IAEA-TECDOC-1652

# Small Reactors without On-site Refuelling: Neutronic Characteristics, Emergency Planning and Development Scenarios

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# SMALL REACTORS WITHOUT ON-SITE REFUELLING: NEUTRONIC CHARACTERISTICS, EMERGENCY PLANNING AND DEVELOPMENT SCENARIOS

FINAL REPORT OF AN
IAEA COORDINATED RESEARCH PROJECT

INTERNATIONAL ATOMIC ENERGY AGENCY
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## FOREWORD

Small reactors without on-site refuelling have a capability to operate without reloading or shuffling of fuel in their cores for reasonably long periods of time consistent with plant economy and considerations of energy security, with no fresh or spent fuel being stored at the site during reactor operation. In 2009, more than 25 design concepts of such reactors were analyzed or developed in IAEA Member States, representing both developed and developing countries. Small reactors without on-site refuelling are being developed for several reactor lines, including water cooled reactors, sodium cooled fast reactors, lead and lead bismuth cooled reactors, and also include some non-conventional concepts.

Most of the concepts of small reactors without on-site refuelling reactors are at early design stages. To make such reactors viable, further research and development (R&D) is necessary, inter alia, to validate long-life core operation, define and validate new robust types of fuel, justify an option of plant location in the proximity to its users, and examine possible niches that such reactors could fill in future energy systems.

To further research and development (R&D) in the areas mentioned above and several others, and to facilitate progress in Member States in design and technology development for small reactors without on-site refueling, the IAEA has conducted a dedicated Coordinated Research Project (CRP) entitled 'Small Reactors without On-site Refuelling' (CRPi25001). The project started late in 2004 and, after a review in 2008, was extended for one more year to be ended in 2009. The project has created a network of 18 research institutions from 10 Member States, representing both developed and developing countries.

Over the CRP period, collaborative results were achieved for many of the abovementioned research areas. Some studies highlighted new directions of research to be furthered after the CRP completion. Some studies remained the efforts of particular research groups but produced results of common interest.

Upon the advice and with the support of IAEA Member States, the IAEA provides a forum for the exchange of information by experts and policy makers from industrialized and developing countries on the technical, economic, environmental, and social aspects of SMR development and implementation in the 21st century, and makes this information available to all interested Member States by producing status reports and other publications dedicated to advances in SMR technology.

The objective of this report is to document reference points and conclusions achieved through coordinated research conducted within the CRP on 'Small Reactors without On-site Refuelling' and to suggest R&D activities to be furthered after the CRP completion. Being documented, the outputs of this CRP may foster further R&D and increase the capability of Member States to achieve progress in development and deployment of small reactors without on-site refuelling.

The report is intended for designers of advanced small and medium sized reactors and officers responsible for planning and implementation of R&D programmes on advanced technology development for nuclear power. Section 2 and Annex I to this report are intended also for regulators in Member States.

The IAEA officer responsible for this publication was V. Kuznetsov of the Division of Nuclear Power.

## CONTENTS

# SUMMARY

Small reactors without on-site refuelling are reactors of 300 MW(e) or less power rating that are designed for infrequent replacement of well-contained fuel cassette(s) in a manner that impedes clandestine diversion of nuclear fuel material. Small reactors without on-site refuelling are being developed for several reactor lines, including water cooled reactors, sodium cooled fast reactors, lead and lead bismuth cooled reactors, and also include some non-conventional concepts.

Most of the concepts of small reactors without on-site refuelling reactors are at early design stages. To make such reactors viable, further research and development (R&D) is necessary to validate long-life core operation, define and validate new robust types of fuel and examine possible niches that such reactors could fill in future energy systems.

To further R&D in areas mentioned above and to achieve progress in design and technology development for small reactors without on-site refuelling, in 2005-2009 IAEA has conducted a coordinated research project (CRP) titled ‘Small Reactors without On-site Refuelling’ (CRPi25001). The project has created a network of 18 research institutions from 10 Member States, representing both developed and developing countries.

The overall objective of the CRP was to increase capability in Member States to develop and deploy small reactors without on-site refuelling. The specific objectives were:

(1) To develop a vision statement for small reactors without on-site refuelling

(2) To develop a methodology to revise the need of evacuation and relocation measures beyond the plant boundary unique to NPPs with innovative SMRs and advanced reactors of larger capacity;

(4) To review the approaches to ensure long-life core operation without refuelling and to perform a comprehensive coordinated study of long-life cores for small reactors of various types with a focus on neutronics, thermal-hydraulics and new robust types of fuel;

(5) To identity possible niches and applications for small reactors without on-site refuelling and to outline pathways for commercialisation of plants with such reactors.

