, in the case of the SWR 1000, the integral frequency of core hazard states resulting from plant-internal events occurring during power operation is approximately $5 \times 10^{-8}$/a and the frequency of core damage states arising after plant shutdown is around $6 \times 10^{-8}$/a. These figures are well below the targets specified by the IAEA.

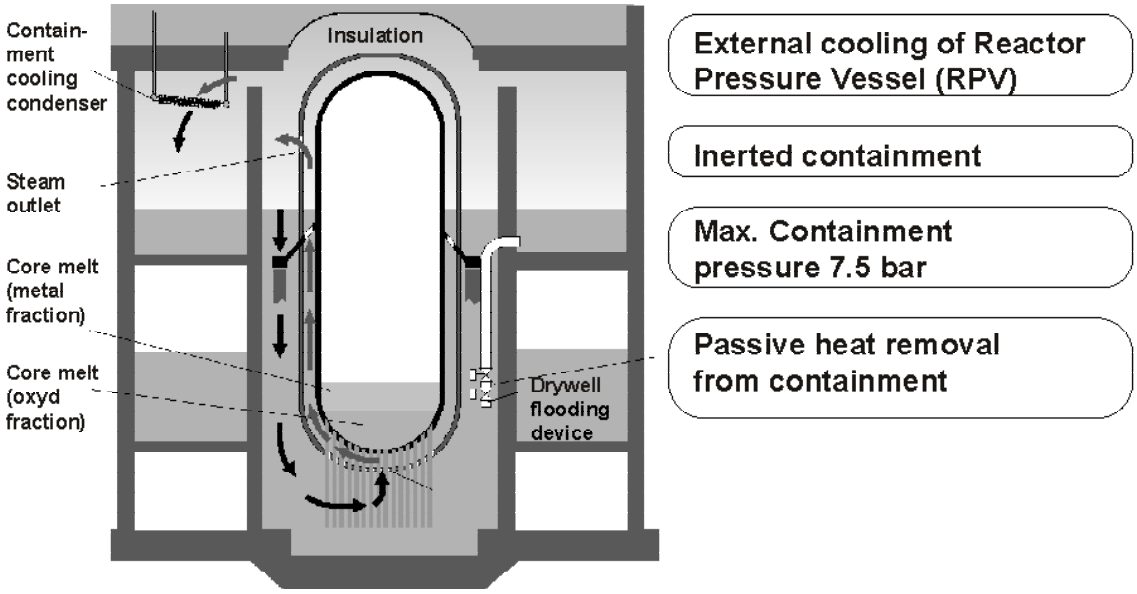
## Control of Core Melt Accidents

Despite the extremely low probability of occurrence of a core melt accident, this hypothetical event is nevertheless postulated and the following design features are provided for controlling an accident of this kind:

- Drywell flooding for cooling of the RPV exterior and retention of the molten core inside the reactor vessel.
- Nitrogen inerting of the containment to prevent hydrogen deflagration or detonation.

- Design of the containment with sufficient capacity to accommodate the hydrogen released by 100% zirconium oxidation.
- Passive heat removal from the containment by containment cooling condensers.

Figure 4. Concept of in-vessel core melt retention




Analyses of the impact of a core melt accident on the environment have shown that, even in the conservatively postulated case of an unfiltered release of radioactive materials from the containment atmosphere via the plant vent stack, the lower dose level of ICRP Publication No. 63 for evacuation of the population will not be reached. This verifies that the consequences of a core melt accident will remain restricted to the plant and that there would be no need for emergency response actions such as evacuation or relocation in the plant vicinity.

**Protection of buildings against natural and external man-made hazards**

The plant is designed in accordance with the European Utility Requirements to withstand the effects of natural and external man-made hazards such as seismic events, aircraft crash and explosion pressure waves. One of the goals in designing the plant was to accommodate the systems and components that require protection against these hazards in such a way inside the plant buildings that as few buildings as possible would have to be designed to withstand the loads from such events. Since all safety-related systems and components as well as those containing a high activity inventory are housed in the reactor building – except for the redundant emergency diesels along with their 690-V switchgear and the two safety-related closed cooling water and service water systems – the concept implemented for building protection is as follows.

The reactor building is the only building protected against all three major postulated hazards (seismic events, aircraft crash and explosion pressure waves). The buildings containing the emergency

diesels and safety-related cooling water systems are protected against the effects of aircraft crash through physical separation (by a distance of more than 40 m) and are designed to safely accommodate the loads imposed by a seismic event or an explosion pressure wave. Since none of the other buildings contains safety-related equipment or components with a high activity inventory, they are only designed to withstand seismic loading according to standard industrial practices.

Figure 5. **Protection against natural and external man-made hazards regarding activity inventory**

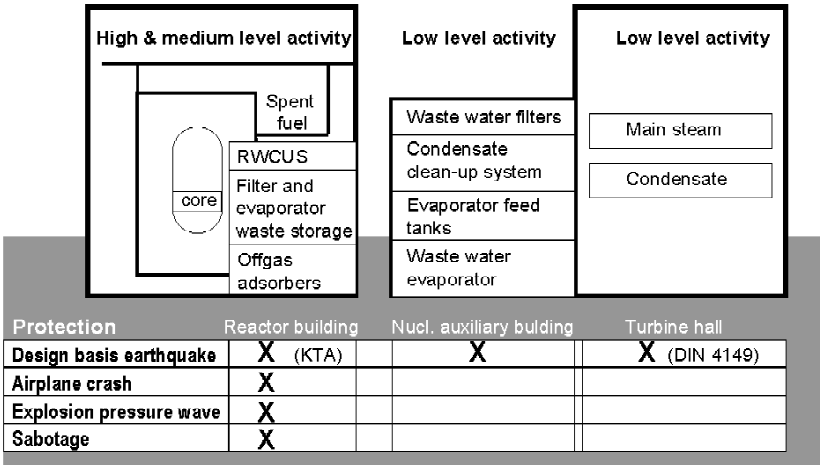
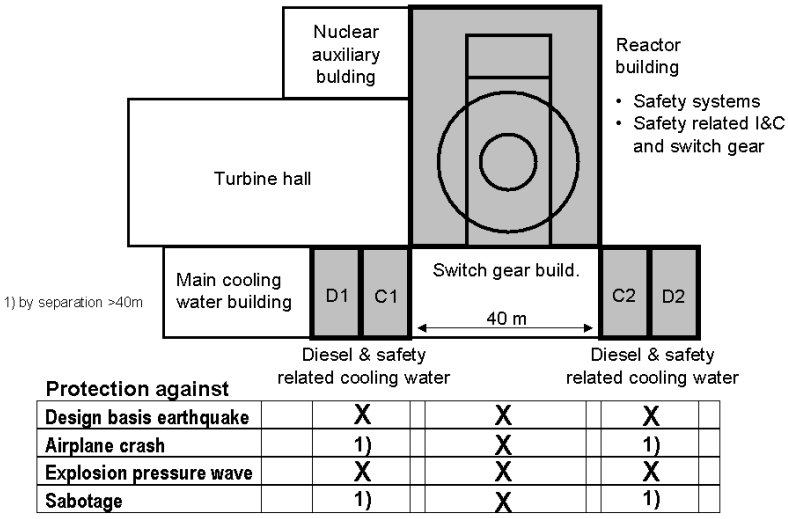



Figure 6. **Protection against natural and external man-made hazards regarding safety equipment**



As a result of the recent terrorist attacks in the USA, new aircraft crash requirements are currently under discussion, aimed at providing protection not only against impact by a military jet but also against the loads induced by large commercial jet airliners with their larger fuel inventories. The design concept must be specifically reviewed in terms of the following:

- Protection against penetration.
- Structural stability of the reactor building.
- Induced shock and vibration.
- Fire.

It is however expected that, given the basic design concept of the plant, even higher loads could be sustained after modifying the wall thicknesses and layout of the buildings. Any requirements that may arise in respect of severe fires would be less problematical for the SWR 1000 on account of the fact that the passive means provided for accident control preclude any immediate need for emergency power supplies.

## Conclusion

The safety concept of the SWR 1000 satisfies the requirements to be met by advanced reactor designs. The combination of active and diverse passive safety systems for both initiating and performing safety functions ensures full compliance with the principle of diversity. If all active systems such as I&C equipment, emergency power supplies and RHR/LPCI systems should fail, all postulated accidents (whether transients or LOCAs) can be controlled by just the passive systems alone without resulting in heatup of the core and with a grace period of more than 72 hours available before operator intervention becomes necessary.

The frequency of occurrence of core damage states has been able to be reduced by more than one order of magnitude compared to that of existing light water reactors.

A postulated core melt accident can be controlled by the passive systems alone in such a way that the consequences of the accident remain restricted to the plant itself and there is no need for large-scale emergency response actions in the vicinity of the plant.

By introducing passive systems which are less expensive than active systems and combining these with a reduced number of active systems, this improved safety concept additionally meets requirements for a competitive plant cost.

# DEVELOPMENT OF ABWR-II AND ITS SAFETY DESIGN

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## Abstract

This paper reports the current status of development project on ABWR-II, a next generation reactor design based on ABWR, and its safety design. This project was initiated over a decade ago [1,2] and has completed three phases to date. In Phase I (1991-92), basic design requirements were discussed and several plant concepts were studied. In Phase II (1993-95), key design features were selected in order to establish a reference reactor concept. In Phase III (1996-2000), based on the reference reactor concept, modifications and improvements were made to fulfill the design requirements.

By adopting large electric output (1 700 MW), large fuel bundle, modified ECCS, and passive heat removal systems, among other design features, we achieved a design concept capable of increasing both economic competitiveness and safety performance.

Main focus of this paper will be on the safety design, safety performance, and further research needs related to safety.

## 1. Introduction

In the early 1990s the project to develop a next generation reactor was launched at a time when the first ABWR was still under construction at Kashiwazaki-Kariwa Nuclear Power Station. Initiating this project was not considered premature since replacement of operating power plants were anticipated in the next twenty years and sufficient lead-time was required to develop a new reactor.

At the initial stage of this project, developing a "user-friendly" plant design was the most important objective since shortage of human resources was predicted in the 21$^{st}$ century. Thus, main focus was placed on selecting design for easy operation and maintenance. In the meantime, two important changes arose in Japanese nuclear industry. One was the delay of fast breeder reactor development due to trouble of Monju, and the other was the onset of deregulation in the electric power business. The delay of FBR development suddenly bolstered up the role of light water reactors, and the deregulation of electric power business highlighted the urgency of improving economics of nuclear power generation. For these reasons, economical competitiveness became one of the most important objectives of developing ABWR-II, while achieving the highest standards of safety was another

important objective. In order to shave R&D cost, the project focused on improving current ABWR design rather than pursuing revolutionary technologies, but succeeded in coming up with a design compatible from both economical and safety points of view.

## 2. ABWR-II Design Requirements

### *2.1 Improvement of Economical Competitiveness*

In order to improve the economical competitiveness of ABWR-II, the following issues were set as major design targets: low capital cost, high availability factor, and short construction period.

### *2.2Conformance to Global Safety Standards*

To meet the highest global safety standards, the following safety requirements were set.

*Grace Period*

"The grace period shall be one day for accidents and approximately one hour for transient events."

As we have learned from major accidents such as TMI and Chernobyl, it is imperative that we design a reactor that is tolerant of operator errors: ensuring safe shutdown without any operator intervention, and providing sufficient time for operators to take appropriate actions in case of accidents.

*Severe Accidents*

"Design shall incorporate countermeasures against severe accidents."

Although current reactors resort to complex accident management (AM) procedures by operators to cope with severe accident, a next generation user-friendly reactor should incorporate severe accident countermeasures in the design so as to eliminate the necessity of AM procedures to the extent feasible from a rational design point of view. This is also in accord with the trend of next generation reactors in both Europe and in the U.S.

*Probabilistic Safety Assessment*

"The core damage frequency (CDF) shall be equal to or smaller than that of ABWR."

"The conditional containment vessel failure probability (CCFP) shall be equal to or smaller than that of ABWR after adoption of AM."

A sufficiently low CDF of ABWR, meeting IAEA's requirement of below $10^{-5}$/reactor-year by a large margin, is considered an appropriate target.

A CCFP smaller than 0.1 – proposed as a supplementary objective in the Containment Vessel Design Guideline compiled by the Japanese nuclear industry [3] – will further restrict the frequency of releasing radioactive materials even if core damage accident occurs.

*Combined Active and Passive Systems*

"The active system shall be responsible for short-term measures after the accident, while the passive system shall be considered as a backup for long-term cooling."

A design approach of combining both active and passive safety systems with different sources of reliability – the former relying on ample artificial driving force (motors/turbines) and the latter relying on human-error-resistant natural driving force – serves as a holistic way of ensuring safety.

## 3. ABWR-II Design Features

### *3.1 Overview*

To conform to the above design requirements, various design features were adopted as illustrated in Figure 1.

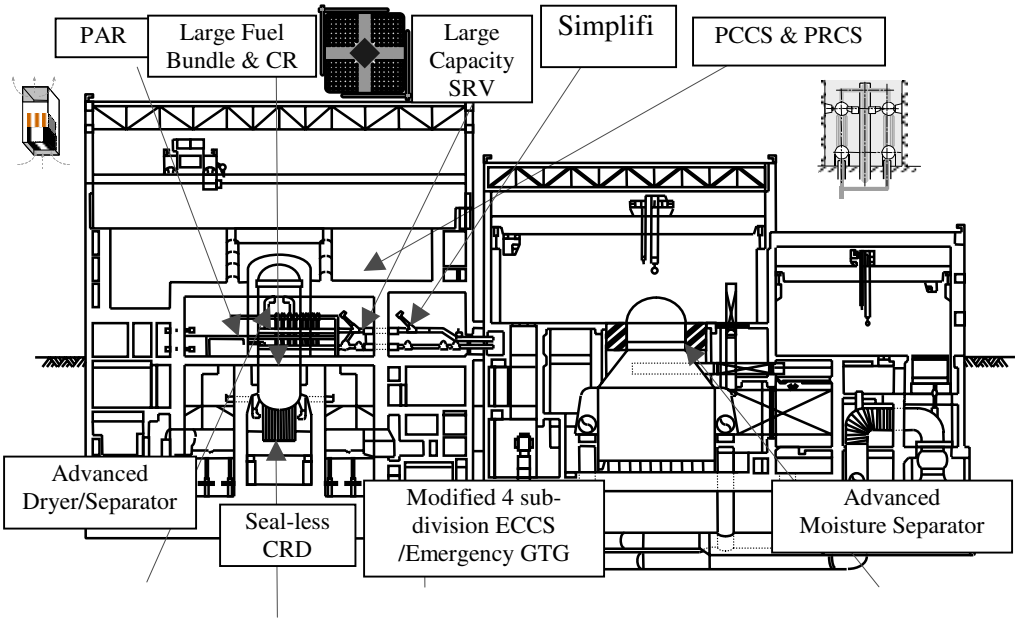
From an economical point of view, 1700MW electric output, large fuel bundles (1.5 times large K-lattice), low-pressure-loss MSIV, and large-capacity SRV were adopted.

The output of 1 700 MWe was determined considering the compatibility with Japanese grid capacity and manufacturing capacity for components such as reactor pressure vessels and generators.

The 1.5 times large K-lattice concept was selected as the reference design due to its capability of increasing fuel inventory since the bypass flow region is smaller than that of conventional design. The resulting improvement in thermal margin can be used for power uprate, higher burnup, and longer cycle operation. Large fuel bundles will also benefit to shorten refueling outage duration and provide capability of adopting a more flexible fuel cycle.

For enhanced safety, the reference design implements modified ECCS – with four sub-division RHR, diversified power source incorporating gas turbine generators (GTG), and advanced RCIC (ARCIC) –, passive heat removal systems – the passive containment cooling system (PCCS) and the passive reactor cooling system (PRCS) –, and passive auto-catalytic hydrogen recombiner (PAR). Modified ECCS configuration also enables on-line maintenance contributing to higher availability factor.

Figure 1. **Overview of the ABWR-II design features**




### *3.2 Safety Design*

*Modified ECCS*

The four sub-division configuration of RHR and emergency AC power sources is adopted from both economical and safety perspective (Figure 2) [4].

The four sub-division configuration has 4 x 50% RHR pumps, heat exchangers, and valves, while there are only 2 loops of piping for the supporting systems – RCW (Reactor Cooling Water System) and RSW (Reactor Sea Water System). This configuration enables to cut cost and increase safety performance at the same time.

From a safety point of view, the increased redundancy and diversity of power sources – equipped with two emergency D/Gs and two GTGs – effectively boosts the reliability especially against external events such as earthquakes. The modified ECCS also includes advanced RCIC (ARCIC) – a self-standing RCIC that can continue to operate without battery depletion at SBO – enhancing the safety performance even more.

From an economical point of view, maintenance for RHR, RCW, RSW, and emergency D/G can be conducted on-line, enabling shorter outage duration. On-line maintenance of the above-mentioned systems enables them to be effective during outage, leading to reduced shutdown risk. Also, due to the fact that ABWR-II only has 2 loops of cost dominant long piping of RCW/RSW – compared with 3 loops for ABWR – construction cost related to safety system is minimized.

The four sub-division concept is a cost-effective and high-safety-performing ECCS configuration that can be implemented without any major testing-and-development effort.

*Passive Heat Removal Systems*

In accordance to the safety design requirement of combined active and passive safety systems, two passive heat removal systems – PCCS and PRCS – were introduced (Figure 3). Besides providing a passive safety function, another virtue of PCCS and PRCS is that they are not reliant on seawater, thus providing diversity in terms of heatsinks.

PCCS is designed to passively condense steam released to the containment by heat exchangers (Hx) located in the water pool at the top of the containment vessel [5]. Following an accident, high-pressure steam is released from the reactor pressure vessel to the drywell (D/W) and/or steam is generated in the D/W, raising the D/W pressure. The resultant pressure difference between the D/W and the suppression-chamber (S/C) drives the mixture of steam and non-condensable gas in the D/W into the PCCS-Hx through the steam line. The mixture enters the inlet header of the Hx and is distributed to the multiple heat transfer tubes in which the steam is condensed. The condensate and non-condensable gas is released to the S/C through the outlet header of the Hx, and the drain and vent line. The decay heat is transferred to the PCCS pool and subsequently diffused to the atmosphere.

PRCS is designed to passively remove heat from the primary cooling system through heat exchangers during a reactor isolation event. It is manually activated on severe accident conditions in which ARCIC and RHR becomes inoperable.

Figure 2. **Configuration of the modified ECCS**

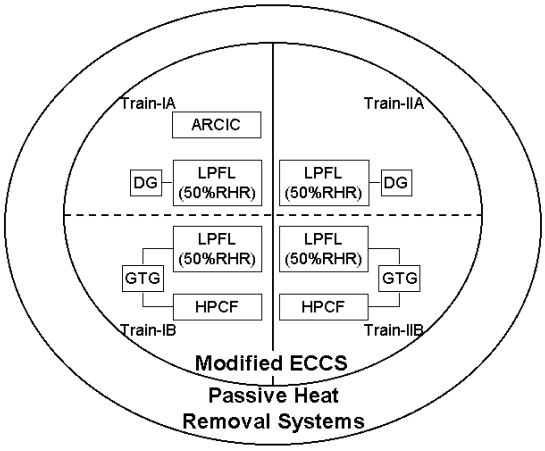
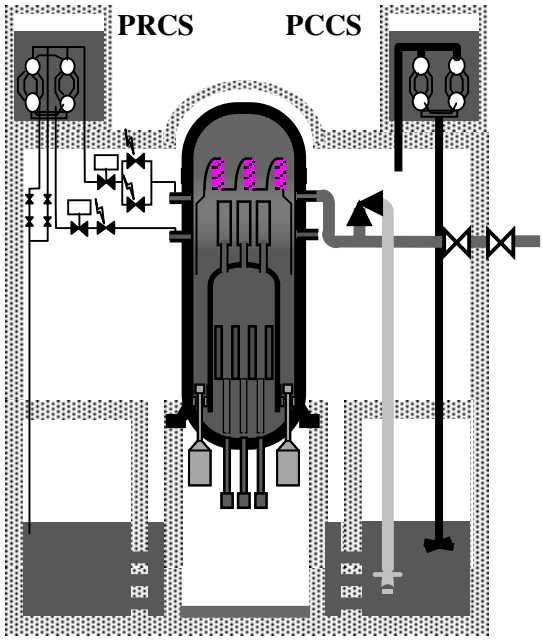



Figure 3. **Schematic of PCCS and PRCS**

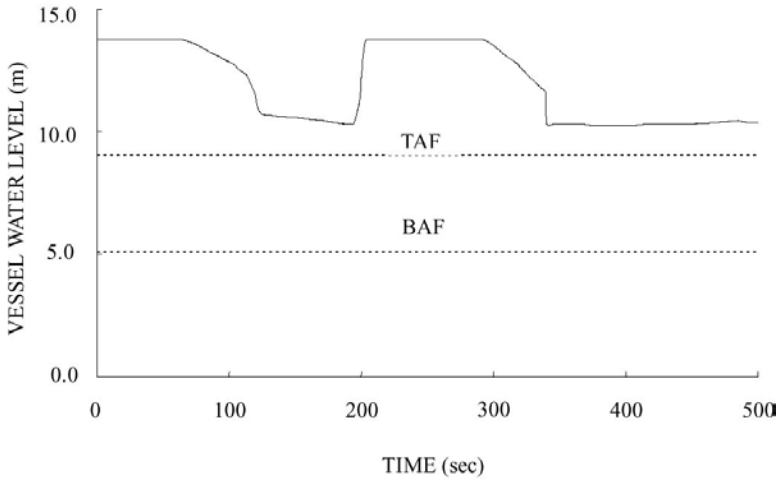


## 4. Safety Performance of ABWR-II

### *4.1 Deterministic Safety Performance*

The severest combination of accident conditions for the modified ECCS is a combination of pipe break LOCA of one of the HPCF injection line, single failure of a GTG, and on-line maintenance of an RSW sub-train and an emergency D/G. In this situation, the ARCIC and two LPFL sub-trains are still available. Figure 4 shows the result of safety analysis for this case run on SAFER code [4]. The core is always covered during any accident period after the LOCA. This is one of the ABWR heritages boasted as "no-core-uncovery" for DBA LOCA.

Figure 4. **The result of DBA LOCA analysis by SAFER code**




### *4.2 Probabilistic Safety Performance*

Level 1 PSA results revealed that CDF for internal events during power operation was reduced by a factor of about 3 compared with ABWR (Figure 5) primarily due to reduction of previously dominant SBO sequence by virtue of ARCIC and PRCS [5].

In a simplified seismic PSA (Figure 6), in which a loss of off-site power due to seismic event and no off-site power recovery was assumed, the effectiveness of diversified power sources and passive systems in terms of reducing seismically induced SBO sequence was revealed [4].

Simplified shutdown PSA (Figure 7) also demonstrated that modified ECCS with on-line maintenance capability serves to decrease shutdown risk by maintaining RHR availability during an outage [4].

It should be noted that in order to take credit of the effectiveness of PCCS, whose primary mission is to prevent containment vessel failure due to overpressure by steam release after core damage, level 2 PSA is required.

Figure 5. **Level 1 PSA for ABWR-II and ABWR**

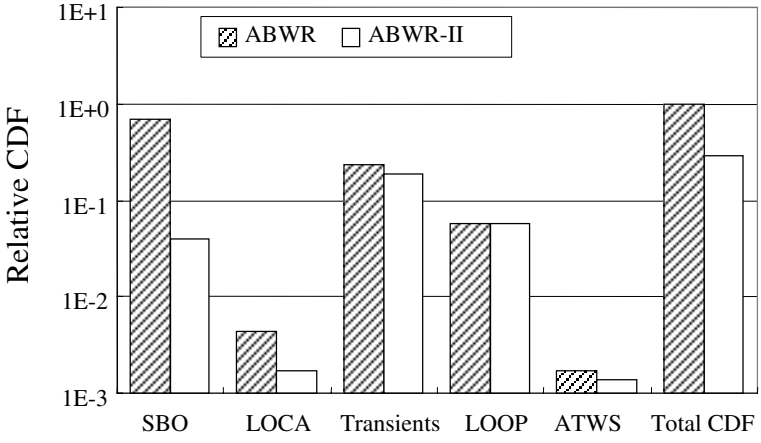



Figure 6. **Simplified seismic PSA for ABWR-II/ABWR**

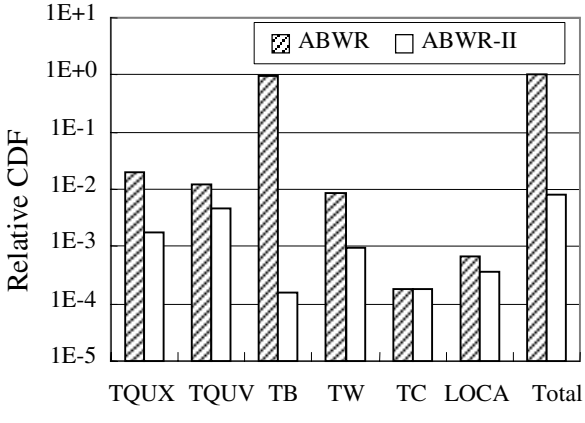
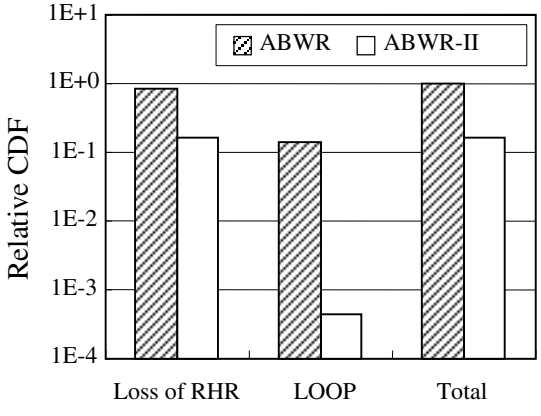



Figure 7. **Simplified shutdown PSA for ABWR-II/ABWR**




## 5. Further Research Needs

In July 2001, the Japanese nuclear industry published a guideline for assessing the containment event tree (CET) of advanced light water reactors [6]. Branching probabilities in the CET of ABWR was studied and one of the major issues identified was that uncertainties of severe accident phenomena, especially those related to molten core-concrete interaction (MCCI) had the largest impact on the assessment the containment failure frequency. MCCI is a severe accident phenomenon that

takes place when molten core drops on the concrete pedestal and erodes it through pyrolysis, releasing non-condensable gas such as $H_2$ and CO, and subsequently pressurizing the containment vessel. MCCI terminates if molten core is cooled below the erosion temperature of about 1500K, but various experiments related to molten core coolability have shown large uncertainty with regards to the formation of hard crust that inhibits the ingression of water. Thus, whether or not MCCI terminates is presently subject to a large degree of uncertainty.

In-vessel retention (IVR) of molten core is another severe accident phenomenon in need of further research. If IVR is proved to be highly successful, uncertainty of severe accident phenomena such as MCCI that take place after RPV rupture and molten core release to the containment vessel could be eliminated.

TMI-Vessel Investigation Program (TMI-VIP) and corresponding various analytical researches showed that IVR would be achieved if water ingresses into the narrow gap formed between reactor pressure vessel wall and debris crust [7]. Analytical models that employed this gap cooling process showed agreement with estimated maximum vessel wall temperature in TMI-VIP. However, the existence of such gaps and the gap formation process, i.e. the mechanism of debris non-adherence and strain of vessel wall when it is exposed to high temperature debris crust, have not been demonstrated or confirmed experimentally.

Figure 8 illustrates the following key phenomena governing debris coolability in the lower plenum of BWR: debris jet breakup and cooling in water pool of lower plenum, thermal interaction between debris and lower plenum structures, heat transfer from accumulated debris, and lower head cooling mechanism. Table 1 depicts technical questions to be answered with regard to each phenomenon.

Figure 8. **Key phenomena of IVR in BWR**

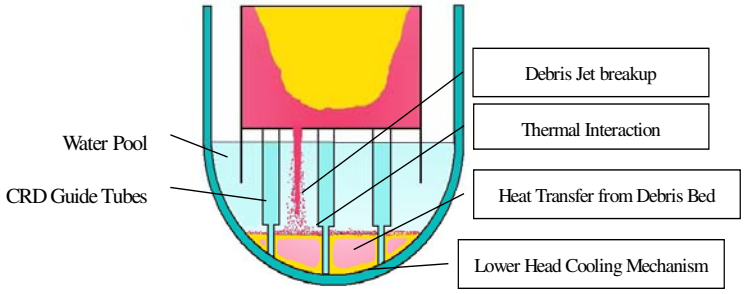


Table 1. **Technical questions with regard to key phenomena of IVR**

| Key phenomena | Phenomenology concerned | Technical Questions |
|---|---|---|
| Jet break up and quenching in a deep pool | a. melt relocation uncertainty<br>b. Break-up and quenching<br>c. Fuel Coolant Interactions | a. Drainage or blockage<br>b. Effect of structures<br>c. Effect of structures |
| Thermal interactions between lower head and debris | d. Thermal interaction between debris and penetration | d. Effect on CRD penetrations on additional heat sink and delivery of coolant to lower head |
| Heat transfer behavior from accumulated debris | e. Multi-Layer (metal/oxide)<br>f. Debris bed cooling<br>g. Cooling in upper surface<br>h. Molten pool natural convection heat transfer | e. Melt composition<br>f. Formation of debris bed<br>g. Applicability of model<br>h. N/C H/T model for high Ra number |
| Cooling mechanisms in gap | i. Creep of RPV wall<br>j. Gap formation and cooling in a gap | i. Applicability of model<br>j. Gap cooling heat transfer in narrow gap with curvature |

## 6. Summary

Current reference design of ABWR-II adopts the following features among others: large electric output (1700MW), large fuel bundle bundles (1.5 times large K-lattice), modified ECCS (four sub-division), and passive heat removal systems (PCCS and PRCS).

ABWR-II is designed to provide economical competitive edge in a deregulated electricity market as well as to meet global safety standard that is a prerequisite for next generation reactors.

Study on safety performance of ABWR-II has elicited its strength together with some future research needs.

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# FINNISH SAFETY REQUIREMENTS FOR FUTURE LIGHT WATER REACTORS

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This paper gives an overview of the Finnish safety requirements for future Light Water Reactors (LWRs). Emphasis is on developments during the last fifteen years; this has been summarised in the Preliminary Safety Assessment that was recently completed for the application to build a new power reactor in Finland. Starting from historical development of the legislation, Finnish safety guides and licensing process, the paper proceeds through safety philosophy into generic requirements of the safety case, including discussion of recent developments regarding external threats. Concluding remarks are made also on the basis of experiences from the Preliminary Safety Assessment so far, including feedback from the society.

## 1. Introduction: history and brief legalia

Historically, the development of indigenous nuclear safety requirements in Finland started in the early seventies when the first nuclear power plant construction projects (the two Loviisa VVER 440s) commenced. This first project was an import which was to meet the requirements of both the vendor and the customer country; as there was practically no domestic nuclear industry in Finland, then state-of-the-art safety requirements were initially largely borrowed from other countries, notably the United States, that had a relatively well documented legislation and guidance system in place already at that time. In order to apply these criteria to the Loviisa project, which had originally been conceived on a different basis, necessitated the development of much indigenous know-how of the technical substance and thinking behind the formal requirements. Major modifications in the plant design were introduced, and many of them were subsequently implemented by the vendor to their later projects elsewhere in Europe. All this provided for a smooth path for the next project (Olkiluoto), which began while construction of Loviisa was still going on. The Olkiluoto BWRs were originally designed to meet the U.S. criteria. The two plants were commissioned in 1977-1981 timeframe.

From those days on, the Finnish safety criteria have evolved together with accumulation of operational experience worldwide, advances in science and technology, and industry efforts to expand the nuclear generating base. The utilities have filed applications for a new plant in 1986, 1993 and in 2000; the almost continuously open possibility of a new construction has been a strong incentive to further develop the safety requirements and continuously improve domestic safety analysis and review capacity. Advances in requirements have also been put to practical tests at the periodic license renewal reviews for current plants, done in mid-eighties and mid-nineties (the Finnish operating license for a nuclear power plant has typically been issued for a fixed duration of ten years).

The Finnish safety requirements have evolved under two simple constraints: focus on Light Water Reactor safety and formulation in terms of safety goals. Other reactor types have not so far been

seriously considered (even as candidates for future construction), and goal formulation has been the easiest way to ensure as broad as possible applicability to both Pressurised and Boiling Water Reactor technologies. An obvious advantage of goal-oriented criteria formulation has been the automatic avoidance of undue prescriptivity; another is the self-evident independence of any one vendor or technological tradition.

The licensing framework for nuclear facilities, including the responsibilities and tasks of regulatory organisations and the licensees, is provided by the Nuclear Energy Act and corresponding Decree, issued in 1988. These legislative documents also grant the necessary authorities for issuing safety regulations. Safety requirements have been codified in Decisions of the Council of State, and regulatory guidance issued as the YVL Guides. These are available in Finnish and English on the STUK website (http://www.stuk.fi/english/regulations). Shortly put, the Nuclear Energy Act states that "the utilisation of nuclear energy shall be safe and shall not harm humans, property or environment" – this establish the needs of the society in lawmakers' terms. General safety requirements for nuclear power plants are established in the Decision of Council of State 395/1991, which defines fundamental technical requirements such as defence-in-depth implementation, safety functions, quality assurance, safety culture, demonstration of safety, radiological acceptance criteria for various event categories, and a requirement to constantly improve safety as warranted by operational experience and advances in science and technology. This forms the basis on which the various YVL Guides build more detailed technical requirements. (Other Decisions of Council of State define requirements for nuclear plant physical protection, emergency response arrangements, and reactor waste disposal facilities.)

Finnish licensing process for major nuclear facilities proceeds in three steps: decision in principle, to be made by the government and confirmed by the parliament, followed by a construction permit, followed by a (fixed duration) operating license. Construction permit and operating license are issued by the government. The government decision is prepared in each stage by the Ministry of Trade and Industry on the basis of an utility application. A necessary prerequisite for the government decision is STUK's statement of approval; in case of decision-in-principle, STUK's statement confirms (if appropriate) that no issues remain that would prevent fulfilment of safety requirements during subsequent steps, in case of other permits or licenses STUK's statement confirms (if appropriate) that no unresolved safety issues remain to be addressed further before granting the permit or license. This politically controlled system is in place to address and resolve in advance all possible concerns of political or societal nature, and thus to minimise (to the extent practically possible) the risk of delays and/or termination due to extraneous reasons.

## 2. Technical safety philosophy

The basic philosophy written into the Decision of Council of State 395/1991 is, of course, to implement defence-in-depth in both functional and physical interpretations. The functional interpretation consists of prevention unwanted events, capability to tolerate their occurrence without undue consequences, and mitigation of consequences; physical interpretation focuses on maintaining the integrity of main barriers against fission product release, that is, the fuel cladding, primary system, and the containment. The main safety functions in turn are the capability to shut down the reactor, to remove decay heat, and to maintain the containment function. The Decision of Council of State 395/1991 explicitly refers to severe accidents as the design basis events for the containment.

For practical purposes these general terms are translated into detailed technical (performance) requirements in individual YVL Guides. From design/review viewpoint the most important of these are Guides YVL 1.0, General design requirements, YVL 2.0, System requirements (to be issued soon), 2.2, Transient and accident analysis requirements, 2.4, Overpressure protection,

2.7, Failure criteria, 2.8, Probabilistic safety analysis, and 6.2, Fuel design limits; other central YVL guides such as 3.0, Mechanical design, 4.3, Fire protection, 5.5 Instrumentation and control system also address respective safety factors, but they are at the fringe of the present discussion. In well established technological areas these guides are quite unambiguous, but there are some areas where their requirements intentionally leave some room for interpretation; where this is the case, it is the Finnish practice to apply the SAHARA principle (Safety As High As Reasonably Achievable). What is "reasonable" within a technical community changes much faster that laws are written, even in Finland. The requirement of constant improvement has been written into the Decision of Council of State 395/1991, because experience shows that declaring any given state of affairs "good enough" tends to result in complacency and creeping deterioration of actual performance standards.

In the following subsections the Finnish safety criteria are discussed on the basis of the Preliminary Safety Assessment concluded a year ago, with a few added details of technical interest. In the Preliminary Safety Assessment, this discussion is preceded by a few important background remarks:

"When setting for the safety requirements, the intention has been to take the planned operational lifetime of a new plant, in principle 60 years, into account. The long operating lifetime requires preparedness to technical renovations/improvements and changes in key social community infrastructures. Readiness to meet changes shall exist, even if specific requirements for the readiness cannot be presented. Changes at the international level may comprise restructuring and/or ownership changes of the plant vendor and the equipment and fuel suppliers. In Finland, the changes relate to training and research arrangements in the field as well as other societal aspects, which might affect the long-term (tens of years) sustainability of know-how in the field. The social stability and alterations in existing values can change the technical and other services available on the market, which then might affect nuclear power operations. During the past years increasing investments have been directed to initiate international projects to study new nuclear reactor concepts. A technological breakthrough, in which the focus of electricity production carried out by nuclear energy moves towards technologies essentially different from light water reactors, may occur during the operating lifetime of the possible new nuclear power plant. In this kind of a situation, the importance of maintaining sufficient domestic know-how of light water reactor technology will further increase."

### *2.1 Safety design*

The design of a nuclear power plant to be safe is fundamentally a technical design undertaking. On one hand, all technical design is based on the fact that the causal (deterministic) relation between the cause and consequence is understood for practical purposes in a sufficiently accurate way and, on the other hand, that all available knowledge is limited and incomplete. The limited scope of knowledge becomes apparent when technical equipment malfunctions or is damaged in unexpected ways or when plant operators make unexpected errors. This will be taken into account in safety design of the nuclear power plant by the application of the defence-in-depth principle.

According to the defence-in-depth principle the release of radioactive substances hazardous to people, the environment and property is prevented by multiple independent barriers. Barriers are as independent of each other as possible, so that the failure of one barrier would not endanger the others. This provides security against the knowledge-related uncertainty and other imperfections related to the design and implementation of the barriers.

The barriers are dimensioned to maintain their integrity with the best possible certainty even if the worst imaginable barrier-specific threat were directed against them. If necessary, the threat is limited by design solutions and safety systems affecting the behaviour of the plant.

Threats to barrier integrity are mainly limited by designing the nuclear reactor and the other main processes and systems self-regulating, slowly reacting to disturbances and by dimensioning safety margins related to physical phenomena large. Factors restricting the size of safety margins are technical-economic viewpoints and possible mutually contradicting different safety objectives – for example, the reactor core emergency cooling must not jeopardise the integrity of the reactor pressure vessel.

In addition to the safety margins, the barrier integrity is ensured by different protection and safety systems designed to limit the possibilities of transients to develop more serious and to mitigate the consequence of events. These systems are designed to perform their functions irrespective of various presumed failures or failure combinations. Thus the safety of the plant is maintained also in the case of failures and failure combinations, which in practice have never occurred at nuclear power plants. The key objective in managing transients and accidents is to uphold the integrity of the first barrier, the reactor fuel cladding, sufficiently well.

According to the defence-in-depth principle, also failures in mitigating transients and accidents are taken into account. Such failures result in large-scale loss of the reactor fuel cladding integrity. For this kind of an event, a severe reactor accident, the nuclear power plant will be equipped with a containment building dimensioned to withstand the loads resulting from the accident and to keep the hazardous radioactive substances inside the building.

### *2.2 System design*

The possibility of equipment failures will be taken into account in the design of safety systems. As concerns the failure, it is required in Finland that the most crucial safety systems must be able to carry out their functions even if any single device of the system fails, and any other device of the system is simultaneously out of use due to maintenance or repair. This so-called N+2 failure criterion affects the structure of some systems implementing safety functions: the complete system comprises of several almost identical separate subsystems, i.e. redundancies. It is further required that for the so-called common cause failures, i.e. similar equipment gets failed from same reasons, the safety functions are secured by systems and/or devices functioning on different principles (diversity). For external threats (such as fire) systems ensuring each other and their redundant parts are physically separated from each other. Balanced operation between different systems is to be ensured with probabilistic methods, described below in more detail.

Systems taking part in the implementation of safety operations are classified, based on their safety significance, in safety classes 1, 2, 3 and 4 in declining order significance according to the Guide YVL 2.1. If the system has no nuclear safety significance, it is classified in the category EYT.

When dimensioning the safety functions, a larger group of events than that used for the original design of the existing power plants has to be taken into account as possible initiating events or other dimensioning factors. The requirements concerning this are presented in the Guide YVL 2.2. Some examples are presented below:

- operational transients during which the reactor shutdown with the control rods is assumed to fail completely (so-called ATWS) have to be included among the initiating

events to be considered as postulated accidents. This is a means to ensure that systems designed for reactivity control have adequate redundancy and diversity.

- a possible leak from the pressurised water reactor's primary coolant circuit to the secondary coolant circuit must not lead to the coolant discharge into atmosphere; otherwise primary-to-secondary leaks could involve significant radiological consequences, and potential for irrecoverable coolant loss. In line with the functional philosophy of Finnish guidance, the designer is free to choose how to accomplish this (i.e. whether to route the discharge into a closed space of the plant capable of receiving it, or whether to set the secondary side design pressure so that the relief/safety valves are not challenged, or whether to develop some other solution)
- also indirect (inherent) threats are to be recognised in the initial event analyses, and precautions against them are to be taken into account when designing the systems and devices. An apparent direct threat to the barrier integrity is always connected to every initial event: for example, loss of coolant resulting from a pipe break always disturbs directly cooling of the fuel cladding, thus threatening the integrity. In addition to direct threats, indirect threats may be connected to initial events: for example, in connection with loss of coolant, materials damaged by a pipe break (a LOCA initiator) may clog up filter structures of the emergency cooling system cooling the fuel and thus disturb the emergency cooling (designed precisely for mitigating a LOCA). As another example, a natural process can be related to loss of coolant in pressurised water reactors, during which the boron dissolved in the coolant for reactor power control purpose concentrates to the reactor core while clean water pockets are formed elsewhere in the primary circuit. The clean water getting into the reactor core at a later stage can cause reactor recriticality, which would be unsafe during the accident.

The acceptability concerning the design of safety functions, the reactor and the system dimensioning is proved with deterministic safety analyses. The so-called conservative (including disadvantageous assumptions in view of end result) and the so-called best-estimate computer programs can be used as analytical methods. Irrespective of the method, the safety analyses have to take always into account the uncertainties related to the analyses e.g. by carrying out a sufficient number of sensitivity studies. A sufficient safety margin must remain between the result of the analysis and the acceptance criterion to cover uncertainties. The acceptance criteria to be used in the analyses of every initiating event are determined in the Guide YVL 6.2 for the reactor fuel, in the Guide YVL 2.4 for overpressure protection and in the Guide YVL 2.2 for other parts. In addition, the mutual independence of barriers partly covers the uncertainties resulting from limited scope of knowledge and incompleteness of analytical methods.

The Finnish design basis event categorisation in Guide YVL 6.2 includes three categories, anticipated transients (frequency > $10^{-2}$/year), "minor" postulated accidents (frequency between $10^{-2}$/year and $10^{-3}$/year), and "major" postulated accidents (frequency < $10^{-3}$/year). It may be of interest to list here respective fuel performance acceptance criteria:

- Regarding anticipated transients fuel cooling shall remain well established, with 95/95 confidence with respect to heat transfer crisis (DNB or dryout), and (internal) fuel melting may not occur, nor damage due to pellet-cladding mechanical interaction.
- Regarding "minor" postulated accidents the number of rods reaching the heat transfer crisis may not exceed 1% of the total number of fuel rods in the reactor, the maximum temperature of the fuel cladding may not exceed 650°C, and it has to be shown that the

probability of fuel damage caused by the mechanical interaction between fuel and cladding remains extremely low.

- Regarding "major" postulated accidents: the higher the initial event frequency of a postulated accident in this frequency class, the smaller the number of damaged fuel rods shall be. The number of damaged fuel rods may not exceed 10% of the total number of fuel rods in the reactor. The consequences of the postulated accident may not endanger the coolability of the fuel, either.

Some countries apply the so-called leak-before-break principle (LBB) to the dimensioning of the emergency cooling systems. The idea of the principle is to ensure the integrity of the primary circuit pipelines by accurate and carefully controlled manufacture, inspections during operations and continuous monitoring of leakages. The objective is to discover possible leaks in the primary circuit at their initial stages and thus avoid the possibility of a large break. Based on this principle some countries have considered acceptable to simplify the plant structure e.g. by removing the break supports of main circulating pipelines, which improves the possibilities to inspect the pipelines during operation. The application of the leak-before-break principle to the Finnish nuclear power plants will be dealt with in the Guide YVL 3.5 to be published in the near future. STUK applies the principle in the following way: if its preconditions are fulfilled, the pipelines can be constructed without pipe whip restraints, however, without modifying the dimensioning of the emergency cooling system.

### *2.3 Passive safety systems*

The passive safety systems refer to systems implementing safety functions without external source of power. The requirements defined in the YVL guides, originally developed mainly for active systems, are applied for the passive safety systems as they concern the safety objectives of systems and demonstration of their reliability. Passive systems must be based on experimentally well-founded proof of functionality and reliability especially where there is no comprehensive operational experience from earlier corresponding technical solutions.

If the safety function of a passive system is ensured (applying the diversity principle) by an active system, which in the first place is designed as a system of normal operation, the active system in question shall be safety classified.

### *2.4 Management of severe reactor accidents*

The design of the new nuclear power plant takes into account the possibility of extensive reactor core degradation, the so-called severe reactor accident. The requirement primarily concerns the design of the containment building, because a severe accident in itself signifies loosing of the integrity of the innermost barriers (fuel cladding, primary circuit).

Successful management of a severe reactor accident requires a strategy, which systematically takes into account the plant characteristics and the phenomena threatening the containment. The strategy must include well-founded methods to prevent or mitigate the energetic phenomena related to development of the accident (e.g. hydrogen burn, high-pressure core melt discharge, energetic core melt coolant interaction). It has to guarantee also the coolability of core melt and the decay heat removal of the containment in such a way that the containment remains leak tight during the accident and long after it.

In case of a severe reactor accident the designed systems shall perform their functions even if any single device of the system fails (N+1 failure criterion). The systems to be designed to mitigate the severe reactor accidents are to be independent of other safety systems.

The severe reactor accident shall be mitigated in all nuclear power plant operational states, i.e. not only during the power operation but also during the shutdown periods.

### *2.5 Use of probabilistic safety analysis*

The probabilistic safety analysis as prescribed in the Guide YVL 2.8 has to be used as a tool in the nuclear power plant safety design. Its main purpose is to identify the factors reducing the reliability of designed systems and thus ensure the reliability technical balance of the system design. The probabilistic safety analysis methods are also used for the classification of initiating events based on their estimated frequency.

The active safety systems are generally quite complicated and often the successful functioning of the system depends on many auxiliary or supporting systems. The reliability of various parts of the whole must be in correct proportion to the importance of that part in ensuring the success of the entire safety function. The passive safety systems do not as a main rule need auxiliary or supporting systems to function.

The basis of the probabilistic safety analysis is the physical analyses and studies concerning the plant behaviour with which the ability of systems to perform safety functions in different transient and accident situations are studied. The logic models concerning the implementation of safety functions and the reliability of systems are drawn up based on the physical analyses. The frequency of an (undesired) final event, for example a core meltdown, can be calculated with these logic models. Estimated frequencies of various initiating events and reliability data of components and the operator actions are needed for the calculations. The reliability data are mainly acquired from the operating experience of existing plants. The calculation also gives some understanding of probabilities of chain events and failure combinations leading to end state. The worst failure combinations, which do not yet prevent the implementation of the safety function, are especially charted. Various risk measures can be further calculated on the basis of the results. The Guide YVL 2.8 presents numerical design objectives related to risk values: the mean value of the probability of core damage shall be less than 1E-5/a, and the mean value of the probability of a release exceeding the target value for severe accidents must be smaller than 5E-7/a.