Over the CRP period, collaborative results were achieved for many of the abovementioned research areas. The project outputs are documented in this report to foster further R&D and increase the capability in Member States to achieve progress in development and deployment of small reactors without on-site refuelling. A short summary of the outputs is provided below.

The CRP participants have developed a vision statement for the small reactors without on-site refuelling in which they defined them as reactors of 300 MW(e) or less capacity designed for infrequent replacement of well-contained fuel cassette(s) in a manner that impedes clandestine diversion of nuclear fuel material. Small reactors without on-site refuelling could be:

- Factory fabricated and fuelled transportable reactors; or
- Reactors with a once-at-a-time core refuelling at a site performed by a special team that brings and takes away the core load and refuelling equipment

Specifically, small reactors without on-site refuelling are being developed to reduce the obligations of the user for fuel manufacture and for spent fuel and radioactive waste management.

The CRP participants agreed that distinctive features of small reactors without on-site refuelling are the following:

- A key feature is absence of the refuelling equipment present permanently in the reactor or at the site;
- Another key feature may be that no fresh or spent fuel is being stored on the site during reactor operation;
- Small reactors without on-site refuelling incorporate increased refuelling interval (from 5 to 30 years) consistent with plant economy and considerations of energy security;
- Such rely strongly on inherent and passive safety features and have a potential for simplified operational control, such as passive load follow operation;
- Small reactor does not necessarily mean a small nuclear power station; many small reactors without on-site refuelling are being designed as modules capable clustered of operation within a power plant of medium, large or even very large (1600 MW(e)) capacity.

The CRP participants have suggested that small reactors without on-site refuelling could be used as energy sources for:

- Cities in developing countries with small electricity grids and insufficient infrastructure;
- Autonomous small settlements;
- District heating and seawater desalination;
- Advanced non-electric applications, such as hydrogen production, coal liquefaction, etc.
- Remote areas in the North with complicated access and high cost of energy from fossil fuel;
- Dispersed or remote islands; mountainous areas, etc.

The CRP participants identified common themes among diverse concepts of small reactors without on-site refuelling as the following:

- For technical approaches, reduced core power density and use of (i) burnable absorbers in thermal neutron spectrum reactors or (ii) high core conversion ratio (CR~1) in fast neutron spectrum reactors to achieve long operation without on-site refuelling;
- For institutional approaches, reduction of the emergency planning zone extent and requirements to locate small nuclear power plants closer to the users; operation with centralized regional or interregional fuel cycle centres;
- For plant safety, strong reliance on inherent and passive safety features, taking an advantage from the resulting simplification and/or elimination of certain systems; reduction of stored potential energy hazard through implementation of new low-temperature types of fuel and chemically inert, ambient pressure coolants;

- For plant economy, transition from the economy of scale to the economy of serial fabrication of multiple reactor modules; taking an advantage from the economy of scale at large factories for fabrication of small reactor modules and at centralized fuel cycle centres; reduction of a financial risk to the buyer through increased project reversibility and high quality of factory production, with a potential to spread risks by involving additional stakeholders in the overall energy architecture.

The CRP participants reviewed the current bases for the definition of the emergency planning zone with the intent of re-introducing the concept of risk, previously ruled out due to technical considerations but especially due to public concern and distorted risk perception.

The CRP participants developed a conceptual methodology which would allow relating the size of the emergency planning zone to the safety performance of a plant design, thus recognizing the enhancement in safety attained by new plant designs during the last thirty years, i.e. since when the basis for emergency planning have been defined.

The methodology, which allows for a bridge (i.e. applicable to a relatively early design phase) towards the use of a full scope Level-3 PRA as the reference supporting tool in the definition and sizing of the emergency planning around a nuclear power plant, builds on the fundamentals of the concept of risk, i.e. a potentially complete probabilistic approach to the entire spectrum of accident scenarios and the deterministic evaluation of consequences through dose and dispersion analysis.