The accuracy requirements of the physical analyses forming the basis of probabilistic studies may vary from relatively coarse to very detailed. In addition to the physical analyses all initiating events and sequences have to be identified as accurately as possible in order to make the analysis comprehensive. The uncertainties included in the final results can be quantitatively estimated only for the uncertainties of the reliability data (known) used in the analyses. Other sources of uncertainty are the structure of models, some factors difficult to estimate, such as the frequencies of failure combinations (including also common cause failures) or the human reliability as well as the choices made at the beginning of the study concerning the scope of analysis, for example, to what extent the external events (flood, fire, meteorological conditions, seismic phenomena) are dealt with in addition to the internal initiating events of the plant. Due to these reasons substantial differences occur in the calculated risks between identical plants. According to the Finnish requirements the scope of analyses shall cover all the phenomena mentioned above.

### *2.6 A few qualification items*

There are discussions going on regarding the qualification of safety classified mechanical equipment and digital automation systems. The utility perceives certain old mechanical nuclear grade quality outdated, anticipates availability and/or cost concerns, and has expressed a desire to probe the possibilities of satisfying the quality requirements using industrial grade hardware. There are a few examples in Finland of this type of approach, with mixed results – it is viable, provided the applicant/utility is competent enough to demonstrate (by any appropriate means) that industrial grade equipment meets the intent of nuclear grade requirements. (As far as procedural matters go, the Finnish YVL Guides explicitly allow procedures different from the Guide if the applicant / utility can demonstrate that the alternative procedure meets the intent of the Guide.)

## 3. Recent developments: external threat re-evaluation

In the terror acts on September 11, 2001 in the United States passenger aircraft were used as weapons for the first time in history. The STUK started immediately after the event to reassess hazards to nuclear safety connected with aircraft crashes and also hazards caused by other terror acts. The assessments consider from one side the safety of the existing plants and from the other side the design requirements for new plants.

Based on the work performed until now (with focus on current plants) the STUK has reconsidered the design criteria of a new nuclear power plant regarding aircraft crash and other corresponding external hazards. Generally, preventive measures have priority for ensuring safety, but regarding aircraft crashes they are beyond the direct authority of STUK.

For future plants, the consequences of crash shall be evaluated both for large passenger aircraft and military aircraft. The target shall be technical solutions, which do not need any modifications even in future although aviation technology or air traffic frequencies would change during the expected operational lifetime of at least 60 years.

A new nuclear power plant shall be designed against a possible aircraft crash or other external attack so that:

- the event does not cause damages which would lead to immediate release of significant amount of radioactive substances to the environment;
- in spite of the direct consequences of the event (penetration of structures by impacting parts, vibration, explosion, etc.) the most important safety functions can be started with adequate certainty;
- in spite of later consequences of the event (e.g. fire at the plant site) the most important safety functions can be maintained with adequate certainty for such a long time that the consequences of the crash can be repaired, without release of significant amount of radioactive substances to the environment.

Release of “significant amount” of radioactivity means in this context a release, which is evaluated to lead to the dose limit value for a postulated accident. When evaluating the population dose realistic assumptions can be used (so called best-estimate assumptions) and the public protection actions, which are easily performed, can be taken into account.

In practice evaluation for aircraft crash require consideration of:

- the structural capability of the structures protecting safety critical equipment and of the structural capability of components, systems and storage facilities (e.g. storage for spent fuel) containing radioactive substances;
- the physical separation of the safety systems;
- how the operation of heat sink and external supply of electric power are ensured.

The aforementioned measures against aircraft crashes protect the plant also against other external hazards and against possibilities of damaging acts but not against effects of full war. In Finnish practice, measures against effects of state level war do not belong to the technical design basis of nuclear power plants.

Protective measures against contingencies of other external hazards and damaging acts shall be taken also by other than structural measures as in the in the Decision of the Council of State 396/1991 on the general regulations for physical protection of nuclear power plants is prescribed. Among other things, unauthorised access is forbidden, and this means that also endangering the plant safety from outside the plant must be considered, in addition to conventional prevention of unauthorised entrance or infiltration to the nuclear power plant site area. The general design of the plant shall consider in addition to use of explosives and fire arms also, among others, electromagnetic radiation (High Power Microwave, HPM) targeted deliberately against nuclear power plant and the possible use of chemical and/or biological weapons endangering the working conditions in the control room.

## 4. Summary and conclusion

As is obvious from the outline above, the Finnish safety requirements are relatively stringent in that they demand reasonably good (i.e. demonstrable with confidence) safety margin against frequent threats and, at the same time, emphasise the capability to withstand low-frequency (potentially) high-consequence events. This approach is expected to provide for future light water reactors a reasonably uniform and acceptably low risk profile; anything less ambitious would be unsatisfactory from both public acceptance point of view, and from the need to ensure to the extent possible the safety of the hopefully very long-term investment already at earliest possible stage.

As of this writing, the Finnish government has just decided in favour of the application to build a new light water reactor in Finland, and Parliament discussion on the subject are commencing. The government decision has spurred considerable media and public interest in nuclear safety, and many of the discussions involve questions based on past accidents (Three Mile Island and Chernobyl), as well as the recently prominent terrorist threat. So far the Finnish approach has fared very well in such interactions.

## References

www.stuk.fi/english/regulations contains links English translations of the Nuclear Energy Act (990/1987), Nuclear Energy Decree (161/1988), and five Decisions of Council of State (395-398/1991 and 478/1999) that relate to various aspects of nuclear safety, as well as links to regulatory Guides lists.

www.stuk.fi/english/publications/yvl-guides.html contains links to all YVL Guides available in English.

www.stuk.fi/ydinvoimalaitokset/ydinvoimalaitosluvat.html contains links to the (original Finnish texts of) Preliminary Safety Assessment concluded on February 7, 2001, statement by the Advisory Committee on Nuclear Safety on the application and STUK's assessment, Addendum to the Preliminary Safety Assessment concluded on January 7, 2002 (in the wake of September 11, 2001 events) and the statement by the Advisory Committee on Nuclear Safety on the Addendum.

# SAFETY OBJECTIVES AND SAFETY PRINCIPLES FOR FUTURE PWRs

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## 1. Development of the joint French-German safety approach

In 1992 the safety authorities of France and Germany decided to jointly develop a new safety approach for future PWRs. This safety approach, consisting of safety objectives, technical principles and requirements, has been developed for the design of future PWRs, to be built in the first decade of the 21st century. Despite the maturity of existing LWR concepts in France and Germany there was an incentive to develop a safety approach aiming at a further improvement of safety, driven by several arguments:

- The number of nuclear power plants will continuously increase world-wide, despite this increase the potential for a major accident with serious off-site consequences must be reduced in the long term and on a global scale.
- Progress in research and development can be and should be utilised to improve safety.
- Present operating experience (more than 1 000 reactor years in France and Germany) and results from PSA for existing plants are an important source for further design improvements

The development of the joint safety approach can be grouped into 4 periods which are also interrelated with the design development steps of the industry.

The first period ended with a set of common recommendations, issued in May 1993 as "GPR/RSK Proposal for a Common Safety Approach for Future Pressurised Water Reactors". (hereafter called the "Basic Safety Approach of 1993"). It contains general safety objectives and technical principles, and it is the basis for all subsequent more detailed and refined recommendations. This document also served as a guidance to the industry during its development of the conceptual design, presented in September 1993.

The second period (1993 to 1995) is characterised by GPR/RSK recommendations for some key safety issues of high time priority (relative to the needs within the design development). The main subjects were: primary circuit integrity, general system design principles, use of PSA, external hazards and severe accidents. This period is described in more detail in /QUE 95/.

A third period, which started in 1995, is characterised by a further refinement of issues treated earlier in a more general way (e.g. system design) and by recommendations on new items such as core design, and man-machine-interface. Results of this phase are presented in /QUE 97/ and /FRI 97/. The results of the second period and part of the third period were available to the industry when continuing the design development, presented in the Basic Design Report in October 1997. In parallel with the second part of the third period industry performed the "Basic Design Optimisation Phase", resulting in a "Basic Design Report Update" of February 1999.

Partly overlapping with the third period was the fourth period which was characterized by a transformation of all objectives, principles, recommendations and requirements into Technical Guidelines. These Technical Guidelines use a new structure, reflecting the defence-in-depth principle, without changing the content and meaning of all previous recommendations.

Work in the near future depends on the continuation of the work by the designer in the frame of a future PSAR which could be ready in 2003.

## 2. The basic safety approach of 1993

The basic approach of 1993 mentions the need for a significant safety improvement at the design stage for future pressurised water reactors despite a high safety level already achieved with present plants. The main direction of the approach is characterised by the choice of an "evolutionary" strategy for the design of future reactors enabling an improvement of safety by taking into account existing operating experience and results from in-depth safety studies performed for present plants. However, the approach also stimulates innovations.

Three important general safety objectives characterise the common approach:

- A further reduction of the core melt frequency.
- The "practical elimination" of accident situations which could lead to large early releases of radioactive material. If those situations cannot be considered as physically impossible, provisions have to be taken to "design them out".
- For low pressure core melt situations the design has to be such that the associated maximum conceivable releases would necessitate only very limited protective measures in area and time (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long-term restrictions in the consumption of food).

The first and second objectives are in line with the past safety strategy, the third one can be interpreted as an extension of the defence-in-depth principle by adding at the design stage an additional level of defence and asking for new and innovative technical solutions.

The Basic Safety Approach of 1993 also contains several technical safety principles such as

- Strengthening and extending the defence-in-depth principle.
- Utilisation of experience feedback from plant operation and safety studies in both countries.
- Deterministic design basis, supplemented by the use of probabilistic methods.

The recommendations derived later for some key issues and in the following refinement and extension phase always make reference to the basic approach and are in agreement with it.

## 3. Key issues and further refinement of the safety approach

Starting in 1993, some subjects have been treated as key issues in more detail and in a more refined way in order to give more detailed recommendations as a guidance to the designer.

During that period detailed recommendations have been given on: containment design, various system design issues, radiation protection during normal operation, primary circuit integrity, internal and external hazards, severe accident R&D needs, shutdown states and man-machine-interface. They are partly presented in different publications /QUE 97/, /BIR 98/, /FRI 98/, some of the more recent recommendations are briefly explained here.

### *3.1 Integrity of the primary circuit*

Considering the state of technology, it appears feasible to design and operate future PWRs so as to exclude the complete guillotine break of a main coolant line.

This “break preclusion” approach, already adopted in Germany, will essentially allow to reduce the number of whip restraint devices as compared to current French practice allowing to improve the accessibility and inspectability of each point of the lines.

GPR/RSK underline that the use of a “break preclusion” concept implies provisions for the efficient primary leak detection and thorough in-service inspection.

This evolution was deemed acceptable for the French side, considering operating experience and international practice on the one hand and the extensive improvements of the containment function for future nuclear power plants on the other hand.

To provide extra margin in the design, GPR/RSK recommended to size the safety injection system and the reactor containment building for the complete guillotine rupture of a main primary coolant line.

On the German side, a modification to its own practice was also admitted: on the basis of defence-in-depth considerations, the German side accepted to include the combination of a safe shutdown earthquake and a primary break in the design basis for some structures and equipment, in conformity with the current French practice.

Under these conditions, GPR/RSK admit that the loads to be considered for the design of the internal structures in the vessel and of the structures in the containment building could be limited to those resulting from a break equivalent to the complete guillotine rupture of the largest pipe connected to a main coolant line (surge line).

### *3.2 Event and system classification*

Various system design issues have been treated at several occasions, e. g. in connection with event classification , system classification and requirements, shutdown states etc. A detailed classification concept has been set up in agreement with the defence-in-depth principle.

For information the plant states used in the French-German safety approach are listed here:

- State A power state as well as hot or intermediate shutdown state with all the automatic reactor protection functions available;
- State B intermediate shutdown above 120°C, residual heat removal system not connected, some automatic reactor protection functions might be deactivated;
- State C intermediate and cold shutdown with the residual heat removal system in operation and the primary coolant system closed;
- State D cold shutdown with the primary coolant system open;
- State E cold shutdown with the reactor cavity flooded;
- State F cold shutdown with the reactor core completely unloaded.

Internal events are classified according to their frequency of occurrence and are called plant conditions categories (PCC):

| | |
|---|---|
| PCC-1 | Normal Operation |
| PCC-2 | Reference Transients |
| PCC-3 | Reference Incidents |
| PCC-4 | Reference Accidents |

A list of events to be analysed in each category is specified. According to the extension of the defence-in-depth principle two categories in the area of risk reduction are defined (Risk Reduction Categories RRC):

| | |
|---|---|
| RRC-A | Prevention of core melt after a complete failure of a safety function (multiple failure conditions) |
| RCC-B | Limitation of radiological consequences after low pressure core melt scenarios. |

The events to be analysed to prevent core melt are specified and some examples are given of sequences to be considered for the limitation of the consequences.

The classification of safety functions and safety systems is arranged in such a way, that sufficient diversity is guaranteed in order to minimise the effect of common cause failures, e. g. Safety Functions F1 are specified to cope with PCC events, while Safety Functions F2 (diverse from F1 functions, less stringent requirements than on F1 functions) are required to cope with multiple failure events (RRC-A). The entire list of events and the system classification will be checked by a PSA at the end of the detailed design.

### *3.3 Role of PSA*

The design should be made on a deterministic basis, supplemented by the use of probabilistic methods. Generally speaking, quantitative probabilistic targets are not to be seen as requirements; they are essentially meant to be orientation values for checking and evaluating the design.

A probabilistic safety assessment must be carried out during the design stage to support the choice of design options. Particular attention has to be paid to the possibilities of multiple failures stemming from a single cause and human interventions, including diagnosis and maintenance.

Probabilistic objectives can be used as guidance in order to determine the most suitable combination of redundancy and diversity for the safety systems. It is recommended that a target value for the core melt frequency of $10^{-6}$ per year could be used for internal events (both power states and shutdown states) with due consideration of the associated uncertainties.

### *3.4 External Hazards*

The subject **"earthquake"** had been treated as a key issue. Later on some complementary recommendations have been specified, mainly with respect to the superposition of earthquake and internal events. Some examples are:

Concerning the combination of the design basis earthquake with a loss of coolant accident for the design of components and structures of future pressurised water reactors, GPR and RSK has recommended "to consider the complete guillotine rupture of the largest pipe connected to the main coolant lines" and more precisely, to consider for the design of the internal structures of the reactor vessel, "a load case combining the design basis earthquake and the rupture of the largest pipe connected to a main coolant line, using" the square root of the sum of the squares "methodology".

It has been asked also for a margin assessment to demonstrate that no cliff-edge effect in terms of radiological consequences would occur for acceleration values postulated beyond the site specific acceleration values; the corresponding methodology has to take into account the actual behaviour of representative equipment and the possibilities of simultaneous failures of equipment.

Concerning the emergency power supplies, it has been indicated that they can be constituted by four main identical diesel generators supplemented by two small diversified diesel generators; to cope with the potential long-term loss of off-site electrical power, all emergency power supplies have to be seismically designed and qualified.

Systems necessary to cope with reference transients, incidents and accidents have to be designed and qualified for the combination of loads resulting from the corresponding transient, incident or accident and the design earthquake.

Concerning **explosion pressure waves,** initial GPR/RSK recommendations indicate that "it appears necessary to take into account, for the design of future plants as a standard load-time function, a steep front triangular pressure wave with a maximum overpressure of 100 mbar and a duration of 300 ms ...".

GPR and RSK add later on that, for an adequate protection of future pressurised water reactor plants, the reactor building, the fuel building, the safeguard buildings and the diesel buildings must be protected as well as site specific structures and ducts related to the service water supply.

With regard to the protection against **aircraft crashes,** several in-depth technical discussions showed that relevant parameters have changed in the past and will change in the future. One example is the significant reduction of the number of military aircraft movements in Germany after the political changes in Eastern Europe and the German reunification.

There was an early agreement that the design has to be made on the basis of reference load-time functions which also include crashes of military aircrafts. The detailed discussions resulted in a common GPR/RSK recommendation of the load time functions to be applied. The safety functions to be ensured were fully defined (reactor shutdown and prevention of core melt, no dewatering of spent fuel in the pool), and the methods to be used to calculate the various aircraft crash effects (perforation, vibrations) were specified.

Furthermore, the design must be balanced and generally speaking, design provisions must be taken with respect to external hazards, consistently with provisions for internal events and internal hazards ; that is to say, external hazards must not constitute a large part of the risk associated to nuclear power plants of the next generation.

### *3.5 Severe accidents and containment design*

In the Basic Safety Approach of 1993 it is stated that: „It must be a design objective to transfer **high pressure core melt sequences** to low pressure core melt sequences (less than 15 to 20 bar primary system pressure at time of vessel failure) with a high reliability so that high pressure core melt situations can be "excluded". The designer was asked to propose depressurisation means with due consideration of the expected reliability of the valves; in particular these means must be clearly qualified under representative conditions. The use of specific valves – to be actuated only in case of core melt sequences – should be investigated. Upon this recommendation, the designer has proposed a design solution with a dedicated bleed valve for primary system depressurisation in case of a failure of the pressuriser valves, which are supposed to be used for depressurisation as a first choice. For this solution more detailed GPR/RSK recommendations have been given. Their discharge function must be available in case of loss of off-site power and unavailability of all diesel generators. Once open, the bleed path should stay fully open with high reliability. Sensitivity studies regarding the discharge capacity, the hot gas temperature and the initiation criteria have to be performed for specified relevant scenarios considering delayed bleeding and late reflooding as well as uncertainties of the code models.

Concerning the risk related to hydrogen explosion GPR and RSK stated that the containment volume and the mitigation means must be such as to prevent the possibility of a global hydrogen detonation. The possibilities of high level hydrogen concentrations must be prevented as far as achievable by the design of the internal structures of the containment; besides, specific provisions, such as reinforced walls of the compartments and of the containment, have to be implemented as far as necessary to deal with such phenomena as fast local deflagrations or deflagration to detonation transition sequences.

Concerning the **ex-vessel molten core cooling**, the designer has presented a concept with a large spreading compartment separated from the reactor pit and protected from the thermo-mechanical loads resulting from the reactor pressure vessel failure. Design provisions prevent any flow of condensate into this compartment. Moreover, a steel gate physically separates the reactor pit from the spreading compartment. Sacrificial concrete layers are implemented in the reactor pit and in the spreading compartment to obtain adequate characteristics of the melt. To prevent basemat penetration a protective refractory layer covered by a cast iron layer is foreseen. The cooling of the melt is

achieved by melt flooding from above by water coming from the inner refuelling water storage tank. Thermal loads on the basemat would be limited by a thick steel plate under a protective layer (refractory ZrO2), with cooling channels linked to the containment heat removal system. GPR and RSK underline that the validation of such a strategy would require extensive research and development work. The robustness of this concept has to be checked for various scenarios, including late reflooding and low residual power; specific attention has to be paid to the possibility of an early or partial failure of the steel gate. Specific provisions have to be implemented to ensure that the reactor building basemat would remain leak-tight in order to prevent contamination of soil and groundwater.

### *3.6 Man-Machine-Interface*

Early recommendations of GPR and RSK indicate that "due consideration has to be given to human factors throughout the design stage, taking into account aspects of operation, testing and maintenance with special emphasis on operating experience. The general aim is to take advantage of the human abilities, while minimising the possibilities for human errors and making the plants less sensitive to these errors ... Improving the man-machine interface shall be applied in all the locations where men interact with technical equipment."

After complementary discussions on these topics, GPR and RSK consider that the designer has to elaborate at an early stage of the design, a comprehensive human factors engineering programme which covers also maintenance and testing activities in order to ensure consistency and tracking of human factor issues and design choices in a well-structured and state-of-the-art human factors approach.

## 4. Development of Technical Guidelines

The recommendations given by GPR and RSK within the French-German safety approach consist of the Basic Safety Approach of 1993 with main objectives and technical principles for all important items, followed by a series of recommendations on key issues and many additional subjects. In many cases recommendations on the same subjects have been given successively, starting with general principles followed by an extension and refinement. Due to this continuous development the complete set of recommendations does not automatically contain a structure which is appropriate for a set of guidelines.

As one goal of the French-German co-operation was to develop joint guidelines for future PWRs, the transformation of the recommendation into technical guidelines was started in 1997. The technical guidelines are structured according to the defence-in-depth principle:

- A Principles of the Safety Concept
- B Conceptual Safety Features
- C Accident Prevention and Plant Safety Characteristics
- D Control of Reference Transients, Incidents and Accidents
- E Protection against Multiple Failure Situations and Core Melt Accidents
- F Protection against Hazards
- G System Design Requirements and Effectiveness of the Safety Functions

They were written to present in a logic and comprehensive way the requirements resulting from all recommendations of GPR/RSK and GPR / German experts respectively, structured according to the above mentioned scheme, however without any changes of the content. (In 1999 BMU has

terminated the participation in all projects related to future reactors for political reasons. This included also the involvement of RSK. From this time on, joint recommendations were given by GPR and a group of German experts.)

Part A on principles of the safety concepts will contain to a large part the basic approach of 1993 especially safety objectives, defence-in-depth principle, safety demonstration, classification concept, various general design principles and quality of design, manufacturing, construction and operation.

Part B concentrates on the design of safety features, comprising design of barriers, requirements on safety functions, systems, components and I&C, and on building arrangements.

Parts C, D and E are set up according to the defence-in-depth principle with part C on accident prevention containing requirements on quality of operation, inspection and supervision, reduction of frequency of initiating events and radiological protection of employees and the public. Part D concentrates on determistic requirements on barriers and safety functions and on the proof to be furnished. Design basis accidents and the corresponding accident analysis rules are defined. Part E addresses the prevention and the mitigation of core melt accidents. The first part deals with the safety assessment of multiple failure situations ; it defines the proofs to be furnished, including probabilistic consideration, and describes in detail the requirements on the design of the protection measures designed to prevent certain sequences. The second part addresses core melt accidents and the limitation of their consequences; it contains many new element compared to existing licensing requirements and reflects the extension of the defence-in-depth principle.

Part F contains requirements related to hazards. Firstly, internal hazards to be considered are presented as well as the safety approach and the proofs to be furnished consistently with the rules described in chapter D. Secondly, external hazards are listed with a focus on the load cases to be applied to earthquake, airplane crash and external explosion. Part G address some particular design concerns necessary to demonstrate the effectiveness of some specific safety functions as well as the use of Technical Codes for the design of equipment.

## 5. Final remarks

In the French context, these Technical Guidelines, issued on the basis of seven years of French/German cooperation are considered by the French Safety Authority as the safety basis to be used by the designer to continue its work for instance in the frame of the development of a PSAR which would have to be assessed in a French German scheme.

In the European context, these guidelines are one basis used by European Utilities to propose a set of requirements for future LWR and a part of the discussion with safety authorities.

In a wider international context, these guidelines are already used as a reference in the frame of discussion with others TSO and Safety Authorities involved in the assessment of new designs.

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# LICENSING OF FUTURE LWRs

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## Abstract

The Nuclear Regulatory Commission (NRC) expects that advanced reactors will provide enhanced margins of safety and that new nuclear power plants will achieve a higher standard of severe accident safety performance than previous reactor designs. Also, the NRC requires that the performance of new safety features be demonstrated by tests, analyses, or a combination thereof. Accordingly, the NRC developed new review standards for future nuclear plant designs based on operating experience, including the accident at Three Mile Island; the results of probabilistic risk assessments; early efforts on severe accident rulemaking; and research conducted to address previously identified generic safety issues. These new standards were used during the design certification reviews performed by the NRC in the 1990s and the resolutions are documented in the NRC's final safety evaluation reports.

## Enhanced safety

The NRC has determined that future reactor designs should achieve a higher level of safety for selected technical and severe accident issues than currently operating nuclear power plants. In its policy statement on the resolution of severe accident issues [1], the Commission stated its expectation that vendors would achieve a higher standard of severe accident safety performance for new standard plant designs than for previous designs. The policy affirmed the Commission's belief that reactor designers could show that new designs are acceptable with regard to severe accident concerns if the design meets the current NRC regulations, including requirements stemming from the accident at Three Mile Island; if the design demonstrates technical resolution of unresolved safety issues and the medium- and high-priority generic safety issues; and if the design considers the severe accident vulnerabilities from a probabilistic risk assessment (PRA) and the insights from severe accident research. The Commission also indicated its intent to continue a defense-in-depth philosophy and to maintain an appropriate balance between accident prevention and consequence mitigation [1].

In its policy statement on advanced nuclear power plants [2], the Commission stated its expectation that advanced reactors will provide, as a minimum, at least the same degree of protection of the public and environment that is required for current generation light-water reactors (LWRs). Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. The Commission requires proof of performance of certain safety-related components, systems, or structures prior to issuing a license on a new nuclear power plant design. For LWRs this proof has traditionally been in the form of analysis, testing, and research and development sufficient to

demonstrate the performance of the item in question. Similar proof of performance for certain components, systems, or structures for advanced reactors will also be required. The requisite proof will be design dependent.

**Severe accidents**

On the basis of the Commission's severe accident policy guidance [1], a new review approach was developed by the NRC staff for future LWR designs. This approach used the NRC's standard review plan[3] along with additional standards that were developed to address events beyond the design basis of a nuclear power plant. These events are commonly referred to as "severe accidents." These additional review standards for selected technical and severe accident issues [4,5,6,7] departed from or went beyond existing requirements and were based upon operating experience, including the Three Mile Island accident; the results of PRAs of current and future reactor designs; early efforts on severe accident rulemaking; and research conducted to address previously identified generic safety issues. These issues included station blackout, fire protection, intersystem loss-of-coolant accidents, and severe accidents. Other issues also evolved as a result of the design certification reviews for the U.S. Advanced Boiling Water Reactor (ABWR), System 80+, and AP600 standard plant designs. The goal of this approach was to resolve these issues with design features and not rely on analyses. The additional standards were used during the design reviews and the resolutions are documented in the NRC's final safety evaluation reports (FSERs) [8,9,10].

The NRC's new review approach recognized the wealth of information from severe accident research that had been generated since the accident at Three Mile Island. General agreement on the major severe accident challenges to the light-water reactor and containment designs had been reached, based upon extensive international and NRC research. However, uncertainties remained regarding initiation and progression of severe accidents. Therefore, the challenge for these design reviews was to resolve severe accident issues, notwithstanding these uncertainties. Severe accident research and knowledge of light-water reactors has increased to a level such that a plant designer can include measures to further reduce the risk from severe accidents. As a result, the approach to closure of severe accidents was to review these designs for severe accident requirements that apply to current operating reactors, such as hydrogen control requirements, and for additional severe accident challenges to the nuclear reactor and containment. These challenges include design features for ex-vessel core coolability to provide an adequate means of spreading core debris and for flooding the reactor cavity or drywell, and maintenance of containment integrity, or leak tightness, for a specified period following the onset of reactor core damage. Challenges to containment integrity considered during these design reviews included steam explosion, core-concrete interaction, high-pressure melt ejection, hydrogen detonation, and containment bypass.

Information used by the NRC in its new review process also included PRA findings, deterministic evaluations, and existing technology relative to experimental data. The NRC required a supporting PRA for each design certification application. For the first time, the PRA would be available early in the development of the design when modifications could be most effectively implemented. The insights developed during the design-specific PRA review resulted in changes to the design to reduce risk and also identified important information to be considered during construction, testing, and operation of the facility. These PRA insights were captured and documented in the FSERs [8,9,10]. As a result of these PRAs, the NRC focused on those issues that have an impact on the overall safety of the design.

Deterministic analyses were used in addition to the PRAs to develop a better understanding of the phenomena of a severe accident event. From a containment performance perspective, an

assessment of ex-vessel sequences was essential to determine the ability of the containment to withstand the anticipated thermodynamic as well as hydrodynamic loads. To support the deterministic analyses, the NRC used the experimental database to aid in the understanding of the loads associated with the various ex-vessel core-melt phenomena. In situations in which the database was incomplete, the NRC assumptions are believed to bound the phenomena in question. The NRC concluded for the ABWR and System 80+ designs that sufficient understanding of severe accident phenomena was available to resolve severe accident issues for these designs. That is not to say that all aspects of each individual response of the design to a postulated severe accident is fully understood. Rather, there is sufficient understanding of the phenomena for the NRC to conclude that these designs are acceptable. The following discussion provides examples of how severe accident issues were resolved during the reviews of the ABWR and System 80+ (evolutionary) standard plant designs.

The first issue pertains to the resolution of postulated steam explosions occurring external to the reactor vessel. The ABWR reactor cavity is designed to be dry at the onset of the ex-vessel release. The ABWR has fusible plugs within its drain lines to assure that the quench water will not be released until there is a substantial amount of corium on the floor to cause melting of the fuse plugs. Therefore, the steam explosion issue was determined to be resolved based on a design feature that prevents water in the reactor cavity before ex-vessel release. However, in an effort to fully explore the capabilities of the design, the ABWR was analyzed to determine if the critical structures could survive a steam explosion. The analysis demonstrated that the reactor vessel and containment would survive intact. The System 80+, on the other hand, is designed to allow a flooded reactor cavity before ex-vessel release, thereby maximizing cooling of the entering corium. The decision on whether to do this will depend on the accident management procedures implemented at that time (which would factor in the latest knowledge). For this approach, water is expected to be present and, therefore, a steam explosion is more likely. However, the construction of the reactor cavity walls is unique in that the immediate walls are not the major reactor vessel supports. The structural loads analysis of a steam explosion demonstrated that these immediate walls could be partially destroyed, but the primary support structures would remain intact. In fact, these immediate walls protected the primary structures. Therefore, it was demonstrated that if a steam explosion were to occur, the System 80+ design would accommodate the expected loads. This example demonstrates the flexibility of the NRC's review approach because the NRC has been able to determine that the different designs can both accommodate external steam explosions.

Core-concrete interaction (CCI) involves the decomposition and chemical interaction of core debris with the concrete containment floor. The extent of CCI and its effect on the containment are influenced by many factors, including the amount of core debris, core debris superheat, core debris composition, the amount of metals within the core debris and concrete floor, the availability of water, the type of concrete, heat transfer mechanisms, and the thickness of the core debris layer. Although many of these factors are specific to the accident sequence and are dependent upon core-melt progression, several of them can be controlled or optimized through the containment design. These factors include providing for actively and passively flooding the reactor cavity, optimizing the reactor cavity floor space by furnishing a large unobstructed area for core debris to spread, supplying a thick layer of concrete to prevent containment liner melt-through in the event of continued CCI, and selection of a type of concrete that either decreases the amount of noncondensible gases generated during decomposition or inhibits radial and axial erosion. In the review of the evolutionary designs, all these factors were evaluated through a combination of analyses and design review. All analyses were performed to achieve the best estimate, and consideration was given to uncertainties in severe accident progression and phenomena. Analytical modeling took into account uncertainty and sensitivity analyses.

High-pressure melt ejection (HPME) and direct containment heating are associated with severe accident sequences at high reactor coolant system pressure that result in vessel failure and core debris ejection, fragmentation, and entrainment into the containment atmosphere. The fragmented core debris mixes and reacts with the atmosphere, causing large pressure and temperature increases that may challenge containment integrity. In addition, the core debris may relocate into contact with the containment shell, leading to melt-through. HPME has generally been associated with pressurized-water reactors because they lack the depressurization ability associated with boiling-water reactors. Therefore, to eliminate HPME as a credible threat to containment integrity for the evolutionary reactor designs, NRC required installation of reliable depressurization systems on all designs. In addition, design features were provided to ensure that a direct pathway did not exist for core debris to be transported from the reactor cavity to the upper containment. The NRC's evaluation of HPME focused on ensuring a reliable depressurization system. This evaluation included an assessment of the power sources (both electrical and air), capacity, valve design, operations and controls, and incorporation of the findings into the emergency operating procedures.

Hydrogen generation and control for these future LWR designs followed the precedent set by the NRC regulation that was developed after the accident at Three Mile Island [10 CFR 50.34(f)(2)(ix)]. This regulation specifies the amount of hydrogen generated that must be accommodated through a control system. The hydrogen control system is typically either an inert atmosphere to preclude hydrogen ignition or an engineered ignition system to control the effects of a hydrogen burn. The ABWR design uses inertion, which ensures that regardless of the hydrogen concentration, hydrogen recombination will not occur. The System 80+ and AP600 designs use an ignitor system, which deliberately ignites the hydrogen at a low-enough concentration such that sufficient hydrogen cannot accumulate and recombine in a manner that could challenge containment integrity.

**Testing requirements**

The NRC's new licensing process for nuclear power plants [11] requires both proof of performance (qualification) testing and verification testing. The NRC may also chose to perform some confirmatory testing to support its licensing review. Confirmatory tests are not required for licensing nuclear power plants, but the NRC may sponsor these tests in order to confirm its understanding of certain phenomena or the performance of a new safety feature. Verification tests are required by 10 CFR 52.79(c) and performed in accordance with section XI, "Test Control," of Appendix B to 10 CFR Part 50. Verification tests are used to provide assurance that construction and installation of equipment (as-built) in the facility has been accomplished in accordance with the approved design.

Qualification tests for licensing nuclear power plants are required by section III, "Design Control," of Appendix B to 10 CFR Part 50[12] and 10 CFR 52.47(b)(2) and are performed in accordance with section XI, "Test Control," of Appendix B to 10 CFR Part 50. On the basis of its advanced reactor policy statement [2], the Commission codified requirements for qualification testing[11] to prove that new safety features will perform as predicted in an applicant' s safety analysis report, that effects of systems interactions have been found acceptable, and to provide sufficient data for analytical code validation. The NRC has determined that this proof of performance for each safety feature of the design can be demonstrated through either analysis, appropriate test programs, experience, or a combination thereof. For new nuclear power plant designs, the NRC may require separate effects testing, integral system testing, or prototype testing [13].

The AP600 design relies on new (passive) safety systems that were not previously licensed by the NRC. There is very little experience with these types of safety features that operate at low

pressure differentials and, while conceptually simpler, the NRC believed they were potentially more susceptible to systems interactions that can upset the balance of forces on which these so-called passive systems depend on for operation. Therefore, the NRC required both separate effects and integral system testing programs to qualify the AP600 standard plant design [14,15]. The following discussion provides examples of the testing that was performed to qualify the AP600 design.

Westinghouse performed separate effects tests on selected safety systems, such as the core makeup tanks (CMTs), automatic depressurization system (ADS), and passive residual heat removal system (PRHR). The CMTs provide direct vessel injection for emergency core cooling in many transients and accidents. These tanks are filled with cold, borated water and maintained at reactor coolant system (RCS) pressure. The CMTs inject the borated water into the RCS by either recirculation or gravity drainage. The NRC staff had concerns regarding the recirculation and gravity drain behavior, including steam condensation during draining; thermal stratification in the CMTs; and the effects of system depressurization on heated CMT behavior. Through its testing program, Westinghouse adequately demonstrated that the AP600 CMTs would operate as designed. Westinghouse also performed numerous containment tests using the Large Scale Test program at the Westinghouse Science and Technology Center to investigate the thermal-hydraulic phenomena of the passive containment cooling system. Westinghouse demonstrated the capability of passive containment heat removal via evaporation, sensible heating, convection and radiation, including effects of different levels of water coverage, various external air flows, and the effects of non-condensables."

Westinghouse performed integral system tests at Oregon State University' s Advanced Plant Experiment (APEX) in Corvallis, Oregon and at the Societa' Informazioni Esperienze Termodrauliche (SIET) laboratories in Piacenza, Italy (the SPES-2 test program). APEX was a low pressure, 1/4-height presentation including the RCS and related components and all safety systems in direct communication with the primary system. Most of the tests run at the APEX facility simulated design-basis accidents, primarily small break loss of coolant accidents of various sizes and at different locations. All of the tests included an extended time period after the loop was fully depressurized to investigate integral system thermal-hydraulic behavior during injection from the in-containment refueling water storage tank (IRWST), transition from the IRWST to sump injection, and long-term recirculatory cooling from the simulated sump. The two major variables affecting system behavior were break size and location. Other effects studied included interactions with non-safety-related systems and the effects of the elevated containment pressure. The NRC staff concluded that the proposed integral system test at the APEX facility would not be sufficient to qualify the AP600 design and determined that full-height, high-pressure integral system tests would be needed. The SPES-2 integral system test facility could operate at pressures and temperatures up to prototypic AP600 values and was approximately full vertical scale. Therefore, Westinghouse performed tests at SPES-2 that focused primarily on integral systems behavior in the period from accident initiation to the establishment of stable injection from the IRWST. In addition, non-LOCA transients and other parameters were simulated at SPES-2, such as steam generator tube ruptures (SGTR), main steamline break, and inadvertent ADS actuation during a SGTR event. Both of these test programs provided integral systems data for validation of the AP600 safety analyses computer codes. The APEX test program demonstrated integral systems behavior at low pressures, while the SPES-2 test program was able to demonstrate integral systems behavior at elevated pressures, such as system response and systems interactions in the early stages of design-basis accidents, including transition from CMT recirculation to CMT draining, accumulator injection, and the effects of early stages of depressurization. The SPES-2 tests also demonstrated that for non-LOCA transients the AP600 could stabilize at elevated pressures.

The NRC staff performed tests at the Rig of Safety Assessment/ Large Scale Test Facility (ROSA/LSTF) loop at the Japan Atomic Energy Research Institute to independently confirm selected Westinghouse test results. This confirmatory testing was important in the resolution of NRC concerns with the PRHR heat exchanger performance. It was determined that the PRHR system had a significant effect on RCS behavior over a wide range of design basis accidents. The PRHR separate effects testing program at the Westinghouse Science and Technology Center used a heat exchanger with straight tubes. Subsequently, Westinghouse changed the AP600 PRHR heat exchanger design to use C-tubes. This design change had a significant affect on the predictions of both the primary and secondary side of heat transfer behavior. As a result, the NRC staff questioned the correlation of data from the straight tube heat exchanger test program to Westinghouse' s predicted performance of the C-tube heat exchanger used in the AP600 design. Because the PRHR heat exchanger at ROSA/LSTF used C-tubes, the NRC staff provided data from its confirmatory testing to Westinghouse and requested "blind" calculations to predicted heat exchanger performance. From these blind calculations, the NRC staff was able to conclude that Westinghouse' s heat transfer model used in the AP600 calculations and analytical codes adequately predicted PRHR C-tube heat exchanger performance.

## Conclusions

The NRC has completed its reviews of three applications for design certification in the 1990s and it is expecting an application for certification of the AP1000 standard plant design in April 2002. During these LWR design reviews, the NRC demonstrated its ability to resolve complex safety issues and to provide a more stable and predictable licensing process. The interactions between the nuclear industry and NRC during the design certification reviews provided an effective means for resolving severe accident and selected generic safety issues, and for qualifying new safety features through testing. Site-specific issues were bounded to allow for separation of siting reviews from the design reviews. Based on these reviews, the NRC has determined that the ABWR, System 80+, and AP600 standard plant designs have achieved a higher level of design safety than the operating fleet of light-water nuclear power plants in the United States.

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# MAJOR SAFETY ASPECTS OF NEXT GENERATION CANDU AND ASSOCIATED RESEARCH AND DEVELOPMENT

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## Abstract

The Next Generation (NG) CANDU design is built on the proven technology of existing CANDU plants and on AECL's knowledge base acquired over decades of nuclear power plant design, engineering, construction and research. Two prime objectives of NG CANDU are cost reduction and enhanced safety. To achieve them some new features were introduced and others were improved from the previous CANDU 6 and CANDU 9 designs.

The NG CANDU reactor design is based on the modular concept of horizontal fuel channels surrounded by a heavy water moderator, the same as with all CANDU reactors. The major novelty in the NG CANDU is the use of slightly enriched fuel and light water as coolant circulating in the fuel channels. This results in a more compact reactor design and a reduction of heavy water inventory, both contributing to a significant decrease in cost compared to CANDU reactors which employ natural uranium as fuel and heavy water as coolant.

The reactor core design adopted for NG CANDU also has some important effects that have a bearing on inherent safety, such as a significantly negative power reactivity coefficient. Several improvements in engineered safety have been made as well, such as enhanced separation of the safety support systems.

Since the NG CANDU design is an evolutionary development of the currently operating CANDU plants, limited research is required to extend the validation database for the design and the supporting safety analysis. A program of research and development has been initiated to address the areas where the NG CANDU design is significantly different from the existing CANDU designs.

This paper describes the major safety aspects of the NG CANDU with a particular focus on novel features and improvements over the existing CANDU reactors. It also outlines the key areas where research and development efforts are undertaken to demonstrate the effectiveness and robustness of the design.

## Introduction

The NG CANDU reactor design is based on the modular concept of horizontal fuel channels surrounded by a heavy water moderator, the same as with all CANDU reactors. The major novelty in

the NG CANDU is the use of slightly enriched fuel and light water as coolant circulating in the fuel channels. This results in a more compact reactor design and a reduction of heavy water inventory, both contributing to a significant decrease in cost compared to CANDU reactors which employ natural uranium as fuel and heavy water as coolant. (Reference 1).

The design also features higher pressures and temperatures of reactor coolant and main steam, thus providing a larger thermal efficiency than the existing CANDU plants. These thermal-hydraulic characteristics further emphasize the NG CANDU drive toward improved economics.

The above changes and other evolutionary design improvements are well supported by the existing knowledge base and build on the traditional characteristics of the CANDU system, including: available simple, economical fuel bundle design; on-power fuelling; separate cool, low-pressure moderator with back-up heat sink capability; and relatively low neutron absorption for good fuel utilization.

The safety enhancements made in CANDU NG encompass safety margins and performance, reliability and separation of safety related systems. Passive features additional to those already present in the operating CANDU plants are also being considered. Sections 3 through 5 provide a description of these safety aspects. (Reference 2).

NG CANDU is based on proven technology. As such, the demands of the design on additional research and development (R&D) are limited and well supported by an AECL's R&D program. An outline of these R&D areas is given in Section 6.

**Reactor design**

The use slightly enriched uranium (SEU) with light water coolant flowing through the horizontal fuel channels allows a tighter $D_2O$-moderated lattice and results in a more compact reactor core, a smaller calandria vessel containing the moderator and simpler reactor internal components (see Figure 1). The compact core sharply reduces the inventory of heavy water in the moderator, giving a major cost reduction.

The reactor core design adopted for NG CANDU also has some important effects that have a bearing on inherent safety. The core exhibits a very even flux shape across all fuel channels, achieved without the need for local absorbers, while maintaining exceptional flux stability. This is driven by a high generation of thermal neutrons in the reflector. The result is a core which has a very flat power distribution, with the outer fuel channels having a maximum time-averaged power that is only about 20% lower than that of the peak central channel. This extends the traditional CANDU advantages of consistent control characteristics throughout the fuel cycle, to include very low reliance on external reactivity control and to minimise the demands on the reactor control system during operation.

The equilibrium core will have a significantly negative power coefficient. Moreover, the use of 43-element CANFLEX fuel bundles in lieu of the traditional 37-element fuel bundle increases fuel operating margins. For the same fuel channel power, NG CANDU fuel element ratings are reduced by about 20%. In addition, the CANFLEX fuel design increases the critical heat flux of the fuel elements, therefore the margin to fuel sheath dry-out during transients and accidents.

The use of SEU allows an increasing in the thicknesses of both the Zr-2.5%Nb pressure tubes and the Zircaloy calandria tubes, thus improving their capability to withstand loads during

normal operation and upset conditions. The thickness of the pressure tube also extends the pressure tube design lifetime, with respect to limits determined by both creep and corrosion.

The use of SEU fuel also enables increasing fuel burnup. The CANFLEX fuel bundle has been developed and irradiation-tested specifically with the intent of high burnup use. The average fuel burnup in NG CANDU is about 20 MWd/kg(U), which gives a factor-of-three reduction in the quantity of spent fuel bundles per unit of electricity generated, compared to CANDU reactors employing natural uranium.

In addition to fuel channel development, AECL has also made improvements to key core components, in particular reducing component sizing to enable a more compact core configuration. For example, the end-fitting sections, which connect each end of the fuel channels to feeder pipes, have been adapted to a smaller outer diameter, and to include a channel closure seal that resides inside the bore of the end fitting. This closure, together with a smaller, more efficient connecting "snout" on the fuelling machines, enables the lattice pitch (distance between fuel channel centres) to be reduced to 80% of current CANDU cores. This has several advantages: calandria vessel size, weight and complexity can be reduced; the inventory of heavy water for the reactor moderator and reflector is almost halved; and further improvements to the fuelling machines can be incorporated, such as simplified ram design.

**Safety design**

Several improvements in performance and reliability of safety related systems and in protection against common cause events have been made in NG CANDU. These improvements were identified from insights gained from preliminary probabilistic safety assessments, the results of preliminary safety analyses and operating feedback from existing plants.

Separation, reliability and diversity of safety systems, which are the fundamental safety design requirements of CANDU reactors, remain the pillars of the NG CANDU design as well. NG CANDU features: two separate, fast-acting and fully capable shutdown systems operating on different physical principles; complete separation of the special safety systems (shutdown systems, emergency core cooling system and containment) from the process systems and from each other (no sharing of equipment all the way through each system, from sensors to actuating devices); and availability of each special safety system to be better than 0.999 during operation and to be proved by means of at-power testable features. Separation and independence also includes the provision of a secondary control centre as a backup to the main control centre for certain emergency conditions.

Because of the smaller core size, the two shutdown systems can introduce negative reactivity into the core more quickly. The gravity-driven shutoff rods of Shutdown System 1 (SDS1) have a smaller distance to travel into the core, while the liquid poison injected by Shutdown System 2 (SDS2) has to mix with a smaller moderator volume to render the reactor subcritical.

Each shutdown system and all other reactivity mechanisms, are located in the low pressure and temperature moderator, eliminating the possibility of accidents such as rod ejection.

The reliability of the Emergency Core Cooling (ECC) system which supplies light water coolant to the reactor and maintains fuel cooling in the event of a loss-of-coolant accident, can be significantly increased by simplifying the interface between the ECC and the Heat Transport (HT) system since they are both light water systems. The ECC system consists of a high-pressure injection

subsystem with pressurized tanks and a recovery subsystem recirculating water from the floor of the reactor building back into the HT system by means of pumps.

The CANDU reactor type also features the inherent ability of the cool, low-pressure moderator to act as an emergency heat sink, in the event of a severe accident resulting from a loss of coolant coincident with postulated complete failure of the ECC system.

The containment system includes a prestressed concrete containment structure (the reactor building) with a prestressed concrete dome and an internal steel liner, access airlocks, equipment hatch, building air coolers for pressure reduction, and a containment isolation system consisting of valves or dampers in the ventilation ducts and certain process lines penetrating the containment envelope. Hydrogen igniters and recombiners are included to limit hydrogen content to below the deflagration limit in each containment compartment following an accident.

To achieve appropriate grouping and separation, the NG CANDU is divided into two groups of systems, similar to the current plants. Each group can independently perform the essential safety functions to shut down the reactor and to maintain the plant in a safe shutdown condition. Further separation within the two groups is achieved by locating redundant subsystems or components in physically separate areas, called quadrants. Fire resistant walls separate the quadrants from one another. Since the quadrants in each group contain redundant equipment, it will be possible for one quadrant to provide all the required safety functions to shut down the plant in a safe and controlled manner and maintain it in a safe shutdown condition. The result of this grouping and separation approach is a highly robust safety design.

The spatial independence arising from the quadrant-based layout of safety support systems also increases seismic capability. The result is a plant capable of operation on most sites around the world with an increased margin of protection against potential seismic events.

Further safety characteristics of NG CANDU include:

- Natural coolant circulation removes decay heat from the fuel if pumping power is lost. This is effective even if, following a small loss-of-coolant accident, the HT system coolant inventory is somewhat depleted. To enhance stable thermosyphoning following higher probability events, the NG CANDU pressurizer is sized so that the HT system remains filled with water after events such as a loss of forced circulation, a loss of main heat sinks, or a spurious cooldown.
- The shutdown cooling system can remove decay heat from the fuel at full HT system coolant temperature and pressure conditions, and is therefore a backup to the steam generators for emergencies as well as for normal low-power heat removal. NG CANDU has a further method of emergency high-pressure heat removal, through the Group 2 feedwater system supplying the steam generators.
- Separate, seismically-qualified Group 2 water and power systems assure heat removal after an earthquake. For a loss of main feedwater and/or main electrical power, the Group 1 auxiliary feedwater and Class III power systems are effective; these are backed up by the Group 2 systems.
- Distributed control systems control the plant routinely, freeing the operator of mundane tasks and reducing the likelihood of operator error.

Passive safety is another area in which the design efforts of NG CANDU are concentrating. The current CANDU plants have a number of passive safety design features, for example, gravity-driven spring-assisted shut-off rods in SDS1, pressure-driven poison injection in SDS2, thermosyphoning capability in the heat transport system and high-pressure (gas-driven) emergency core cooling.

NG CANDU builds on the existing base of passive safety design elements. As part of AECL's generic engineering program to advance the current product line, options for increasing the reliability and degree of passivity of heat removal from important systems have been studied. For NG CANDU, additional passive systems are being evaluated, including gravity driven emergency feedwater to the steam generators and passive containment cooling capability after an accident.

Probabilistic safety assessment methods are used in the design development of NG CANDU to identify systems where reliability could best be enhanced to ensure that safety targets are met with ample margin. Engineering assessments will then be conducted on the merits of active, passive, and combination active/passive systems or components to achieve those targets.

**Safety analysis**

As part of the early assessment of the feasibility of the NG CANDU design concepts, preliminary analyses of important accident sequences have been carried out to ensure that the safety margins in the response of the plant to upset conditions will be increased. The results from the preliminary large break loss-of-coolant accident (LOCA) analyses show that the design will have substantial improvements in margins to fuel and pressure tube overheating, and other safety related parameters during a large LOCA transient. These preliminary analyses indicate that the internal strain on the pressure tube will be small and result in very little deformation during the course of a LOCA event.

A preliminary probabilistic safety assessment (PSA) is also being conducted with an examination of event trees and severe core damage sequences. The purpose of the preliminary PSA is to identify reliability targets for accident mitigating systems so that the plant design will meet overall safety objectives for prevention and mitigation of severe core damage accidents, consistent with international practice.

There is enough information available from the probabilistic safety assessments for current CANDU designs and other nuclear plants to provide the bases for the NG CANDU preliminary PSA. The moderator backup heat sink remains a fundamental, inherent safety characteristics to deal with severe accidents involving the loss of normal and emergency heat removal systems. Additionally, the preliminary PSA is being used to confirm areas where improvements are possible that would yield substantial dividends in terms of safety margin. For example, early consideration has been given to the layout of equipment and service water systems such that failures of components in one system cannot propagate to other systems.

The availability and reliability of back-up emergency electrical power supplies is another important factor that affects the likelihood of many severe accident sequences. An increased reliability for the back-up emergency electrical power is a design target for NG CANDU. Assessments of the plant design indicate that it is possible to achieve a reliability improvement, without large increases in the cost of the diesel-electric generating systems, by improved load management.

Overall, comprehensive assessment of PSA 'lessons learned' from previous designs and the application of preliminary PSA in parallel with safety design definition will assure that the design probabilistic safety objectives are met.

**Research and development**

AECL has a comprehensive research program in place to maintain and extend the technology database that supports the safety design and performance of the operating CANDU reactors. Because of the evolutionary nature of the NG CANDU design, most of the products of this research program are directly applicable to supporting the design and safety basis of the NG CANDU. In addition to this generic program, AECL is also carrying out specific R&D that addresses the key innovations in the NG CANDU design.

The NG CANDU R&D program includes the following major elements:

- extension of the database on fuel channel performance to NG CANDU operating conditions;
- qualification of CANFLEX SEU fuel to the NG CANDU target performance level;
- development and qualification of new reactor equipment (notably fuel handling equipment);
- extension of the safety analysis technology base to address NG CANDU operating conditions and design features.

The first three elements of this program are very important for the overall development of the NG CANDU reactor, but the remainder of this section will focus on the fourth element and highlight some of the R&D topics of particular relevance to this paper.

At the beginning of the conceptual design of the NG CANDU, the safety impact of the NG CANDU design innovations was reviewed, the impact of the innovations on safety margins was evaluated and R&D requirements were identified to address gaps in the knowledge base which would affect the confidence level of the safety analyses or the safety margins. The review found that the NG CANDU design would provide for substantial improvement in some safety margins with the support of a limited R&D program. The following are examples of the R&D activities.

Since the NG CANDU reactor design is an evolutionary extension of the existing CANDU designs, safety analyses will be performed using the existing set of validated safety computer codes. Each of these codes has been reviewed for its applicability to NG CANDU. Where warranted, R&D work will be carried out to extend the validation database to cover the NG CANDU operating conditions or design features. For example, to address the impact of the increased temperature and pressure of the HT system for loss-of-coolant accidents, experiments will be carried out under the NG CANDU conditions in the RD-14 thermalhydraulics test facility. This extends the validation of the CATHENA safety thermalhydraulics code. To address the tighter lattice pitch and new fuel channel design, experiments will be carried out in the ¼-scale Moderator Test Facility to validate the moderator thermalhydraulics code MODTURC. The validation basis for the core physics code (WIMS-AECL) will be extended to include the NG CANDU lattice geometry and coolant/moderator properties using experiments conducted in the ZED-2 zero energy lattice facility.

The NG CANDU reactor design has a slightly modified fuel channel design (e.g., thicker calandria tube and pressure tube) and new fuel with different power ratings and burnups. To confirm the availability of the moderator as a reliable heat sink for loss-of-coolant plus loss-of-emergency-coolant accident sequences, experiments will be performed to measure heat transfer rates from overheated channels. Related tests will also address the behaviour of the NG CANDU fuel channel for other accident sequences. Experiments will also be performed to extend the ongoing R&D program on CANDU fuel channel degradation phenomenology under severe accident conditions to include NG CANDU design features and operating conditions.

In addition to the work described above, the planned safety-related R&D program will address issues such as margins against component failure under accident conditions and the performance of passive engineered systems.

## Conclusion

NG CANDU design is built on proven technology and is being developed to achieve cost reduction and enhanced safety. The use of light water as coolant, coupled with slightly enriched fuel in the fuel channels, while providing a substantial economics improvement over the existing CANDU plants, enhances inherent safety by giving a flat neutron flux profile which minimizes the demands on the reactor control system and a significantly negative reactivity power coefficient.

Adoption of CANFLEX fuel permits some key improvements such as a larger margin to fuel sheath dryout under transient and accident conditions.

The development of the new design has offered the opportunity to make improvements also in other areas such as layout and separation of safety related systems. Consideration is being given to include further passive systems in addition to those already present in all CANDU plants.

Preliminary safety analyses have confirmed that larger safety margins can be achieved for the response of the plant to accidents and a preliminary probabilistic safety assessment has been undertaken to set reliability targets for the safety related systems so that safety objectives for prevention and mitigation of severe accidents, consistent with international practice, can be met by the design.

Limited additional R&D is required for NG CANDU, basically to address the key innovations in the design. Most of the R&D in place at AECL to maintain and extend the technology base of the existing CANDU designs, is also directly applicable to supporting the NG CANDU design.

## References


1. J.M. Hopwood, J.W. Love and D.J. Wren, “The Next Generation of CANDU: Reactor Design to Meet Future Energy Markets”, proceedings of the 22nd Annual Canadian Nuclear Society Conference, Toronto, June 2001.

2. D.J. Wren and F. Doria, “The Next Generation CANDU Evolution in Safety”, proceedings of the 22nd Annual Canadian Nuclear Society Conference, Toronto, June 2001.

Figure 1. **Comparison of calandria sizes**

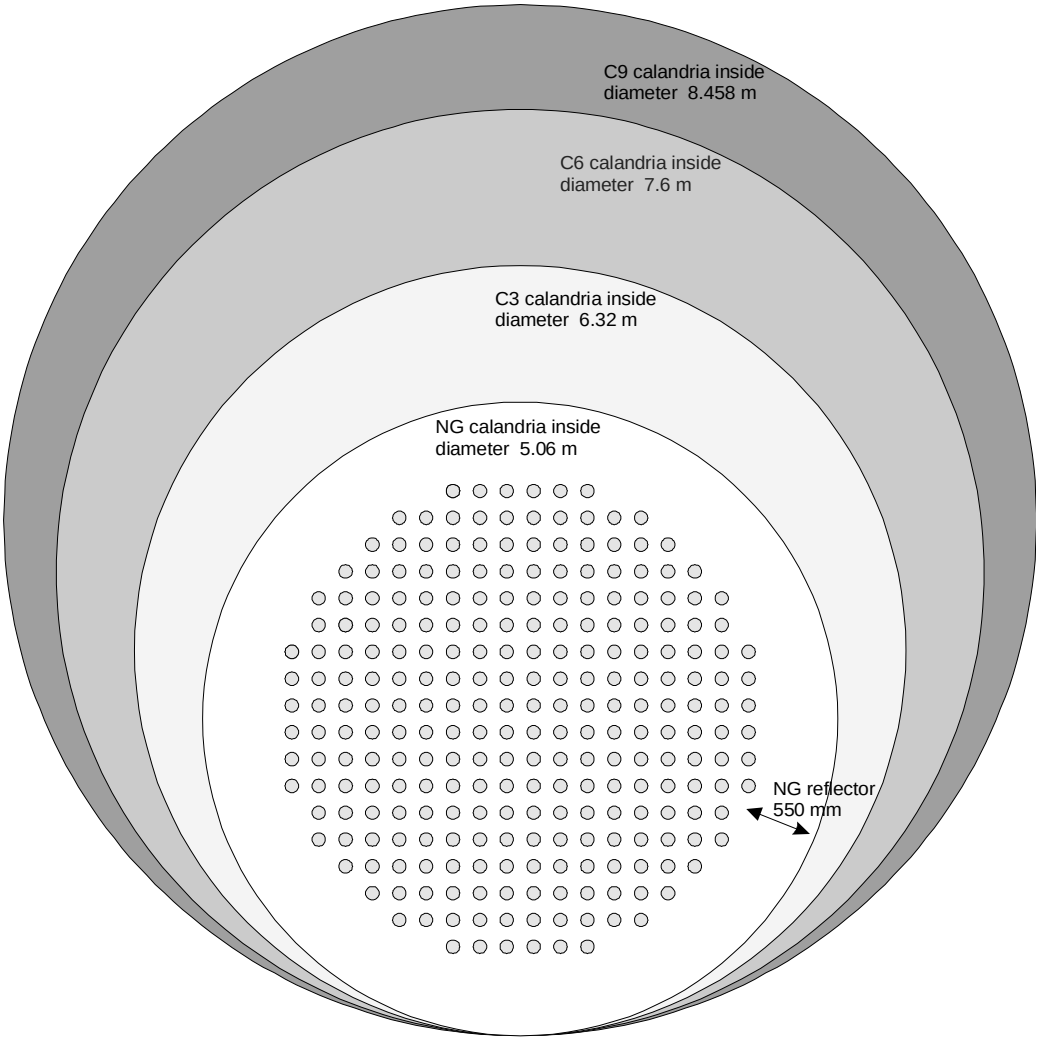
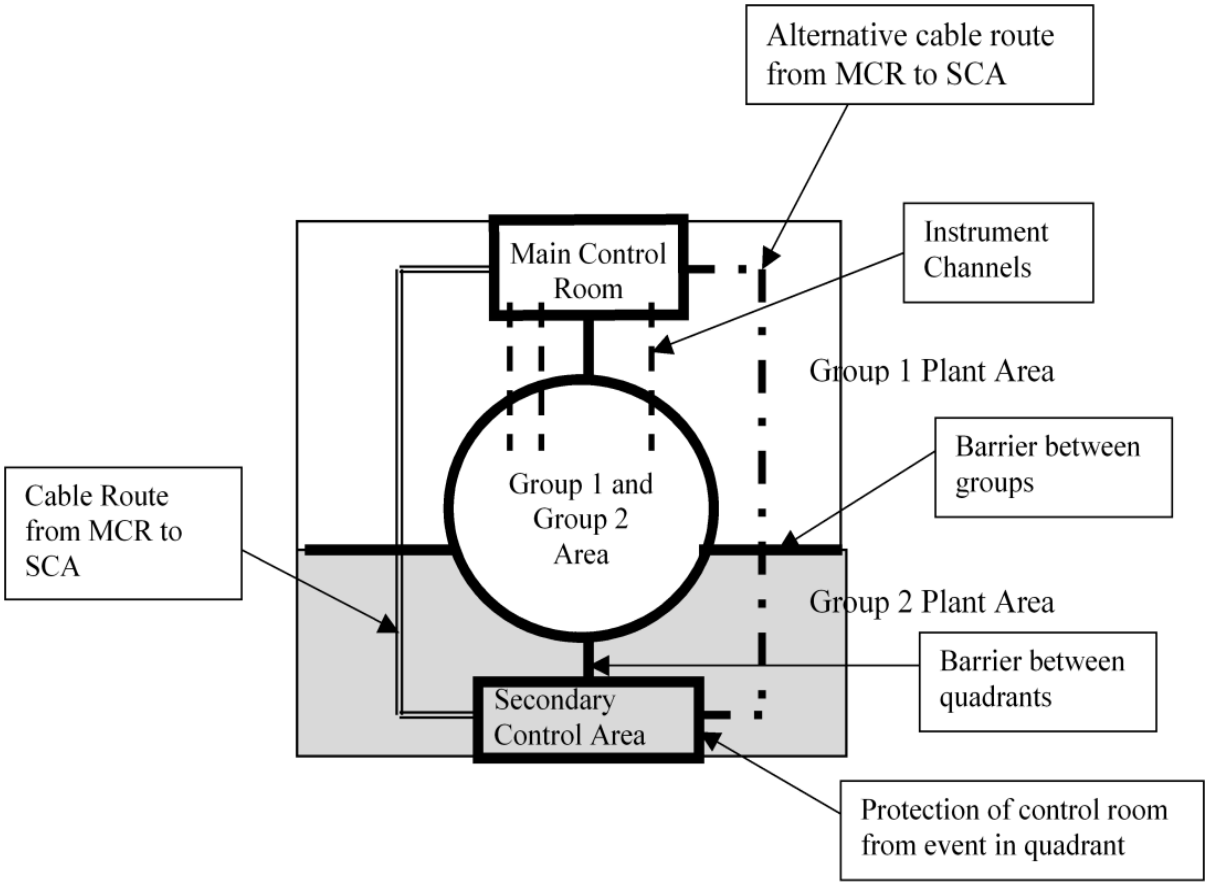


Figure 2. **Layout of group 1 and group 2 areas**



# SAFETY ISSUES FACED DURING THE HTR-10 DESIGN AND LICENSING AS REFERENCE FOR SAFETY DESIGN REQUIREMENT CONSIDERATION FOR INDUSTRIAL MHTGRs

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## 1. Introduction

The HTR-10 is a high temperature test reactor with a thermal power of 10 MW. This test reactor has been erected in the suburb of Beijing and reached its initial criticality in December 2000. It is currently under commissioning tests. The test reactor incorporates the key technical and safety features of modular high temperature gas cooled reactor (MHTGR) designs. The main safety issues faced during the design and licensing review process should be of relevant reference value for the safety design requirement consideration for industrial MHTGRs.

## 2. HTR-10 test reactor and its safety case

The HTR-10 is a 10 MWth pebble bed test reactor, using graphite as moderator and helium as primary coolant. Its conceptual design was developed in the late 1980's by the Institute of Nuclear Energy Technology (INET) in cooperation with German institutions (Siemens and Research Center Jülich). The reactor is intended to test and demonstrate the key technical and safety features of the MHTGR designs firstly proposed by German industries and to serve as a base for further research and development of gas cooled reactor technologies.

The reactor has an active core volume of 5 $m^3$ and it contains 27 000 spherical fuel elements in its equilibrium core. Each fuel element contains 5 g uranium in $UO_2$ form, 17% enriched, in about 8 000 TRISO coated fuel particles. This kind of spherical fuel element demonstrates excellent fission product retention capability, when its temperature is limited to about 1600°C during accident conditions.

Helium coolant is circulated by the primary helium blower and works at 3.0 MPa during operation at full power. Average helium temperature at reactor inlet and outlet is respectively 250°C and 700°C. A steam generator is coupled to the reactor. Reactor nuclear heat is transferred in the steam generator to the secondary water/steam, producing main steam which drives a steam turbine cycle. Figure 1 shows the primary system of the HTR-10 reactor, while Figure 2 shows the simplified process flow of the test reactor plant.

In the following, the safety case of HTR-10 is discussed from the viewpoint of the three basic safety functions, namely, control of reactivity; decay heat removal; and confinement of radioactivity.

### *2.1 Control of reactivity*

There are two shutdown systems, namely the control rod system and the small absorber ball system, both designed in side reflector. Any of these two systems alone can bring the reactor from operating conditions to cold shutdown state and maintain the state. The control rod system also serves for reactor power regulation and reactor trip. Because of the continuous refueling, there is only about 1% excess reactivity reserved for power regulation. The fuel and moderator demonstrate negative temperature reactivity coefficients. The overall temperature reactivity coefficient of fuel, moderator and reflector at power operation is about $-5x10^{-5}$ (1/K). The maximum fuel temperature is calculated at 864°C. Considering a uniform core temperature rise of 500°C, which is allowable from fuel temperature limit point of view, would bring a negative reactivity of more than 2% which can override any possible positive reactivity transient.

Several mechanisms, such as abnormal core temperature transients, inadvertent rod withdrawal, water ingress into reactor core due to steam generator leakage or tube break, earthquake, etc., can cause abnormal reactivity transients or positive reactivity insertion. The design is such that all kinds of possible positive reactivity insertion can be covered by the limiting one which is the inadvertent withdrawal of control rods.

In summary, reactivity control or reactor shutdown is normally realized by engineered shutdown systems, but the inherent mechanism, i.e. the negative temperature coefficient, can also override any positive reactivity transients when the primary helium blower is stopped which is always one action item of any reactor trips, without the limiting fuel temperature being exceeded.

### *2.2 Decay heat removal*

Due to the design selection of low volumetric power density and the large heat capacity of core material which are typically common to all MHTGR designs, the HTR-10 test reactor does not require, for the purpose of decay heat removal, any flow or even the presence of helium coolant in reactor core. Decay power in the pebble bed would then dissipate by heat conduction and radiation, and natural convection if coolant is present, to the outside of the reactor vessel.

On the wall of the concrete compartment which houses the reactor pressure vessel, a reactor cavity cooling systems is designed to take the decay heat dissipated through the pressure vessel away to the ultimate heat sink. The HTR-10 cavity cooling system consists of two redundant water cooling panels which work on natural circulation principle. It transfers the decay heat to atmospheric air via air coolers. The cavity cooling systems is always in operation when the reactor is in operation to take away the heat loss from the reactor pressure vessel. The maximum fuel temperature due to decay heat after reactor trip is not very much dependent on the operation of the cavity cooling system. In fact, the function of the cavity cooling system, as its name indicates, is more to protect reactor vessel and cavity concrete from being over-heated.

In summary, decay heat removal from reactor core is realized inherently. The core heat-up and cool-down process usually takes place over a very long time period, allowing reactor operator to monitor accident progression and take appropriate measures. For the HTR-10 reactor there is in fact no

core heat-up process after reactor trip because of the overriding heat dissipation relative to decay power generation.

### *2.3 Confinement of radioactivity*

The barriers against fission product release consist of the coated particles, the spherical fuel elements, the primary pressure boundary, and the confinement. Since the reactor design is such that reactor shutdown and decay heat removal can all take place inherently, no challenging conditions are expected which would lead to considerable damage to fuel elements. Therefore, coated particles and fuel elements become the primary barriers. As long as the fuel meets it specified quality, fission products will be retained within the fuel elements.

The confinement of HTR-10 consists of concrete cavities which house the primary pressure boundary components and systems, including the reactor, the steam generator, the helium purification system, and parts of the fuel handling system. The confinement is a low-pressure-containing concrete boundary which is kept sub-atmospheric by the ventilation system during normal reactor operation. When primary helium leaks into the confinement due to pipe break and would cause unacceptable pressure increase, the leaked helium would be directly released to environment through a blow-off path without filtering. This would not lead to unacceptable dose rate on the plant site and for the public. The helium blow-off path of the confinement will be closed when the confinement internal pressure comes down to atmospheric, and the confinement will be ventilated with iodine filtering.

In summary, confinement of radioactivity is designed to be basically realized within the fuel elements. It can be said that the integrity of primary pressure boundary and confinement is of less significance in terms of serving as radioactivity release barriers.

## 3. Main safety design issues

### *3.1 Safety classification*

Safety classification of systems and components is a key issue in the plant design. Appropriate quality requirement on systems and components can enhance the safety level of the plant, while over-emphasized quality requirement would lead to unnecessary plant cost increase. Component safety classification in HTR-10 design approach has been done basically by following the well-established national and international safety standards and by making significant reference to the practice of large water reactor plants. Therefore, systems or components in the HTR-10 plant which perform seemingly similar functions have been classified similarly to large extent. Examples could be named such as the primary pressure boundary components including primary isolation valves and their actuators, main steam isolation valves and associated safety valves, electrical power supply system, protection system and so on. As usually practiced, design principles of redundancy, diversity, fail-safe, physical separation, etc., are followed where necessary and applicable.

Certain types of components and equipment are more or less specific to the technology of high temperature reactors, e.g. helium blowers or electrical penetrations. When such components are classified as nuclear grade, their qualification requirements have to be established specifically because of the limited availability of market supplies. The function of certain equipment with MHTGR is, roughly speaking, similar to that with large water reactors, but the functional requirement could be different, e.g. the diesel generators. The start-up requirement for diesel generators with MHTGR is

usually much less demanding. In the HTR-10 practice, specific qualification requirements on diesel generators have been established.

Safety classification, same as accident analysis discussed in the next section, is one integral part of the overall safety design approach. The HTR-10 safety design has taken the deterministic approach, as practiced for other reactors. In fact, the performance of the basic safety functions can be relied on inherent features. Certain engineered systems or components could help perform the safety functions in accident conditions in a more expected manner. Their inoperability or delayed operation would not have strong impact on plant safety. These kinds of systems or components normally have been classified as nuclear grade within the HTR-10 design. This issue could be better addressed for industrial MHTGR designs.

### *3.2 Accident analysis*

Design basis accidents (DBA) are selected and classified in several categories for the HTR-10 reactor. In selecting and classifying the DBA, reference has been made to the licensing experience in the 1980's in Germany [1] and USA [2] of their specific modular high temperature reactor designs. The DBA classification approach is done as normally practiced according to the estimated occurring frequency of the initiating events. The three classified categories are then respectively: middle frequency accidents; rare accidents; and limiting accidents including anticipated transients without scram (ATWS). In Table 1, classification of typical initiating events is given. The HTR-10 reactor is designed against DBA. The analysis of these accidents is done with conservatism. The analysis results show very safe response of the reactor to initiating accident events.

Hypothetical accident scenarios of the HTR-10 plant beyond design basis have been considered in the safety design, in particular in the safety analysis. A number of hypothetical accidents are selected to be analyzed. A list of these hypothetical events is also given in Table 1.

Great attention and effort has been given to the discussion and analysis of these hypothetical events during the stage of the construction permit licensing. Reasonable justification of the consideration of these extremely low probability events could not always be found. Some events are then agreed to be ruled out for further consideration in the detailed design and license stage, e.g. the overlapping of loss of power plus failure of cavity cooling system plus primary depressurization, and the assumption of simultaneous rapture of all steam generator tubes, even though the latter actually does not cause great concern from safety point of view.

Analyses of these events have applied realistic models to the best knowledge. These analyses have led to some specific engineering measures against some hypothetical events in the detailed design stage. For example, one additional means has been designed to trip the primary circulator (which is actually always one action item of reactor trip). And a penetration path has been reserved through the primary system cavity concrete for introducing appropriate medium into the cavity in case the connecting vessel rupture would happen.

Of most interest is probably the scenario of connecting vessel rupture leading to air ingress into the reactor core, because air would react with high temperature graphite structures and fuel elements. Such chemical corrosive reactions could sustain if enough air is available mainly because the reactions are exothermic. Reactor structures would then be damaged and fission product radioactivity in fuel elements would eventually be released to unacceptable levels. But the progression of the air ingress accident and corrosive reactions is rather slow. It typically runs over a few days before appropriate measures have to be taken to terminate the air ingress process.

Table 1. **HTR-10 accident classification**

| Initiating events | Category |
|---|---|
| Inadvertent increase in helium flow rate<br><br>• Inadvertent increase in feed-water flow rate<br>• Fault opening of the Main Steam Safety Valve<br>• Loss of power supply<br>• Loss of or inadvertent decrease in helium mass flow<br>• Loss of or inadvertent decrease in feed-water mass flow<br>• Trip or malfunctions of the turbine and condenser system<br>• Inadvertent withdrawal of one control rod<br>• Rapture of small helium pipes (small LOCA) | Middle Frequency Accidents |
| • Small break of the feed-water pipeline<br>• Rapture of one steam generator tube<br>• Rapture of larger helium pipes (large LOCA) | Rare Accidents |
| • Rapture of the main steam pipeline<br>• Rapture of the feed-water pipeline<br>• Compaction of the pebble bed due to earthquake<br>• Rapture of two steam generator tubes<br>• Loss of power without scram (ATWS)<br>• Inadvertent withdrawal of control rod without scram (ATWS)<br>• Loss of feed-water without scram (ATWS) | Limiting Accidents |
| • Failure of the helium circulator trip<br>• Loss of power plus failure of the cavity cooling system plus primary depressurization<br>• Inadvertent withdrawal of all control rods at cold startup<br>• Rapture of all steam generator tubes<br>• Rapture of the connecting vessel | Hypothetical Plant Scenarios |

### *3.3 Fuel*

Present MHTGR designs are based on fuel elements with TRISO coated particles, so is the HTR-10 reactor. TRISO coated particle spherical fuel elements have been very successfully developed in Germany. Design specifications have been rather well established. The issue of fuel elements for HTR-10 is more fuel fabrication and qualification. The design and fabrication of HTR-10 fuel elements has benefited greatly from German experience. Irradiation qualification tests are being made for the Chinese made fuel elements. Real maximum burn-up level in the HTR-10 reactor will be always behind the burn-up realized in irradiation tests, as required by regulatory authority.

## 4. Research needs and concluding remarks

The safety design approach of the HTR-10 reactor should be of relevant reference value for the safety design requirement consideration for industrial modular high temperature reactors. In particular, the real operating performance of HTR-10 should provide valuable experience in this regard. The safety design approach of HTR-10 has been practiced to greatest extent in accordance with Chinese regulatory documents and practice which are established so far basically based on the

experience of other reactor types. The shortage of a set of logically established regulatory standards, which take more account of the technical and safety features of modular helium reactors, certainly has led to difficulties or complexities for the designers to derive a logic, coherent and consistent safety design approach, which could easily go through the licensing process.

It is therefore believed that, considering the worldwide broad interest in the MHTGR technology, there is a strong need to investigate on the issue of starting to establish regulatory documents which take specific account of the technical and safety features of MHTGRs and of the latest advances in nuclear technology. The IAEA has started effort in this regard, and strong support from member states is definitely necessary, in particular from those member states who have gathered a lot of expertise and experience in the helium reactor technologies.

There exists worldwide much knowledge on high temperature gas cooled reactors. Both Germany and the USA have spent a lot of money on the development of this technology. Both test and demonstration plants were built and operated in these two countries. China and Japan have recently erected test reactor plants. Advanced gas cooled reactors are now still in operation in a number of countries. Pooling and preservation of relevant knowledge and technology deserves international cooperation effort. The IAEA has also such initiatives in its gas cooled reactor program through Coordinated Research Programs (CRP), and this certainly also requires strong support from member states. Operational test reactors such as HTR-10 should expectedly contribute to the pooling of knowledge. For example, the HTR-10 could serve as a test bed for spherical fuel elements. A series of accident simulation tests are also planned to be carried out on the HTR-10, which could help demonstrate to some extent the safety case of MHTGR designs.

MHTGR technology has not come to being fully market mature. There are certainly many areas where dedicated research and development is still necessary. To start from the basics, development is necessary for example of verified code systems for core physics calculations, which is important for both plant safety design and licensing. Accident analysis represents another area for research needs. Definition of the design basis accident spectrum and consideration of severe plant conditions needs to be addressed and have a certain kind of consensus. Analysis of certain accident scenarios, such as water or air ingress into reactor core, needs to have improved models and computer codes. Fission product behavior in coated particle fuels and in the primary system needs to be better understood. Hardware technical development is surely also necessary. This is particularly true for MHTGR designs which involve direct close cycle helium turbines.

## Reference


[1] Siemens, High-Temperature Reactor Module Power Plant Safety Analysis Report, April 1987.

[2] Williams, T.L. King, J.N. Wilson, Draft Pre-application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor, NUREG-1338, March 1989.

Figure 1. **The HTR-10 primary system**

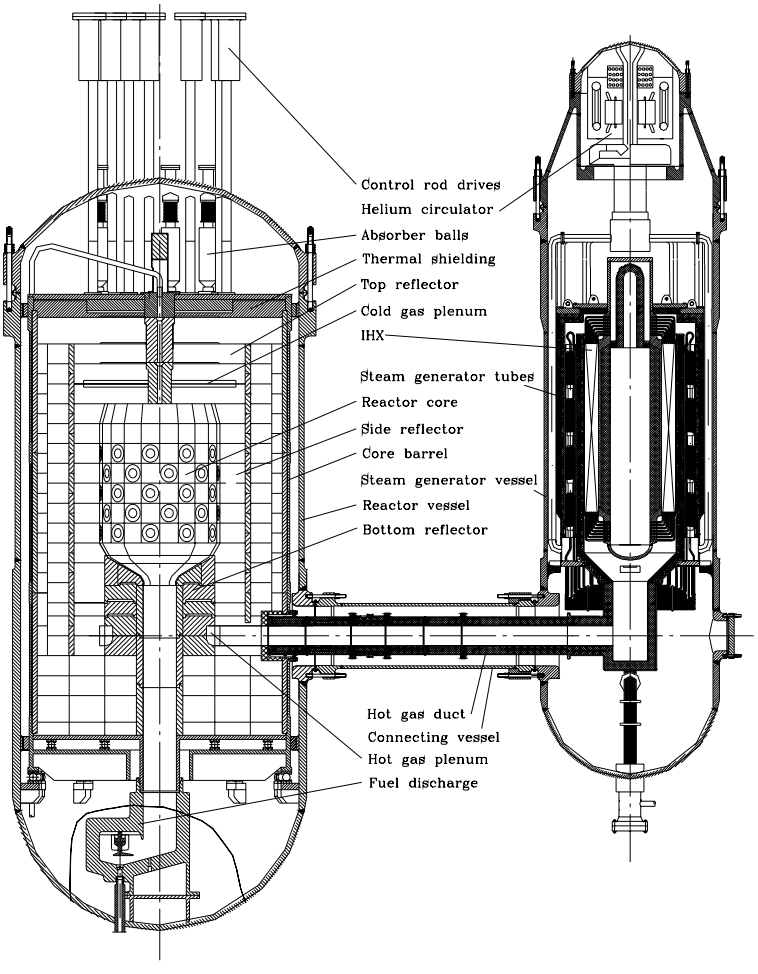
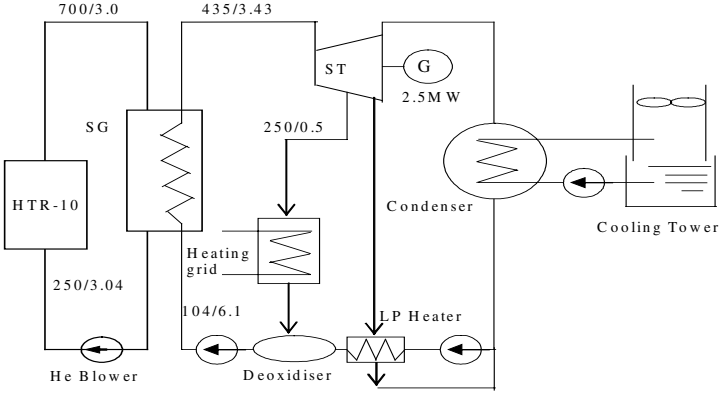



Figure 2. **Simplified process flow of HTR-10**



# THE FIRST STAGE OF LICENSING OF PBMR IN SOUTH AFRICA AND SAFETY ISSUES

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## 1. Introduction

The National Nuclear Regulator (NNR) has received a nuclear installation licence application from Eskom (the South African electricity utility). The Application is made in accordance with the National Nuclear Regulator Act for a nuclear installation licence for the demonstration module of a 110 Mwe Class Pebble Bed Modular Reactor (PBMR) electricity generating power station.

It is proposed to locate the installation on Eskom property within the owner-controlled boundary of Koeberg Nuclear Power Station situated in the Western Cape, subject to *inter alia* a favourable Environmental Impact Assessment (EIA) record of decision, which is currently being undertaken under the requirements of another legislation the Environment Conservation Act.

The PBMR is a graphite moderated helium cooled reactor using a direct gas cycle to convert the heat, generated by nuclear fission in the reactor and transferred to the coolant gas, into electrical energy by means of a helium turbo-generator. By design, provision has been made to accommodate the storage of spent fuel in the buildings for the 40-year design life of the plant and thereafter for a further period if so required. Radioactive material and waste will be managed and disposed of in accordance with Regulatory and Government legal requirements.

## 2. Licensing process

In terms of the complexity of this Project a multi staged licensing process has been adopted by the NNR. This is to acknowledge the developmental nature of the PBMR Demonstration Unit. The approach adopted entails that, following a satisfactory Regulatory Review of the application by the NNR, an initial Nuclear Installation Licence, (NIL) will be issued to the applicant for the first stage of the process and a Variation to this NIL will be requested by the applicant, and issued by NNR following its satisfactory Regulatory Review, at each of the subsequent agreed Licensing Stages. A programme of staged licensing submissions will coincide with the application for a NIL variation (by means of a NIL Change Request) to proceed to the next phase, which will need to be supported by a comprehensive safety justification e.g. Safety analysis Report to demonstrate compliance with the NNR Regulatory safety requirements.

Each stage of the licensing process will indicate the NNR Hold & Witness Points that will form the prerequisites to proceed to the next licensing stage. The Quality Assurance (QA) Programme

will ensure traceability and credibility of results of the previous licensing stage, before issuing the next stage licence variation.

The Licensing Programme currently includes, *inter alia*, the following major licensing stages:

- Limited construction activities (PBMR NIL issued for the first stage).
- Construction and manufacturing phase (NIL Variation).
  - Civil works.
  - Installation of auxiliary systems.
  - Installation of main power system.
- Nuclear Fuel on Site/ Commissioning and Start-up (NIL Variation).
  - Cold commissioning testing.
  - Fuel load.
  - Initial criticality.
  - Low power testing.
  - Full power testing.
- Operation (NIL variation).
- Decommissioning.

As indicated above a comprehensive safety justification e.g. Safety Case must accompany and support the application for the initial NIL and for each subsequent applicant for a NIL Variation. The framework of such safety case is presented below under Chapter 3.2.

The first stage of the licensing process, which is currently being undertaken, is the regulatory review towards issuing the initial NIL, which will be issued for limited site construction activities prior to issuance of the licence variation authorising the construction of the civil structures, auxiliary systems and main power system of the Pebble Bed Modular Reactor.

Typical activities authorised under the scope of the first stage of the licensing process, which is being discussed with the applicant, would be as follows:

- Preparation of the site for construction of the facility (including such activities as clearing, grading, and construction of temporary access roads).
- Installation of temporary construction support facilities (including such items as warehouse and shop facilities, utilities, concrete mixing facility, unloading facilities, and construction support buildings).
- Excavation for facility structures.
- Construction of service facilities (including such facilities as roadways, paving, fencing, exterior utility and lightning systems, transmission lines, and sanitary sewerage treatment facilities).

The following typical documentation to be submitted in support of the application for authorisation of the above activities of the first stage of the licensing process:

- Approved safety case philosophy.
- Updated site safety report.
- Environmental impact assessment report (EIR).
- Licensing programme for the multi licensing stages.
- Site redress plan.
- Safety case.

The Site Redress Plan provides an assurance that the activities performed in the first stage will not result in any significant environmental impact that cannot be redressed.

Each licensing stage shall be scoped by making reference to the construction programme, and to the test and commissioning programme, to ensure alignment of the expectations of both the applicant and the Regulator.

## 3. Licensing requirements for the PBMR and the PBMR safety case

### *3.1 Licensing requirements for the PBMR*

The "Basic Licensing Requirements for the PBMR" [1] describes the fundamental safety standards adopted by the National Nuclear Regulator and provides some insight into their basis and establishment. It presents the derived standards in terms of design and operational principle and in terms of quantitative risk criteria both of with which the design must comply. The document then describes the processes that the licensee must undertake in demonstrating compliance with the standards, essentially the requirements for licensing of the reactor.

The licensing requirements defined by the NNR for the PBMR cover all general safety requirements needed to protect individuals, society and the environment from radiological hazard. In this sense for the purpose of this workshop, licensing requirements and safety requirements should be considered as synonyms.

The philosophical basis for the current safety standards set down by the National Nuclear Regulator for licensing any nuclear installation or activity involving radioactive materials is presented in a set of fundamental principles referred to as the fundamental safety standards. From these standards, quantitative criteria and qualitative requirements are derived for a particular installation or activity and the licence applicant must demonstrate that the installation or activity in question will comply with these regulatory requirements.

In order for the applicant to demonstrate that the reactor will be acceptably safe, it is required that he demonstrate that the design and operation of the plant:

- respects good nuclear safety design practise;
- that it will make use of appropriate internationally recognised design and operational rules;

- will comply with the risk and radiation dose limitation criteria.

With regard to good nuclear safety design practice, of prime consideration are the principles of defence-in-depth and of ensuring that risks and radiation doses to members of the public and workers will be maintained as low as reasonably achievable (ALARA) below laid down radiation dose limits.

The "Basic Licensing Requirements for the PBMR" and the licensing process adopted requires the applicant to identify all events that will be associated with the normal operation of the reactor (referred to as category A events with a frequency up to $10^{-2}$ /y), which will or could give rise to radiation exposure to workers or members of the public. The design of the plant must be demonstrated to ensure that such exposures will not give rise to the applicable dose limits being exceeded and will be maintained as far below these limits as reasonably achievable by optimal provision of engineered and operational safety features. In undertaking the assessment to demonstrate compliance with dose limits, conservative assumptions must be used.

The applicant is also required to identify all those events associated with the design which could reasonable be anticipated to be possible and which may give rise to accidental exposure of workers or members of the public (referred to as category B events with a frequency from $10^{-2}$ to $10^{-6}$ /y). The applicant must demonstrate that such events will either be prevented from occurring or that the design will mitigate the consequences such that radiation doses will not exceed laid down criteria and will not give rise to any serious off site radiation hazard. Again conservative assumptions must be made in this assessment.

According to the NNR requirements all events even with very low probability of occurrence or complex events of equally small likelihood, which could give rise to accidental exposure (referred to as category C events) must also be identified. A probabilistic risk assessment must be conducted which includes these and the other events (identified in categories A and B) and a demonstration provided that the risk from the reactor will comply with the criteria laid down for workers and members of the public. It is acceptable for best estimate assumptions to be made for this assessment

For this analysis a cut off criteria for the consideration and analysis of low probability events also needs to be established.

In addition to demonstrating that the reactor will be safe in terms of meeting good design and operational requirements and will comply with the risk and radiation dose criteria, the applicant must also demonstrate that the radioactive waste arising from operation and decommissioning of the reactor will be safely managed. This requires all sources of waste to be identified and characterised and that the design makes provision for collection and treatment of the waste, for control over effluent discharges and for safe storage of waste at the facility. The adequacy of these proposals will be evaluated against prevailing internationally endorsed standards for radioactive waste management and Governmental policies in terms of radioactive waste management in South Africa.

The applicant must also demonstrate that arrangements will be in place to deal with any accident that may occur. The arrangements must enable the operator to recognise the occurrence of an accident or incident, which may degrade levels of safety. Accident management procedures will be required to minimise the consequences of any accident and arrangements in place to ensure that the public and workers will be adequately protected.

### *3.2 PBMR safety case*

The licensing process requires the licensee to present a safety case to the National Nuclear Regulator which is a structured and documented presentation of information, analysis and intellectual argument to demonstrate that the proposed design can and will comply with the licensing requirements. In order to demonstrate that the PBMR design will meet the licensing requirements, Eskom has, in consultation with the NNR, developed and implemented a structured process to develop the PBMR safety case. This process also provides a logical link between the various steps of the design process, the safety assessment and the development of operational support programmes.

The two main components of the Safety Case are 1) the Safety Case Philosophy (SCP) and 2) the Safety Analysis Report (SAR) and Development/Support Documents.

The Safety Case Philosophy provides the high level intellectual safety argument, and demonstrate the linkage between the various elements of the licensing basis, clearly linking the Licensing requirements, the plant design basis, the plant safety assessment and the plant General Operating Rules (GORs) while the SAR provides the detailed justification for the demonstration of safety as presented in the safety case philosophy.

The PBMR Safety Case development framework is illustrated in Table 1.

As indicated in the table the following nine main elements, briefly explained below, have been identified for the development of the PBMR safety case.

a) **Fundamental Safety Design Philosophy** – The key safety objectives and fundamental safety principles on which the PBMR will be designed, constructed, commissioned, operated and ultimately decommissioned are defined.

b) **Quality Management Programme** – Over its entire lifecycle the PBMR must be supported by a quality management system.

c) **Technical Description & Key Safety Characteristics** – presentation of the PBMR plant technical description and key safety characteristics

d) **Identification & Classification of Events** – Licensing basis events. Identification and classification of all potential challenges (events) to the plant which could give rise to radiation exposure to workers or members of the public.

e) **General Design Criteria (GDC)** – Identification and development of the General Design Criteria against which the plant will be designed to prevent/mitigate the consequences associated with the identified events.

f) **SSC Classification – Classification of Systems Structures and Components (SSCs).** Present the safety classification of the plant SSCs, which provides the rationale for determining the relative stringency of design requirements and rules applicable to the SSCs as derived from the above GDC.

g) **Design rules** – Development of the set of rules, codes and standards which will be applied to the PBMR design, construction (including manufacturing), commissioning, operation and maintenance.

h) **Safety Assessment** – an appropriate safety assessment must demonstrate that the PBMR design is in line with the PBMR fundamental safety design philosophy and meets the associated regulatory requirements.

i) **Support Programmes** – As derived from the safety assessment a set operational programmes e.g. General Operating Rules (GORs) is developed to support the operation of the PBMR.

For the development and review/assessment of each of these nine elements the following approach has been implemented:

- **Column a: Safety Case Philosophy** – the philosophical approach of each element is presented.
- **Column b: Safety Case Route Map** – This basically provides the link between the Safety Case Philosophy (SCP) and the SAR. The Safety Case Route Map defines the following:
  - HOW will the assumptions and assertions made in the SCP be substantiated.
  - WHERE they will be substantiated e.g. in the main body of the SAR or/and in supporting documentation.
  - WHEN will they be substantiated in the licensing process and to what level of detail and completeness.
- **Column c: Development documentation** – giving additional detailed information e.g. safety analyses, in depth design calculations etc. Development/Support Documents are required to provide further details to the information submitted in the SAR.
- **Column d: Safety Analysis Report** – the Safety Analysis Report (SAR) documents the output of a, b and c in presenting the safety demonstration of the PBMR. Compliance with the NNR licensing requirements and safety criteria must be demonstrated by way of formalized safety analyses. These safety analyses shall be presented in a Safety Analysis Report (SAR), which shall substantiate the statements, made in the Safety Case Philosophy and be carried out in an auditable fashion under the appropriate QA regime. The SAR is the principle document submitted with the various licence variation applications as part of the staged licensing process. Specific licensing issues may require addressing by means of focused supporting licensing submissions, but these will be the exception rather than the rule.

This systematic "matrix" type process provides a framework for efficient project management and reporting mechanisms for both Eskom and the NNR.

## 4. Major safety issues/concerns identified

At this stage of the licensing process the NNR has not carried out an in depth review of the design and safety analysis of the PBMR. However the following concerns and issues important to safety have been identified

### *Application of Defence-in-depth in the PBMR*

As indicated above one of the main consideration in the NNR safety requirements is the application of defence-in-depth. As an internationally adopted principle defence-in-depth requires that there should be multiple layers (structures, components, systems, procedures, or a combination

thereof) of overlapping safety provisions. Accident prevention and accident mitigation are natural consequences of the defence-in-depth principle.

The application of defence-in-depth for High Temperature Gas cooled Reactors (HTGR) is currently not supported by international guidelines and therefore there are some views amongst the PBMR designers that the PBMR fuel balls will provide sufficient levels of defence-in-depth. The NNR does not accept this kind of approach and considers that defence-in-depth principles are generally applicable and required in assuring the safety of any Nuclear Power Plants. In this respect the NNR recommended to the applicant to use, as a guideline, the approach developed in the draft document prepared by the IAEA Consultancies on "Safety and Licensing Aspects of the Modular HTGR". The applicant has taking cognisance of the NNR recommendation and has subsequently accordingly updating their Safety Case Philosophy to reflect the application of the 5 levels of defence-in-depth (as per IAEA INSAG 10) to the PBMR.

Completion of the IAEA document, mentioned above, is seen as an important milestone in the process of establishing and harmonizing international safety standards for advanced Modular HTGR reactors.

***PBMR design basis***

One of the major safety requirements is the credibility of the PBMR design basis. Unlike for Pressurised light Water Reactors (PWRs) such as the Koeberg Nuclear power Station design, for which well-researched and documented design criteria and rules are readily available, broad international consensus has not been developed in terms of general design criteria and design rules for the PBMR. No international "off the shelf" package is available for defining the design basis of the PBMR. The establishment and documentation of a credible PBMR design basis is thus an important issue, which shall be resolved during the licensing process.

***Requirements for the confinement structure***

Internationally there are many philosophical discussions around the acceptability of building a new reactor without having a conventional type of PWR containment. The approach of the NNR in this regard is that the design requirements of the confinement structure will be defined by the capability of the structure towards accident mitigation. Should the results of the accident analyses, demonstrate with adequate safety margins, that the PBMR design has low radiological consequences during accidental conditions, the PBMR may not require a conventional PWR type containment when considering plant faults, and therefore a conventional type of confinement structure designed to withstand external events e.g. earthquake, aircraft crash etc. might be adequate; the detailed analyses will have to support this conclusion.

In terms of these external events taking into account the recent events, which happened on 11 September 2001 in the USA, consideration is being given in terms of the criteria for the analysis of the aircraft crash.

***Use of passive safety features and systems in the PBMR design***

A main feature of the PBMR design is the elimination of most of complex active systems that rely on a large number of safety grade support systems as for example used in PWR type reactors

and the extensive use of passive safety features and systems to perform the required safety functions. This "new" approach requires the applicant, as part of the safety case, to demonstrate the capability and the reliability of these passive safety features and systems in particular for the long time response required during some transient or accident scenarios. This extensive use of passive components, could lead to the case that by sound design the safety of the PBMR is determined by initiating events of very low probability. Therefore taking into account the advanced reactor type of the PBMR and its expected inherent safety characteristics, the main change which has been made in the NNR licensing requirements in comparison with the existing requirements for PWR (as based on ANSI/ANS-51.1-1983 ) is in area of event categorization. As indicated above in 3.1 all design basis events have been combined into two categories A and B up to a frequency of $10^{-6}$ /y.