The methodology can provide a risk-informed definition of the emergency planning zone (EPZ), once the basic acceptance criteria in terms of limiting dose and limiting frequency have been provided (i.e. agreed upon with regulatory bodies). The limiting dose could be taken from the current regulations, but then it remains to define the limiting frequency. A reverse application of the methodology was proposed as a way to determine the level of risk associated with currently accepted EPZ size for the existing generation of nuclear power plants. Even though risk is not retained as the main defining basis for the EPZ size in most of the national regulations, a level of risk can actually be retrieved by measuring the frequency at which a pre-defined consequence is manifested at the distance from the plant which is currently adopted as the EPZ size. If measured with this approach and on the basis of the rationale that was selected for the EPZ size, the level of risk associated with the currently accepted EPZ size will also factor in the additional margin associated with the unique emotional perception of the nuclear risk. Such a risk value could then be used as the reference baseline for the definition of an EPZ for a new NPP design. The methodology proposed, supported by a performance-based licensing approach adopted in national regulations, could in this way allow a new NPP design to maintain the implied acceptable risk, while reducing the EPZ size.

The CRP participants performed trial application of the developed methodology. The simplified approach used for a first test case investigated, which was a NPP with the IRIS-like reactor (typical of several medium sized integral design PWRs) considered for the Caorso site in Italy. The test case was geared towards a feasibility and conceptual test of the methodology rather than towards the details of the analysis implementation. Nevertheless, the very preliminary results show the potential for a significant reduction in the size of EPZ for a small/medium sized nuclear power plant.

A detailed description of the practical aspects involved in the enforcement of the EPZ requirements was beyond the scope of the work performed within the CRP. Even without entering in the details, the beneficial impact on the economics of a hypothetical utility managing the Caorso IRIS-like NPP is easily understandable noticing the two relatively big population centers of Piacenza and Cremona (with up to 180 000 people in these two cities alone) being excluded from emergency planning by the newly re-defined EPZ approach.

While the benefit for such a reduction for the utility and the nuclear industry may be apparent, the main benefit for the final stakeholder (i.e. the public) is a reduced impact of the presence of the NPP from the economical and social points of view, due to the increase in safety and a corresponding reduction of the burden associated with outside emergency planning.

To illustrate this point, the CRP participants conducted a second test case for the real energy planning situation in Lithuania. The positive impact that a reduced emergency planning zone radius could have on deployments of medium sized integral type PWRs considered in an electricity/district heating co-generation mode was displayed parametrically against the assumed EPZ radius values.

For water cooled small reactors without on-site refuelling the CRP participants examined several advanced fuel options aimed at reducing hazard by reducing heat energy stored inside the system. Reducing stored heat energy; stored potential energy of chemical reactions; and stored mechanical (pressure) energy means that there is less to be dissipated should an off normal event occur, making it easier for passive (rather than active engineered) features to handle the dissipation tasks. The fuel forms considered were particle bed, pebble bed and cladded particulate fuel. The fuel types included tri-isotropic fuel (TRISO) originally developed for high temperature gas cooled reactors, but with outer coating layer made of SiC, and cermet fuel. Particulate types of fuel and, specifically, particulate bed fuel forms allow maintaining fuel temperature only a few degrees above the coolant temperature and reduce the characteristic time during which heat is transferred from fuel to the coolant. They also offer a huge heat exchange surface practically eliminating the issue of heat exchange crisis.

First, the issues of fuel performance were reviewed. As the first candidate for particulate-bed fuel, tri-isotropic (TRISO) fuel was investigated in the early phase of the CRP efforts. Out-of-pile corrosion testing of the TRISO particles with Si-C outer layer in hot water and in steam environments typical of reactor service conditions was shown to produce mixed results. Testing at VNIIAM (Russian Federation) was showing excellent corrosion resistance but testing at PNNL (USA) suggested corrosion issues. This led the PNNL fuel developers to consider an alternative particulate fuel form – cermet spheres made of $UO_2$ kernels in a Zr matrix coated with an outer Zr-1Nb layer impermeable to fission products.

Subsequently, during in-pile testing of the TRISO particles conducted by VNIIAM it was found that pyro-carbon and Si-C could experience integrity problems under low temperature irradiation, related to the accumulation of atomic displacements in the graphite lattice structure (Wigner energy), owing to insufficient annealing at temperatures below ~260°C. The open sharing of these testing results facilitated the designers to investigate whether or not their TRISO-fuelled designs could use the newly proposed cermet fuel form. The concepts of small light water reactors with micro fuel elements (MFE) were then re-designed using this new cermet type of particulate fuel. Preliminary studies performed during the project demonstrated suitability of such fuel for small water cooled reactors without on-site refuelling.