***The PBMR annual core***

One of the new design features of the PBMR compared to the previous HTGR design e.g. the German AVR is that to reduce the peak fuel temperatures the PBMR has an annular core design with a central graphite balls column. There is currently limited experimental data, which supports this core geometry. At this stage this data is however deemed to be insufficient to give confidence that the assumed annular core geometry would remain uniform during the life cycle of the power plant, especially taking into account that the current proposed PBMR design does not cater for an instrumentation monitoring system, which could provide an on line verification of the annual core geometry. The following options could be considered to address this issue:

- establishment of an experimental program to verify that under the PBMR operational condition the core geometry can be maintained uniform; this may however prove to be quite difficult to achieve with a high level of confidence.
- Consideration for deviating from the assumed annular core geometry. Two scenarios can be envisaged: the first implies that no credit is given in the accident analyses for the annular core; this option could most probably impose some restriction on the thermal power that can be generated by the core and the second is to consider this deviation as a design basis accident in the PBMR design.

This issue is currently being discussed with the applicant in terms of which options (or mix of options) will be most appropriate.

***Use of computer codes in safety analyses***

The design and safety calculations are performed using evolutional models incorporated in complex computer codes that simulate many different phenomena.

One of the safety rules requires that the licensee demonstrates in the safety case that the safety analysis performed for the PBMR design is comprehensive and sufficient and that all models used are robust and benchmarked against experimental data. The NNR licensing guide LG 1038 [2] presents the NNR requirements for licensing submissions involving computer codes and evaluation models for safety calculations. The document defines the following specific requirements:

- A complete description of each evaluation model which is sufficient to permit technical review of the analytical approach, empirical correlations, the equations used, their approximations in difference form, the assumptions made and included in the programs,

procedure for treating program input and output information, specification of those portion of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

- Solution convergence shall be demonstrated for each computer program, by studies of system modelling or nodalisation and calculational time steps.
- Sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in nodalisation, time step size and phenomena assumed in the calculation to predominate. For items for which results are shown to be sensitive, the choices made shall be justified.
- The empirical models and correlations used in the evaluation model shall be compared with relevant data. Predictions of the entire evaluation model shall be compared with applicable experimental information. If an evaluational model for evaluating the behaviour of the reactor system during a postulated accident includes one or more computer programs and other information, overall code behaviour must be checked against results from standard problems or benchmarks.

There are two main approaches used in evaluational models: Conservative and Best-estimate.

The Conservative approach is a well-known traditional one that provides assurance that calculational results and values of the critical parameters are conservative from the safety point of view. The approach has to be used carefully because in some cases the conservatism that was incorporated for calculating one critical safety parameter could introduce non-conservatism with regard to the other.

The Best-estimate approach requires the use of phenomenological models for realistic calculations of processes and systems behaviour. This kind of evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behaviour of the reactor system during postulated accidents. Comparison with applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that when acceptance criteria are satisfied, there is a high level of probability that the criteria will not be exceeded.

As indicated in 3.1 above, the NNR requires to use conservative approach for the design basis accidents (category A+B) and best-estimate approach could be used for category C or risk analysis.

At this stage, it is evident that the validity of models and data used for the PBMR is not adequate to provide the safety evaluation of accident conditions which is comparable to for example what is currently available in terms of the safety evaluations of existing reactor designs e.g. PWRs.

In this respect an important part of the licensing process will be the validation and assessment of code quality and uncertainties of the results. An insufficient validation and verification of the codes may be compensated by extra margins, limitation of power etc.

There are many examples of computer codes verification and validation around the world. The approach used by US NRC for its Camp agreement to validate the Relap5 code is a good example

of international cooperation, which could be applied in principle for the verification and validation of the PBMR computer codes.

Important experimental data needed for computer codes validation will need to be obtained during plant testing. Therefore the demonstration plant will be subjected to a comprehensive step by step testing and commissioning programme for the acquisition of such. The expected major testing areas are: fuel design, reactor physics, in-vessel-flow distribution and flow-induced vibration, the reactor vessel, the passive heat removal system, and the safety analysis.

The main objectives of this test programme will be:

- to resolve safety questions in order to proceed to the next licensing stage;
- decrease and justify uncertainties in the design and safety analysis;
- validate evaluation models and computer codes;
- demonstrate and validate expected inherent safety features of the PBMR design.

To fulfil these objectives a comprehensive testing programme must be prepared and justified. Each step of the testing programmes shall be supported by a safety assessment. As part of this safety justification reactor transient response shall be investigated and assessed over the maximum practical range without significant challenging the NNR safety requirements.

In order to capture the relevant data required, during the commissioning phase, the PBMR demonstration plant will be fitted with additional instrumentation and controls and other systems important to safety as identified in the safety assessment supporting the proposed testing programme, which will not necessarily be required in subsequent plants. The test programme should be directed towards internal events and conducted step by step from lower power and low decay heat to higher power and decay heat conditions.

This type of testing for the demonstration module, outline above, is one the most important part of the concept in some circles referred as " licensing by test", which will very likely be applied for the licensing of the PBMR demonstration plant should the NNR assessment concludes that the qualification and verification of computer codes as well as plant equipment in the safety case is not adequate. As indicated above as part of this concept additional margins will need to be built in the safety assessment at each step of the testing programme and additional plant hardware will be required, at least on the first demonstration module under review. This concept, which is being discussed between the NNR and the applicant, is relatively new and it requires serious consideration both from a philosophical and practical development point of view.

***PBMR fuel issues***

The Fundamental Safety Design Philosophy of PBMR is based on the premise that the fuel adequately retains its integrity to contain radioactive fission products for all normal operating and design basis accident conditions, thereby allowing radiological safety to be assured. This is achieved by relying on fuel, whose performance has been demonstrated under simulated operating and accident conditions, and whose integrity, therefore, is not compromised even under accident conditions.

Fuel design limits are required to be established since one of the key safety features of the PBMR is based on the fuel design and performance.

The limiting value of about 1600°C for the coated fuel particles is widely accepted by the HTGR international community, and since this limit plays an important role in the safety analyses, it must be adequately justified for the current fuel design by the applicant.

To justify the fuel design limits, the applicant is required to address fuel system damage mechanisms and provide justification for limiting values for important parameters such that damage and radioactive fission product release be limited to acceptable levels.

Considering aspects of PWR fuel, related to the re-evaluation of Reactivity Insertion Accident following the CABRI REP-Na tests in France, it seems that there is insufficient experimental data currently available to justify the PBMR enthalpy limit for severe reactivity accidents, which could result in the PBMR fuel fragmentation.

Taking the above points into consideration the fuel characteristics and its quality is one of the main factor defining safety characteristics of the PBMR, therefore information concerning fuel requires very serious consideration and the NNR has identified the following issues which need to be addressed:

- The local manufacturing process and associated quality system to confirm the equivalence of locally manufactured fuel with the referenced German one and to ensure that the process will deliver fuel of a quality standard, at least equal to that of the German reference fuel, with a high confidence level.Level of applicability of German fuel results to the PBMR fuel to be substantiated as the conditions in the PBMR core are sufficiently different in terms of power density, power gradients, etc.
- Detailed fuel manufacturing and qualification programme must address qualification of referenced German fuel including reliability and auditability of data used for this qualification.
- The requirements for further test regime must be determined and justified. The relevance of earlier Proof Tests to in-core performance in practice is in need of clarification. It is felt that although Proof Test irradiations can test certain important parameters, it can never be fully representative of in-core conditions, particularly regarding variable mechanical and presumably thermal stresses to which fuel elements are exposed.
- Justification of enthalpy limit for severe reactivity accidents which could result in the PBMR fuel fragmentation.
- Define other fuel design limits, and address fuel system damage mechanisms.

With reference to the above issues raised it is also difficult at this stage to see how the extreme fuel parameters in the PBMR core can be calculated when the dynamics of fuel element flow through the core and the extent of mixing zones have not yet been satisfactorily proven.

## 5. Conclusion

As presented above the PBMR licensing process is currently at the first stage of the process. The NNR has accepted the Safety Case Philosophy as being an acceptable basis to review the Safety Analysis Report, against the NNR licensing requirements, which have been formulated for the PBMR. The NNR is currently undertaking the review of the first SAR. As indicated above although the NNR has not, at this stage carried out an in depth review of the PBMR design and safety analyses, which

will still require a substantial amount of work from both the applicant Eskom and the NNR ( in terms of review), there are a few very important safety issues/concerns, which have been discussed above, which need to be resolved not only for South Africa but also to a certain extent for the international nuclear community as well.

From NNR point of view the following are to be addressed:

1. Completion of the draft document prepared by IAEA Consultancies on "Safety and Licensing Aspects of the Modular HTGR" is an important milestone in the process of establishing and harmonizing international safety standards for advanced HTGR reactors.
2. Containment or confinement issue taking into account heavy aircraft crash.
3. The establishment and documentation of the PBMR design basis including classification of structures, systems and components and General Operating Rules.
4. Licensing Basis Events selection and classification.
5. Reliability of passive systems in particular for the long time response.
6. Probabilistic Safety Assessment (PSA) for advanced reactors with extensive use of passive components and events of very low probability.
7. Cut off criteria for selecting and analysing low probability events.
8. PBMR annual core and core geometry.
9. Computer codes, used in safety analyses, validation and verification.
10. Testing and commissioning programme for the first demonstration module e.g. "Licence by test" philosophy and acceptance criteria.
11. Fuel qualification program including Reactivity Insertion Accident and fuel design limits.
12. Develop and qualify fuel system damage mechanisms.

This preliminary list of safety issues is currently receiving some serious attention in South Africa and new issues might be identified as the licensing process progresses.

## References


[1] National Nuclear Regulator document "LG 1037 Rev 1 – Basic Licensing Requirements for the Pebble Bed Modular Reactor."

[2] National Nuclear Regulator document "LG 1038 Requirements for licensing submissions involving computer codes and evaluation models for safety calculations."

Table 1. **Safety case development framework**

| | a | b | c | d |
|---|---|---|---|---|
| **Sect** | **Safety Case Philosophy** | | **Developmental Documentation** | **SAR** |
| A | **Fundamental Safety Design Philosophy** | **Safety Case Route Map** | Non derived numbers<br>❒ Good Engineering Judgement<br>❒ "Customer parameter file" | Link to safety case |
| B | **Quality Management Programme**<br>❒ Quality Management Philosophy and Approach | | ❒ PBMR QA Policy Manual<br>❒ PBMR QA Procedures & WIs<br>❒ QA of computer codes | QA Programme description and demonstration of adequacy |
| C | **Technical Description & Key Safety Characteristics**<br>❒ High level explanation of how it works<br>❒ Identification of Key Safety Characteristics of design<br>❒ Explanation of how safety design philosophy is followed | | ❒ Incremental development of Technical Package & Supporting Design Documentation | Technical description and demonstration of adequacy of safety design characteristics |
| D | **Identification & Classification of Events**<br>❒ Basis for identification<br>❒ Classification and grouping of events & link to Aa<br>❒ Basis for establishing Initial list of LBEs | | ❒ Procedure for identification & Classification of LBEs<br>❒ Initial List of LBEs.<br>Final List & Classification of LBEs analysed | Identified, fully analysed set of LBEs |
| E | **General Design Criteria**<br>❒ Purpose & Application<br>❒ Establishment of GDCs & Preliminary listing<br>❒ Grouping to provide links to Aa | | ❒ Procedures for development & application of GDCs<br>❒ List of GDCs )<br>❒ Basis for each GDC i.e. why selected | Demonstration of compliance to GDCs |
| F | **SSC Classification**<br>❒ Classification philosophy<br>❒ Process of classification<br>❒ Links to Da, Ea &Ha | | ❒ Definition of classifications<br>❒ Procedures for classifying SSCs<br>Definition of corresponding QA levels, & design requirements (environmental, seismic)<br>❒ Classification Listing | Demonstration of design compliance with classification system |
| G | **Design Rules**<br>❒ Philosophy<br>❒ Design process<br>❒ Scope | | ❒ PIDP procedure<br>❒ Selection of Codes, Standards & Rules<br>❒ SSC loading catalogue | Demonstration of compliance with Design Rules |
| H | **Safety Assessment**<br>❒ Philosophy<br>❒ Principles and Approach<br>❒ Deterministic & probabilistic analysis<br>❒ Qualification, Test & Commissioning<br>❒ Computer codes | | ❒ Justification of analysis techniques<br>❒ List of Assumptions<br>❒ Test & Commissioning Plan<br>❒ Ongoing assessment during procurement, construction and commissioning | PRA<br>Event analysis & demonstration of compliance to licensing criteria. Design evaluation. Commissioning & Test Results |
| I | **Support Programmes**<br>❒ Philosophy<br>❒ Scope<br>❒ Process to establish | | ❒ Rules and basis for establishing Operating Programmes<br>❒ Operational Support Programme procedures | Description of Operating Programmes and their technical bases and link to design & safety bases. |

# SAFETY ISSUES OF REDUCED-MODERATION WATER REACTOR AND HIGH TEMPERATURE GAS-COOLED REACTOR DEVELOPED AT JAERI

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## Abstract

As advanced reactors, reduced-moderation water reactor (RMWR) and high temperature gas-cooled reactor (HTGR) are being developed at the Japan Atomic Energy Research Institute (JAERI). RMWR aims at achievement of a high conversion ratio and negative void reactivity coefficients with MOX fuels, based on well-experienced water-cooled reactor technology. Safety issues important for RMWR are thermal hydraulic safety on the tight-lattice core including stability, highly enriched MOX fuel behaviors at high burn-up, and plant system safety including passive components. Related experiments and analysis have been conducted to obtain information required for the design optimization and safety confirmation. Based on the experiences of design, construction and operation of High Temperature Engineering Test Reactor (HTTR), design study on the Gas Turbine High Temperature Reactor 300 (GTHTR 300) has been conducted, with greatly simplified system, including safety systems due to its inherent safety. As major safety issues, severe accident free and demonstrable safety are proposed. This new safety philosophy is to avoid most accidents and to greatly reduce the probability of severe accidents compared with that of current power reactors.

## 1. Introduction

Advanced reactors are being developed at the Japan Atomic Energy Research Institute (JAERI) in order to meet developmental design objectives, such as enhanced safety, long-term energy security, economical competitiveness, minimization of global environmental load, diversification of energy utilization and proliferation resistance. Currently major efforts are devoted to the development of reduced-moderation water reactor (RMWR) and high temperature gas-cooled reactor (HTGR).

RMWR aims at realization of the long-term energy supply with uranium resources, the multiple recycling of plutonium or the high burn-up / long operation cycle achievement, with MOX fuels based on the well-experienced water-cooled reactor technology. To achieve these objectives, the most important design target is a high conversion ratio over 1.0. Such a high conversion ratio can be attained by reducing the moderation of neutrons, i.e. reducing the water fraction in the core with a tight-lattice fuel rod arrangement. The negative void reactivity coefficient is another important design target. At present, several types of basic core concepts satisfying both the main design targets mentioned above have been proposed under both BWR and PWR type concepts. This research activity has been performed in cooperation with the Japanese utilities and LWR vendors.

High Temperature Gas-cooled Reactor (HTGR) has unique safety and high-temperature characteristics, such as large thermal capacity of the core, low power density and negative temperature reactivity coefficient. HTGR shall then be expected to be a safe and economic energy source in 2010s due to inherent safety features and the potential capability of producing high temperature helium gas. In parallel with a power-up test of the High Temperature Engineering Test Reactor (HTTR), JAERI has started design studies on future HTGRs and a new safety philosophy for these reactors. The new safety and acceptance criteria using inherent safety characteristics, such as criteria for coated fuel particle, primary system, passive safety system, and graphite components, are proposed in order to realize severe accident free concept.

In the present paper, major safety issues for both RMWR and HTGR being developed at JAERI are discussed, and the current experimental and analytical efforts to overcome the challenges are described.

## 2. Development of advanced reactors at JAERI

The Long-term Program for Research, Development and Utilization of Nuclear Energy [1], issued by Atomic Energy Commission of Japan on November 24, 2000, described that advanced nuclear reactors with high economic efficiency and safety, and with applicability to diversified energy demands and extended nuclear reactor use are expected as well as next generation Light Water Reactors (LWRs), and research and development on advanced nuclear reactors should be performed under the collaborative scheme of national laboratories, industries and universities.

JAERI proposed the following goals for the advanced nuclear reactor systems to materialize the advanced nuclear reactor concepts specified in the Long-term Program:

1. Enhanced safety: eliminate the need for offsite evacuation by reducing the possibility of severe accident by adopting new technologies such as passive safety equipments.
2. Sustainability: ensure the long-term energy supply through effective fuel utilization by developing high conversion cores.
3. Economical competitiveness: reduce life-cycle cost and financial risk by developing several technologies, such as simplified systems, modular design, and high thermal efficiency.
4. Minimisation of global environmental load: reduce the amount of radioactive waste by recycling fuels and/or transmute long-lived fission products and minor actinides.
5. Diversification of energy utilization: diversify nuclear energy utilization to non-electricity fields such as civilian heat supply, desalination and hydrogen production.
6. Proliferation resistance: prevent diversion of nuclear materials by design barriers as well as institutional barriers through surveillance and inspection by IAEA.

In order to meet these goals, JAERI proposed the concepts of Reduced-Moderation Water Reactor (RMWR) [2] and High Temperature Gas-cooled Reactor (HTGR) [3,4] for near term deployment (2010 to 2030). The safety and economical competitiveness are the most preferential goals for all concepts, while preserving the features of proliferation resistance. In addition to these common goals, the RMWR has the specific aim of sustainability, while the HTGR are specifically for the diversification of energy utilization.

### *2.1 Reduced-moderation water reactor (RMWR)*

The main design goals for the RMWRs are the following two points as already mentioned above:

- High conversion ratio more than 1.0.
- Negative void reactivity coefficient.

The former is indispensable for the long-term energy supply with uranium resources under the multiple recycling of Pu, and is also important for the high burn-up and long operation cycle achievement due to the smaller burn-up reactivity. The latter is common safety characteristic in the current LWRs and is considered to be also required for the RMWRs, because the RMWRs exist on the extension of the experienced LWRs technology following the same safety philosophy.

In order to achieve the high conversion ratio, the volume of the moderator, *i.e.* water, should be reduced. For this purpose, the tight-lattice fuel rod arrangement is commonly adopted. The triangular lattice with a narrow gap between the fuel rods and/or the rods with a large diameter is typical for it. Installation of some special rods for water removal might be another idea. Especially for the BWR-type reactor design, increase in the core void fraction is another realistic technique to be used.

To satisfy the requirement of the negative void reactivity coefficient, neutron leakage should be increased when the void is generated or increased in the core. The short core design is common technique for it. The blanket region could be adequately used to increase neutron absorption. Some streaming mechanism might be also used to promote neutron leakage effect.

The above two design goals are attained by appropriately combining the basic techniques described above and by keeping a balance among them. Conceptual designing of the RMWR core is based on these general basic ideas as described in the following.

In our design study, some different core designs have been investigated based on the different basic ideas described above. That is, there are some design possibility in adopting and combining the above basic ideas. At present, we have investigated three different types of core design under the BWR-type reactor concept [5], and two types for the PWR-type [6]. In this section, only one design out of them is presented in the following to give the general idea on the RMWR with some major core characteristic information.

The present core aims at as high conversion ratio as possible with 1 000 MWe class power output. However, an attainable value was expected around 1.1 at most based on previous research information. In order to achieve a very high conversion ratio, the core consists of hexagonal fuel assemblies with triangular tight-lattice configuration as shown in Figure 1. Y-shaped control rods with the follower structure are introduced as shown in the figure at the ratio of one unit for three fuel assemblies.

The pitch of the assembly is about 220 mm. The diameter of the fuel rods is 14.5mm and the gap between rods is kept at 1.3 mm, which is considered to be a tentative limit from the zircaloy cladding structure and the heat removal points of view. The core average void fraction is significantly increased to 70% in this design. Resultantly, the effective volume ratio of the water to the fuel is reduced extremely to about 0.17 in the present design, trying to attain a high conversion ratio.

To obtain negative void reactivity coefficients, the core or seed is extremely shortened to about 200 mm high and two core parts are piled up with an internal blanket region. Adding the upper and lower blanket regions, the total core region has the five-layer structure in the axial direction as shown in Figure 2.

Major dimensions and characteristics of the core determined by the calculations are summarized in Table 1.

Figure 1. **Schematic of fuel assembly**

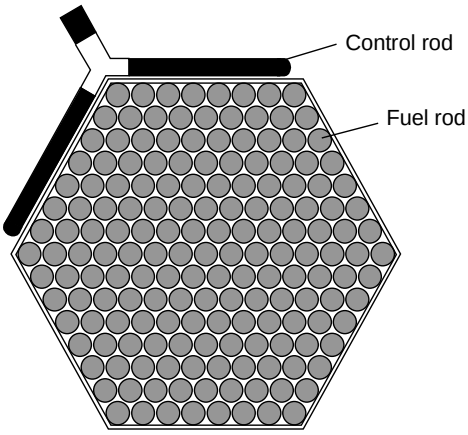



Figure 2. **Vertical cross section of core**

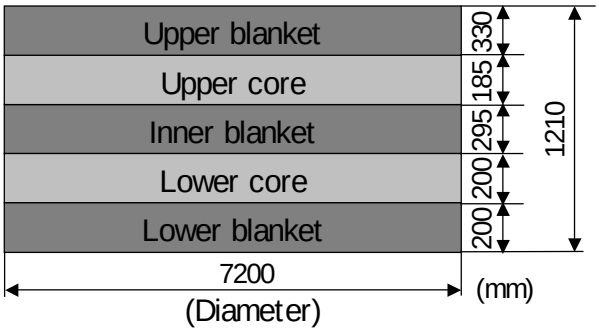



Table 1. **Major dimensions and characteristics of RMWR core**

| Item | Design value |
|---|---|
| Electric power output (MWe) | 1 100 |
| System pressure (MPa) | 7.2 |
| Number of fuel assembly | 924 |
| Discharge burn-up for core part (GWd/t) | 45 |
| Core mass flow rate ($10^4$t/h) | 1.3 |
| Core exit quality (%) | 55 |
| Average fissile Pu content in MOX (wt%) | 18 |
| Conversion ratio | 1.10 |
| MCPR | 1.3 |
| Void reactivity coefficient ($10^{-4}$dk/k%void) | -1 |
| Operation cycle length (EFPM) | 14 |

### *2.2 High-temperature gas-cooled reactor (HTGR)*

Since HTGRs use ceramic coated particle fuels and graphite core structures instead of metal cladding fuels, the reactor core of HTGRs can endure the high temperature conditions during abnormal events including accidents. The transient during abnormal events is slow because of its large heat capacity of reactor core and of its relatively small power density due to graphite moderated reactor. These unique safety characteristics of HTGRs allow simplification of safety functions. For example, the residual heat of reactor core can be removed safely only by passive cooling systems. This point leads a large cost benefit for construction and operation of the plant. On the other hand, high temperature helium gas of HTGRs can be applied diverse energy utilization, such as high efficiency electric power generation using gas turbine system and chemical process for hydrogen production.

JAERI has started design study of gas turbine high-temperature reactor system with electric power of 300 MW(GTHTR300) which has high thermal efficiency of approximately 50%. Plant layout and plant cycle are shown in Figs. 3 and 4, respectively. The reactor core has an annular shape in order to obtain appropriate temperature profile during abnormal conditions. Fuel of the reactor is pin-in-block type which is well developed as a core fuel of the HTTR. Thermal power of the reactor is 600 MW. Turbine system is driven by reactor coolant helium gas of about 850°C and consists of a helium gas turbine, a compressor and an electric generator. Two heat exchangers, that is, a recuperator and a precooler, are installed in this system to obtain high thermal efficiency. A reactor pressure vessel, power conversion vessel and heat exchanger vessel are connected by crossducts for high and low temperature helium gas flows. Major specifications of the GTHTR300 are listed in Table 2.

Table 2. **Specification of the GTHTR300**

| | |
|---|---|
| **Fuel** | |
| - Fuel | LEU(less than 20%) |
| - Fuel Particle | SiC coating |
| **Reactor** | |
| - Reactor Type | Helium coolant/Graphite moderator/ Thermal neutron reactor |
| - Core Type | Pin-in-block and annular type |
| - Thermal Power | 600MW |
| - Power Density | 6MW/m$^3$ |
| - Fuel Burnup | 120GWD/ton |
| - Refueling | Once in 2 years |
| - Coolant | He gas |
| - Core Inlet/Outlet Temperature | 560/850°C |
| - Core Outlet Pressure | 6.83MPa |
| **Electric Power** | 273MW |
| **Net Plant Efficiency** | 45.5% |

Figure 3. **Plant layout of GTHTR 300**

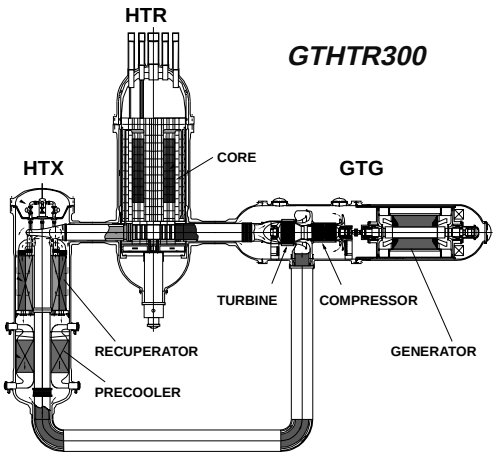
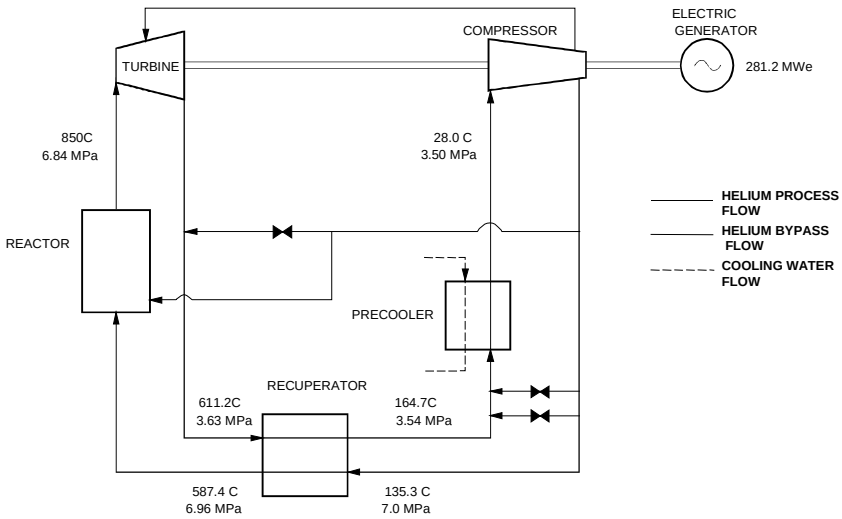



Figure 4. **GTHTR 300 cycle calculation**



## 3. Safety issues of advanced reactors

### *3.1 Reduced-moderation water reactor (RMWR)*

Safety issues important for RMWRs are (1) the thermal hydraulic safety on the tight-lattice core for both the normal operation conditions and the abnormal transients and accidents, including stability, (2) the highly enriched MOX fuel behaviors at high burn-up, and (3) the plant system safety including passive feature.

#### *3.1.1 Thermal hydraulic safety*

RMWR aims at achievement of the high conversion ratio over 1.0 and negative void reactivity coefficient with MOX fuel. As mentioned above, several types of basic concepts have been proposed under both BWR and PWR type concepts. In these concepts, a tight-lattice core is adopted to reduce the water fraction in the core. In general, the tight lattice core causes the critical heat flux to be decreased and the pressure drop to be increased at the same mass velocity. A heterogeneous core arrangement with blanket assemblies and inner blankets is also adopted to attain negative void reactivity coefficient. The heterogeneous core arrangement results in higher local power peaking in core. The core characteristics are different from those of LWRs due to different neutron energy spectrum. The change in the core response function may affect the flow stability in core.

For the thermal hydraulic safety of RMWRs, there are several issues required to be solved, such as critical heat flux in tight-lattice core, core cooling during abnormal transients and accidents, and flow stability during operation.

1. Critical heat flux in tight-lattice core

The critical heat flux (CHF) is a major concern in the thermal design of RMWRs. The fuel rod gap sizes between 1.0 and 1.3 mm in a triangular arrangement are proposed in the current designs of RMWRs. Low mass flow rate at about 500 kg/m$^2$s and short core are also proposed to attain high core void fraction and low pressure drop through the core. A stepwise axial power profile resulting from the inner blankets needs to be considered for the feasibility of particular designs. Because these thermal hydraulic features are quite different from those of conventional light water reactors, it is very important to obtain CHF test data for the thermal design of RMWRs.

CHF tests were performed using a 7-rod test section with gap width of 0.6, 1.0, and 1.5 mm under typical PWR operation conditions, or with gap width of 1.0 mm under typical BWR operation conditions in order to check the CHF correlations used in the core design calculation. Figure 5 shows an example of comparisons between measured and calculated results. The calculated CHF with the design correlation is lower than measured. The test results showed that the CHF correlation used in the design calculation predicted lower CHF than measured under typical BWR and PWR operational conditions. Because these tests were performed using chopped cosine or flat power profiles, it is necessary to perform a CHF test with stepwise axial power profile simulating the inner blanket. In addition, full-scale mockup tests (Figure 6) are planned for safety demonstration including transient tests simulating the abnormal transients during the operation.

Figure 5. **Comparison of CHF between measured and calculated results**

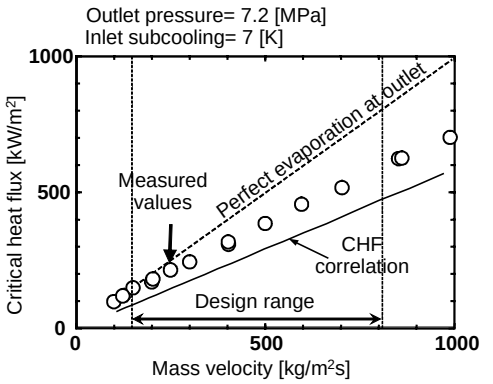
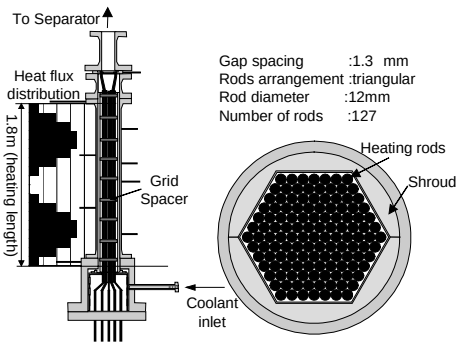



Figure 6. **Schematics of test section for for full-scale mockup test**




2. Core cooling during abnormal transients and accidents

In RMWR design works, it is tentatively intended to use the reactor system of the existing plants as much as possible. If the same system is adopted, the previous experiences can be used for RMWRs as well as the existing plants. It is expected that current safety evaluation methods of abnormal transients and accidents for PWR and BWR can be extended to RMWRs by modifying several thermal-hydraulic correlations specific for RMWRs. The major potential correlations need to be assessed are core void fraction, two-phase pressure drop through core, cross flow in tight-lattice core, and tie-plate counter-current flow limiting. It is necessary to establish test data for the assessment of such correlations.

The thermal-hydraulic feasibility studies are required for core cooling performance in RMWRs during abnormal transients and accidents because of the specific characteristics, such as tight-lattice core, blanket fuel assembly, inner blanket, low core flow rate, high void fraction, and lower void reactivity coefficient. The thermal-hydraulic analyses have been performed for both BWR and PWR concepts to ensure the feasibility of RMWRs. These calculations showed that RMWR cores are feasible against typical design basis accidents, including large-break loss-of-coolant accidents (LBLOCA) in PWRs, pump seizure accidents in BWRs. Figure 7 shows an example of feasibility studies against LBLOCA in a PWR-type RWMR. The calculated peak clad temperature is about 1200 K and much lower than the design criteria (1473K).

Figure 7. **Maximum clad temperature during LBLOCA in PWR-type RMWR**

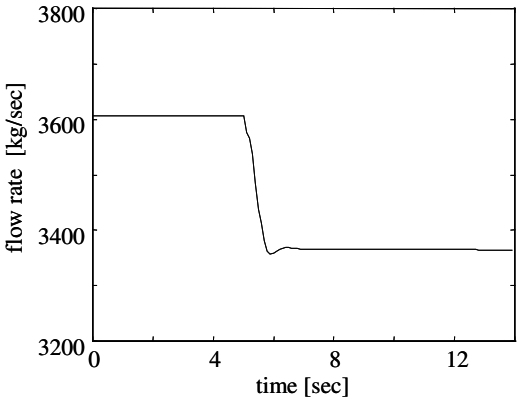
3. Flow stability during operation

The stability is one of major concerns for the safe operation of BWR-type RMWRs, as in conventional BWRs. The oscillatory response of flow and neutron flux in the core should be avoided to ensure the fuel integrity during operation. The stabilities related to the reactor core are generally classified into the following categories; 1) channel hydraulic stability, caused by the hydraulic feedback, 2) core reactivity stability, and 3) core regional stability. The core reactivity stability and core regional stability are induced by the coupled response of core thermal-hydraulics with neutron kinetics.

As previously mentioned, BWR-type RMWRs have specific features, such as tight-lattice core, low flow rate, high void fraction and low void reactivity coefficient. The core height is short compared with current BWRs. In general, the channel stability is influenced by the pressure drop through the core and transit time of coolant through a channel. In RMWRs, high void fraction increases the contribution of the two-phase pressure drop, while the short core makes the transit time short. Preliminary analysis results of RMWR channel flow response against the pressure disturbance under rated operation condition are shown in Figure 8. This indicates the sufficient channel stability.

The preliminary analysis results also showed stable response for core reactivity stability due to the small absolute value of the void reactivity coefficient for the RMWR core. An investigation on the core regional stability is under way.

Figure 8. **Channel flow response**




### *3.1.2 Highly enriched MOX fuel behavior at high burn-up*

MOX fuels of RMWR contain Pu of more than 30% and are irradiated to high burn-up of 100GWd/tHM or over. These tough irradiation conditions make it an essential task to evaluate the mechanical and thermal feasibilities of the fuel. Therefore, the present study conducted a safety evaluation analysis of the fuel behavior using a fuel performance code FEMAXI-RM. The code FEMAXI-RM, which is an advanced version of FEMAXI-V [7, 8], has been developed to cope with the analysis of the RMWR fuel rod with such features as combined structure of MOX and blanket parts.

The present analyses were conducted with a singe rod which is assumed to have the highest power in the RMWR core. The models or materials properties applied to the analysis, such as fuel thermal conductivity, FP gas diffusion and release, and creep rate are derived or extrapolated from those used in the analysis of LWR fuel rods. Particular focus was imposed on the thermal behaviors such as FP gas release and internal pressure increase. They are induced by the fuel temperature rise.

Figure 9 shows the FP gas release rates (FGRs) from MOX fuel. A sharp increase of FGR in the MOX fuel appears around 50GWd, which is induced by the fuel temperature rise. However, after that, the FGR is leveled off due to the gradual decrease of the power. FGR in the blanket part is substantially null.

Figure 9. **FGRs from MOX fuel with the highest power and blanket part with the lowest power**

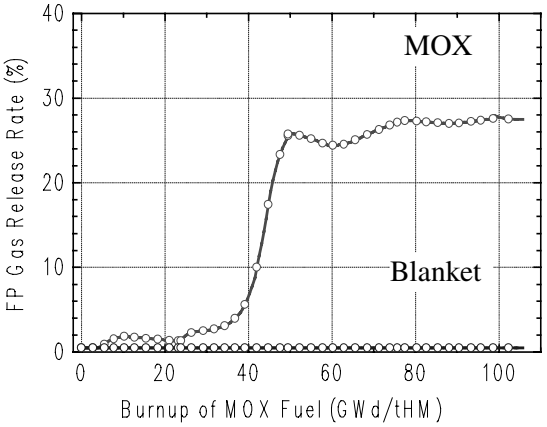
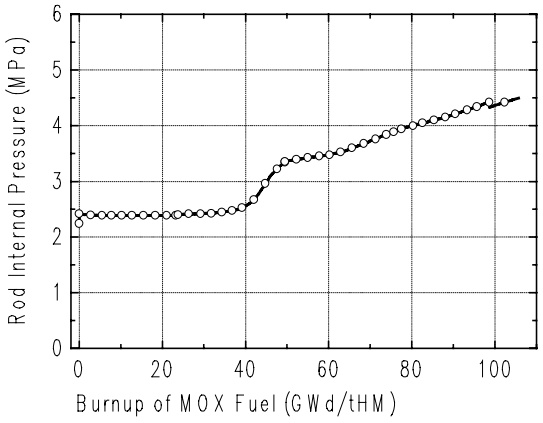



Figure 10 shows the internal pressure rise which is essentially generated by the FP gas release. The pressure at EOL is less than 5MPa, not exceeding the coolant pressure of 7.2MPa. This predicts that the cladding will never cause “Lift-off” even at the very high burn-up.

Figure 10. **Rod internal pressure rise due to the FP gas release and decrease of inner free volume**



The above analytical results suggest that the MOX fuel rod has no particular thermal behaviors that will raise safety and reliability concerns. However, behaviors of the MOX fuel with such a high Pu content have been neither fully understood nor foreseen in very high burn-up region. Therefore, a precise characterization of input data and materials properties models is inevitable to evaluate the fuel safety and reliability on the basis of code predictions.

In addition, modeling of the thermal conductivity degradation with burn-up extension of pellet, swelling by the FP gas pores which are generated around Pu-rich spots, and fuel rod deformation behavior are main issues to be considered in the code analysis. For these issues, irradiation experiment of the MOX fuel is of vital importance.

### *3.1.3 Reactor systems safety including passive feature*

From the use of tight lattice core geometry and its difficulty for the cooling, it is preferred that the plant system does not allow insufficient core cooling conditions, especially when levels of the heat generation and stored energy in the core are relatively high. Since those insufficient cooling conditions occur due to rapid loss of core-flow or coolant, the use of natural circulation cooling and the elimination of large diameter pipe for liquid-water flows are suitable for the RMWR systems.

The development of small advanced reactor systems, as well as the utilization of the RMWR concept for large reactor systems, is also considered, such as an integral type reactor with steam generators contained in the reactor vessel and passive safety systems. Depending on the selections of components, such as type of steam generators, final heat sink, safety injection systems, and a primary depressurization system, various integral reactor systems can be proposed, e.g., MRX [9], and MR-100G [10] by JAERI, IMR by Mitsubishi [11], and SSBWR by Hitachi [12]. One example of the integral reactor systems is illustrated in Figure 11. To maintain safety during accidents in this reactor system, the reactor vessel is cooled by the steam generators, which is cooled by the combined use of the steam generator automatic depressurization system (ADS) and the gravity driven injection system (SGDIS) for loss-of-coolant accident (LOCA) conditions, and the natural circulation decay heat removal (DHR) system for non-LOCA conditions. The passive containment cooling system (PCCS) is used to remove the decay heat to water pool located above the containment. For the mitigation of severe accidents, the outer surface of the reactor vessel is cooled by the gravity-driven injection for the in-vessel retention of the molten core (GDI-IVR). Since a large break LOCA is impossible to occur for this system, the pressure suppression pool is unnecessary, which may significantly reduce the containment volume and thus improve the plant economy.

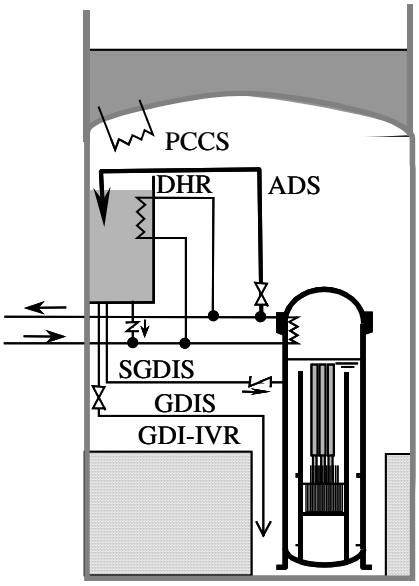
Since the commercial plants are constructed and used by the private industries, JAERI focuses on the development of key technologies to safely use the tight-lattice core geometry, simplify the reactor systems, characterize the natural circulation stability, and predict the performance of safety systems and steam generator. To efficiently develop those technologies, the existing resources used for the safety research at JAERI will be utilized. For example, the ROSA/LSTF facility will be used to obtain the information useful for the safety confirmation and design optimization.

## *3.2 High-temperature gas-cooled reactor (HTGR)*

Although the HTGR has unique safety characteristics, adequately developed criteria do not yet exist in a number of important areas. Even in the design of the HTTR, basic safety principle is quite identical to that of current LWR. Therefore, safety advantage of the HTGR is not well incorporated. The new criteria will have to be developed in order to utilize these characteristics. Key

criteria necessary for the future HTGRs have been identified in the areas, such as coated fuel particle, primary system, passive safety system, and graphite components. Also, safety and acceptance criteria, and various event categories used in the safety evaluation shall be defined considering recent trends for enhancing nuclear safety.

Figure 11. **One example of conceptual designs for RMWR : GDIS for gravity-driven injection system, DHR for decay heat removal, GDI-IVR for gravity-driven injection for in-vessel retention**




The HTTR was successfully constructed, and a power up test including commissioning tests has been carried out since September 1999. It reached the full power in the end of 2001. After having the operational licensing, the safety demonstration tests will be started. These tests are the full-scale simulation of accidents such as reactivity insertion and loss of forced cooling. The important role of these tests is the demonstration of the HTGR safety concept.

### *3.2.1 Concept of new safety philosophy for HTGRs*

1. Defence-in-depth and severe accident free

Defence-in-depth is a basic philosophy for the HTGR as well as LWR. Various layers of requirement are used to ensure safety. However, there are major difference in philosophy between HTGR and LW. The LWR uses highly reliable, redundant and diverse passive or active safety layers. On the other hand, the HTGR safety shall be assured due to inherent safety characteristics and potentially safe components and system, such as coated fuel particle and a decay heat removal system.

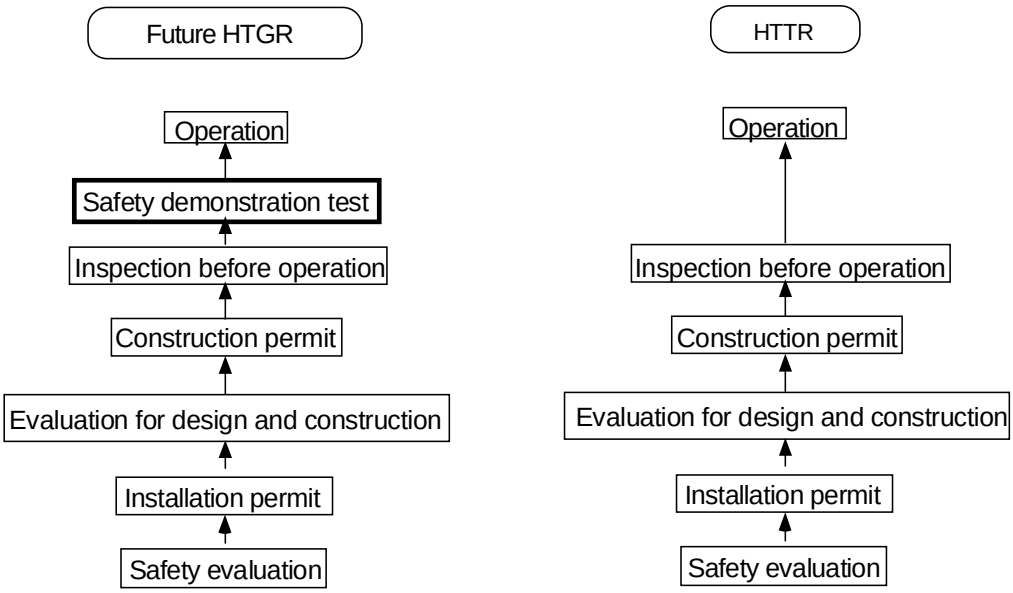
Severe accidents, defined as any conditions beyond design-base accidents, causing core damages with fission product releases to the environment, are very low in probability. The new safety philosophy is to avoid most accidents, and to achieve a probability of severe accidents at least two

orders of magnitude lower than current reactors. Even in the worst event, fuel temperature exceeding its failure limit and excessive fuel oxidation by air ingress can be avoided because of inherent safety features and the passive decay heat removal system.

2. Demonstrable safety

Nearly full-scale worst accident simulation tests can be carried out to obtain licensing before commercial operations because safety assessment by analysis may not enough to convince the public and the regulators. Figure 12 shows the difference between the licensing procedure of the HTTR or current reactors and future HTGRs. In current reactors no accident simulation tests are carried out before commercial operations although inspection and performance tests in normal condition are conducted. On the other hand, safety demonstration by accident simulation tests can, and shall be, a requisite to obtain licensing in the future HTGRs.

Figure 12. **Licensing procedures for future HTGR and HTTR**




3. Mechanistic source term

Mechanistic source term is used to estimate radionuclide releases for plant siting evaluation instead of non-mechanistic source term based on the current safety evaluation guideline for Japanese LWRs or HTTR.

Initial failure of coated fuel particles in the HTTR in manufacturing is only $8\times10^{-5}$ although $10^{-2}$ in design, and no apparent failure was found in continuous irradiation tests up to 6.5% FIMA at Japan Material Test Reactor(JMTR). A post irradiation test is also planned to be carried out in 2003. In addition to these tests, data for long-term integrity of the fuel, lift off and plate out behavior will be accumulated in HTTR operation. These are the reason why the mechanistic source term can be used for the plant siting evaluation.

4. No containment vessel

When the above mentioned mechanistic source term is used for the plant siting evaluation, the effective dose equivalent to whole body in the worst event can meet the dose guideline without the containment vessel. No containment vessel is necessary in future HTGRs due to salient fuel performance, and resuspension and transport phenomena pertaining to the radionuclide within the primary circuit.

5. No need for offsite emergency evacuation and no damage on all offsite assets

Offsite emergency evacuation is not necessary in the worst event selected for the plant siting evaluation. Furthermore, all offsite assets are kept intact and ensured.

6. PSA and event selections for the safety evaluation

The categorisation of the events to be evaluated is the followings. That is in confirming the adequacy of basic design principles of reactor facilities, one must evaluate abnormal conditions "Anticipated operational occurrences (AOO)" and the events beyond AOO "Design basis accidents (DBA)". In addition to these traditional categories, the postulated severest event shall be evaluated among all beyond design basis accident (BDBA). An event with complete loss of forced coolant (depressurization accident) and simultaneous withdrawal of all control rods is selected among all very unlikely events such as anticipated transients without scram (ATWS), station blackout and multiple operator errors. The same worst event is used for siting evaluation. The frequency range for AOO, DBA and BDBA is approximately $10^{-2}$, $10^{-4}$ and more than $10^{-7}$ per year, respectively. The PSA method and engineering judgment is necessary to select these events. For example, Figures 13 and 14 show the fuel temperature transient during loss of forced cooling accident and control rod withdrawal accident in 50MW severe accident free HTR, respectively [3]. The fuel temperature increased the highest in these accidents.

Figure 13. **Maximum fuel temperature during depressurization accident**

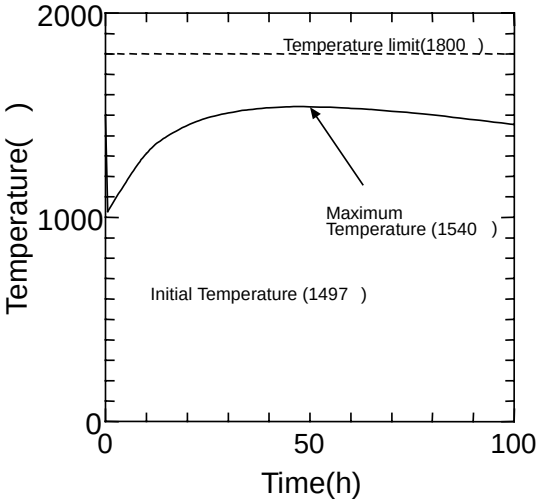
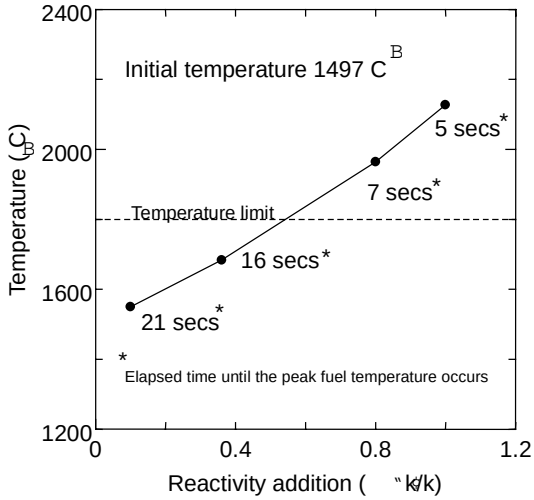



Figure 14. **Maximum fuel temperature during reactivity insertion**



### *3.2.2 Criteria for HTGRs*

Safety design criteria and acceptance criteria proposed here have been determined based on IAEA safety requirements documents [13, 14] and HTTR experiences.

1. Safety design criteria

*a) Coated fuel particle*

The design of fuel elements and assemblies shall be such that they will satisfactorily withstand the intended irradiation and environmental exposure in the reactor core despite all processes of deterioration that can occur under normal operation and in AOO. This means the peak fuel temperature shall not exceed 1 495°C and 1 600°C in normal operation and AOO, respectively. It is also required that "Allowance shall be made for uncertainties in data, calculations and fabrication". In addition, the following effects are considered to design the coated fuel particle; temperature, fission product, irradiation, power variation, mechanical load, oxidation, and thermal hydraulics.

*b) Core graphite structure*

The reactor core is composed of prismatic blocks in the block type HTGR and supported by graphite support structures. The core assembly is provided with a restraint structure which maintains the external configurations. The safety issues with this core graphite structure are as follows.

Entry of the shutdown devices into the core and maintaining the shutdown condition should not be impeded. In addition, the following effects are considered to design the core graphite structure; temperature, oxidation, irradiation, dimensional change and its effect to coolant flow, earthquake, impurity, and deterioration in life time.

*c) Primary components*

The metallic components directly in contact with hot coolant gas shall be designed so as to maintain its function on demand during an entire life of service. To meet this philosophy, the component material shall be selected to keep sufficient mechanical strength, anti-corrosion and stability against the elevated temperatures by taking into account the effect of irradiation. Also the components shall be designed to meet the revised "High Temperature Metallic Component Design Criteria" which was developed for HTTR components.

*d) Passive decay heat removal system*

Passive decay heat removal system indirectly cools the core and keeps the fuel temperature lower that the temperature limit in any loss of forced cooling condition. This system is categorized as a safety system. However, no active and direct core cooling is necessary.

2. Acceptance criteria

*a) Coated fuel particle*

- The peak fuel temperature shall not exceed 1 495°C and 1 600°C in normal operation and AOO.

- The fuel rod and fuel assembly shall not fail in normal operation and AOO.
- The core shall not be seriously damaged and the coolable geometry of the core shall be maintained in DBA and BDBA.

*b) Core graphite structure*

- Core graphite structure shall not be oxidized or deformed so that the core integrity can be kept safely in AOO, DBA and BDBA.

*c) Primary component*

- The pressure of the primary circuit in AOO and DBA shall be less than 1.1 times and 1.2 times of the maximum usage pressure, respectively.
- The peak temperature of the primary circuit shall not exceed the limit temperature in AOO and DBA.

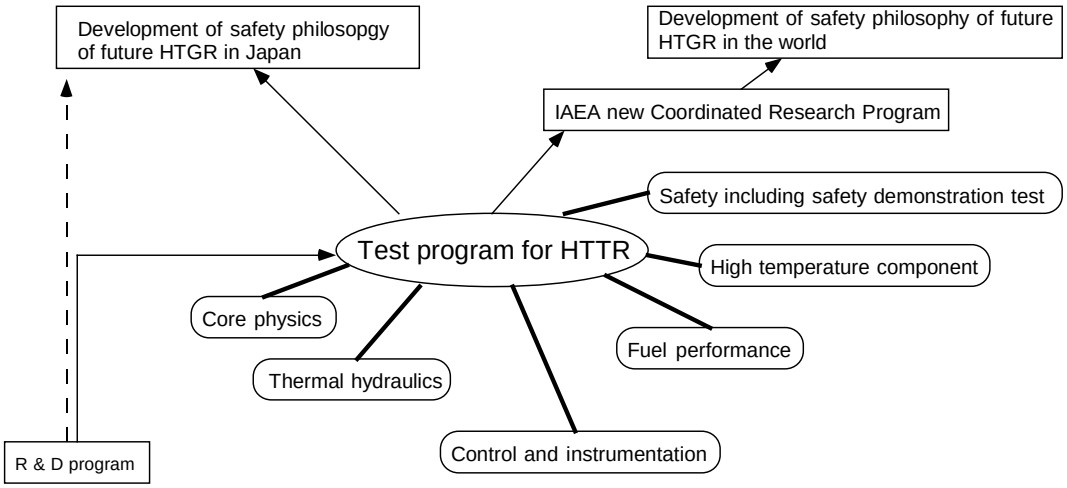
*d) Passive decay heat removal system*

- Passive decay heat removal system keeps the fuel temperature lower than the temperature limit in AOO, DBA, and BDBA.

### *3.2.3 Demonstration of safety concept*

A test program of the HTTR has been established to contribute to the design and development status of the future HTGRS. Figure 15 shows the test program of the HTTR. The test program is divided into six categories and all programs are related to the technology of the future HTGRs. In this section, the test program related to safety is described. In the accident simulation tests using the HTTR, the postulated worst two events in the future HTGRs will be simulated. They are a loss of forced cooling accident (depressurization accident) and control rod withdrawal accident. The accident simulation tests will start in 2002 in order to confirm the safety concept of the other future HTGRs.

Figure 15. **Relationship between HTTR test program and safety philosophy development**



1. Loss of forced cooling accident simulation

This test simulates conduction cooldown behavior during the depressurization accident. In this test, circulators for the primary cooling system stop at a rated power, and no active direct core cooling system is used after that. During the depressurization accident in the future HTGRs, natural circulation occurs in the RPV and transfers heat from the core to outside. On the other hand, this test cannot explicitly simulate the actual natural circulation because a guillotine break of the primary pipe cannot be simulated. However, the flow rate of the natural circulation is low in low pressure condition and heat transfer by the natural circulation is limited during the depressurization accident. Also, natural circulation removes heat from the core. When the fuel temperature is evaluated, the neglect of natural circulation makes fuel temperature higher than that with natural circulation. Therefore, this simulation test makes sense to evaluate fuel integrity.

2. Control rod withdrawal accident simulation

This test simulates the fuel temperature rise by reactivity addition. In a preliminary test, a pair of control rods is withdrawn from the core by a control drive mechanism. In a secondary test, a capsule containing a reactivity absorption material such as $B_4C$ ball is installed in the center of the core and immediately ejected from the core. The maximum fuel temperatures during both simulation tests are measured by the temperature monitoring elements installed in the fuel blocks.

## 4. Concluding remarks

Major safety issues identified to be important for RMWR are thermal hydraulic safety on the tight-lattice core including stability, highly enriched MOX fuel behaviors at high burn-up, and plant system safety including passive components. Several related experiments and analysis have been conducted and planned to obtain information required for the design optimization and safety confirmation.

The design study of the GTHTR300 is now developed and new safety philosophy is applied to this study. This philosophy based on new safety and acceptance criteria using inherent safety characteristics of HTGR can simplify a safety system and shut down system, and eliminate a containment vessel in HTGRs. This new safety philosophy will be proved by the safety demonstration tests in the HTTR.

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# INTRODUCTION TO CEA STUDIES ON FUTURE NUCLEAR ENERGY SYSTEMS AND SAFETY RELATED RESEARCH

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The growth in world energy demand and future prospects of decommissioning of existing power plants stimulate the development of a new generation of nuclear energy systems intended to be more economical, safer and more sustainable. CEA clearly support the idea that nuclear energy has unique assets to be eligible as one of the major energy sources of the sustainable development in terms of natural resources saving, economic competitiveness and public and environmental friendliness, but the possibility of development of nuclear energy is strongly linked with the public acceptance.

Public acceptance of nuclear energy depends strongly on safety, and the public's perception of safety.

Consequently, a major emphasis for future generations nuclear systems is to improve the safety and the robustness of the technology, and thus the public confidence in safety.

## 1. Goals for future nuclear energy systems

Nuclear energy has unique assets to meet the requirements for a sustainable development in terms of economic competitiveness, safety, environmental friendliness and natural resources saving.

This is why future nuclear system studies conducted by the CEA aim at investigating and developing promising technologies for the medium and the long term, for reactors, fuels and the fuel cycle able to cope with these requirements.

Significant progress towards these objectives and also for opening new applications to nuclear power, call for breakthroughs beyond light water reactors, towards hardened neutronic spectra and high temperatures. Moreover, spent fuel processing and recycling of nuclear materials are recognized as essential technologies for minimizing long-lived radioactive waste and for making an efficient utilization of all available fissile and fertile nuclear fuels.

For the sake of consistency of the reflection on future nuclear energy systems, the reactor, the fuel and the associated fuel cycle are seen as integral parts of a nuclear system to be optimised globally.

This effort also aims at maintaining at the best possible level the expertise and the technologies that the CEA will be able to bring to future national and international projects likely to meet market needs in the next decades.

To tackle these challenges, the **CEA objectives for future nuclear systems** are the following:

- **resource saving** through a breakthrough towards high temperatures (high efficiency), spent fuel processing and recycling for minimizing long-lived radioactive waste and an efficient utilization of all fissile and fertile fuel; this last issue by two ways:
  - Near term: Advanced Fuel (CORAIL and APA).
  - Long term: FBR.
- reinforced **economic competitiveness** against other available energy generation means, with a special emphasis put on reducing the investment cost and implementing **potentialities for other applications than electricity production**,

and, concerning the public and environmental friendliness:

- **enhancing the system "safety"**, through:
  - the minimization of radioactive waste production and release both during normal operation or abnormal occurrences as well as an enhanced resistance to proliferation risks;
  - the excellence in daily system operational safety and reliability, preventing and controlling abnormal situations;
  - the improvement of the system capacity for abnormal conditions management and an increased resistance toward (i.e. the prevention of) the plausible severe plant conditions;
  - the improvement of the severe plant conditions management;
  - the elimination of any technical justification for offsite emergency response.

**From safety approach point of view, these safety related objectives lead to practically improve the implementation of the defense in depth**, and this for all the nuclear system including both the reactor and the fuel cycle installations.

We will briefly comment the French vision of the safety goals for future nuclear energy systems, as well as that of the goals contributing to improve the public acceptance. The conclusion will give an insight into the R&D program currently implemented by the CEA to investigate promising technologies, with national and international partners, to meet these goals.

## 2. Safety goals for future nuclear energy systems

### *2.1 The safety requirements for future nuclear systems*

The safety approach should be founded on three essential principles :

- the safety objectives – in terms of radiological consequences – for the different types nuclear plants should be the same;
- the implementation of an adequate safety related architecture and the organization of the safety assessment should be achieved through the integral adoption of the defense in depth principle;
- this approach should be able to integrate the peculiar characteristics of each installation (front and back end cycle, reactor).

### *2.2 The implementation of the defense in depth principle*

As stressed above, the defense in depth concept and its application, remains the basis for the future systems design process as well as for the definition of the operation conditions. The system safety assessment checks its appropriateness.

Following INSAG10 (Defense in Depth in Nuclear safety): "… the defense in depth approach consists of a hierarchical deployment of different echelons of equipment and procedures in order to maintain effectiveness of the physical barriers placed between the radioactive materials and the workers, the public or the environment during normal operation, anticipated operational occurrences and, for some of them, during accidents in the plant…

A consistent implementation needs to consider the accident response characteristics of the plant… These characteristics influence the required number and strength of adequate lines of defense." (LOD), i.e. **the final safety relies on the architecture of the plant**.

This notion of "**lines of defense**" allows to homogenize the approach aims at guaranteeing:

- A **progressive defense** to avoid "short" sequences that can lead to important releases. The objective is **a design with inherent characteristics and automatic implementation of the safety systems** considering the possibility for intermediate operator intervention in order to restore the sequence management. This is coherent with the INSAG 10 recommendations: "The approach for further improvement of defense in depth … includes: improving accident prevention, in particular by optimizing the balance between the measures taken at different levels of defense in depth and by increasing their independence…"
- A **homogeneous defense** to avoid great differences among the contributions of the different event families, and/or the plant status, to the "severe plant condition" frequency.
- A **comprehensive** (extended) **defense. From the very preliminary** design, an effort must be made to identify and to prevent all the plausible initiating events that can lead to the plant degradation. On the other hand the demonstration that the abnormal situations as well as the severe plant conditions can be detected, managed and that the reduction of the consequences is effective, must be made.

### *2.3 General approach for the nuclear safety related design and assessment*

French safety approach for the nuclear safety related design and assessment is basically deterministic. To complete the range covered by the Design Basis Conditions (DBC), the compliance with the first four level of defence-in-depth imposes on the one hand, the integration of a possible lack of exhaustiveness in the deterministic assessment and, on the other hand, as indicated above, the demonstration of the potential of the facility for the prevention, the control and the limitation of the consequences of "severe accident". The situations studied for the prevention, control and limitation of consequences are qualified as Design Extension Conditions (DEC).[1]

Other accidental situations, the release of which is estimated as unacceptable for the environment, it is necessary to demonstrate that they are either excluded by design or "practically excluded"[2] and, so, are rejected in the residual risk. These situations will not be analysed. This part of the approach meets the recommendation of the exclusion of any "cliff edge effect"[3] leading to an early release of a non-admissible source term.As stressed above the final objective is the whole reduction of the "risk" (frequency x consequences). This reduction can be graphically linked to the Farmer curve, asking for a reduction of the allowable risk domain. This shall be done for all the Design Basis Conditions, as well as for the Design Extension Conditions.

This approach is globally coherent with the recommendations of INSAG (Cf. INSAG 10): "Meeting the safety objectives set for the next generation of nuclear power plants will necessitate improving the strength and independence of the different levels of defense. The aim is to strengthen the preventive aspect and to consider explicitly the mitigation of the consequences of severe accidents …".

### *2.3 The link with GEN IV initiative*

As stressed above the general goal considered in CEA R&D is related to the whole improvement of the defense in depth level for all the nuclear plants. The global coherency between the current CEA approach and the one of the GEN IV initiative can be clearly showed identifying the relationships between these levels and the goals retained both by CEA/DEN and GEN IV.

## 3. Improving the public acceptance of future nuclear energy systems

In addition to the improvement of safety, along the lines presented above, the enhancement of sustainability is believed to improve the public acceptance of future energy. This underpins 3 main objectives of progress:

- the **minimization of radioactive waste production** with a clear objective assigned by the Ministry of Research on June 1st, 1999 that "the long term research on nuclear systems capable of minimizing the production of long lived radio-nuclides should be strengthened *(and therefore Plutonium and other transuranic elements)*";

1. The Design Extension Conditions (DECs) is the terminology suggested in the European Utility Requirements document (EUR, 1998).

2. i.e.: …implementing "sufficient preventive design and operation provisions" (ref. TSO Study Project on Development of a Common Safety Approach in EC for Large Evolutionary PWRs)

3. This risk corresponds to the **mobilisation of a potentially unacceptable source term** – severe accident – **with the simultaneous loss of the containment** (releases much higher than those which are acceptable).

- the **efficient utilization of available nuclear fuels**, both from natural resources and from the operation of the present generation of nuclear reactors (e.g. *Plutonium, depleted Uranium and Uranium from spent fuel);*
- **an enhanced resistance to proliferation risks.**

Concerning the latter objective, a special interest is being invested in the integral fuel cycle approach with the spent fuel processing and re-fabrication processes implemented on the nuclear site. This approach indeed permits to minimize the transport of nuclear materials and constitutes an extrinsic barrier to proliferation risks.

## 4. The CEA program on 4th generation systems: a reference concept to substantiate the goals for future nuclear energy systems

The CEA 4th generation research and development program aims at a mastering a new technological range based on:

- high temperature gas cooled reactor,
- with an enhanced simplified safety,
- and a highly confining and temperature resistant fuel.

The range of solutions should offer various evolutionary options concerning the neutron spectrum (including hard neutronic spectrum) and potential for various application beyond electricity production (hydrogen, …).

The CEA has launched a comprehensive R&D program on promising technologies for future nuclear energy systems to make significant progress in safety, sustainability and economics. Among the candidate options likely to afford significant progress towards these goals, those of **high temperature gas cooled reactors** with a **hard neutron spectrum** core with **refractory fuel** possibly compliant with **on-site processing**, re-fabrication and recycling, will be considered as our long term objective.

This main R&D axis partly builds upon current R&D work to support the GT-MHR international project and prepares possible evolutions toward hardened neutron spectra and closed fuel cycles, to develop "high efficiency gas cooled systems". Making use of gas turbines, which gave the fossil fuel plants a decisive gain in efficiency and competitiveness would endow future nuclear systems to also benefit from the same advanced conversion technologies. The hard neutron spectrum assures a highly efficient "nuclear combustion" capable of burning a wide variety of transuranic and fertile nuclides. Other relevant technologies include refractory fuels with a high confinement capability, integrated spent fuel processing techniques, and component technologies of high temperature gas circuits.

**From the point of view of safety**, this technological range based on gas cooled reactors has the following intrinsic advantages:

- the fuel, being refractory and conducting, has a very good ability to fission products confinement especially during incidental or accidental transients,
- low interaction between the coolant (helium) and the reactor physics in general (quasi no reactivity effect, no chemical interaction)

This lead to good passive safety characteristics.

The main objective of the CEA is to achieve the R&D programme on a range of technologies of interest for "high efficiency gas cooled systems" likely to not only aim at promising prospects in the long term, but also to prepare possible contributions to international projects of high temperature reactors likely to meet some needs of the international market around 2020-2030, and also to favor possible spin-offs of general interest for present reactors (especially in the field of materials and fuels).

The potential assets of hydrometallurgical and pyrochemical processes for the different types of fuels considered – including transuranic fuels - are being assessed with a view to selecting the most appropriate options for the next generation fuel processing techniques, taking into account the advantages of on-site processing techniques for minimizing the transports of nuclear materials and enhancing proliferation resistance.

The capability of the considered technologies to meet also the requirements for systems dedicated to transmutation, leads to a strong synergy between both fields of investigation (including ADS).

Even though dedicated to the needs of a reference concept of 4$^{th}$ generation system, several technologies investigated appear to be of generic interest and key to research and development for nuclear systems. Such are metallic, ceramic and composite materials resisting to high temperature and high fast neutron fluences, as well as partitioning technologies by hydrometallurgy and pyrochemistry. Fundamental research and modeling are essential for breakthroughs toward innovative concepts in the fields of fuels, structural materials and spent fuel processing techniques.

The field of future nuclear energy systems studies is a most appropriate subject of collaboration between European countries, and with the USA, Japan and Russia. Therefore the CEA resolutely decided to integrate its R&D effort in European networks and Framework Programs. Therefore also, it decided with industrial partners to be an active member of the Generation IV International Forum launched in 2000 by the US-DOE with a view to organize a roadmapping activity of promising technologies for future nuclear energy systems.

# GENERATION IV INITIATIVE: SAFETY AND RELIABILITY GOALS AND RELATED R&D

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## Abstract

To advance nuclear energy to meet future energy needs, ten countries have agreed to launch the development of a next generation of nuclear energy systems known as "Generation IV." These countries have joined together to form the Generation IV International Forum (GIF). The retained systems would be licensed, constructed, and operated before the year 2030 in a manner that will provide competitively priced and reliable energy products, while satisfactorily addressing nuclear safety, waste, proliferation and public perception concerns.

To practically structure this initiative, the GIF agreed to support the preparation of a roadmap where few systems – the most representative of this Gen IV – will be selected and the corresponding R&D tasks detailed and organized for an international shared effort.

The whole roadmap project is managed by a Roadmap Integration Team. The structure also comprises an Evaluation Methodology Group (EMG), four Technical Working Groups (TWG) for the different technologies and five Crosscut Groups (CGs) covering areas as fuels and cycles, economics, risk and safety, fuels and materials, and energy products.

Since the beginning of the project, a set of consistent goals has been defined on Sustainability, Safety and Reliability, and Economics.

The paper summarizes and discusses the Safety and Reliability goals, as well as the criteria and metrics that have been developed to be used for the practical systems screening and evaluation. The role of the Risk and Safety Crosscut Group (RSCG) is presented; it focuses on the risk and safety aspects of the concept evaluation and corresponding cross-cut R&D areas. Finally a preliminary overview of the R&D items considered as guidelines for the search of the crosscut themes for the Gen IV effort is given.

## 1. Introduction

To advance nuclear energy to meet future energy needs, ten countries[1] have agreed to launch the development of a next generation of nuclear energy systems – including fuel cycles and reactor technologies - known as "Generation IV." These countries have joined together to form the Generation IV International Forum (GIF). The retained systems would be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products, while satisfactorily addressing nuclear safety, waste, proliferation and public perception concerns. The objective for these systems is to be available for wide-scale deployment before the year 2030, when many of the world's existing nuclear power plants will be at or near the end of their operation.

To practically structure this initiative, the GIF agreed to support the preparation of a roadmap where few systems – the most representative of this Gen IV – will be selected and the corresponding R&D tasks detailed and organized for an international shared effort.

Technical Working Groups have been formed to review the proposed systems and evaluate their potential using the tools developed by an Evaluation Methodology Group. Crosscut Groups have also been formed covering fuels and cycles, economics, risk and safety, fuels and materials, and energy products; their main objective is to review the technical evaluation consistency, and to make recommendations regarding the scope and priority for crosscutting R&D in their subject areas.

The first step of the roadmap work has been the definition of a set of consistent goals with three main purposes: to serve as the basis for developing criteria to assess and compare the systems in the technology roadmap, to challenge and stimulate the search for innovative nuclear energy systems and finally to motivate and guide the R&D identification and organization.

The goals are defined in three broad areas of sustainability, safety and reliability, and economics. Sustainability goals focus on fuel utilization, waste management, and proliferation resistance. Safety and reliability goals focus on safe and reliable operation, investment protection, and essentially eliminating the need for off-site emergency response. Economics goals focus on competitive life cycle and energy production costs and financial risk.

The paper summarizes and discusses the safety and reliability goals, as well as the criteria and metrics that have been developed to be used for the practical systems screening and evaluation.

The charter of the Risk and Safety Crosscut Group (RSCG) is presented focusing on the objectives of its work as well as on the preliminary list of cross-cut R&D areas that fall inside its domain.

1. Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the United Kingdom, and the United States

The TWG's work for the identification of the specific safety related R&D needs is still underway. Some specific items that could be confirmed by the TWGs for further research are recalled hereafter. The paper also present, a preliminary overview of the R&D items considered as guidelines for the search of the crosscut themes for the Gen IV effort.

## 2. Risk and Safety Criteria and Metrics

Generation IV nuclear energy systems will be optimized to meet three safety and reliability goals.

***Safety and Reliability–1***

***Generation IV nuclear energy systems operations will excel in safety and reliability***

This goal aims at increasing operational safety by reducing the number of events, equipment problems, and human performance issues that can initiate accidents or, cause them to deteriorate into more severe accidents. It also aims at achieving increased nuclear energy systems reliability that will benefit their economics. Appropriate design requirements are needed to satisfy such operational objectives.

During the last two decades, operating nuclear power plants have improved their safety levels significantly, as tracked by the World Association of Nuclear Power Operators (WANO). At the same time, design requirements have been developed to simplify their design, enhance their defense-in-depth in nuclear safety, and improve their constructability, operability, maintainability, and economics. Increased emphasis is being put on preventing abnormal events and on improving human performance by using advanced instrumentation and digital systems. Also, the demonstration of safety is being strengthened through prototype demonstration that is supported by validated analysis tools and testing, or by showing that the design relies on proven technology supported by ample analysis, testing, and research results. Radiation protection is being maintained over the total system lifetime by operating within the applicable standards and regulations. The concept of keeping radiation exposure as low as reasonably achievable (ALARA) is being successfully employed to lower radiation exposure.

***Safety and Reliability–2***

***Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage***

This goal is vital to achieve investment protection for the owner/operators and to preserve the plant ability to return to power. There has been a strong trend over the years to reduce the possibility of reactor core damage. Probabilistic risk assessment (PRA) identifies and helps prevent accident sequences that could result in core damage and off-site radiation releases and reduces the uncertainties associated with them. For example, the U.S. Advanced Light Water Reactor (ALWR) Utility Requirements Document requires the plant designer to demonstrate a core damage frequency of less than $10^{-5}$ per reactor year by PRA. This is a factor of about 10 lower in frequency by comparison to the previous generation of light water reactor energy systems. Additional means, such as passive features to provide cooling of the fuel and reducing the need for uninterrupted electrical power, have been valuable factors in establishing this trend.

***Safety and Reliability–3***

***Generation IV nuclear energy systems will eliminate the need for offsite emergency response***

The need for offsite emergency response has been interpreted as a safety weakness by the public and especially by people living near nuclear facilities. Generation IV systems have a goal to essentially eliminate that need by reducing or eliminating the potential for offsite radioactive releases from any severe accident. This is in addition to the steps taken for reactors to reduce the likelihood and degree of core damage required by the previous goal.

The strategy for meeting this goal is to identify severe accidents that lead to offsite radioactive releases, and to evaluate design features that eliminate the need for offsite emergency response with regard to their effectiveness and impact on economics.

The three Generation IV Safety and Reliability goals correspond directly to the concept of defense in depth, which has been refined through research and operating experience. Table 1 shows this direct correspondence with the INSAG 10 levels of defense defined by the International Atomic Energy Agency.

Table 1. **Defense-in-depth** (ref. IAEA for the first three columns)

<table>
<tr><th>Levels of Defense in Depth</th><th>Objective</th><th>Essential Means</th><th>Gen IV Goals</th></tr>
<tr><td>Level 1</td><td>Prevention of abnormal operation and failures</td><td>Conservative design and high quality in construction and operation</td><td rowspan="2">Safety and Reliability–1 Generation IV nuclear energy systems operations will excel in safety and reliability.</td></tr>
<tr><td>Level 2</td><td>Control of abnormal operation and detection of failures</td><td>Control, limiting and protection systems and other surveillance features</td></tr>
<tr><td>Level 3</td><td>Control of accidents within the design basis</td><td>Engineered safety features and accident procedures</td><td>Safety and Reliability–2 Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.</td></tr>
<tr><td>Level 4</td><td>Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents</td><td>Complementary measures and accident management</td><td rowspan="2">Safety and Reliability–3 Generation IV nuclear energy systems will eliminate the need for offsite emergency response.</td></tr>
<tr><td>Level 5</td><td>Mitigation of radiological consequences of significant releases of radioactive materials</td><td>Off-site emergency response</td></tr>
</table>

The evaluation and ranking of Generation IV concepts requires a systematic approach to treat uncertainty, since many aspects of potential conceptual designs can not be known until the completion of research and development efforts. To evaluate and rank Generation IV concepts, the Gen IV Evaluation Methodology Group (EMG) developed a set of twelve criteria that examine indicators of potential future performance. These criteria are presented hereafter.

***Safety and Reliability–1***

Evaluation for the first Safety and Reliability goal (SR1) focuses on safety and reliability during normal operation of all facilities in the nuclear energy system, from mining to the final disposal of waste. Thus the focus is on those high to medium probability events that set the forced outage rate, control routine worker safety, and result in routine emissions that could affect workers or the public. Assessment during screening focuses on unique system characteristics that can impact reliability and unusual design aspects that could significantly affect worker safety and routine emissions. Quantitative measures of parameters affecting plant reliability are added in the viability and performance evaluations, when a system design is sufficiently developed. Goal SR1 considers facility attributes operable at the first two levels of defense in depth, as described above in the introduction to "Criteria and Metrics for Safety and Reliability Goals", i.e., those features that can reduce the frequencies of initiating failures for all potential operating conditions and that can control abnormal operation and detect failures.

The SR1 screening metrics examine three areas: (1) the potential reactor forced outage rate, compared to current (Generation II) plants; (2) the potential for unique routine exposures workers; and (3) the potential for worker accidents or accidental exposures. Because these factors are strongly affected by detailed characteristics of the design and operation of facilities, the screening is focused on identifying unique characteristics that might alter the performance of a concept compared to the experience with current plants.

***Safety and Reliability–2***

Evaluation for the second Safety and Reliability goal (SR2) identifies facility attributes that, using models and experiments, create high confidence that all design basis accidents (DBA) are correctly managed and that reactor core damage will have a very low likelihood or can be excluded or practically excluded by design (and in other facilities, that the release of radioactive material from its most immediate confinement or nuclear criticality can not occur.) For performance evaluations much of the information required for a Preliminary Safety Analysis Report (PSAR) will be available, so the likelihood of core (or other facility) damage can be evaluated quantitatively for specific DBAs. Results can be presented as a frequency probability distribution which reflects all sources of uncertainty in models and experiments. For preliminary design information available at final screening, an approximate analysis of the safety related architecture using the level of defense (LOD) analysis can identify conflicts with safety fundamentals. The final screening identifies major design characteristics that are likely to robustly bound potential transient power, temperature, chemical reaction, and mechanical stresses well inside damage thresholds. Equally important, the screening credits design approaches that facilitate the modeling and experiments required to predict quantitatively the uncertainty of safety margins.

The first two SR2 criteria focus on the characteristics of the engineered safety features and/or inherent features (for reactors: power control, heat removal, and radionuclide confinement) that can transparently bound the accessible range of operating and accident conditions and allow the

facility state to be predicted with very low uncertainty, inside this range of conditions. One criteria focuses on the methods used to provide reactivity control, and the other criteria focuses on the methods used for decay-heat removal.

The subsequent three SR2 criteria focus on the potential to model and predict the performance of the reactor safety systems with high confidence, giving credit to system where the models will likely have small and well-characterized uncertainties. The first criteria examines the detailed, dominant phenomena that occur in the reactor safety systems, and credits systems that use phenomena that can be readily and accurately predicted using separate effects experiments and physically based models. The next criteria credits reactors which have large fuel and coolant thermal time constants, since this inertia reduces the importance of the early time, more complex phenomena that commonly occur in reactor accidents. The final criteria credits reactor systems that can be studied in full-scale integral tests, thus giving credit to systems where safety tests can be performed with small or negligible scaling distortion.

***Safety and Reliability–3***

Evaluation for the third Safety and Reliability goal (SR3) considers system attributes that allow demonstration, with high confidence, that the radioactive release from any scenario results in doses that are insignificant for public health consequences. Such confidence must come from the knowledge that reactor core damage has very low probability (SR1 and 2) and that mitigation features provide additional lines of defense to account for any significant residual risk. This confidence comes from three sources: accurate bounding prediction of the timing and magnitude of radioactive source terms and energy releases; accurate assessment of the effectiveness of the confinement system in accommodating the all bounding energy releases and providing holdup of radioactive material; and assessment of the resulting off-site dose probability distribution and comparison against appropriate standards for individual and societal risk. For screening, a concept is qualitatively ranked according to accident and release potential relative to present nuclear energy systems. This screening includes assessment of how well severe-accident phenomena can be characterized and modeled for the concept. For later viability and performance evaluations, quantitative evaluation of damage, release and transport, and comparison of resulting dose relative to public health criteria, are used.

The first two SR3 criteria examine the potential releases that could occur from the primary reactor system from degraded fuel (the source term), and the energy release mechanisms available under severe-plant conditions that could generate damage to structures and safety systems. The source-term criteria credits systems where the fuels are particularly robust to damage at high temperatures, such as some gas-reactor fuels, and credits systems with coolants that tend to chemically absorb released radionuclides. The energy-release mechanism requirement credits systems that have simpler response even under severe plant conditions.

The next two SR3 criteria examine the performance of the independent mitigation systems that confine and hold up any releases that could occur from the primary system. The first criteria evaluates the degree to which any release is delayed, since such delays both reduce the total release, due to increased deposition of radionuclides inside the reactor structures, and provide additional time for off-site response. The second criteria credits the effectiveness of the containment structures, or the filtered confinement structures, in reducing the fraction of radionuclides that can escape from the plant.

The Table 2 summarizes the 12 criteria used for the safety and reliability assessment.

Table 2. **Criteria and weights for safety and reliability goals**

| | |
|---|---|
| **Safety and Reliability–1** Generation IV nuclear energy systems operations will excel in safety and reliability. | |
| **SR1-1** | Reliability |
| **SR1-2** | Public and worker safety – routine exposures |
| **SR1-3** | Worker safety – accidents |
| **Safety and Reliability–2** Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage. | |
| **SR2-1** | Robust engineered safety features |
| **SR2-2** | System models have small and well-characterized uncertainty (physical models / well-scaled experiments) |
| **SR2-3** | Unique characteristics |
| **Safety and Reliability–3** Generation IV nuclear energy systems will eliminate the need for offsite emergency response. | |
| **SR3-1** | Radioactive source/energy release magnitude and timing understood and bounded by inherent features |
| **SR3-2** | Confinement or containment provides robust mitigation of bounding source and energy releases |
| **SR3-3** | No additional individual risk |
| **SR3-4** | Societal risk comparable to competing technology |

## 3. Role of the RSCG

The Risk and Safety Crosscut Group (RSCG) was chartered to examine implementation of the Safety and Reliability Goals, criteria and metrics for the various Generation IV concept sets and to identify common R&D opportunities that would advance Generation IV safety and reliability.

Many aspects of these improved safety and reliability approaches are common to more than one type of advanced concept. The RSCG review and evaluation will develop an integrated perspective on advanced concept safety and reliability approaches and research needs, and identify crosscutting research areas where common R&D could advance safety and reliability technologies for Generation IV systems. In order to evaluate the viability of these concepts from a safety and reliability perspective, and develop a common R&D plan, the RSCG will:

- assess the consistency of scoring the safety and reliability criteria between the diverse advanced concepts presented by the TWGs;
- identify the areas of safety or reliability research needed to advance these goals that are common to many concept sets;
- establish priorities for the crosscutting R&D areas;
- assess the likelihood of success (cost / schedule) for each of these R&D areas;

- provide summary documentation for performance and viability evaluations and safety and reliability research areas and priorities.

To accomplish these objectives, the RSCG is developing a framework for assessing consistency for each of the safety and reliability goals which considers both the approaches used to achieve Generation IV goals (i.e. inherent vs. engineered safety features) and issues of potential vs. uncertainty. The RSCG assessment draws on safety analysis and reliability expertise in areas such as static and transient systems analysis, definition of design basis and beyond design basis events, instrumentation and control, PRA methods, personnel safety, and the regulatory framework for advanced systems. It is anticipated that these areas will be expanded further as the safety and reliability assessment progresses.

## 4. Preliminary list for the R&D areas

Within the context of the current systems screening, the TWGs are identifying, for each assessed concept, the R&D gaps that shall be closed to achieve the expected performances. This is done systematically for all the considered goals. Once these gaps identified the TWGs, with the support of CGs, will characterize the R&D effort needed to achieve the gaps closure (R&D scope and content). The majority of these gaps and the corresponding R&D items are specific to the systems under examination. Some can be common to different systems/technologies. Some new and generic crosscut R&D items could also be identified by the CGs.

It is convened that the search for R&D must be applied to the entire system: fuel cycle facilities (mining, fabrication, reprocessing) and reactors (core behavior, handling, transport).

As indicated, the work is currently underway. For the different reactors technologies, some specific safety related items could be confirmed by the TWGs for further research, e.g.:

- Water: core melting exclusion strategies, in-vessel core retention strategies, hydrogen management.
- Gas: proof for the fuel and material performances at very high temperatures and fluences, search for reliable passive systems for the fast cooled reactors, severe plant conditions management.
- Liquid Metal: core melting management strategies, e.g. minimization of risk from hypothetical core disruptive accidents, and proof for the recriticality free concepts for the Na concepts.
- Non classical: Postulated Initiating Events identification and needs for specific regulatory approaches.

For the safety & reliability crosscut R&D items, a tentative set of crosscut R&D items can be defined using existing documents (e.g.: URD, EUR, European SINTER Network, NERI program, IAEA documents, ...).

The tentative list for the R&D crosscut domains is the following:

***Design basis transients and accidents – Static & transient analysis***

- Material behavior (corrosion, resistance, damage, activation, ageing, ...).
- Fuel behaviour.
- Reactor physics (neutronics and thermohydraulics).
- Passive and active safety system assessment.
- Man machine interface, human factor.
- Instrumentation and control.
- Balance of plant, systems design, normal and abnormal situation (nominal operating conditions and transients).
- Plan decommissioning.
- Radioactive waste management (release to biosphere, contamination, ...).

***Severe plant conditions***

- Core degradation accidents (melt exclusion, corium behavior, core fires, ...).
- Severe accident mitigation (core melt progression, retention, ...).
- Probabilistic risk assessment.

***Environmental impact***

- Workers doses and in-site impact.
- Population doses and offsite impact.
- Exposure consequences assessment.

***Safety related architecture: economical evaluation***

- Passive versus active evaluation.
- Maintenance and repair.
- Modularity.

***Licensing approach: Risk informed and risk based regulations***

- Modeling both from content (e.g. static/dynamic behavior description) and consistency / standardization point of view (e.g. on common causes, human factor, etc.).
- Comprehensiveness, i.e. integration of all the plausible initial plant states (e.g. internal and external events, modes of operation, integration of site specifics).

- Reliability modeling and data for active and passive systems as well as for inherent characteristics.

For each of the previous domain, needs can be defined in terms of

- data;
- codes;
- qualification by experiments;
- qualification of tools;
- methodology.

The table below shows the tentative correspondence between domains and needs. This table is used only as a guideline for the RSCG work purpose; deepest internal discussions are needed to confirm first of all its pertinence within the frame of the Gen IV initiative and secondly to precisely define its detailed content.

| | Data | Codes | Qualification | Experiments | Methodology |
|---|---|---|---|---|---|
| Design basis transients and accidents | X | X | X | X | |
| Severe accidents conditions | X | X | X | X | X |
| Environmental impact | X | X | X | | X |
| Safety system economical evaluation | X | X | | | X |
| Licensing approach:<br>Risk informed and risk based regulation | | | | | X |

## 5. Conclusions

The Generation IV initiative is implementing a framework for international cooperation in research and development for the next generation of nuclear energy systems.

The safety and reliability aspects are among the areas that must be covered in order to meet three specific goals:

- Generation IV nuclear energy systems operations will excel in safety and reliability.
- Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
- Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

A Risk and Safety Crosscut Group (RSCG) has been chartered to examine the implementation of these goals for the different systems technologies submitted for evaluation; beside the work to insure the consistency among the different systems assessment, the RCSG shall identify the areas of safety or reliability research that are common to many concept sets and participate to the definition of the R&D pathways that will be retained within the final Roadmap.

For the different reactors technologies, some specific items could be confirmed for further research, e.g. Water: core melting exclusion strategies, in-vessel core retention strategies, hydrogen management; Gas: proof for the fuel and material performances at very high temperatures and fluences, search for reliable passive systems for the fast cooled reactors; Liquid Metal: core melting management strategies, e.g. minimization of risk from hypothetical core disruptive accidents and proof for the recriticality free concepts for the Na concepts; Non classical: Postulated Initiating Events identification and needs for specific regulatory approaches.

For the crosscut R&D items, a tentative set is established and implemented as guideline under the following main themes:

- Design basis transients and accidents – Static & transient analysis.
- Severe plant conditions.
- Environmental impact.
- Safety related architecture: economical evaluation.
- Licensing approach: risk informed and risk based regulations.

# SESSION 3

## How to Deal with Safety Issues, Questions and Concerns? Identification of Research Needs

*Chairman: J.E. Lyons (USNRC)*
*Co-chairmen: M. Vidard (EDF, France) and H. Wider (EC)*

# ROLE OF RESEARCH IN THE SAFETY CASE OF FUTURE REACTOR CONCEPTS DEFINITE NEEDS, CONFIRMATORY AREAS, ADDED VALUE

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## 1. Introduction

In this paper, we aim to describe the role(s) of research when assessing – and finally demonstrating – the safety of future reactor concepts. The term "research" could be defined in quite a limited fashion, while in this paper, we will use it synonymously to all Research & Design (R&D) work.

First, we will provide a top-down planning perspective by identifying the general set of safety factors related to new reactor projects; i.e., factors that have to be accounted for already in the safety case. The research needs can be based on such a set of safety factors and related challenges, and it is crucial that the research requirements remain reasonable. For this, suitable design choices have to be made, such that they limit the criticality of individual plant functions, related systems/ structures/components (SSC), and human actions. In this context, we will discuss the general ways of limiting excessive research needs. The critical plant functions and SSC, in their turn, will require a strong safety demonstration.

Once major design decisions have been made according to the aforementioned lines, actual systems are designed, using available equipment or developing new. Research support is needed to establish the technological adequacy and confidence level of each decision at each design level; this is relatively easy where conventional technologies are used, but more challenging if novel technologies come into question. Maturity of technology also correlates strongly with the attainable certainty and qualification of analysis tools. Here, we will describe the general phases of R&D from exploratory work to safety demonstration, and also independent confirmation. We will also touch upon the roles of different actors – the vendor, the licensee and the regulator – as well as their strategies in attacking the above-mentioned safety factors through research.

As already mentioned, we will base the above discussions on a top-down approach, but it also should be noted that the research does not end where the top-down planning ends. During the R&D work and also through other experiences (finally the plant operation), bottom-up R&D ideas (problems and possibilities) surface; and the safety research has to continue "with a constantly inquiring mind", "search for excellence". At the end of the paper, we will provide some final remarks

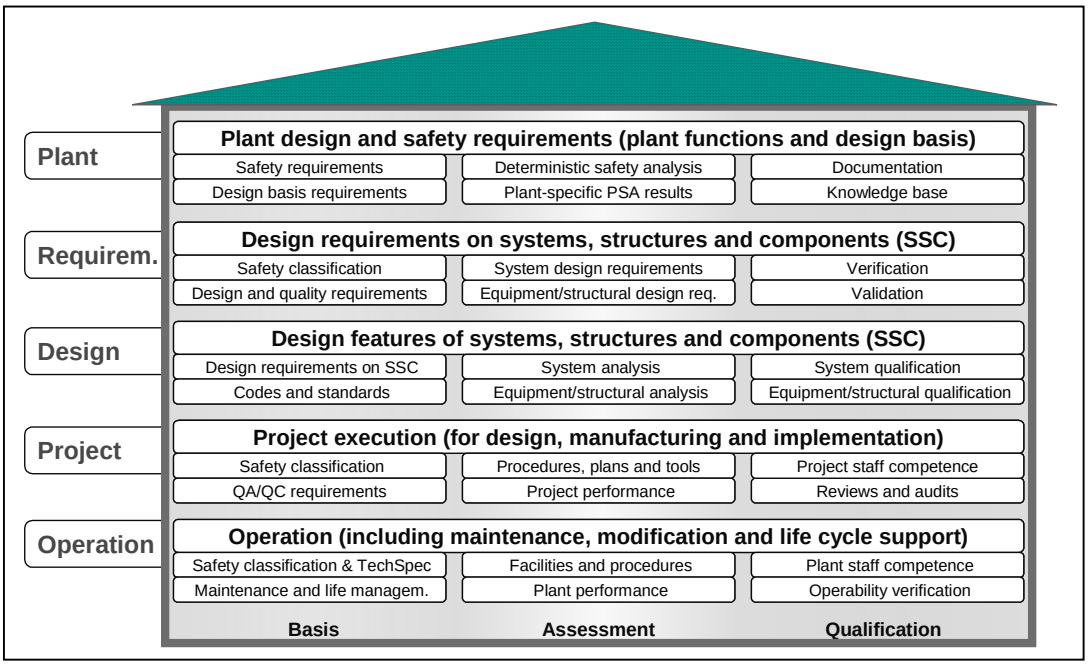
on how to maximize the added value of safety research funding also in the long run; or in other words, how to make choices with limited funds.

It would be much more attractive to examine the subject based on concrete cases, such as those presented in this workshop. At the time of this writing, however, the other papers are not available – and our approach remains quite abstract. We will give brief examples wherever this is possible and potentially useful, at least in the light of light water reactor experiences. We expect the presented principles to be applicable also to other reactor types.

## 2. Safety factors and related research needs

Figure 1 below elucidates the type of factors that are crucial for the safety demonstration of new reactor projects. There are five main groups of safety factors, the first one being related to the *plant concept* and the high-level *safety requirements*. At this plant level, general safety principles such as redundancy, independence and diversity are defined; and the plant safety is assessed in safety and risk analyses. The second group of safety factors highlights the importance of *requirement specifications* derived from the plant level to the next three "implementation" levels below: the technical *design features* and qualification of systems, structures and components; the *project and quality management* in constructing the new reactor plant, as well as the aspects of *plant operation*. The most crucial tool for grading requirements is the safety classification, which is based on the deterministic safety analysis – i.e., postulated initiating events and necessary functions to protect the release barriers. In addition, the safety classification can be tuned, if necessary, with quantitative risk information; although substantial needs for this do not arise if the safety classification is made on the basis of main safety functions such as reactor shutdown and heat removal (functional classification). After all, the main safety functions after different initiating events are important for both the deterministic safety analysis and the PSA results.

Figure 1. **Safety factors related to new reactor projects – "Safety Factory"**



For each group of safety factors, there are three types of safety demonstration efforts: the *basis*, the *assessment* of features against the basis, and the *demonstration* of quality (qualification). There may be research needs associated with all of these. For example, research may be needed at the plant level in order to justify the set of initiating events used for the design basis, to analyse the plant response in them, and to provide the data necessary to qualify (validate) the prediction models used. At the requirement level, quite much research is ongoing in different fields in order to examine the best ways of specifying requirements – it is here most mistakes are made when crossing customer-supplier interfaces. Furthermore, research may be needed to define specific requirements for different functions and related systems/structures/components. As noted above, the requirement specification is the basis for the next three levels, where research may be needed to demonstrate the integrity and quality of certain equipment, for example. The usual research focus is the design features level, while the above figure covers also the plant construction and the plant operational phases. The operational features of the plant, in particular, are of a great safety relevance, including even equipment level aspects such as the verification of structural integrity (e.g., inspection methods) or equipment operability (e.g., programmable devices). The same principal needs for ensuring quality apply also during the reactor construction project, which involves a large number of safety critical equipment being manufactured and delivered to the site.

With the above figure, we would like to emphasise the importance of the full-scope picture: decisions on different technical levels (plant, requirements, design features) and during different life cycle phases (design, construction and operation). The safety research for future reactor designs may involve major issues on different *technical* issues, but the difficulties in demonstrating quality and safety during the different life cycle *phases* should not be forgotten. This is indeed highlighted by the attention given nowadays to the final phases, decommissioning and waste management. In general, it can be stated that the more novel the different plant (including equipment) features are, the more carefully are we forced to examine all the floors in the "Safety Factory" of Figure 1 above.

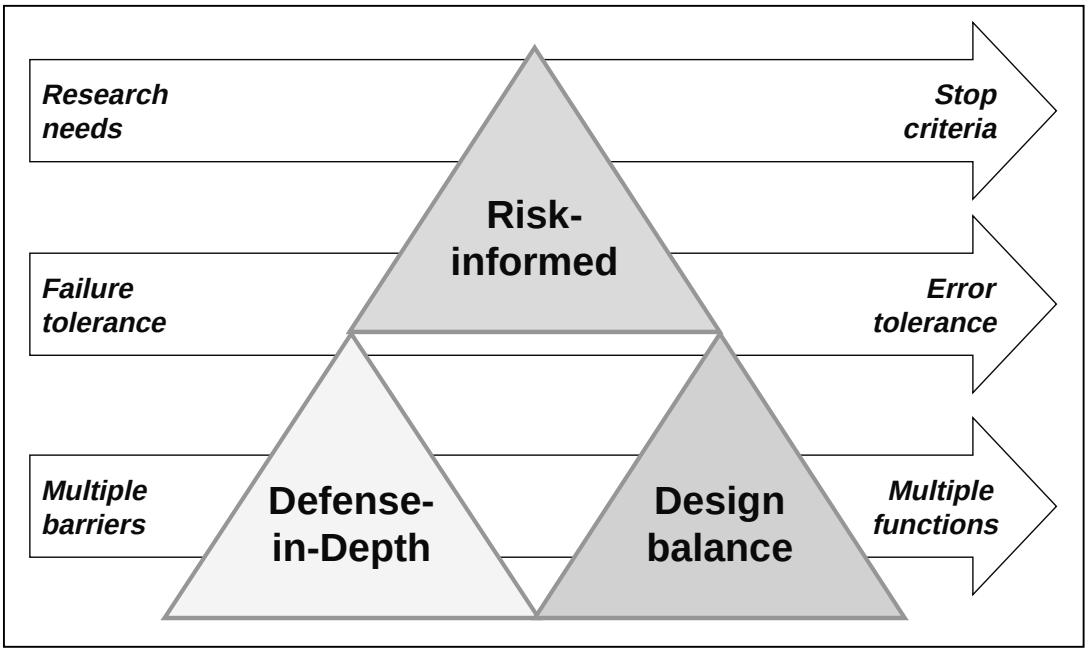
## 3. How to limit excessive research needs?

Tentatively at STUK, we have defined three main principles for the application of *diversity*: the design should be *risk-informed*, including the *defense-in-depth* and the *balanced design* principles; see Figure 2 below. While writing this paper, we noticed that these principles can be used also to limit excessive research needs. The reason for such analogy is of course the main objective of diversity – the limitation of the consequences of Common Cause Failures, which are potentially caused by *errors* during different life cycle phases of a certain system or piece of equipment, for example.

The defense-in-depth principle has to be used to avoid overly dependence on a single part of the plant concept, whereas the design balance requires different parts of the plant to be in balance with each other (no overly weak nor overly strong links in the chain). Here, the traditional release barrier perspective is not sufficient if multiple barriers could fail due to a failure of a single safety function; i.e., a failure of redundant, similar divisions of a specific safety system. Multiple, diverse functions are therefore required within the main safety functions. This is the first main trend required and also seen in advanced reactor concepts, and the second one is the transition from failure tolerance to error tolerance – something we would like to name the *single error criterion*. This goes hand in hand with the defense-in-depth and diversity principles, but can also be seen as a part of the risk-informed principles. On this basis, there should be sufficient general grounds for setting stop criteria for different research efforts, as the criticality of each safety issue can be viewed in the full context. This does not mean "errors can be easily allowed", but it certainly avoids situations where a single error – even on the research side! – puts the plant directly outside the design basis.

The single *error* criterion is basically a "bottom-up" criterion – requiring that a mistake affecting any equipment or function of the plant be tolerated without severe consequences. It combines the criteria usually defined for diversity due to potential design and maintenance errors, for example; or for technical features or administrative controls required to limit the risks of human errors during operation and maintenance, for example. It promotes the use of *simple* safety critical functions and equipment, for which a design or maintenance error is more probably found through independent checks (verification and validation). And it even allows more diversity – or research! – to be invested on managing frequent initiating events; as long as similar investments are made on preventing potential *errors* (such as a major structural crack being missed during inspection) leading to the more remote events. Finally, the single error criterion has a quantitative basis as well, since for any typical safety system consisting of redundant divisions, it is quite generally accepted that the minimum failure probability is of the order of $10^{-4}$ per demand or per year. Consequently, a single system cannot be fully depended upon from the standpoint of quantitative risk targets.

Figure 2. **General ways of limiting excessive research needs**




The single *error* criterion also has a major message to carry concerning the importance of research. Namely, the future reactor concepts can be expected to involve features that limit the criticality of single plant functions and related systems/structures/components (SSC); but this does not relieve the pressure for understanding the details of plant response to different initiating events, including the quality of prediction models utilized for this. At the level of single systems, equipment types, and human actions, the plant designs may – they have to! – become more error tolerant. However, the loadings caused upon structures and the plant conditions in general, require thoroughly validated analysis tools and the application of reasonable safety margins (robustness). Furthermore, the safety critical plant functions and SSC are still required to have a high dependability. The transition from failure to error tolerance does not remove the research needs, but it makes it easier to focus them on issues where errors may still propagate, without mitigation, to unacceptable

consequences. It also makes it easier to accept a certain level of "inherent uncertainty" for complex issues such as the protection system software reliability or the risks associated with human errors.

Most importantly, the single error criterion should be applied also to research! Does the plant design sustain an error here? What it is the dependability level required for this piece of research? Could You put a simple dependability index on the model You are developing and validating, such as "the index 4 goes for an analysis error probability of the order of $10^{-4}$, "3 goes for an order of $10^{-3}$", etc.? How can this be seen in the combination of the model quality and the deterministic criteria applied? How is this communicated between the different parties involved in designing, researching and setting up the general safety requirements (vendor, researcher, regulator)? Are there any points where excessive dependability indexes remain? With our current analysis methodologies and safety principles, we should be able to provide such risk-informed transparency. On one hand, it could mean that a certain "interesting subject" be prioritized lower, while on the other hand, the design-specific vulnerabilities would be even harder to overlook.

Comparing the above discussion with the current trends of risk-informed regulation, the following is noteworthy as well. Both the defense-in-depth principle and the risk-informed practices may facilitate a feasible level for the necessary safety demonstration efforts ("at the plant level"), but they do not remove the heavy engineering and R&D burden on understanding the plant response. This involves the effectiveness of the safety critical functions and related systems/structures/components, in particular ("the devil in the safety critical details").

## 4. General phases of research

As discussed above, hard facts are needed to decide on critical issues and to develop and qualify technological solutions. Hard facts are provided by research, and it can be divided into two main phases: problem *formulation* and problem *solution*.

We underline the significance of the formulation part in reactor safety, because formulation of (many) individual research problems follows from the general design goals (here, demonstration of technological aspects of all safety factors), and sets (for each problem) the scope of all subsequent effort, including the confidence level that the research results must meet. The solution phase in turn provides data, models, and understanding of what the problem is all about and how to handle it in scientific/engineering terms. It should be noted that problem formulation and solution phases may overlap partially, as the formulation may need refining once efforts to solve it provide data and understanding that helps better characterise the original question.

Problem solution in turn is, in effect, a learning process. For a "new" problem – one of which little is known in advance from other fields of science and engineering – this learning process proceeds through three distinct phases, exploration, consolidation (accumulation), and verification (confirmation), as shown in Figure 3 below. For "old" problems part of the efforts required to produce knowledge through various phases already exist, and it remains to establish which of them are valid in the current framework.

### *Exploration*

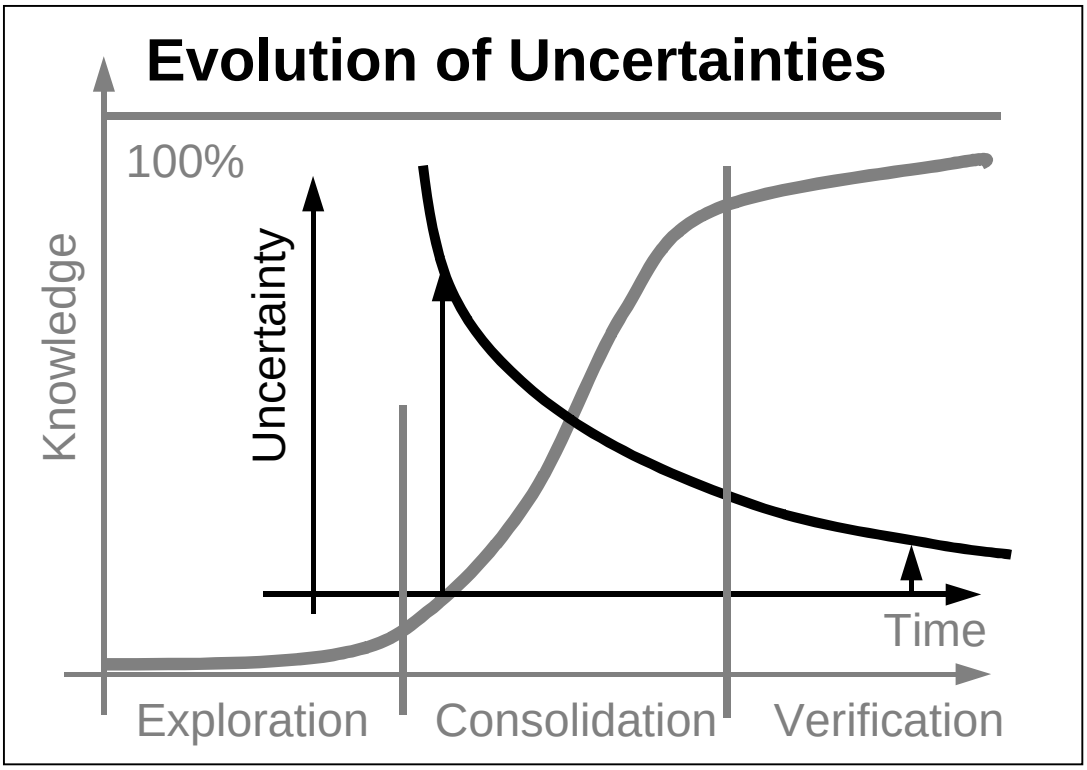
During exploratory phase, research is initially somewhat random, as the main issue is trying to find out what the problem actually is, and trying to establish some initial idea of what the various data and their relations look like. This is scoping around the problem; at this stage it is typical that

research results are scattered and sometimes contradictory, as the proper research methods themselves need to evolve as the work goes on. Typically, uncertainties in data, models and understanding of the problem are very large at this time. Hence this part contributes directly little to the overall "solution", i.e., reaching a near 100% level. However, indirect influence on subsequent work is fundamental: the framing and formulation of the problem dictates most of the scope and methods of all subsequent effort during later stages of the learning process, including definition of the interfaces between the problem, other current problems, related fields, etc. In philosophy of science this process in known as the establishment of the paradigm.

The exploratory stage transitions to consolidation/accumulation stage at about the point where the individual scattered pieces of information collected so far suddenly start to fit together. This is often connected to the maturation of the paradigm; in other words, a formulation of the problem is found that is acceptable to the majority of the workers (and especially opinion leaders) interested in the issue. The transition point is thus marked by *agreement on what the problem really is*, and how it relates to its background and other problems. At this transition point some predictive capability may have been achieved, but uncertainties inherent in the predictions are still typically relatively large as the problem is not yet under control and reliable raw data are still few.

The sump/strainer clogging in a LOCA is a good example of a safety problem still at about the transition between exploration and consolidation. For a few insulation materials (fibrous), there is plenty of data and fair agreement on how to assess their effects (i.e., consolidation is underway), whereas for some others (metallic) there is still major controversy regarding their behaviour, influence and even proper testing methods (i.e., scoping is yet incomplete).

Figure 3. **Research as a three-phase learning process. Prediction uncertainties decrease as knowledge increases. Initially all predictions are impossible, then, highly uncertain; finally, as the field matures, uncertainties become relatively small as the growing competence allows ever better but never perfect control of the problem**



### *Consolidation/Accumulation*

The second stage of learning, consolidation/accumulation, proceeds fast, since all the basic assumptions necessary to do practical work have been established by the problem formulation (paradigm). This stage is characterised by rapid accretion of data, refinement of basic concepts, formation of a knowledge base, followed by models that fuse the concepts into a theory, and development of real understanding through repeated testing of the predictions obtained from the theory. As the understanding improves, uncertainties associated with predictions decrease steadily; as also sketched in Figure 3 above.

During the consolidation/accumulation period the inconsistencies and contradictions of the initiation phase find their natural explanations. Some earlier work will be found useful but limited, some invalid and is discarded. At the end of the accumulation stage stands the point where the field has matured enough to allow *many different but working solutions* to the problem. In engineering problems, this endpoint is marked by the ability to correctly predict (to an accuracy known and adequate for practical applications) the behaviour of the system, given its composition and initial and boundary conditions.

This is the phase during which analysis tools are developed. These days, analysis tools are mostly computer codes that are believed to capture the "essential" features of the physics in question. Code validation to such a degree that their use is generally found acceptable is mostly done towards the end of consolidation phase, because it requires the existence of adequately well established experimental data base which is also developing throughout this phase (in interaction with code development). In general terms, tool accuracy can never exceed the accuracy of the data base, and is normally much inferior due to inevitable simplifications in physical and numerical models. Accuracy of computational tools has received substantial attention especially in the thermal-hydraulic safety analysis community during previous decade, and there are some discussions still going on for this subject, partially because of rather high expectations that some code/result end users foster.

### *Verification/Confirmation*

Over time it becomes evident that most of what is knowable within the established framework (paradigm) has already been worked out, and what remains is to "sweep the nooks", fill in any gaps in the knowledge base, and further refine the accuracy of the predictions. This is where the final verification phase takes over. At this point, rate of knowledge accumulation slows down dramatically: less and less new remains to be known, so it is getting harder to find, and better accuracy can only be achieved with ever larger investments into both theoretical calculation and experimental measurement (each new significant digit costs an order of magnitude more than the previous one to determine). Thus "complete" knowledge within the problem framework is approached asymptotically and never really achieved. However, for engineering applications, finite accuracy is always enough; as far as the safety principles presented in previous sections of this paper, are followed. The push for more accurate data or models usually arises out of the desire to further optimise some aspects of system/process operation (such as system efficiency or economics), with potential implications for the safety demonstration as well.

### *Roles of different actors*

Ideally, the scoping should have been done and accumulation phase well underway for all design/safety significant problems by the time a novel reactor design arrives at a regulator's desk.

These tasks are at the responsibility of the vendor and the industry; the role of the regulator would be to oversee that this work is adequately credible in terms of scientific/engineering research standards, and to contribute by independent confirmatory research to the final part of accumulation phase and through the verification phase. This allows one to develop the recently much sought for termination criterion: once the uncertainties in data, models and understanding are sufficiently well known and under control (see Figures 2 and 3 above), further research is not needed to refine them. However, this is true *only under the crucial assumption* that the assumptions made throughout the work are still valid. Checking the validity of these assumptions is one of the more challenging tasks that a technical regulator needs to face, as amply evidenced by the discussions during recent years regarding LWR fuel safety criteria. If some fundamental assumptions have changed (e.g. due to slow technological evolution without direct relation to the problem), the problem transforms to treatment of an "old" one, with the need to establish what part of the past efforts still remains valid and useful and what needs to be discarded.

In the lucky case that the research conclusions are supported by throughout adequately valid basic assumptions, the technical regulator still needs to satisfy her/himself that both basic data and eventual (computational) tools are properly qualified. We reiterate the fact that all (engineering) knowledge is of finite precision and this must be reflected on the degree of reliance a designer/vendor/utility puts on any individual technological solution. In other words, the more safety critical a design feature is, the more complete knowledge of its characteristics is needed. In nuclear safety this need of knowledge covers the whole plant life cycle, including manufacture, construction, normal operation, all abnormal situations covered by the safety case, decommissioning and final disposal – no small task, but can be rendered manageable by design principles discussed in section 3 above.

## 5. Final remarks

In the previous sections, the subject has been approached from a top-down planning perspective, but this cannot be allowed to "kill the constantly inquiring safety research". Apart from the formal safety case, there will be uncertainties that need further confirmation. This applies equally to reactor concepts under development, as well as to existing plants under operation. And it applies both to the industry and to the regulatory work.

There will be limitations to the safety research funding available, both during R&D and during the operational phase. The set of safety factors defined above for the new reactor projects, may be modified to correspond better to a reactor being operated; with new areas such as emergency preparedness or radiation safety coming into picture, though affecting the original design basis also (see, e.g., the IAEA guidelines for Periodic Safety Reviews, as based on a set of safety factors). The safety challenges related to different safety factors have to be presented in a transparent way, and challenges of a clear safety impact should be prioritized.

This ideal approach is complicated by the need to maintain and cultivate expertise needed in the whole Safety Research Factory (see Figure 1 above). When things seem to have settled, one may start thinking about "closing the doors". From safety perspective it is crucial to maintain both knowledge and appreciation of to what accuracy and, more importantly, *under which basic assumptions*, this "adequate" level has been reached. LWR experience tells that in several cases either (even slow) technological evolution or surprising operational experiences have nullified the validity of basic assumptions in some safety-critical areas. In re-evaluating basic assumptions it is paramount to remember that no amount of research can resolve a poorly formulated problem to any good accuracy. Hence, when facing a novel issue, the safety community needs the resolve and patience to proceed

through the phases of research, from exploration deep enough into consolidation phase, and finally verification, in order to arrive at decision which can stand the test of time.

In fact, *refining the uncertainties and margins* has to be accompanied by deep understanding of what the applicability limits of the models and data are. At present this is a rather pressing problem, because research in the more mature fields related to nuclear safety was initiated decades ago, and many original pioneers have already retired. With them, deep cognizance of the basic paradigms is at the risk of being lost; and at the same time, the economical challenges such as deregulation create additional pressure to cut safety margins and investments. Consequently, the principles of defense-in-depth and diversity have become even more important in limiting the risks of errors.

# EDF VIEW ON NEXT GENERATION REACTOR SAFETY AND OPERABILITY ISSUES

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## I. Introduction

In the foreseeable future, EDF will have to compete in an economically deregulated market. Nuclear currently accounts for more than 80% of the electricity generated by the company, and generation costs are quite competitive compared to that of other competing energies. It is so likely that nuclear units will remain the backbone of EDF generating fleet in the years to come. However, to remain a viable option for electricity generation in the longer term, nuclear will have to maintain both its cost-effectiveness and a very high safety level. This could seem quite straightforward considering the current situation where safety records are at an all time high and Operating and Maintenance costs are under tight control. In fact, it could be a real challenge. Competing fossil technologies progress and there is a concurrent trend to try and improve the performance of future nuclear units. However, in most cases, proposed designs depart from the well-known Light Water Reactor (LWR) technology. They are either new concepts or designs already tested in the past and modified to address some of their perceived drawbacks. Contrary to the prevailing situation where short-term alternatives like the EPR, the ABWR or the AP600 largely build upon experience gathered on operating units, most designs contemplated for implementation beyond 2020 or 2030 cannot be considered proven.

Amongst the characteristics of these contemplated new designs one of the most significant is their output, which is in the range of 100 to 300 MWe. Considering the high overheads generally attached to nuclear, cost-effectiveness could be impaired by such moderate outputs.

Considering the above mentioned uncertainties, EDF have confirmed their preference for proven designs with higher outputs, such as the EPR.

However, it would appear unreasonable to consider that new designs are doomed to fail: they could well turn out to be adequate for specific niches in a deregulated market and provide reasonable alternatives for the utility. Nevertheless, for such an alternative to be considered, additional evidence is needed that utility preferences are reflected in the design, and that all potential technical issues have been identified, adequately addressed and resolved.

Currently, EDF is involved at various degrees in the evaluation of next generation:

- Light-water reactors.

- Gas-cooled reactors.
- Liquid-metal reactors.

Available information is not the same for all concepts, but nevertheless adequate for identifying areas where confirmation of assumptions would be needed.

After discussing some crosscut issues, this paper outlines which areas would have to be clarified for each type of reactor before they could be considered proven by the company. Issues for which R&D programs could be needed are also be identified.

## 2. Crosscut issues

### *2.1 Operator role*

For EDF, a very significant issue to be dealt with in new projects is the role of operators in day to day operation. As pressure increases to control costs, there is a trend, in all companies for re-analyzing personnel tasks in order to detect whether some could be decreased or even eliminated without detrimental consequences for plant safety.

In some cases, this rationalization effort results in staff downsizing. However, operators remain a key component of plant safety as they are relied upon for performing actions allowing bringing units in a controlled or safe state in case of accident.

With the development of new designs relying on passive safety systems, the role of operators in accident management was re-evaluated, and a modified approach adopted. To make the reactor more user friendly and more forgiving, it was decided that all safety criteria would have to be complied with, in case of design basis accidents, without relying on operation action during an extended period of time(e.g. 36 hours in the case of the AP600). With such an option, the role of operators remains essential for preventing actuation of passive systems. If, however, the situation cannot be controlled using non safety-grade systems only, their role is significantly decreased: they are relied upon for a limited number of manual actions aiming at preventing the onset or propagation of coremelt in case passive systems fail to automatically actuate.

For more advanced designs, there could be a trend to decrease their role even further (and eventually eliminate the need for operators). Nuclear units would then be quite similar to some entirely automate chemical units with staff reduced to a minimum. In EDF's opinion, this would be detrimental to cost-effectiveness and, ultimately, to plant safety.

Whatever the simplifications contemplated in the design of future units, a nuclear generation facility will remain a complex installation, involving simultaneous operation of different systems and components interacting with each other. Though component reliability has made significant progress in the past, evidence exist that even for well proven components, malfunctions or failures cannot be excluded. In most cases, automatic action is adequate to control the reactor within its operating prescribed limits. In some cases however, complex sequences resulting for instance from degradation of the electrical supply within the plant require operator insight to limit their consequences for the utility. To take full advantage of these beneficial actions, there is a need to maintain the operator in the loop and give him a role in the conduct of day to day operation and for accident management as long as risks associated with potential human errors remain acceptably low.

This implies that operators be provided with an interface designed to facilitate quick understanding of plant status and evolution. This interface should also allow easy identification of the cause of malfunctions and provide a list of options for coping with them. At last, it should provide for easy use or actuation of systems and components needed to bring the plant into a controlled state. EDF and its partners have developed this approach for years, for example for the EPR.

### *2.2 Inspection and maintenance*

Equally important for the utility is to operate a system whose structures and components can be easily inspected and quickly maintained or repaired if needed. In deregulated markets, peak power has no real meaning for utilities. What is essential is to have the maximum of generating assets connected to the grid when needed, i.e. when retail prices are high. So, a generating company cannot afford to operate reactors where inspection techniques would be too time consuming and repairs would be too difficult. All efforts should so be made to design concepts allowing to limit the inspection and maintenance workforce to a reasonable level (i.e. at most to the current level)

Equally important for inspectability are doses to the personnel. There is a pressure from regulatory bodies to decrease allowable doses in many countries, in particular France. Once inspection time has been adjusted, the only parameter remaining for limiting personnel doses is the amount of fission products released from the core and the amount of activated products in the Reactor Coolant System. A significant effort should be made to identify those materials which could be activated either directly or if released to the reactor coolant, whether they could be released to the coolant through erosion or corrosion mechanisms for example. When proven alternatives exist for providing the same service under expected plant conditions they should be adopted if cost competitive..

### *2.3 Material behaviour*

As most designers emphasize improved plant performance through higher temperatures (and sometimes higher pressures) or through design options requiring specific coolants such as liquid metals, there could be a need to reassess whether the development and qualification of new materials having acceptable characteristics in a nuclear environment is required. Some materials have already been developed to withstand loads resulting from a high temperature or high pressure environment. However, it seems that, in most cases, service conditions were less penalizing than that contemplated for nuclear. Issues to be analyzed for such materials are compatibility with the environment, and full characterization of mechanical properties over the entire spectrum of contemplated temperatures and pressures. For structures operating at high or very high temperatures, creep could also be an issue, and creep behavior should be fully investigated, in particular when a pressure load is added to the thermal loading and when there is a potential for creep-fatigue interaction.

Once these metals have been characterized, design rules will have to be derived and validated in order to guarantee that structures as designed have the capability to withstand all loads contemplated during plant lifetime with adequate margins.

At last, manufacturing and construction procedures (e.g. welding techniques) will also have to be developed and validated to provide adequate confidence that the material is proven for use in harsh nuclear environments.

### *2.4 Wastes, decommissioning*

At last, it is important that the amount of waste generated by the new designs be limited during normal operation, and could be satisfactorily handled during the decommissioning phase. As is the case for radiation exposure, some regulatory bodies are emphasizing reducing releases to the environment, and due to the difficulty of opening new sites, the current trend is to start decommissioning as soon as possible to allow reuse of land property in a reasonably short delay. All efforts should so be made to eliminate materials which could prove extremely difficult to deal with during the decommissioning phase.

## 3. Light Water Reactors

Two types of LWRs are currently proposed for implementation in the middleor long-term:

- Small-size integrated-type reactors such as IRIS.
- Reactors using "super-critical" water.

### *3.1 Integrated type reactors*

Integrated type reactors largely building upon knowledge and operating experience feedback coming from either large commercial or smaller military units. The general orientation seems to be an improvement of plant economics through using a more compact design, and extending the fuel cycle length to increase plant availability. There are still issues to be addressed such as contemplated enrichment for fuel, integrated component reliability or dynamic behaviour of the reactor in case of transients or accidents. Some of them could well justify some demonstration tests, but at the time being, EDF has not identified specific issues needing a very large amount of R&D work for complete validation.

### *3.2 Supercritical LWRs*

For such reactors, the main design option is increasing both pressure and temperature to improve plant thermal efficiency. Though supercritical water be used in some fossil fire plants, and some data have already been collected on component behavior and thermal efficiency, addition information would be welcome on the following:

**Thermalhydraulics:** it seems there is a need for experimental data on fluid behavior under supercritical conditions in case of accident. Issues of interest are critical flow in case of break, flow regimes during the transition between sub-critical and critical flow conditions, or fuel cladding dryout in case of high heat fluxes. Also, reassessing scaling condition for test facilities and verifying that phenomena having little importance in current integral test facilities are not amplified under supercritical flow conditions could be interesting.

Special attention should also be given to numerical modelling to make sure that the transition regime does not create problems similar to that encountered in current modelling at low pressure (oscillations due to rapidly changing fluid properties)

Another specific issue to be looked at could be the mechanical behavior of overpressure protection components when discharging supercritical fluid.

**Component and structure reliability:** this could have a significant impact on plant and system design as initiating event frequencies could be modified. It is of course always possible to contemplate using alternate components in case of limited reliability, if they exist. However, the technology is not yet widespread, and the number of components capable of operating under such harsh conditions could be limited. In any case, the design basis should be reassessed for these designs

**Fuel behaviour:** fuel would have to be re-qualified for anticipated operating conditions. In particular, it should be verified that fresh fuel cladding has been adequately conditioned to accommodate the high pressure it will have to withstand at the beginning of the fuel cycle. Mechanical loadings on the cladding in case of power increase will also have to be assessed to better guarantee cladding integrity.

## 4. Gas-Cooled Reactors

For Gas-Cooled Reactors, in particular those contemplating direct cycles, the most important challenge seems to be fuel behavior during normal operation and accident conditions. This issue is of utmost importance as it could affect:

**Primary circuit contamination**. Depending on the initial number of failed particles, there will be an 'inherent' release of fission products to the Reactor Coolant System. Some of these fission products will be trapped on piping and component surfaces and result in doses to the personnel during maintenance operations. In addition to this inherent release, fuel temperature variations during transients will result in expansion and contraction of the fuel particles. This could mechanically affect the structural strength of the first coating layer for example through the initiation of cracks. At last, the behavior of some fission products such as silver or palladium, which could migrate through the various coatings layers for the former, react with silicon carbide for the latter will need better understanding to allow further quantification of personnel doses.

Another consequence of this potential contamination, is activity release during normal operation, and the type of protective measures needed to address this issue. For direct cycle reactors in particular, it is to be expected that there will be a permanent leak of helium to the reactor pit or the turbine pit. Associated doses to maintenance personnel will have to be evaluated and countermeasures provided, as worker irradiation could become a very contentious issue with regulatory bodies, and result in loss of cost-effectiveness for the utility (extended delays for maintenance, shortage in maintenance work force).

At last, there could be a double impact on fuel cost if fuel manufacturing techniques needed some upgrading for guaranteeing adequate fuel quality. The first one would be the direct cost impact of more severe control procedures. The second could result from the reduction of the number of fuel manufacturers willing to compete in the market.

**Releases during accidents.** Up to now, most designs have adopted a low power density fuel. This allows limiting the maximum fuel temperature below a threshold corresponding to no coating failure during accidents. Further experimental evidence of excellent fuel behavior is probably needed using ''industrial type'' fuel elements, i.e. whose quality would correspond to that obtained in industrial processes implemented for supplying tens of reactors. Moreover, fuel behavior should also be studied beyond the current threshold to show that no cliff-edge effect is to be anticipated if this temperature limit is exceeded.

**System architecture and component classification.** As low power density and limited output could be detrimental to plant economics, there is a need to reassess the safety value of all components to try and reduce plant overheads. Depending on the assumptions made on fuel behavior, the number and type of components and systems which need to be safety-related could be dramatically reduced. In some cases, this could turn out to be a real barrier for implementation for some utilities: in most countries, utilities are responsible for plant safety and are used to relying on safety-related systems and redundant distributed barriers for mitigating accident consequences and preventing fission product release to the environment. Substantial guarantee on intrinsic fuel robustness and quality of the fuel manufacturing process should thus be provided to overcome this potential problem.

A second important issue is **graphite behaviour**. Impurities in the graphite matrix could have a non negligible impact on core reactivity and thus govern the design of the plant control rod system. Industrial development of the concept thus implies adequate knowledge of required material specifications allowing to guarantee an acceptable core design.

In addition graphite has exhibited specific mechanical behavior due to its anisotropic structure in some gas-cooled reactors. It should be verified that experimental evidence available to date allow to cover all operating conditions contemplated for the next generation of reactors, in particular those using Helium as a coolant. A significant effort should be made, in this perspective, to consolidate all available data on graphite corrosion at high temperature. Considering the temperature level contemplated in Helium cooled High Temperature Reactors, even traces of impurities in Helium (e.g. $O_2$, $H_2O$) could be corrosion initiators and result in corrosion assisted mechanical problems. Potential consequences could be valve leakage (valve seats having a tendency to behave as cold traps for impurities), and graphite shroud degradation, which could create problems for vessel mechanical design and plant availability (e.g. shroud replacement).

A third issue has to do with **plant thermalhydraulics behavior**. The limited specific heat of the coolant results in the need to operate HTRs with significant temperature differences between core inlet and outlet. In case of reactor transient, the amplitude of thermal cycles will be very high and could pause a significant problem when designing structures. This problem could be further increased if materials had to operate in temperatures ranges where creep damage could be significant. At last the top-down circulation of the coolant in the core during normal operation could lead to complex flows during some transients, in particular those resulting from a loss of forced convection flow.

Attention should also be paid to the behavior of the reactor pit during normal operation. Though the vessel be thermally protected, its normal operating temperature remains high and heat losses to the reactor pit have to be expected. This should result in complex natural convection flows in the reactor pit and a transfer of energy to the upper parts of the vessel and the reactor pit closure. The temperature field in these areas should be carefully evaluated as this could result in malfunctions of some components (including control rod drive mechanisms) if not adequately evaluated.

In EDF's opinion, so, the capability of currently available models to adequately simulate all transient or steady-state situations should be assessed. In particular, it should be checked that the detailed physical models implemented in the computer codes are consistent with the anticipated result accuracy.

At last, there seems to be a need for further clarification of the **safety assessment process**. It is obvious that gas-cooled reactors (and liquid-metal reactors) exhibit specific characteristics compared to LWRs, and that the safety approach applicable to LWRs should be modified to reflect HTR specifics and eliminate LWR ones. At the time being, however, it seems that emphasis should be put on the following to minimize regulatory uncertainty:

- Definition of the Design Basis: should it be based on deterministic principles, probabilistic objectives, or a combination of both? In particular, external events to be addressed in the design should be discussed as this could influence important design decisions such as the need for containment.
- Definition of Severe Accidents, whether they should be considered in the design process, and, in case yes, how. It seems that the definition given for LWRs, in which coremelt is emphasized, has also been adopted in some cases for HTRs where the core is not likely to melt. This could be misleading as the real issue is fission product release to the environment, which could be quite significant even in the absence of coremelt. Issues to be discussed include:
  - the definition of Source Term(s), including release timings, release fractions, and chemical speciation if relevant;
  - fission product behavior inside buildings and release paths to the environment;
  - mitigative devices and their safety pedigree;
  - need for and nature of Emergency Planning provisions.

Only the first of the four above-mentioned issues might need to be addressed through further R&D programs

- confirmation of probabilistic safety objectives adopted for other types of reactors (e.g. INSAG3). As compliance with these objectives would have to be demonstrated, further R&D allowing to assess component reliability or initiating event frequency (e.g. for piping failure) could well be needed

## 5. Liquid Metal Reactors

Liquid Metal Reactors (LMRs) are better known to EDF, at least those using sodium as a coolant. Amongst the issues which would have to be carefully studied, the following could be emphasized:

**Coolant melting and transfer procedures:** this has been a very delicate and time consuming process in the case of Superphénix. It is to be expected that similar difficulties would have to be addressed if other coolants such as lead were to be used. The following problems would have to be solved, and adequate procedures validated when appropriate: thermal conditioning of storage tanks, pipings components and reactor vessel, melting procedure, coolant transfer to the reactor vessel and the primary system, keeping the coolant liquid in the absence of nuclear heating.

**Coolant melting temperature:** melting temperature could be adversely affected by the presence of impurities. The influence of impurities should be studied, and instrumentation allowing to adequately monitor the evolution of impurity content in the coolant developed and validated. If

eutectics were contemplated (e.g. Pb-Bi mixtures), the influence of chemical additives on eutectics equilibrium should also be assessed.

**Inspection of structures:** this has already been mentioned as a crosscut issue, but one specific issue must be stressed for liquid metal. The coolant being opaque in any case, and emptying the reactor is practically excluded, some components or structures will have to be inspected using specific techniques. These techniques should be validated to show that signals remain accurate in case of temperature gradients or impurities in the coolant (e.g. traces of gas significantly modify sound velocity in sodium).

**Thermalhydraulics:** many liquid metals require high temperature differences between core inlet and outlet for core heat removal. This could lead to the existence of complex flows, in the vessel or in pipings. Fluid stratification could exist for both steady-state and transient conditions, leading to some difficulties for designing structures, in particular near the transition between hot and cold fluid. At last, these complex flows could influence plant global behavior, modifying the timing and amplitude of temperature evolution during transients. The development of sodium cooled LMRs in France had led to the development of sophisticated computer codes and implementation of an extensive test program.

Similar developments are to be expected for adequate understanding of plant behavior and design of structures. It is also to be noted that diffusion heat transfer is not negligible for some liquid metals, and that special attention has to be paid to scaling for experimental programs, and to numerical modelling for computer codes (minimize numerical diffusion).

**Coolant interaction** with the fuel, air or water also needs careful assessment as this could result in severe loadings on structures, including buildings) and components. Additional development could be needed to better evaluate subsequent loadings.

**Decommissioning:** this is also a crosscut issue, but specific problems to be looked at are how coolant temperature can be maintained during decommissioning operations, and which specific provisions have to be implemented for coolant disposal.

## 6. Conclusion

EDF is currently evaluating small reactors contemplated for implementation for 2020 and beyond. To be a credible alternative for electricity companies, convincing evidence showing that these new designs can be considered proven will have to be provided. Moreover, it should also be shown that utility needs and concerns are adequately addressed.

At the time being, EDF thinks that additional R&D could be needed on the following:

- Thermalhydraulics for LMRs, HTRS and Supercritical LWRs.
- Fuel behavior under both normal operating conditions and accident situations including severe core degradation.
- Qualification of materials under harsh conditions.

Other issues could exist and emerge when the company has collected more information on these designs.

# CURRENT AND FUTURE RESEARCH ON CORROSION AND THERMALHYDRAULIC ISSUES OF HLM COOLED REACTORS AND ON LMR FUELS FOR FAST REACTOR SYSTEMS

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## Abstract

Heavy liquid metals (HLM) such as lead (Pb) or lead-bismuth eutectic (Pb-Bi) are currently investigated world-wide as coolant for nuclear power reactors and for accelerator driven systems (ADS). Besides the advantages of HLM as coolant and spallation material, e.g. high boiling point, low reactivity with water and air and a high neutron yield, some technological issues, such as high corrosion effects in contact with steels and thermalhydraulic characteristics, need further experimental investigations and physical model improvements and validations.

The paper describes some typical HLM cooled reactor designs, which are currently considered, and outlines the technological challenges related to corrosion, thermalhydraulic and fuel issues.

In the first part of the presentation, the status of presently operated or planned test facilities related to corrosion and thermalhydraulic questions will be discussed. First approaches to solve the corrosion problem will be given. The approach to understand and model thermalhydraulic issues such as heat transfer, turbulence, two-phase flow and instrumentation will be outlined.

In the second part of the presentation, an overview will be given of the advanced fuel types that are being considered for future liquid metal reactor (LMR) systems. Advantages and disadvantages will be discussed in relation to fabrication technology and fuel cycle considerations. For the latter, special attention will be given to the partitioning and transmutation potential. Metal, oxide and nitride fuel materials will be discussed in different fuel forms and packings.

For both parts of the presentation, an overview of existing co-operations and networks will be given and the needs for future research work will be identified.

## 1. HLM cooled reactor designs

During recent years, several heavy liquid metal (HLM) cooled reactor designs have been proposed: fission reactors, breeder reactors and accelerator driven systems (ADS). Here, only the designs using lead (Pb) or lead-bismuth eutectic (Pb-Bi) as coolant will be considered.

The main reason to use a lead alloy are the neutron-physical characteristics, such as for example the low absoption cross sections both for thermal and for fast neutrons and the high neutron gain of the spallation reaction. Due to the very low molecular Prandtl number of lead alloys that is two orders of magnitude smaller compared to water, lead alloys show excellent heat transport capacities which allows their applicability as coolant even with high surface heat fluxes present. The shielding characteristics of lead alloys in comparison to for example sodium are simplifying the construction of a reactor significantly.

In comparison to other liquid metals such as sodium, lead alloys show great advantages in the area of system safety as highly exothermic reactions such as burning in contact with ambient, humid air or water can be excluded. The thermodynamic properties of lead alloys such as vaporisation point, evaporation rate and saturation pressure at prototypic temperatures are much more favourable in comparison to sodium.

Disadvantages when using lead alloys are the high density that results in high dynamic pressure losses and that requires a more robust construction in order to cope with earthquakes. Another disadvantage is the production of polonium during the spallation reaction, especially in the presence of bismuth. The most severe disadvantage of lead alloys is the liquid metal corrosion of structure materials such as steels that are in contact with the liquid metal: the corrosion rates can be in the order of several millimeters per year at temperatures above 500°C, if no adequate precautions are taken as is proposed by [1].

- Pre-oxidation of the structure material in air prior to contact with the liquid lead alloy.
- Corrosion resistant coating on the surface of the structure material based on silicium or aluminium.
- Possibly surface modification and alloying (Si or Al) by using a pulsed electron beam.
- Application of an oxygen control system to control the oxygen potential in the liquid lead alloy.

First Pb-Bi cooled reactors were operated by Russia in their Alpha class nuclear submarines, [2]. Russia has gathered an enormous amount of practical experience in Pb-Bi technology, which resulted in the design of small-size transportable units (NPP Angstrem and SVBR-75), medium-size reactors (BRUS series) and large-size units (BREST series). An overview on the Russian work is given in [3] and in the proceedings of the HLMC-98 conference “Heavy Liquid Metal Coolants in Nuclear Technology” held in Obninsk in 1998.

Word-wide, several HLM cooled reactor designs are proposed and investigated: in the US, within Generation-IV, an encapsulated nuclear heat source (ENHS) reactor is proposed which is extremely proliferation-resistant and which solely uses natural circulation features for heat removal [4]. Another US project designs a low cost electricity producing as well as actinide burning HLM cooled reactor, presently focussing on the key technical issues of core neutronics, materials, thermal-hydraulics, fuels, and economics [5]. In Japan, a feasibility and strategy study on fast reactor cycle

systems is under way, evaluating – among others – small-size, medium-size and large-size HLM cooled reactor designs of pool-type and loop-type, [6].

Another field of application of HLM coolants are the ADS systems which are dedicated to nuclear waste transmutation, thus reducing the long-term radiotoxicity of the nuclear waste. In Europe these research and development activities are co-ordinated within the ADOPT thematic network (ADvanced Options for P&T) of the Euratom FP5 ($5^{th}$ Framework Programme), [7, 8]. These activities are based on the "European Roadmap for Developing ADS for Nuclear Waste Incineration", [9]. One optional ADS design which is investigated in more detail in the PDS-XADS Project (Preliminary Design Study of an eXperimental ADS) of FP5, is a pool type system that uses Pb-Bi as coolant in the primary system and Pb-Bi as coolant and spallation material in the spallation target. In figure 1 a sketch of this optional design, developed by [10], is given.

Research and development work into corrosion and thermalhydraulics of lead alloys are done in three FP5 projects: TECLA being "TEChnologies, materials and thermalhydraulics of Lead Alloys", and SPIRE being "SPallation and IRradiation Effects in martensitic steels under neutron and proton mixed spectrum". A third more application oriented project is MEGAPIE-TEST (MEGAwatt PIlot Experiment-TESTing), the objective of which is to develop, test and operate a liquid lead-bismuth cooled spallation target of 1 MW of proton beam power in the SINQ accelerator complex at PSI, [7].

Similar activities into ADS systems and thus the development of HLM technologies are summarized in "A Roadmap for Developing Accelerator Transmutation of Waste Technology", [11], and in the AAA Project (Advanced Accelerator Applications) [12] for the USA, in the Omega Project [13] for Japan, and in the HYPER Project [14] for the Republic of Korea.

Currently, various ISTC Projects (International Science and Technology Center), [15], are under way that stimulate a close cooperation and information exchange between Russian scientists and western laboratories. Two important ongoing projects are ISTC 2048, investigating the improvement of corrosion resistance of steels by surface modification and protective coatings up to a temperature of 650°C, and ISTC 559, developing and testing a spallation target of 1 MW of proton beam power.

All these HLM cooled reactor designs have some major technological challenges in common which still need further research and development work. These are fuel (composition, fabrication and reprocessing), core design (neutronics, kinetics, performance, fuel integrity, subcriticality with ADS), heat removal (forced convection/natural circulation heat transfer, thermal cycling, coolant characteristics), liquid metal (corrosion and erosion, purification and oxygen control system, online instrumentation) and in case of an ADS the target module (thermalhydraulics, materials compatibility, beam window loads, maintainability).

This paper concentrates on corrosion / materials, thermalhydraulics and fuel issues only.

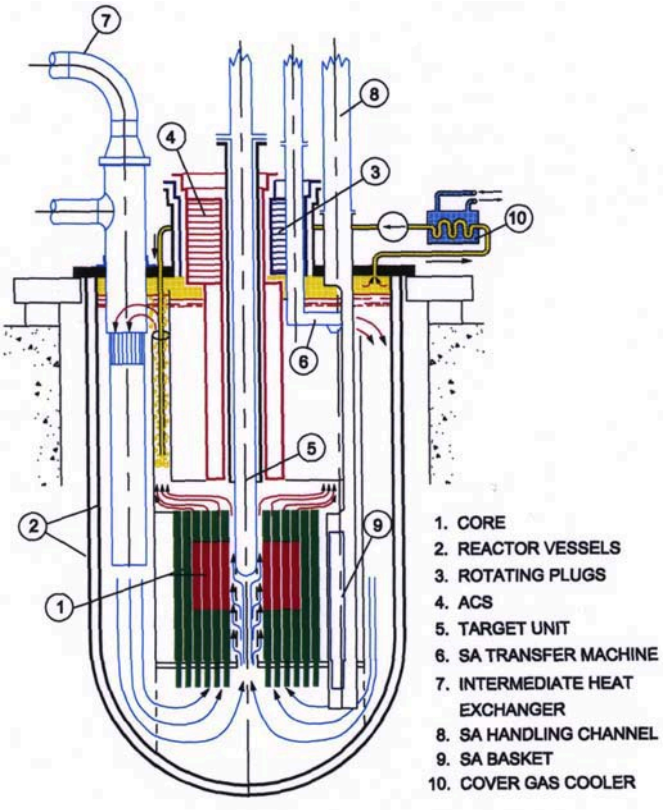


Figure 1. **Sketch of the XADS Pb-Bi Cooled Experimental Accelerator Driven System from [10] with lead-bismuth as coolant and spallation material**.

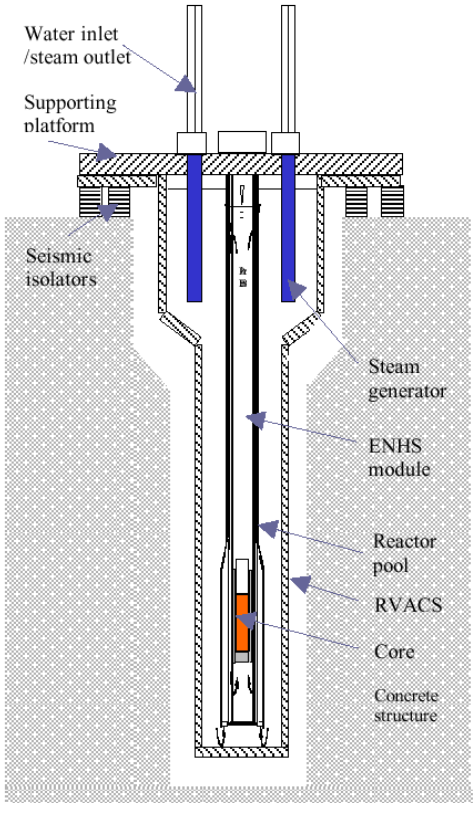


Figure 2**. Sketch of the encapsulated nuclear heat source (ENHS) reactor from [4] with lead-bismuth as coolant.**

## 3. Current research issues: materials and thermalhydraulics

### *3.1 Corrosion and materials*

It is well known from literature that lead and lead-bismuth have a high corrosion potential relative to steels that are used as structural material. The fundamental mechanism of this physical-chemical interaction (liquid metal corrosion) between the structure material and the liquid metal is characterised by:

- Solution of some metal components of the structure material, e.g. nickel, in the liquid metal.
- Mass transport of structure material within the system due to temperature gradients, e.g. dissolution in the hot part and precipitation in the cold part of the loop.

- Change in the structure and the morphology of the surface.
- Influence on the mechanical properties of the structure material.
- Reaction of the structure material with non-metals that are dissolved in the liquid metal, e.g. oxygen, and its influence on the long-term stability.

In general, austenitic steels with a high amount of nickel can be severely corroded as long as they are unprotected. Ferritic and low-alloyed steels, however, show a much more favourable corrosion resistance. For tungsten and tungsten-rhenium, which are optional materials for the beam window of an ADS, only little corrosion is expected due to their low solubility in lead alloys.

The strategy and the objectives of future research and development must be to elaborate the scientific-technical fundamentals of the corrosion and erosion behaviour of metallic structural materials that are in contact with flowing lead alloys. The main interest has to be focused on, first, nickel-free, ferritic-martensitic steels of the 12% Cr-type and, second, due to the high temperature strength, on the high chromium/high nickel austenitic steels. As a result the following issues should be addressed in more detail for a wide range of structural materials and, in case of an ADS, beam window materials:

- Corrosion of steels in lead and lead-bismuth:
  Extensive corrosion tests in stagnant and flowing Pb or Pb-Bi have to be performed to characterise the materials and to evaluate the importance of various parameters involved in the corrosion process such as temperature, flow rate, surface treatment, chemistry of coolant, spallation products. The kinetics of the corrosion process and the corrosion rate have to be determined for unwelded and welded materials. The parameter range for temperature has to be extended up to 650°C, the liquid metal velocity has to be as high as 3 m/s in order to cover extreme operational values of a HLM cooled reactor.
- Protection mechanisms and corrosion resistance enhancement:
  There are three protection mechanisms which are currently investigated to enhance corrosion resistance: in-situ protection by formation of an oxide layer on the structure material and interaction between oxygen, PbO and structure material, protection by coatings, surface restructuring and alloying (e.g. aluminium or silicium forming oxidic corrosion barriers), and protection by addition of inhibitors.
- Mechanical behaviour of steels being in contact with lead or lead-bismuth:
  The mechanical characterisation of various steels has to be investigated in tensile, creep and liquid metal embrittlement tests. In parallel, modelling approaches under a wide range of parameters such as temperature, flow velocity, liquid metal chemistry have to be addressed. Description of intergranular penetration effects. Setup of a thermo-dynamic database for multi-constituent systems based on Pb-X and Pb-Bi-X where X is a transmutation product.
- Irradiation effects on mechanical properties:
  Assessment of the effect of neutron / proton irradiation on mechanical properties of structure materials at temperatures below 400°C. Investigation of long-term evolution of mechanical properties and of loss of dimensional stability due to a possible onset of swelling above 400°C.

- Irradiation effect on liquid metal corrosion:
  The evaluation of the neutron / proton irradiation effects on corrosion and protection mechanisms of steels and the effects on liquid metal embrittlement have to be done under prototypical conditions.
- Effects of spallation elements (in case of ADS):
  Assessment of the influence of spallation elements on the physical metallurgy, microstructure and mechanical properties of structure materials.
- Purification system and oxygen control system:
  Characterisation of impurities, sampling systems and analytical methods, and qualification of purification systems. Development of an active oxygen control system for prototypical applications which incorporates oxygen sensors and calibration instructions.

### *3.2 Thermalhydraulics*

The thermalhydraulic design of a HLM cooled reactor basically requires three tools:

- A numerical code with physical models which are validated for reactor relevant geometries and conditions.
- Single-effect and integral effects model experiments for fundamental flow configurations and reactor relevant geometries, in order to benchmark the numerical codes.
- Measurement techniques.

In general, computational fluid dynamic (CFD) codes are lacking validation, especially in the fields of turbulence, buoyant flows, free surface and two-phase flow, for heavy liquid metal applications.

The physical description of a liquid metal flow, in principle, does not differ from that of other Newtonian fluids, as long as the correct physical properties are used in the conservation equations. However, in standard CFD codes, turbulence models are implemented that are not suitable to describe buoyant turbulent HLM flows: these models assume the Reynolds analogy, i.e. the turbulent transport of heat is coupled with the turbulent transport of momentum using a constant turbulent Prandtl number. This approach is not valid for liquid metal flow; the influence of the molecular Prandtl number on the heat transfer is not modelled correctly. More advanced models such as the low-Reynolds number k-ε model, the TMBF (turbulence model for buoyant flows) model or second-order heat flux models, which explicitely model the turbulent transport of heat, need a number of model coefficients. The values of these coefficients have to be determined by both dimensional and empirical analysis for low Prandtl number fluids and flow configurations, and they have to be validated in benchmark experiments that provide a data base of both mean and turbulent quantities. Moreover, new model relationships are necessary to extend the validity of the modelled equations to low Peclet numbers.

For the case of free surface flow, which is inherent in the windowless design of an ADS spallation target, the modelling of the cone-like shape of the free surface between the liquid metal and the evacuated beam pipe, the stagnation point, and free surface instabilities (resulting in droplet formation) have to be described, in order to be able to predict the position of the free surface and

possible boiling / evaporation phenomena of the HLM. The position of the free surface is important for the location of the spallation area and thus of the external neutron source.

Finally, two-phase flow phenomena such as bubbly flow have to be both experimentally investigated and modeled for HLM in order to correctly design gas-assisted natural circulation cooling as is foreseen in reactor designs of [4] and [10]. When applying a two-fluid model, great emphasis has to be given to the modeling of the interfacial forces and to the transport equation of the interfacial area concentration, and to the experimental validation of the coefficients.

Besides the experiments, the development of measurement techniques for HLM under prototypical conditions (especially high temperature and irradiation) has to be intensified: at present, there is a variety of existing technologies, however, all of them have to be adapted to HLM flow. Most important is the development of pressure gauges, flowmeters (magnetic, ultrasonic and turbine principle), wall heat flux (e.g. heat emitting temperature sensing surface or thermocouples), void (impedance and conductivity probes), and velocity probes (permanent magnet flowmeters). Another promising non-intrusive velocity measurement technique is the ultrasonic Doppler velocimetry (UDV) which urgently needs the development of high temperature sensors and coupling devices (waveguides).

Within the concerted action ASCHLIM of FP5 [7] an assessment of the state of the art CFD tools is made on the basis of existing and planned experiments in liquid metals, which address the fields mentioned above. In order to continue and strengthen this effort an international data base of HLM experiments has to be set-up. This data base can be used to improve and validate physical models and CFD codes.

Typical flow configurations and geometries which still have to be experimentally investigated under well-defined chemical and material specific boundary conditions are:

- Heat transfer along thermally highly-loaded surfaces (e.g. pipe flow, channel flow).
- Flow mixing under forced flow, transition and buoyant flow conditions.
- Bubbly two-phase flow in a vertical pipe.
- Flow field and heat transfer in a fuel element.
- Flow field and heat transfer in heat exchangers.

The experiments should consist both of small-scale tests to address fundamental questions and of real-size tests to demonstrate application oriented problems.

### *3.3 Test facilities*

Within the projects and programmes mentioned in chapter 2 a large variety of test facilities are either operated or are being built. Most of the test facilities are located in Russia: there are corrosion loops (e.g. IPPE Obninsk, CRISM Prometey Saint Petersburg) designed to operate at temperatures of up to 650°C, at flow velocities below 2 m/s and various oxygen levels, and thermalhydraulic loops (e.g. IPPE Obninsk) designed to operate at temperatures of up to 520°C, at electrical powers up to 8 MW under forced (maximum flowrate of 120 m³/h) or natural circulation, [16]. During the last few years, similar test facilities were erected and are operated in western countries. The largest efforts are the following:

- Lead-bismuth eutectic Materials Test Loop (MTL) at LANL, dedicated to investigate corrosion and natural circulation flow phenomena.

- KArlsruhe Lead LAboratory KALLA at Forschungszentrum Karlsruhe, [17], dedicated to do technology development (oxygen control system, measurement techniques, fundamental heat transfer and turbulence measurements) in THESYS Loop, to do corrosion experiments in CORRIDA Loop, and to do thermalhydraulic experiments in THEADES Loop.
- LECOR Loop, dedicated to corrosion, purification and mechanical properties tests, and CHEOPE Loop, dedicated to Pb-Bi chemistry and technologies, at ENEA.
- CIRCE Loop at ANSALDO Nucleare, dedicated to do large-scale natural circulation and two-phase flow experiments for ADS relevant geometries.
- LiSoR Loop (Liquid / Solid interaction under Radiation) at PSI in Switzerland.
- Various small-scale corrosion and/or thermalhydraulic loops at CEA and CNRS in France, CIEMAT in Spain, Tokyo Institute of Technology and JAERI in Japan.

## 4. Current research issues: fuels

### *4.1 Introductory remarks*

The fuels for future fast reactor systems that are studied at present are generally based on successful developments from the past (metal fuel or oxide fuel) and as much as possible incorporate elements from the Partitioning and Transmutation research programmes that are being conducted since the last 10 years. The focus is at present on metal, nitride and oxide fuels which all have their advantages and disadvantages. It is too early to rule out or select any of these materials, also because the lay-out of the future generation fast reactors is not yet clearly defined. Though many aspects are of interest, the coolant (gas or liquid metal) is a key issue. Also the fuel strategy is very important for the fuel design, as it can have an influence of the amount of actinides recycled per element. In a dual stratum strategy, in which special actinide burners are used, the concentration of minor actinides in the fuel is always high. In a single stratum cycle, in which the minor actinides are recycled together with plutonium in power reactors, the quantities in the fuels are more likely smaller, although in such a scenario also high content targets can be foreseen. In addition, there is a trend of using uranium-free fuels for transmutation of actinides to avoid production of new actinides during the transmutation process. This has opened a complete new field of inert matrix fuels.

In recent years several reviews of advanced reactor systems have been made and a well-documented collection of research topics can be found in the proceedings of the OECD-NEA Workshops of Advanced Reactors with Innovative Fuels (ARWIF) [18,19]. A more specific study of fuel aspects was made for the “European Roadmap for Developing ADS for Nuclear Waste Incineration” [9,20], in which the current development for fuels for transmutation are discussed.

In the present paper the advantages and disadvantages of the most promising fuel forms will be briefly described: metals, nitrides and oxides; a more extensive discussion can be found in [21]. These three fuel forms are currently under investigation in Europe, USA, Japan and Russia.

## *4.2 Review of most important fuel forms*

### *4.2.1 Metal fuels*

Metal fuels have interesting thermal and mechanical properties, in combination with a high minor actinide (MA) content. However, U metal fuels showed significant swelling at relatively low burn-up as a result of anisotropic growth. Research in the USA has shown that this could be reduced by addition of zirconium. With increasing burn-up, also the accumulation of fission gases and, in case of minor actinides, helium in bubbles contributes to the swelling. At a fuel swelling of about 30%, the gas bubbles in the fuel start to interconnect to form paths for release. Therefore the swelling can be accommodated by increasing the space between the fuel and the cladding (lowering the smeared density), in combination with the use of a metal bond (Na or Pb), and increasing the plenum. This aspect makes metal fuel very interesting. But metal fuel has some serious drawbacks when containing high amounts of minor actinides and no uranium:

- The melting point of the fuel decreases significantly and addition of a non-fissile metal in a significant quantity is required. Zirconium seems to be the most likely candidate but there is considerable uncertainty about the (mutual) solubility of neptunium and zirconium and the maximum amount of zirconium that can be added to the alloy.
- The volatility of especially Am metal, which is quite high, will complicate fabrication and may lead to unwanted re-distribution during irradiation.

Research on these topics is mainly conducted in the USA. The US researchers proposed a metal-metal composite with Zr as matrix to deal with the compositional problems and a trapping and recycling of Am metal to deal with the Am volatility [11], but the experimental and technological realisation of this must be proven.

### *4.2.2 Nitride fuels and targets*

The actinide nitrides show a mutual miscibility in a wide range of compositions. They have a good thermal conductivity and a high melting point thus resulting in a large margin to melting. Nitride fuel is also compatible with sodium bonding. These properties allow low-smeared density fuel designs to accommodate swelling, like metal fuel. For uranium-free fuels ZrN can be used as matrix. However, nitride fuel requires enrichment of nitrogen in $^{15}N$ to avoid the $^{14}C$ production, which makes the fabrication process more demanding and has an impact on the fuel cycle costs. In addition to this these are significant technical drawbacks of nitride fuel with high concentrations of minor actinides:

- The Am vapour pressure over AmN is predicted to be high, which will complicate fabrication (carbothermic reduction requires high temperatures) and may lead to unwanted re-distribution during irradiation.
- At high temperature that can occur during accident conditions, the actinide nitrides will dissociate in $N_2$ gas and actinide metal (which melt or volatilise at low temperature).

The nitride fuel options is studied already for a long time in Japan and the researchers from JAERI have fabricated and measured the properties of many actinide nitride compounds [22]. Also in US, Russia and Europe research programs for nitride fuels have been started in which the above mentioned drawbacks are being addressed.

### *4.2.3 Oxide fuels*

Mixed transuranium oxide fuel is a logical extension of the current MOX fuel technology since the oxides of the minor actinides Np, Am and Cm can be dissolved in the fluorite lattice of $UO_2$ (or $ThO_2$). The SUPERFACT experiment, performed in the Phoenix Reactor in the 1990s, has shown that this type of fuel behaves well during irradiation, though the burnup in this experiment was relatively low. When the fuel must be fertile-free, zirconium oxide can be used as a matrix. Because oxide fuel operates at relatively high temperature due to the poor thermal conductivity, it can be expected that mixed oxide fuel with high concentrations of minor actinides has a higher operating temperature than $UO_2$ or MOX fuel. Since the melting points of the actinide oxides decrease along the actinide series, the margin to melting will be small if operated with high linear power. As a result the smeared density of the fuel must be high, allowing only for little swelling of the fuel. This forms a major disadvantage of this fuel type. This can be overcome in composite fuel designs in which the fissile actinide phase is mixed with a good conducting matrix which can be a metal or a ceramic.

## *4.3 Ongoing European research programmes, research needs and test facilities*

An inventory of the research needs and test facilities was made in the frame of the European Roadmap for ADS [9], the conclusions of which equally apply to fuels for critical fast reactors. It was concluded that:

- Specific irradiation experiments are needed in combination with a dedicated out-of-pile programme on the determination of the basic properties of mixed oxide fuels (with and without matrix) and a in-depth safety analysis should establish the limits of the use of this fuel form. The European FUTURE (FUels for Transmutation of transURanium Elements) project in FP5 was initiated to start such an activity in the coming years, [7]. Parallel to that, the activities of the EFTTRA collaboration will contribute to this goal [23].
- Nitride fuel must be considered as back-up solution. Because MA-containing nitride fuel is studied extensively in Japan and Russia, it should not become a major topic of the European research. It is recommended to seek a collaboration with the Japanese and Russian activities in this field. The CONFIRM (Collaboration on Oxide and Nitride Fuel IRradiation and Modelling) project in FP5 (irradiation of (Zr,Pu)N fuel) should be the starting point for such a collaboration.

The facilities needed to perform such programmes are limited in Europe: there are only two fabrication laboratories able to handle americium and curium for research and development on fuels and fuel processing: ITU (JRC, Karlsruhe) and Atalante (CEA, Marcoule). For the testing of the irradiation behaviour of fuels and targets for transmutation, no fast reactor will be available in the EU after 2004-2006. Taking also into account the increasing complexity of transports of nuclear materials, the outlook is not positive and actions need to be taken.

# 5. Summary

Heavy liquid metals such as lead or lead-bismuth eutectic are currently investigated world-wide as coolant for nuclear power reactors and for accelerator driven systems. In Russia an extensive practical experience exists in operating lead-bismuth cooled small-scale reactors and a large number of test facilities.

However, and this holds especially for western countries, corrosion and thermalhydraulic issues still need a comprehensive research and development programme to solve remaining questions such as:

- Corrosion mechanisms and mechanical behaviour of steels in contact with lead and lead-bismuth.
- Protection mechanisms and corrosion resistance enhancement.
- Irradiation effects on mechanical properties and on liquid metal corrosion.
- Effects of spallation elements (in case of ADS).
- Purification system and oxygen control system.
- Thermalhydraulic measurement techniques.
- Well-instrumented model experiments on turbulence, buoyant flows, free surface, two-phase flow.
- Improvement of physical models and validation of CFD codes.

Oxide and nitride fuels are considered as main candidates for fast reactors in Europe. A dedicated fuel development programme has been started but it must be stressed that the research infrastructure (laboratories that can handle minor actinides in significant quantities, irradiation facilities) must be strengthened.

The co-ordination of these research and development activities is embedded in various national and international programmes. In order to make the best benefit possible out of these programmes and in order to avoid doubling of work, all results should be collected and assessed in one common data bank.

Within the ADOPT thematic network the effort is currently under way to establish a continuously updated internet homepage (http://www3.sckcen.be/adopt/) for the compilation and exchange of the activities of the partitioning and transmutation community.

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# CURRENT AND FUTURE RESEARCH ON PASSIVE SAFETY DEVICES AND ADVANCED FUELS FOR INNOVATIVE LWRs

## Part 1: Research on Passive Safety Devices

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## 1. Introduction

To verify the effectiveness and capacities of the passive safety systems of the SWR 1000 they must be tested under conditions which are as realistic as possible – for this is the only way to confirm that they will, on demand, successfully perform the functions for which they have been designed.

The tests carried out for the passive safety systems, together with their results, are described in the following.

## 2. Tests performed on the passive systems and components for the SWR 1000

### *Emergency condenser (EC)*

In order to investigate the conditions for the Emergency Condenser experimentally, the NOKO test facility was erected at the Jülich Research Center (NOKO is a German acronym for "emergency condenser"). Eight condenser tubes having a geometry identical to those of the ECs designed for the SWR 1000 were installed in the test facility, although only four of these tubes were normally used in the tests. System pressures of between 3 and 70 bar were employed.

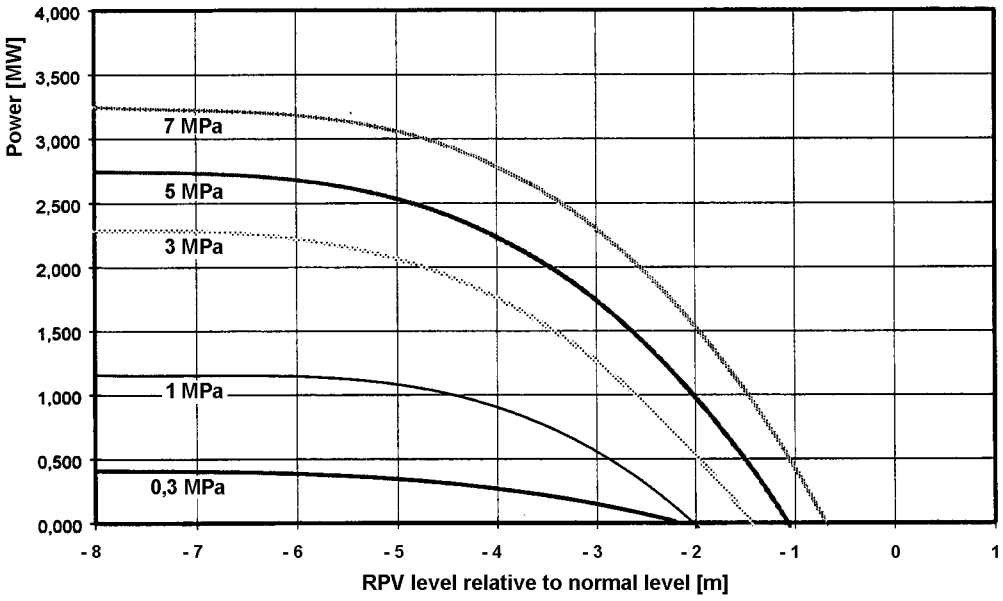
Figure 1 shows heat transfer as a function of reactor water level and system pressure. The maximum amount of heat that could be removed by four condenser tubes was approximately 3.25 MW. When system pressure is reduced, the rate of heat transfer decreases much more slowly than the pressure.

The tests did not reveal any instabilities. The heat transfer rate is dependent to only a small degree on the temperature of the cooling water outside the condenser tubes since nucleate boiling is the predominant mode of heat transfer in the boundary layer on the exterior of the tubes. Hence,

regardless of the temperature in the core flooding pool, the temperature prevailing on the exterior of the condenser tubes will only be slightly above the saturation temperature of around 110°C.

Moreover, the heat fluxes determined in post-test calculations performed using computer codes (e.g. ATHLET) were found to agree quite well with the measured heat fluxes [1].

Figure1. **Heat flux through four condensation tubes of emergency condenser at NOKO test facility as a function of reactor water level and system pressure**




The EC is thus an entirely passive component. Failure of an EC is only conceivable as the result of a pipe or tube break or blockage. However, these failure mechanisms can be avoided through regular in-service inspections.

### *Containment cooling condenser (CCC)*

The CCC is a tubular heat exchanger designed to remove heat from the drywell as soon as the temperature of the containment atmosphere rises above the normal value. Finned tubes were originally planned for the condenser to enhance heat transfer on the outside of the tubes. The condenser tubes are hydraulically connected to the water inventory of the shielding/storage pool via non-isolatable lines.

Assessment of the CCC's heat transfer performance is made difficult by the fact that, depending on the type of accident that has occurred, the fluids present on the primary side (drywell) of the CCC as well as on its secondary side (condenser tubes) may vary considerably. In the event of an accident not involving loss of coolant, the fluid on the primary side will initially be nitrogen. As the temperature of the water in the core flooding pool rises and the water starts to evaporate, the partial pressure of the steam in the drywell atmosphere steadily increases over a period of several hours until the condenser is surrounded by an atmosphere of pure steam.

In the event of a loss-of-coolant accident (LOCA), the CCC may be surrounded by an atmosphere predominantly comprised of steam after only a few seconds or minutes, depending on the size of the break. In the event of a severe accident involving core melt, a steam-hydrogen mixture may be released into the containment atmosphere.

The release of hydrogen to the drywell atmosphere causes a reversal in the direction of flow on the primary side of the CCC. When a containment atmosphere comprising a mixture of nitrogen and saturated steam is cooled, the specific density increases and the cooler mixture flows downwards (the standard case in physics). The opposite occurs, however, with a mixture of saturated steam and hydrogen. The cooler steam-hydrogen mixture therefore rises.

Integral system tests performed with all of the various boundary conditions described above were carried out at the PANDA test facility of the Paul Scherrer Institute in Switzerland. In this test facility the volumes of the SWR 1000 containment are modeled to a scale of approximately 1:25, with the elevations being largely the same as those in the actual SWR 1000 design. The condenser was equipped with the same finned tubes as originally planned for the SWR 1000, except that they were shortened from 4 m to 3 m so that they would fit inside the PANDA test vessel used to model the containment.

The test program comprised six tests in which various types of accidents were simulated. In two of the tests the release of hydrogen from the RPV (severe accident scenario) was simulated by feeding large quantities of helium into the test facility along with the steam.

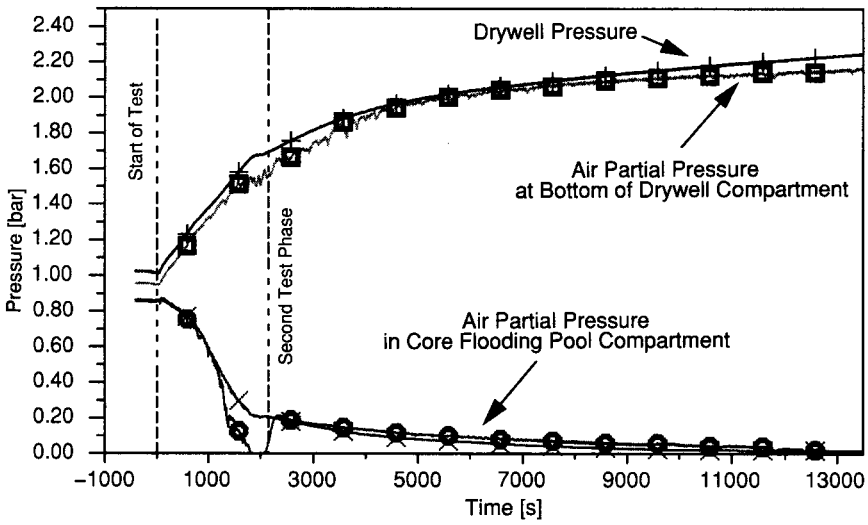
As the water inventory of the simulated shielding/storage pool was approximately 7.5 times too small, the water heated up at a correspondingly faster rate than would occur under real plant conditions. This was not a disadvantage, however, as it also meant that the test sequence proceeded approximately 7.5 times faster than the corresponding accident sequence in a real plant. Hence a test duration of approximately 3.3 hours was sufficient in the PANDA test facility to simulate a 24-hour accident sequence. Only the two most important of the many test results shall be presented and discussed here:

- When an accident not involving loss of coolant is simulated, the CCC is first mainly surrounded by air (see Figure 2; air was used in the test in place of nitrogen). But the non condensable gases together with the surplus steam are washed out into the wetwell more and more. Approximately one day after onset of the accident (timeframe based on actual SWR 1000 design), their concentration is negligible. The maximum heat transfer of the CCC is necessary about 20 hours after begin of accident because in the time before the heat will be stored mainly in the flooding and in the condensation pool. If the concentration of non-condensable gases is negligible, the fins on the condenser tubes are useless.
- When helium is fed into the atmosphere (in place of hydrogen), a condition of stable stratification likewise arises in the air space above the core flooding pool. In this case, however, the colder layers are above the hotter ones. The transition from a pure steam atmosphere to a steam-helium mixture in the air space above the core flooding pool is so abrupt that a thermocouple installed in the vicinity of the CCC recorded a temperature of around 160°C (virtually equal to the saturation temperature of the steam), while another thermocouple located 4 cm higher measured a value of approximately 120°C (corresponding to the temperature of the water on the secondary side of the CCC).

The PANDA tests showed that a significant role is played by stable stratification of fluids of different temperatures in the containment atmosphere. The conclusion must thus be drawn that the results of accident analyses performed using computer codes will always contain an element of uncertainty as long as stable stratification is not accounted for in the codes.

Furthermore it was also found that smooth tubes can be used in the CCC instead of finned tubes without having any adverse effect on the CCC's medium- and long-term capacity for removing heat from the containment.

Figure 2. **Total pressure and air partial pressures at different locations in drywell during PANDA test BC 1 (simulation of accident without loss of coolant)**




***Passive pressure pulse transmitter (PPPT)***

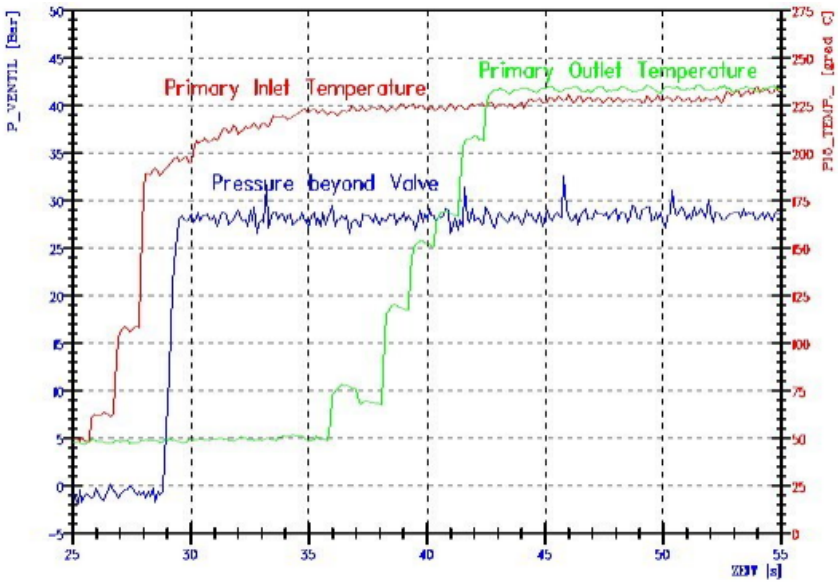
The efficiency of the PPPTs is measured by their so-called "response time", defined as the time between the first contact of the tubes with steam on the primary side of the PPPT and actuation of the pilot valves. The maximum specified response time is 20 s at a system pressure of 70 bar.

The PPPTs were also tested in the NOKO test facility in the Jülich Research Center under realistic boundary conditions. System pressures of between 10 and 70 bar were employed. The first design variants that were tested (No. 1 and 2) were so slow in terms of thermal performance that their response times exceeded 20 s. Variant 3 achieved just under 20 s and Variant 4 around 12 s. Further design improvements culminated in a Variant 5 which was also tested. Figure 3 presents actual printouts of the primary inlet and outlet temperatures as well as the pressure downstream of the pilot valve for a system pressure of 30 bar. These clearly show the sharp increases in temperature which occur when steam having a temperature of approximately 233°C reaches the uppermost and the bottommost tubes of the PPPT.

The response time for pilot valve actuation is approximately 3 s, although around 16 s elapse before the steam reaches the bottommost tubes. At a system pressure of 70 bar the response time is likewise around 3 s, while at 10 bar it is approximately 10 s.

The reduction in response time from the original figure of 20 s to less than 5 s means, for example, that reactor scram would be initiated by the PPPTs at least 15 s earlier. As a result, the coolant inventory available inside the RPV for the remainder of the accident sequence, assuming loss of all active safety systems, is approximately 23 Mg larger.

Figure 3. **Primary inlet and outlet temperatures of PPPT Variant 5 and pressure downstream of the actuated pilot valve**




***Passive outflow reducer (POR)***

A guillotine break in one of the condensate return lines of the Emergency Condenser would result in a break discharge flow of approximately 620 kg/s. The water level in the RPV would drop so rapidly that core uncovery and heatup would have to be expected if only passive safety systems were available.

The passive outflow reducer was developed for the purpose of significantly restricting the maximum flow that could be discharged in the event of such a break. It is a device without movable parts which generates high swirl when water is leaving the RPV and only a small swirl flow component in the opposite direction.

A POR with a nominal diameter of DN 50 (2”) was tested in Siemens’ Large Valve Test Facility in Karlstein. Depending on the specific test being performed, it was repositioned to allow fluid to pass through it in either the normal direction or the reverse direction. The fluids used for the tests comprised saturated steam, a steam-water mixture, saturated water and subcooled water with inlet pressures of up to 70 bar. The pressure in the line downstream of the POR was approximately atmospheric.

Figure 4 shows the mass flow as a function of inlet pressure for cold water flowing in the normal direction. The curve is approximately parabolic and corresponds to a flow resistance of $\zeta = 5$. At an inlet pressure above 1.3 bar, the curve abruptly changes into a virtually horizontal line. This is due to saturation pressure being reached in the narrowest cross section of the POR, leading to cavitation.

Figure 4. **Mass flow through passive outflow reducer DN 50 in normal direction using water of different temperatures**

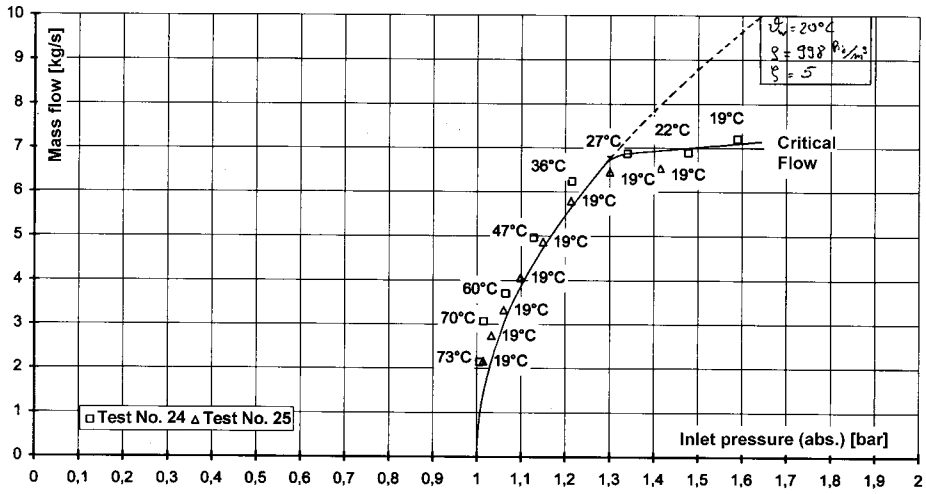
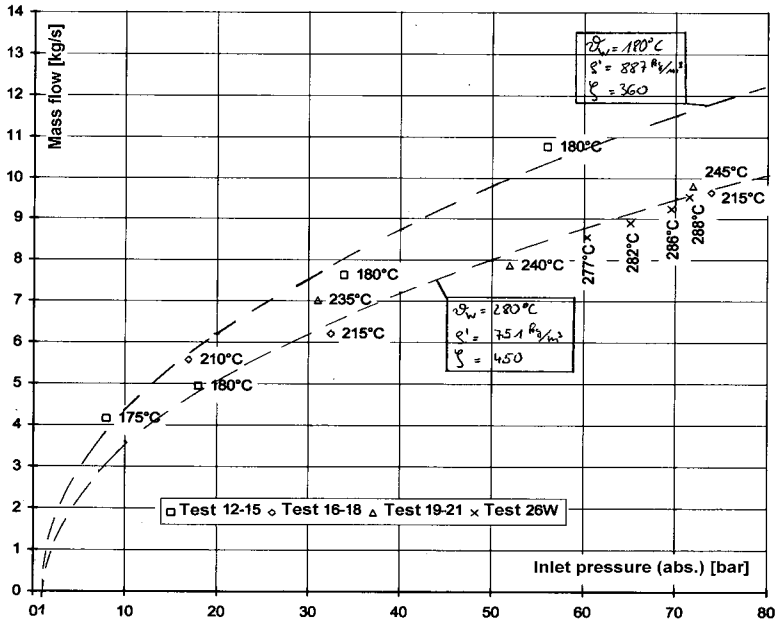



Figure 5 shows the results for flow through the POR in the reverse direction. Once again the curve is roughly parabolic, corresponding to flow resistances between $\zeta = 360$ and $\zeta = 450$. The wide scatter of data points can be explained by the fact that water with varying degrees of subcooling was used for these measurements.

For saturated water at 70 bar flowing in the reverse direction through the tested DN-50 POR, a resistance coefficient of $\zeta = 450$ was determined. The break discharge flow is then 74 kg/s, approximately 12% of that which would result without the POR.

Figure 5. **Mass flow through passive outflow reducer DN 50 in reverse direction using water of different temperatures**



***Passive core flooding system***

The four core flooding pools are joined to the outlet headers of the ECs (and therefore to the RPV) by four core flooding lines, each equipped with a check valve. These lines are designed to permit water to discharge from the core flooding pools into the RPV by gravity flow in the event of a LOCA once reactor pressure has dropped to such an extent that the check valves open.

Standard check valves, however, would not open at a sufficiently early point in time. Furthermore, the pressure differentials available for opening these valves are relatively low. To overcome this problem, the check valves in the flooding lines are designed for spring-assisted opening in order to ensure that they open in good time.

As the outlet headers of the ECs are filled with steam at the time when the spring-assisted check valves open, steam initially flows through the flooding lines into the core flooding pools. This has the desirable effect of causing a faster reduction in reactor pressure. As the reactor pressure drops, the flow of steam decreases and comes to a complete stop as soon as the same pressure prevails on both the inlet and outlet sides of the check valves. If there is now a further drop in reactor pressure as a result of steam blowdown via the safety-relief valves, the direction of flow in the flooding lines reverses and cold water is drawn from the core flooding pools into the EC outlet headers and from there to the RPV to flood the core.

These processes were also investigated experimentally in the NOKO test facility at Jülich Research Center using a model in which the volumes were scaled approximately 1:25. The test results were in very good agreement with the theoretical investigations. Also, no condensation shocks were found to occur when steam entered the cold flooding line after opening of the (simulated) check valve or when cold water again entered the flooding line after flow reversal.

***In-vessel retention of core melt by cooling of RPV exterior***

The aim of the exterior RPV cooling in the case of a core melt accident is to keep the RPV intact. So all molten material remains inside the RPV.

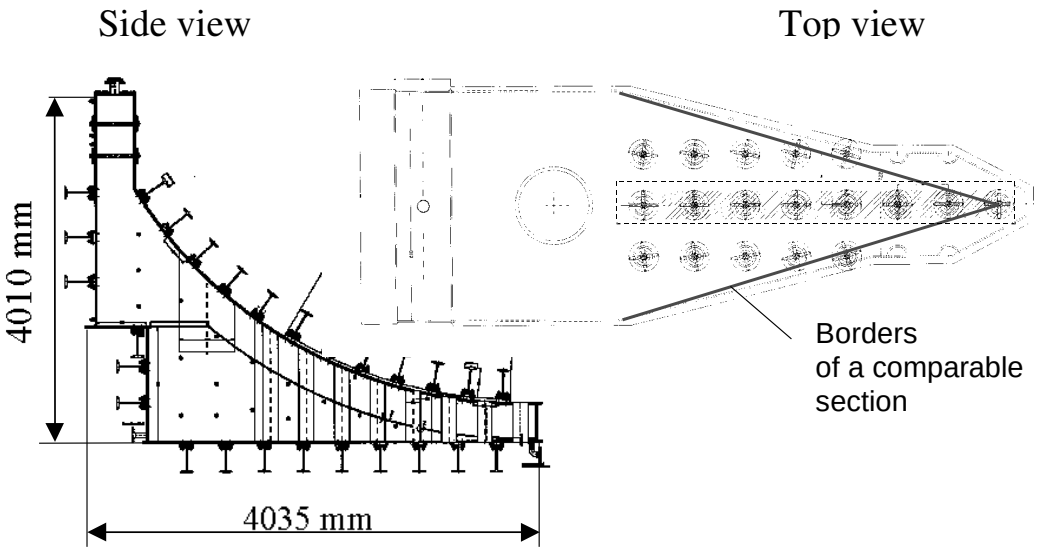
In-vessel core melt retention by cooling of the RPV exterior had already been verified experimentally for the AP 600 pressurized water reactor [2]. However, whereas the reactor vessel of the AP 600 has an uninterrupted hemispherical bottom head, the bottom head of the SWR 1000 RPV contains approximately 170 nozzles of varying sizes. Although some positive test results have been obtained for a model RPV bottom head containing a few nozzles [3], it is not clear whether these nozzles enhance cooling of the bottom head due to the larger effective heat-exchange surface that they provide, or whether cooling may be impaired because the nozzles prevent steam voids from being swept away from these areas quickly enough by the flow of water, possibly resulting in localized film boiling.

To investigate this problem an extensive test program was started in our own test facilities in Erlangen. First of all, in a 1:10 scale model operated with air and water, impedance probes were used to determine the void fraction in the flow of water between adjacent nozzles. It was found that, in any given quadrant of the bottom head, the removal of air bubbles by the water was obstructed most in the diagonal direction.

In a further series of tests, to be performed using part of the same 1:10 model, the void distributions in just this nozzle region alone are be investigated in greater detail.

Finally, this section of the RPV bottom head is also to be tested in a full-scale model operated at elevated temperatures. The tests will be done in the Framatome test field in Erlangen. Figure 6 shows a sketch of the test facility. The electrical heaters to be used for this test should enable heat flux densities to be achieved that are three times higher than those calculated for the actual SWR 1000. Using this model, investigations are to be conducted at various system pressures to determine the boundary conditions under which local film boiling and thus a significant increase in temperature on the outside of the RPV bottom head may occur.

Figure6. **Design of the test vessel to prove the feasibility of the exterior cooling concept**




The tests concerning cooling of the RPV exterior will begin in some weeks and are expected to be completed in the first half of 2002.

## 3. Further research program

Actually further R&D activities related to the development of the SWR 1000 are initiated:

- Fast Acting Boron Injection System (FABIS) program

  The Fast Acting Boron Injection System is a diverse measure for a fast reactor shutdown supposing that the control rod drives (CRD) fail due to a common cause failure. It consists of analyses and test and is a co-operation between the Technical Research Center of Finland (VTT), the Finnish Lappeenranta University of Technology and Framatome AP GmbH. The program is funded by the EU.

- Spring supported check valve for passive internal RPV flooding

  Owing to the pool elevation, the water in the core flooding pools is used for passive flooding of the reactor core following RPV depressurization in the event of a LOCA. In this function, check valves designed for spring-assisted opening open the flooding lines automatically. Passive flooding serves as a diverse supplementary function to the active

injection systems for core cooling. This check valve designed for spring-assisted opening will be tested in our own test facilities in Karlstein.

- Test of quenchers and vent pipes

  The geometry of the quenchers (safety & relief valve discharge lines) and vent pipes is different compared to existing Framatome BWR's. It is expected that the forces acting at the bottom and the walls of the flooding pools can be reduced. The quenchers and vent pipes will be tested in our own test facilities in Karlstein.

- Investigations on control rod drives (CRD)

  The control rod drives – activated by SCRAM are a passive device – shall be mounted from above. In this connection further investigations concerning reduction of piston diameter, gland sealing free of asbestos, reduction of acceleration forces and improved coupling of the control rod will be investigated and tested if necessary.

## 4. Conclusion

The already conducted investigations, analyses and tests were finished successfully at several European test and research facilities and by Framatome ANP itself. The new program is initiated to verify the passive safety equipment further.

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# CURRENT AND FUTURE RESEARCH ON PASSIVE SAFETY DEVICES AND ADVANCED FUELS FOR INNOVATIVE LWRs

## Part 2: Fuel Development for Advanced Light Water Reactors

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## Abstract

Presently an enrichment limit of 5% $^{235}U$ is fixed in licenses all along the fuel cycle. Although working groups (e.g. DOE, WNA) are investigating incentives and prerequisites to increase the limit, it is widely expected that at least for the mid term (10-20 years) the limit will hold. For the EPR and SWR 1000 development the 5% enrichment limit was not a serious restriction, since for higher enrichment each fuel assembly has to be shimmed with fixed burnable absorbers anyway for criticality or reactivity reasons. Thus the fuel safety issues for these ALWR are the same as for currently operating LWR, essentially the influence of high burn-up on safety analyses (fuel response to transients e.g. RIA, LOCA). Operational behavior is of minor concern since the burn-up increase is slow enough to allow sufficient experience feedback. Worldwide the LWR fuel failure rates have not increased with burn-up. At present, and expected mid term, uranium prices are such that there is little incentive to look into other fissile materials like thorium. Nevertheless, on a small scale, R&D is ongoing to prepare a longer term alternative. Some R&D is also considered for the end of NPP lifetime fuel cycles, however there is little economic incentive for such (due to lifetime extension) rarely needed specific solutions. If nuclear fission energy is to be used in the real long term (beyond 50-100 years), then R&D on breeder reactor fuel will have to be resumed.

## Introduction

All six advanced LWR designs (including Framatome ANP's EPR and SWR 1000) which are expected to be offered in Finland, if the Finish Parliament gives green light for unit 5, are of evolutionary design with regard to features affecting operational availability. Utilities consider nuclear fuel as a product, which must be robust and reliable in operation, provide flexibility and increased margin for operation and is easy to license. Potential failures tend to increase outage duration and cost, and any new or open licensing issues can cause delays or downtime. Calculated fuel cycle cost advantages of new designs can easily be eliminated by just a few days of additional downtime or reduced power level. On the other hand, utilities have established competition between the OEMs and other fuel vendors and have benefited from declining prices and increasing burn-up. Utilities want to maintain this situation and have little or no economic interest in drastic fuel design changes, for example, like inert matrix fuel. In the present licensing environment it is expected to take more than 10 years from the development of an inert matrix pellet to the insertion of fuel reloads.

## Past development experience

Since the introduction of commercial PWRs and BWRs the discharge burn-up has been increasing significantly worldwide. However, in different countries the economic incentive for utilities was quite different depending on relevant fuel cycle backend costs. In Germany utilities pushed very much, the resulting burn-ups of the leading PWR and BWR reload batches over time are shown in Figure 1. However, the rate of burn-up increase was low enough to ensure sufficient experience feedback and the defect rates decreased in the same period as shown in Figure 2. On the PWR side e.g., the burn-up increase was limited for a long time by waterside cladding corrosion. Figure 3 shows as an example the cladding development of Framatome ANP which allowed for a steady increase in the burn-up limitation caused by waterside corrosion. Our best throughwall cladding, M5$^{TM}$, an essentially ZrNb based proprietary alloy, is expected to fulfill PWR operational requirements even under most demanding plant conditions (temperature, void, power history etc.) up to the maximum burn-up one can achieve with 5% $^{235}$U. The decreasing defect rates for both PWR and BWR demonstrate that fuel performance related operational safety was overall improved in the last 30 years. To demonstrate that we are not operating close to a defect threshold we have operated pathfinder rods to much higher burn-ups as shown in Figure 4.

The present licensing limit of 5% $^{235}$U is established all along the fuel cycle: enrichment plants, UF6 transportation containers, fuel manufacturing, fresh fuel transportation containers, storage and handling in the NPP and storage and transportation of spent fuel. Sometimes the limits are even still lower. One of the major objections to raise the enrichment limit is the lack of measured criticality data between 5 and 10% $^{235}$U enrichment. Is there a utility incentive to go beyond 5% enrichment? Figure 5 shows the structure of the fuel cycle cost as a function of burn-up, it demonstrates the big influence of the backend assumptions. Figure 6 shows that in countries, which build reserves based on kWh produced, there is no incentive to spend much money to change the licensing limits. The present burn-up levels in Scandinavia and Spain are comparatively low. This was also the case for many years in the USA. Only the recent shortage in spent fuel storage capacity favored rapid burn-up increase in order to have fewer spent fuel assemblies.

Recently working groups were established by DOE in the USA and by WNA both with Framatome ANP participation which will look into the incentives, the barriers and the investment needed to lift the 5% enrichment limit all along the fuel cycle. Figure 7 shows an excerpt from the WNA agenda. During the conceptual and basic design phases of the EPR and SWR 1000 development the participating utilities showed no interest to go beyond 5%. However, with the flexibility to shim each fuel assembly with fixed burnable absorbers and to go to enriched boron 10 as soluble absorber in the coolant, fuel enriched to higher than 5% can later be accommodated without plant design modifications.

## Advanced fuel for EPR and SWR 1000

For the EPR the traditional 14 ft 17x17 fuel design as used e.g. in the French P4 and N4 plants was selected. This gives zero additional operational and licensing risk to the utilities compared to the fuel for NPPs already in operation. All design improvements and burn-up increases established in the operating plants can be transferred to the future EPR without the need for lead assemblies, inspection programs and significant additional licensing efforts.

For the SWR 1000 the fuel rod design, the pitch and the spacer grid was taken from Framatome ANP's most advanced ATRIUM 10 fuel assembly. Also the skeleton structure with the internal water channel as load chain was taken from the ATRIUM 10, since the avoidance of tie rods is

of special benefit at high burn-ups which are achievable with 5% maximum enrichment. Just two rows of rods were added and the fuel assembly was shortened to optimize the core design to the passive safety system features. Figure 8 shows a comparison of the SWR 1000 ATRIUM 12 and the present ATRIUM 10. Larger fuel assemblies require lower plant investment costs e.g. fewer (but larger) control blades and allow shorter refueling times. Again the SWR 1000 fuel assembly can directly benefit from future advanced design features developed for the ATRIUM 10.

**Ongoing R&D with regard to safety analyses**

Contrary to the operational safety issues, which can be resolved by cautious step by step burn-up increase, the transient safety issues require a combination of safety analyses and experimental tests. The most important issues considered in recent years are Reactivity Insertion Accidents RIA (Figures 9 and 10 summarize the present status in Germany) and Loss of Coolant Accidents LOCA (see Figure 11).

In addition we started to look again into other potential burn-up dependencies. Fission gas release is of relevance for operational as well as for transient conditions. Figures 12 and 13 in combination show that the burn-up dependence is highly sensitive to the power histories and thus the plant operation and fuel management modes.

**Thorium fuel**

Framatome ANP's first experience in thorium fuel use in LWR was gained in the irradiation of thorium/plutonium fuel rods in a German BWR from 1970-1977. We also had later a cooperation with Brasilian entities. Today Th/Pu test rods are under irradiation in the German PWR Obrigheim. They are expected to be discharged in 2005 with 40 MWd/kg (Figure 14) [1]. The results so far show Thorium fuel can be designed, manufactured and operated. However, there is practically no economic interest by utilities. Large scale use would require a completely separate fuel cycle, a long lead assembly test phase with expensive post irradiation examinations and extensive reanalysis of the plant transients in order to check for potentially necessary plant modifications. At present and expected mid term future uranium prices neither at the utilities nor at the fuel vendor side a reasonable return on investment is imaginable for any change away from the uranium base. It has for the foreseeable future to be considered as basic research outside industrial application.

**Non-Proliferation issues**

Non-proliferation of atomic weapons is fully supported by the nuclear industry. The use of HEU designated by the weapon states as no longer required for defense purposes is easily usable in operating reactors and is well underway. The comparable use of the "surplus" weapons Pu was extensively discussed at the P8 expert meeting in Paris in October 1996. The fastest way to denaturate weapons Pu, i.e. to change the isotopic vector, is to burn it as MOX assemblies in operating LWRs, for which MOX fuel experience is already available. Of course there are potentially more efficient fuel designs and new reactor designs. To develop them takes huge financial resources and decades of development and thus postpones the reduction of the proliferation risk. Unfortunately little real progress has been made since 1996, although many such studies were produced.

However, the so-called "Spent Fuel Standard" defined by the US National Academy of Sciences is now widely accepted. Therefore to develop more proliferation resistant fuel designs seems

not to be the best use of limited R&D resources. None of the existing or suspected weapon states went the route via a commercial NPP program with LWRs since there are obviously less expensive routes. The US initiated KEDO project in North Korea supports this point of view.

## Conclusion

From an industry point of view LWR fuel safety research should be concentrated in the mid term on such burn-up dependent aspects which are of relevance under a $^{235}U$ enrichment limit of 5%. Lead assemblies and reloads with 4.95% are already under irradiation. In order not to limit unnecessarily the reload burn-up increase it is recommended to try to complete these investigations in the foreseeable future.

The fast breeder reactor programs including the relevant fuel programs came to a virtual standstill in many OECD countries due to a combination of being economically not competitive with LWRs and because of public acceptance problems. There are no insurmountable safety issues. If nuclear fission energy is to be used in the real long term (beyond 50-100 years), then R&D on breeder reactor fuel will have to be resumed.

[1] NEA/NSC/DOC(2001)21 Thorium fuel in LWRs – An option for an effective reduction of plutonium stockpiles – D. Porsch, Framatome ANP, D. Sommer, NPP Obrigheim.

Figure 1**. Peak reload batch average discharge burnup of Framatome ANP fuel assemblies in Europe**

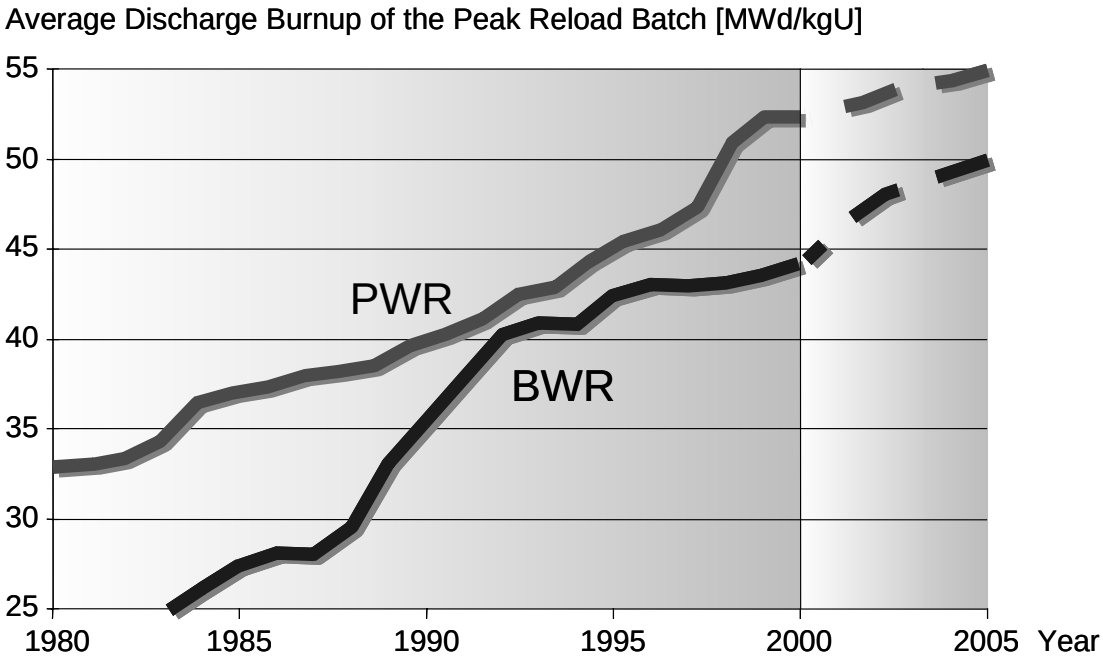



Figure 2. **Evolution of the discharge burnup of Framatome ANP PWR fuel assemblies and the percentage of reactor cycles with failed fuel**

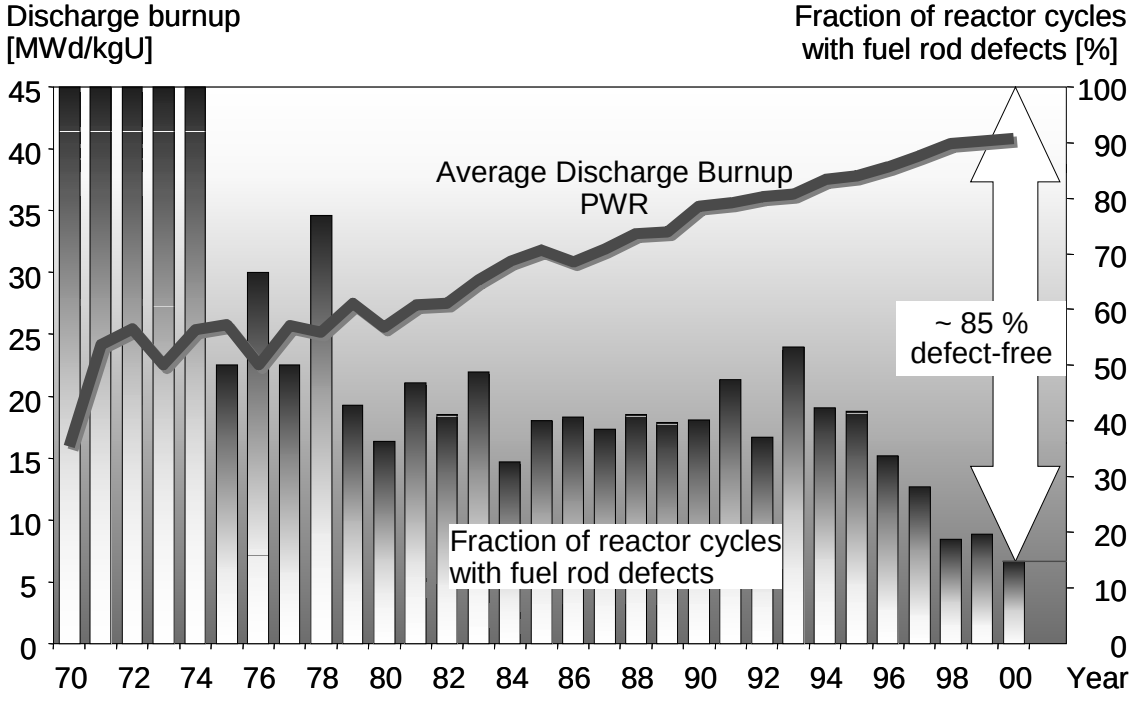


Figure 3. **Continued development of PWR materials reduces oxide layers at high burnup**

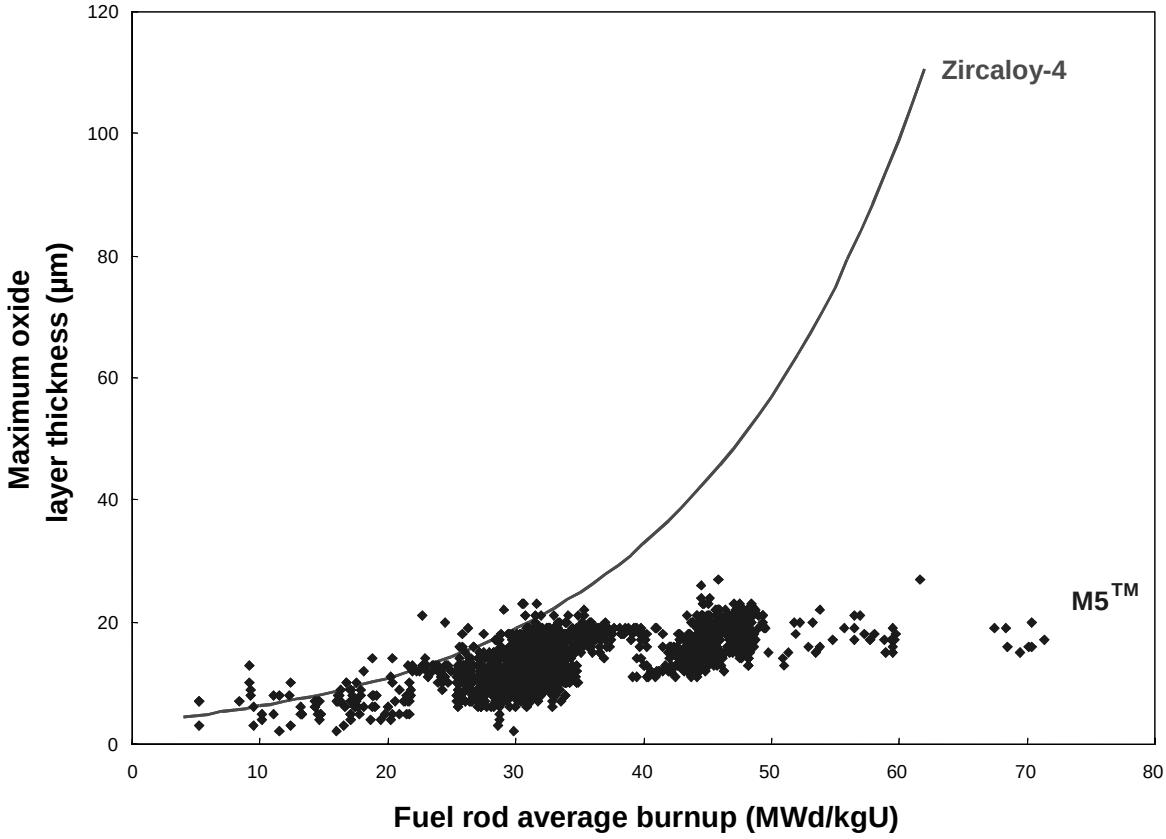



Figure 4. **Framatome ANP irradiation experiences with high burnup fuel rods – status 07/2001**

| Cycles | Maximum burnup (MWd/kgU) | Number of inserted fuel rods | Number of measured fuel rods (at max. burnup) | Cladding type |
|---|---|---|---|---|
| 5 | 59-74 | 73 | 59 | Alternative Alloys |
| 6 | 70-81<br>62<br>67-70 | 23<br>2<br>8 | 19<br>2<br>8 | Alternative Alloys<br>M4 (Zr 0.5 Sn 0.6Fe 0.3-0.45V)<br>M5™ (Zr 1Nb 0.14O –Fe-S) |
| 7 | 79-84 | 10 | 8 | Alternative Alloys |
| 8 | 89-91 | 2 | 2 | DX Zr0.25Fe0.25V<br>Zr0.5Sn0.6Fe0.3V |
| 9 | 97-99 | 3 | 3 | Zr0.25Sn0.5Fe0.05Cr<br>DX Zr2.5Nb |

Figure 5. **Effect of the individual components of the FCC on the FCC-curve over the discharge burnup**

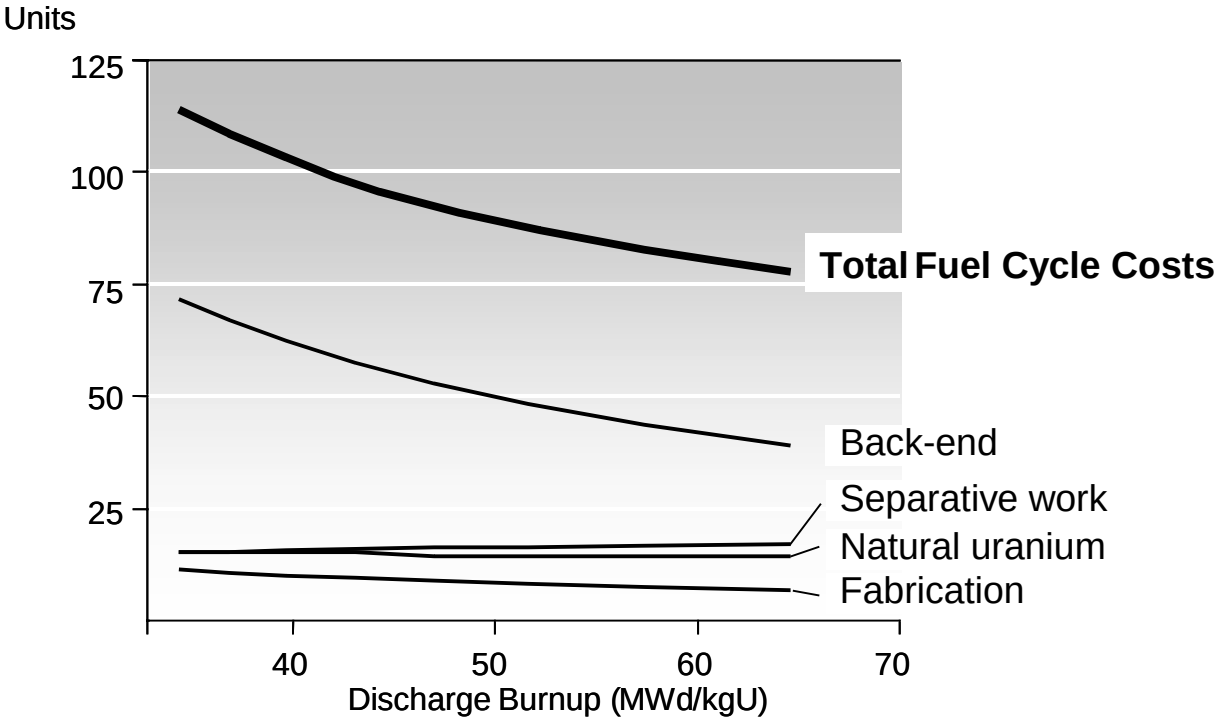



Figure 6. **Important parameters influencing the economic effect of burnup increase**

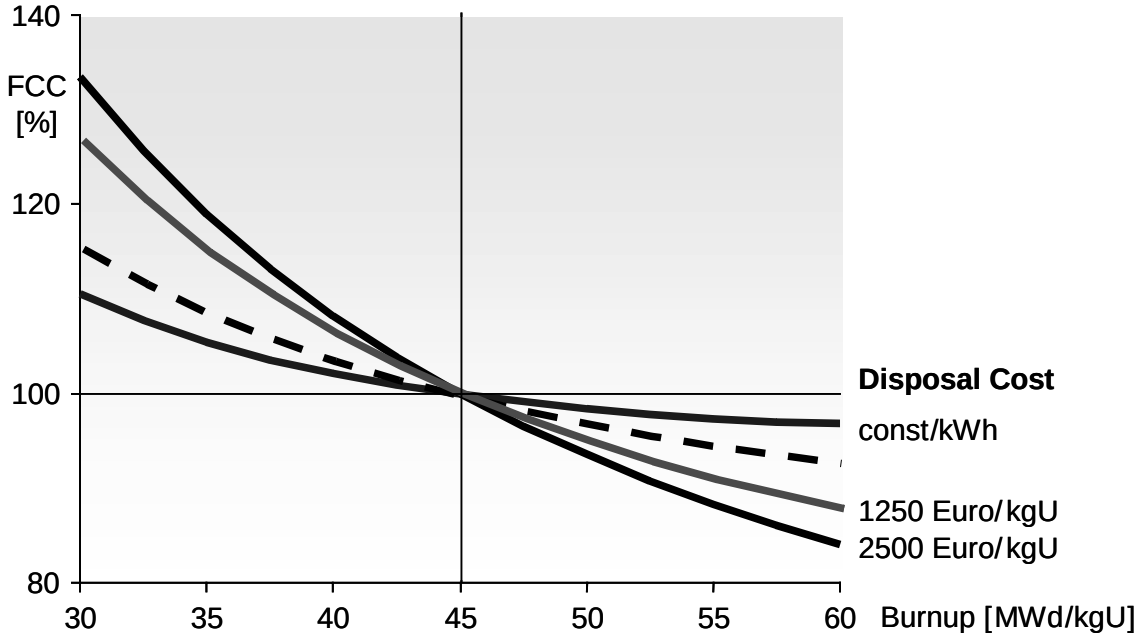
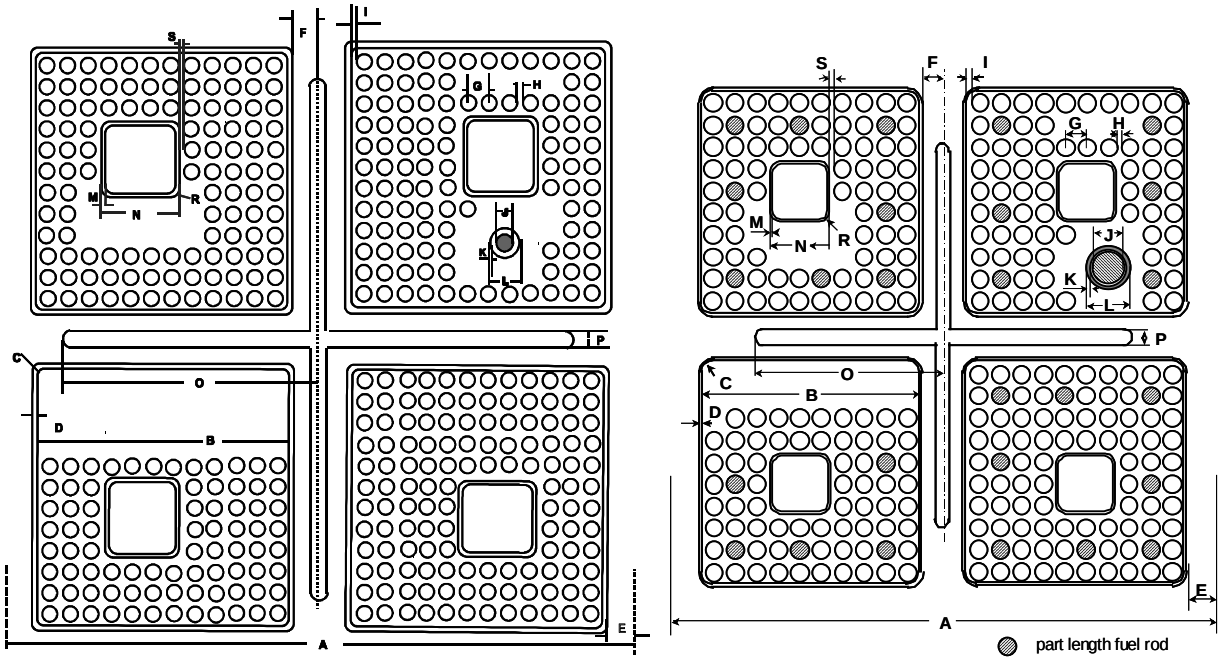


Figure 7. **WNA working group program ">5% $^{235}$U (excerpt)**

- Where are we today?
- Implications for reactor operators
- Fuel Assembly design implications
- Issues for Enrichers
- Issues for Fuel Fabricators
- Fresh fuel transport
- Spent fuel issues
- Recycle Considerations

- Technical capability
- Need for modifications
- Regulatory issues
- Investment issues

Figure 8. **Comparison of core cell geometry – BWR 1000 FA and conventional BWR**




SWR 1000 Conventional BWR

Figure 9. **RIA – Framatome ANP/Germany proposal of an integrity limit for fuel rods under RIA conditions**

| fuel rod burnup [MWd/kg] | enthalpy increase [cal/g] with oxide spalling | enthalpy increase [cal/g] without oxide spalling |
|---|---|---|
| 0.0 | 170 | 170 |
| 31.8 | 100 | 100 |
| 50.0 | 60 | 100 |
| 80.0 | 60 | 100 |

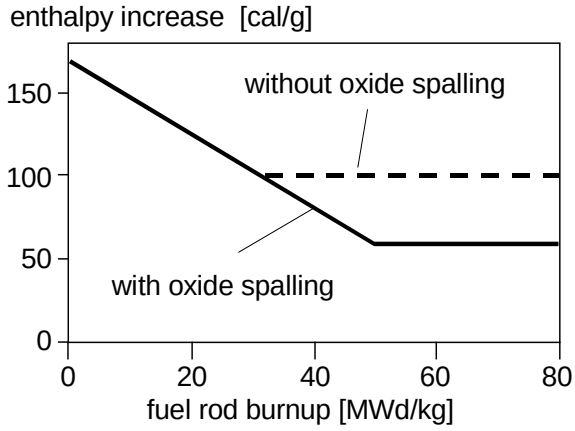



Figure 10. **RIA – actual status of the discussion in Germany**

- No final decision from the Licensing Authorities on RIA limits
- Self-limitation of some utilities with respect to burnup to avoid restrictions by the Regulators in some cases
- Agreement on participation of the utilities group VGB at the CABRI water loop

Figure 11. **LOCA**

Criteria and Discussion

International: Several programs to review the validity of the established LOCA safety criteria
- 17 % equivalent cladding reacted
- 1200 °C peak cladding temperature

for high burnup and high rod power as well as for advanced cladding (M5, ZIRLO, Duplex)
(e.g. ANL, HALDEN, TAGCIR, CODAZIR, HYDRAZIR, EDGAR)

Germany: In addition to the international discussion the restriction of the core damage (max. 10 %) as a limiting criterion.

Figure 12. **Fractional fission gas release of PWR fuel rods increases from 7 % at a burnup of 50 MWd/kgU to about 28 % at 100 MWd/kgU**

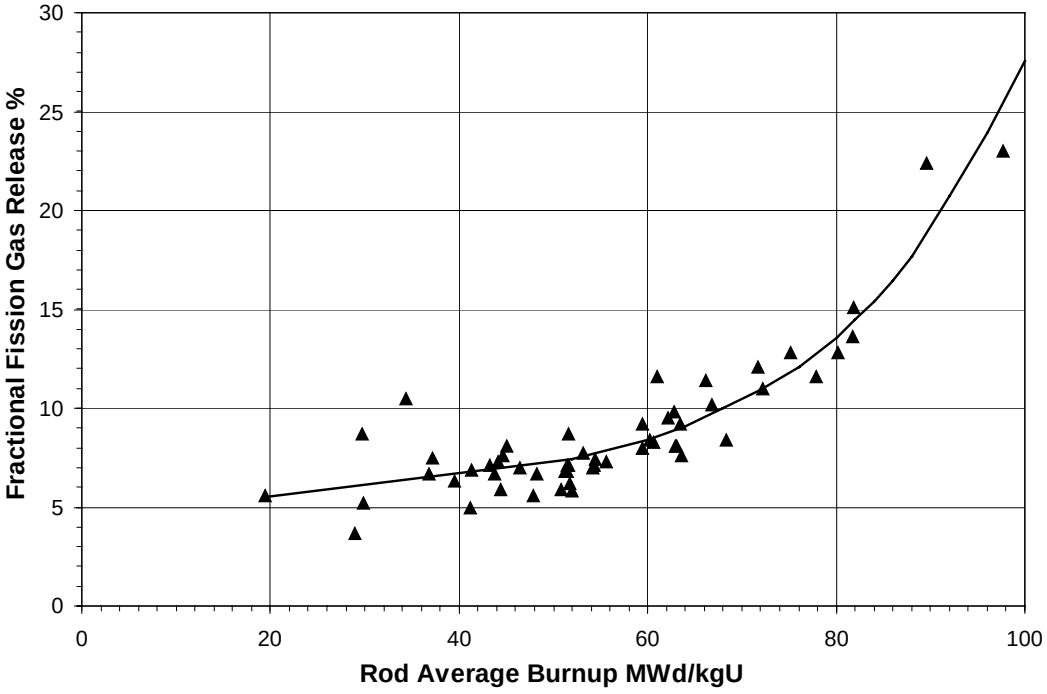



Figure 13**. Fractional fission gas release vs. rod power (burnup <40 GWd/tM))**

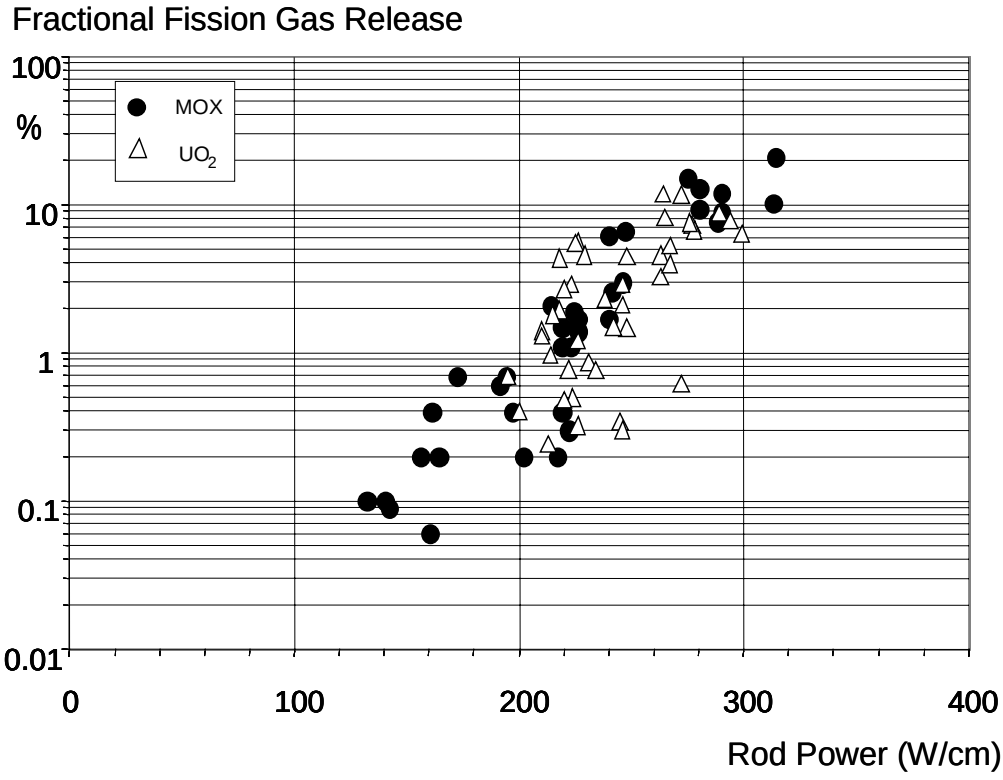


Figure 14. **Thorium fuel in German LWRs**

- 15 Th/Pu fuel rods irradiated 1970 - 1977 in a BWR
  Incentive: Significantly increased Pu reduction rate in Th/Pu fuel compared to U/Pu fuel

- 1979-1988 German-Brasilian research studies (Nuclebras, KFZ Jülich, Siemens, Nukem)

- 2001-2005 Th/Pu test rods under irradiation in a PWR
  Incentive: Significantly increased Pu isotopic degradation rate

⇒ Manufacturing works properly on a laboratory scale

⇒ Design codes work properly

⇒ Thorium fuel can be used in modern PWRs without major changes of the safety systems

# CURRENT AND FUTURE RESEARCH ON HTR SAFETY ISSUES

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## 1. Framatome ANP involvement in HTR development

The development of a new generation of high-temperature reactors (HTR), modular concepts with direct cycle, brings new opportunities for small and medium size reactors, economically competitive, with potential for other applications than electricity generation: Hydrogen production, cogeneration, district heating, desalination…

Framatome ANP has been involved in HTR development for several years through the participation to different activities and programs:

- In the past Siemens/Interatom (now Framatome ANP GmbH) designed and licensed the HTR Module in Germany. This concept has been the starting point for the development of the PBMR in South Africa.
- At the present Framatome ANP activities in HTR field is performed within different frames.
- Framatome ANP has been a partner of the GT-MHR project since 1996; the project is devoted to the development of an HTR with the capability of burning weapon grade plutonium in Russia.
- An other part of international activities is done in the European Union Framework Program, a 50% funding of the European Commission. The European HTR programme concerns basic developments of HTR technology in the fields of reactor physics, fuel, materials, key components of the power conversion system, and safety approach. Around 25 different organisations, representing research centres, universities, regulators, utilities and industries from 9 EU member states and Switzerland are involved. In particular, Framatome ANP is the leader of the HTR-TN (European High Temperature Reactor – Technology Network), gathering around 20 industrials and R&D organisations all over Europe.

Besides these activities performed within an international frame, Framatome ANP and the French CEA have established a cooperation program in order to address the main R&D issues related

to the HTR technology (calculation tools, fuel design and manufacturing, high temperature materials, specific equipment, Helium technology...).

## 2. Development of a safety approach for HTR

A working group composed of several safety experts from Framatome ANP has been gathered to develop a safety approach for HTR consistent with approaches developed for modern reactors.

The objective is to provide the main features of the safety approach to be implemented for a direct cycle modular HTR and to assess the possible strategies for licensing, in particular in the United States.

This exercise is performed on the basis of Framatome ANP safety experts background (EPR, EFR, HTR-MODULE), using qualitative judgement and existing quantitative results.

A review of former HTR main licensing issues in Germany and the United States is first performed.

The progress of licensing activities in South Africa (PBMR project), in the United States (American PBMR project) and in Russia (international GT-MHR project) is then proposed as well as IAEA activities in HTR licensing and safety area.

Relying on the working group knowledge about HTR safety behaviour, the main features of the safety approach are assessed regarding modern nuclear plant licensing issues and regulatory requirements. The consequences on the design will be identified. European Utility Requirements for Light Water Reactors and IAEA recommendations will be considered also.

The proposed safety approach will be compared to the one developed for PBMR and GT-MHR licensing. The resulting consequences on the design will be then assessed.

A work program will be proposed to support the safety demonstration.

### *2.1 Safety approach main features*

The safety approach has to be defined at the early steps of a project. It is then presented to the Safety Authorities for evaluation with the purpose of providing guidance early in the design process on the regulatory acceptability of the design. The safety impact on the design derived from the safety requirements has to be also assessed for the project cost evaluation.

The safety approach developed for the licensing of a nuclear installation should provide at least the following information:

- the objectives and principles for the public, workers and environment protection against radiological impact of the installation (including reference to national and international rules and recommendations);
- the description of the safety demonstration with emphasizing on the defence-in-depth principle application, the design basis and beyond design basis approaches, and the use of deterministic and probabilistic analyses;

- with regard to the events considered in the design, the description of the following items:
  - the classification principles for initiating events and deriving operating conditions together with the definition of associated radiological (workers and public) and equipment requirements;
  - the determination and classification of initiating events for the studied design using the results of preliminary studies on the main possible risks and their consequences, and the review of the state of the art;
  - the rules for deterministic safety analyses (demonstration that the consequences of operating conditions meet the requirements): initial conditions, uncertainties, delay for operator action, final states, delays for system recovery, aggravating failure, combination with loss of station service power, external hazards consideration, source term determination, exposure assessment;
- the rules for probabilistic safety analyses: definition of the probabilistic targets, identification of the requested systems;
- the rules for design analyses: definition of the loadings to be considered in the equipment design, selection of codes and standards used for the design;
- the rules for determination and classification of safety related components;
- the provisions to workers protection against radiation, the provisions with regard to fire protection, the provisions with regard to in service inspection and repair;
- the provisions for the plant decommissioning.

This approach should be submitted to Regulatory Bodies in the early phase of the design.

### *2.2 Methodology for developing a safety approach*

A common terminology should be defined and used, as far as possible in consistency with the international practice. Different glossaries are available: IAEA, EPR project, EUR.

A glossary is also in preparation in the frame of the European programme aiming at establishing the main features of a European safety approach for modular HTR. It is based on the IAEA one, HTR specific definitions are provided in addition.

The approach implemented is as follows:

- Assessment of the safety issues within the frame of modern projects (EPR, EUR, EFR) and international recommendations (IAEA, INSAG) to provide the approach main features and the objectives if any.
- Applicability and application to a modular direct cycle HTR relying on the existing knowledge about these reactors.

The definition of a design and the calculation of the plant behaviour during normal and accidental conditions will later confirm the different safety issues.

### *2.3 Defence-in-depth application*

Defence-in-depth is considered as the basis of the safety demonstration and licensing of existing and future nuclear plants.

INSAG 10 recommends the following for future nuclear plants:

"Defence-in-depth must continue to be the basis for the safety of nuclear plants…. Its improvement will continue to be the essential basis for further advances in safety."

INSAG-10 provides the objectives related to each level of protection against unacceptable releases and the means of achieving them.

The application of defence-in-depth principle for the design of future LWR and LMFBR relies on the consideration of both severe core damage prevention (third level) and severe core damage mitigation for the confinement function assessment (fourth level).

For modern reactors, the prevention level with regard to the risk of unacceptable releases can be assessed with probabilistic analysis or with the Line of Defence method. This approach may be used where the lack of experience feedback cannot allow to assess reliable probabilistic quantitative data. For modern reactor, the risk of unacceptable radiological releases is prevented by 2a + b (two strong and one medium) lines of defence. This is achieved differently for LWR and LMFBR:

*For LWR*

The severe core damage is the core melting resulting from identified sequences such as a loss of coolant accident combined with loss of Emergency Core Cooling System, reactivity insertion (e. g. heterogeneous boron dilution)…

The prevention of core melting could be achieved with 2 a (two strong) lines of defence and 2 a + b (two strong and one medium) lines of defence for early containment failure sequences for which only prevention is existing.

The mitigation of core melting is achieved with b (one medium line of defence) (accident management and containment provisions).

*For LMFBR*

The severe core damage is the core melting resulting from identified sequences such as unprotected loss of flow, unprotected transient overpower, unprotected loss of heat sink, fuel sub-assembly blockage, large reactivity insertion (e. g. voiding) …

The characteristics of LMFBR allow to enhance the prevention of core melting: 2 a+ b (two strong and one medium) lines of defence.

Emphasis on prevention only of core melting has been judged not to be sufficiently consistent with the defence-in-depth approach and despite the high level of prevention, mitigation measures against core melting have been implemented.

*Modular HTR*

Severe core damage: core melting and subsequent drop on the reactor cavity through the reactor vessel can be excluded for a modular HTR. The core graphite structure (fuel element and reactor internals) allows to withstand very high temperatures and to keep core configuration. The severe core damage is therefore defined as the degradation of the confinement capability of a large number of fuel particles. The challenges (thermal, chemical, mechanical…) on the fuel particles, first containment barrier, are overcome mainly by the reliance on simple natural phenomena such as heat transfers by conduction and radiation, and no severe core damage is expected as a result of any initiating event.

Thus the development of a safety approach excluding severe core damage is expected to be possible. This would allow:

- To avoid complex R&D actions for the analysis of severe core damage accidents.
- To eliminate the need of additional equipment for severe core damage mitigation and particularly the implementation of a leaktight containment withstanding high pressure.

The challenges for acceptance of such a safety approach are:

- To demonstrate the non-credibility of severe core damage with convincing arguments, (e. g. an exhaustive list of fuel particle challenging sequences) and with tools well qualified and data validated for modern materials.
- The simple passive phenomena such as conduction and radiation to remove decay heat and the time provided for corrective actions will help for a convincing demonstration.
- The air and water ingress issue and the graphite fire prevention should be particularly well assessed.

Complementary analyses would allow assessing the confinement function might be performance with regard to postulated beyond design situations.

## 3. R&D activities

### *3.1 R&D needs*

The R&D needs as identified by the designer are as follows:

- establishment of the list of events to be studied within the safety demonstration and determination of the associated limits (radiological, barriers);
- identification of the associated computer tools needs, assessment of the existing ones and definition of the development and qualification needs;
- determination of the needs for physical data (temperature, burnup…) for materials (particularly for new ones) since the safety demonstration will rely on the assessment of the plant inherent behaviour in any imaginable circumstance including the internal and external hazards;
- assessment of uncertainties.

More specific needs are listed hereafter:

- The fuel is a key issue in the safety demonstration as it is the main confinement barrier. Emphasis is put on fuel fabrication and fabrication quality control and also on qualification of fuel behaviour during normal and accidental conditions.
- The analysis of unprotected transients requires efficient modelling of annular core physics, good quantification of temperature coefficients and assessment of uncertainties (particularly for Pu cores needing erbium), characterisation of thermal properties of core materials, and modelling of the overall direct cycle (calculation of the equilibrium point).
- The passive decay heat removal requires a good knowledge of graphite properties and reactor vessel emissivity.
- In case of Helium break the radio-elements and graphite dust transport mechanisms (source term, depressurisation phase) must be analysed.
- Air or water ingress events require an assessment of corrosion of graphite and potential for graphite fire.

## *3.2 R&D programmes*

Framatome ANP is participating to several R&D programmes within different frames:

- Cooperation between Framatome ANP and CEA.
- R&D programme in support to the GT-MHR project.
- 5th European Framework programme: the nine HTR-related projects selected by the European Commission (EC) form a consistent and structured cluster covering both fundamental research and technological aspects. They were selected after two calls for proposals with deadlines 4 October 1999 and 22 January 2001.

Following is a brief description of the objectives as well as the main experimental and analytical activities foreseen within the above-mentioned projects. Around 25 different organisations, representing research centres, universities, regulators, utilities and vendors from 9 EU member states and Switzerland are involved.

### *3.2.1 Fuel (Projects HTR-F and HTR-F1)*

These projects are “shared-cost” actions to be carried out by a consortium of 7 organisations (CEA, FZJ, JRC-IAM, JRC-ITU, BNFL, Framatome and NRG) under the co-ordination of CEA. The duration foreseen for the combined projects is 48 months.

The objectives of HTR-F are: (i) to restore (and improve) the fuel fabrication capability in Europe, (ii) to qualify the fuel at high burn up with a high reliability and (iii) to study innovative fuels that can be used for applications different from former HTR designs. The project started in October 2000 and its Work Programme includes the following activities:

- to collect data from the various types of fuels tested in the past in European reactors (e.g. HFR, THTR, DRAGON, OSIRIS, SILOE, etc.) and to analyse them in order to better understand the fuel behaviour and performance under irradiation;
- to define experimental programmes (in-pile and out-of-pile) in order to qualify the fuel particle behaviour under irradiation and high temperatures. A first irradiation test is planned in the HFR reactor on pebbles from the last German high quality fuel production with the objective to reach a burn-up of 200 000 MWd/t. Concerning the heat-up tests, the Cold Finger Furnace (KÜFA) facility, in which temperatures can reach up to 1 800°C, was transferred from Jülich (FZJ) to Karlsruhe (JRC/ITU) where it will be commissioned after having tested one irradiated pebble;
- to model the thermal and mechanical behaviour of coated fuel under irradiation and to validate it against the experimental results available. The models in existing codes (e.g. PANAMA, FRESCO, COCONUT, etc) will be used to develop a common European code;
- to review the existing technologies for fabrication of kernels and coated particles, to fabricate first batches of U-bearing kernels and coated particles, to characterise them and to study alternative coating materials (e.g. ZrC and TiN). Kernels and particles will be fabricated in different laboratories (two at CEA and one at JRC/ITU) and the first coatings tests will be performed on simulated and depleted uranium kernels.

The programme of HTR-F1, started in November 2001, is fully complementary of HTR-F.

It will enable to complete the irradiation of the German pebbles in the HFR in Petten, to carry out their post irradiation examination (PIE) and to perform heat-up tests under accident conditions in the modified KÜFA facility at JRC/ITU. Also, the code developed in HTR-F to modelling the thermal and mechanical behaviour of the coated fuel particles should be validated. Finally, the production of coated particles and kernels should start at CEA and JRC/ITU.

### *3.2.2 Core Physics (Projects HTR-N and HTR-N1)*

These projects are "shared-cost" actions to be carried out by a consortium of 14 organisations (FZJ, Ansaldo, BNFL, CEA, COGEMA, Framatome ANP SAS and GmbH, NNC Ltd., NRG, JRC-ITU, Subatech, and the Universities of Delft, Pisa and Stuttgart) under the co-ordination of FZJ. The duration foreseen for the combined projects is 54 months.

The main objectives of HTR-N are: to provide numerical nuclear physics tools (and check the availability of nuclear data) for the analysis and design of innovative HTR cores, to investigate different fuel cycles that can minimise the generation of long-lived actinides and optimise the Pu-burning capabilities, and to analyse the HTR-specific waste and the disposal behaviour of spent fuel. The project started in September 2000 and its Work Programme includes the following activities:

- to validate present core physics code packages for innovative HTR concepts (of both prismatic block and pebble bed types) against tests of Japan's High Temperature Test

Reactor (HTTR) and to use these codes to predict the first criticality of China's HTR-10 experimental reactor;

- to evaluate the impact of nuclear data uncertainties on the calculation of reactor reactivity and mass balances (particularly for high burn-up). Sensitivity analyses will be performed by different methods on the basis of today's available data sets (ENDF/B-VI, JEFF-3, JENDL 3.2/3);
- to study selected variations of the two main reactor concepts (i.e. hexagonal block type and pebble-bed) and their associated loading schemes and fuel cycles (i.e. the static batch-loaded cores and continuously loaded cores) in order to assess burn-up increase, waste minimisation capabilities, economics and safety;
- to analyse the HTR operational and decommissioning waste streams for both prismatic block and pebble bed types and to compare them with the waste stream of LWR;
- to perform different tests (e.g. corrosion, leaching, dissolution) with fuel kernels such as $UO_2$ and $(Th,U)O_2$ and coating materials of different compositions (e.g. SiC, PyC) in order to evaluate and generate the data needed to model the geo-chemical behaviour of the spent fuel under different final disposal conditions, i.e. salt brines, clay water and granite.

The HTR-N1 project proposes to: extend the nuclear physics analysis of HTR-N to the hot conditions of Low-enriched Uranium (LEU) cores with data from HTTR and HTR-10; to investigate the potential to treat or purify specific HTR decommissioning waste (e.g. structural graphite) on the basis of samples taken from the AVR side reflector and to continue the leaching experiments for disposed spent fuel with irradiated fuel (instead of dummies) for initial commissioning of the test rigs.

### *3.2.3 Materials (Projects HTR-M and HTR-M1)*

These projects are "shared-cost" action to be carried out by a consortium of 8 organisations (NNC Ltd., Framatome, CEA, NRG, FZJ, Siemens, Empresarios Agrupados and JRC-IAM) under the co-ordination of NNC Ltd. The duration foreseen for the combined projects is 54 months.

The objectives of HTR-M are to provide materials data for key components of the development of HTR technology in Europe including: reactor pressure vessel (RPV), high temperature areas (internal structures and turbine) and graphite structures. The project started in November 2000 and its Work Programme consists of the following basic activities:

- review of RPV materials, focusing on previous HTRs in order to set up a materials property database on design properties. Specific mechanical tests will be performed on RPV welded joints (Framatome facilities) and irradiated specimens (Petten HFR) covering tensile, creep and/or compact tension fracture;
- compilation of existing data about materials for reactor internals having a high potential interest, selection of the most promising grades for further R&D efforts, and development and testing of available alloys. Mechanical and creep tests will be performed at CEA on candidate materials at temperatures up to 1 100°C with focus on the control rod cladding;
- compilation of existing data about turbine disk and blade materials, selection of the most promising grades for further R&D efforts, and development and testing of available alloys. Tensile and creep tests (in air and vacuum) from 850°C up to 1 300°C

and fatigue testing at 1 000°C will be performed at facilities at CEA while creep and creep/fatigue tests in Helium will be performed at JRC;

- review the state of the art on graphite properties in order to set up a suitable database and perform oxidation tests at high temperatures on: (i) a fuel matrix graphite to obtain kinetic data for advanced oxidation (THERA facility at FZJ) and (ii) advanced carbon-based materials to obtain oxidation resistance in steam and in air respectively (INDEX facility at FZJ).

The HTR-M1 project complements HTR-M, as it concentrates on the long-term testing of the materials for the turbine and irradiation tests for the HTR graphite components. Special attention is put on the fact that previous graphites are no longer available because the coke used as the raw material has either run out and the manufacturer's experience lost, or production techniques and equipment do no longer exist. The work programme includes verification of models describing the graphite behaviour under irradiation and screening tests of recent graphite qualities.

### *3.2.4 Direct cycle equipment Project (Project HTR-E)*

This project is a "shared-cost" action to be carried out by a consortium of 14 organisations (Framatome ANP SAS, Ansaldo, Balcke Dürr, CEA, Empresarios Agrupados, Framatome ANP GmbH, FZJ, Heatric, Jeumont Industrie, NRG, NNC Ltd., S2M, University of Zittau and Von Karman Institute) under the co-ordination of Framatome ANP SAS. The duration foreseen for this project is 48 months and the project started in December 2001.

This project addresses the innovative key components, systems and equipment related to the direct cycle of modern HTRs. These include turbine, recuperator heat exchanger, active and permanent magnetic bearings, rotating seals, sliding parts (tribology) and the helium purification system. The programme contains both design studies (e.g. Computer Fluid Dynamics and Finite Element analyses) and also experiments (e.g. magnetic bearing tests at Zittau facility, validation tests of the recuperator at CEA's CLAIRE loop or tribological investigations at Framatome's Technical Centre).

### *3.2.5 Safety approach Project (Project HTR-L)*

This project is a "shared-cost" action to be carried out by a consortium of 8 organisations (Tractebel, Ansaldo, Empresarios Agrupados, Framatome ANP SAS, Framatome ANP GmbH, FZJ, NRG, and NNC Ltd.) under the co-ordination of Tractebel. The duration foreseen for this project is 36 months and the project started in October 2001.

The project proposes a safety approach for a licensing framework specific to Modular High Temperature Reactors and a classification for the design basis operating conditions and associated acceptance criteria. Special attention will be put on the confinement requirements and the rules for system, structure and component classification as well as a component qualification level being compatible with economical targets.

### *3.2.6 Coordination (Project HTR-C)*

This is a "concerted action" to be carried out by a consortium of 6 organisations (Framatome, FZJ, CEA, NNC Ltd., NRG, and JRC) under the co-ordination of Framatome. The duration foreseen is 48 months.

This project, which started in October 2000, is devoted to the co-ordination and the integration of the work to be performed in all the above-mentioned projects. Moreover, HTR-C should organise a world-wide "technological watch" and develop international co-operation, with first priority to China and Japan, which have now the only research HTRs in the world. In order to promote and disseminate the achievements of the EC-sponsored projects, HTR-C will organise presentations in international conferences.

### *3.2.7 Fuel irradiation program*

HTR-TN is also preparing a programme of irradiation of HTR fuel in the High Flux Reactor (HFR) of the Joint Research Centre of the European Commission located at Petten (Netherlands). The objective of this programme is to verify if HTR fuel keeps its outstanding leak tightness in normal operating and accident conditions up to ultra high burnups (the target is 200 000 MWd/tHM). Moreover a co-operation is under preparation with INET in China and with JAERI in Japan, which operate the two only high temperature test reactors presently existing in the world, HTR-10 for the first one and HTTR for the second one.

## 4. Conclusions

It is intended to implement a safety approach consistent with international recommendations and with the approaches developed for current HTR projects and future reactors (GCFR, ADS…). This is achieved through:

- Close follow-up of IAEA activities about future reactors, particularly HTR.
- Close follow-up of international licensing activities of HTR in US (PBMR, GT-MHR), in South Africa (PBMR) and Russia (GT-MHR).
- Participation in European programs aiming at defining safety approaches for modular HTR (activities aiming at providing for HTR a document equivalent to the European Utility Requirements report/ Safety performed for LWR), GCFR and ADS.

This safety approach should include an extremely convincing demonstration of severe core damage exclusion.

The associated R&D is determined in consistency with the needs required for the safety demonstration, focussing on the fuel and the determination of physical properties needed for the analysis of passive behaviour, and relying on the past experiments and international programmes.

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