Second, benchmarking of the neutronic depletion codes on cell and fuel assembly models of small water cooled reactors with particulate-based fuel was performed, first for TRISO fuel option and then, for the reactor concepts re-designed for the cermet fuel. As it could be predicted, the differences in calculation results produced by different codes and data libraries for the reactors with new types of fuel were quite significant, reaching 2% $\Delta k/k$.

Specifically addressed were the differences between neutronics performance of TRISO versus cermet fuelling of the several concepts. On a general level, the unit cell results showed a higher $k_\infty$ and larger attainable discharge burnup owing to a much larger thermal component in the neutron spectrum for the TRISO fuelling as compared with the cermet. This issue can be addressed by adjusting the initial enrichment of fuel.

Collaborative activities on heavy liquid metal cooled fast-spectrum small reactors without on-site refuelling included a depletion benchmark exercise. A numerical whole core depletion model of a Pb-Bi reactor was developed by the Russian Research Centre 'Kurchatov Institute' and used as a benchmark to perform verification of the neutronic codes and data libraries. The calculations were carried out using different code systems and nuclear data derived from different libraries. Both deterministic and Monte Carlo methods have been used.

The results of calculations displayed notable differences – especially in $k_{\mathrm{eff}}$ – among the participants, with the spread reaching 1.5% $\Delta k/k$ at different moments during burnup cycle. The inter-comparison study has been aimed at identification of the sources of the discrepancies between the different methods and libraries.

Despite the noted differences, especially in $k_{\mathrm{eff}}$, a satisfactory level of consistency among different design teams was displayed on this first of a kind Pb-Bi alloy cooled fast spectrum depletion benchmark. While the degree of consistency lends some confidence to predictions of design performance at the conceptual and preliminary stages of design, the large spread in $k_{\mathrm{eff}}$ predictions makes it clear that critical experiments would ultimately be needed as the concepts progress toward advanced design stages.

Several studies of both near-term and of longer term nuclear park deployment approaches for parks containing significant share of small reactors without on-site refuelling were conducted within the CRP. The studies addressed attainable growth under constraints on internally generated and external fissile mass availability, on internally-generated and external capital financing availability, on mix of reactor types in the nuclear park, and on timing considerations for closing the fuel cycle. Some of the studies were based on idealized models and were intended to gain strategic insights to guide future higher-fidelity modelling.

The conclusions from the deployment approach studies include the following main insights:

- Fissile mass availability should not constitute a limit on quite significant growth of a nuclear park so long as:
    - the park contains significant market share of fast breeder reactors and fast spectrum small reactors without on-site refuelling that are fissile self sufficient; and
    - $^{235}U$ fuelling of these reactors can be used to accelerate the early introduction of these reactor types.
- On the other hand, capitalization for financing of an aggressive growth is much more confining than is fissile mass availability. Even given non-negligible reinvestment of profit (~25%) for self-financing of the new deployments, massive external cash flows (~100 billion US$/year) would be required for important but still only moderate (up to 5000 GW(e) within 100 years) growth. Small reactors without on-site refuelling can be effective in mitigating this financing challenge if they offer shorter (than that of the

economy of scale LWRs) on-site construction time which both hastens revenue generation and reduces interest during construction.

- The ability of fast spectrum small reactors without on-site refuelling to accommodate fuels of various isotopic composition (and to eventually, upon repeated recycle, convert any feed into an asymptotic mix of transuranic isotopes) provides valuable flexibility for the timing of closing of the fuel cycle. In this, $^{235}U$ fuelling may be used until such time as a cost advantage accrues to closing the fast reactor fuel cycle.

- A symbiotic fuel cycle for feeding LWR used fuel into the fast reactor closed fuel cycle need not require LWR fuel reprocessing. Instead, the LWR used fuel can be crushed and injected into fuel fabrication for fast spectrum small reactors without on-site refuelling as is, e.g. in a DUPIC type process. Harvesting of fissile mass from fast reactor spent fuel is ~10 times more efficient than from LWR spent fuel because the used fast reactor fuels have a ~10 times higher fissile content per unit mass. So it pays to wait for recycle until fast reactor fuel recycle is required.

The CRP established collaboration with OECD-NEA in benchmark thermal-hydraulic calculations of forced and natural convection modes of lead-bismuth simulating the tests performed in the HELIOS loop at the Seoul National University of the Republic of Korea. Several participants of the CRP contributed to these activities and continued their involvement after the completion of the CRP. It was agreed that the report on this exercise will be produced by OECD-NEA.

# 1. INTRODUCTION

## 1.1. Background

### *1.1.1. Rationale and developments in Member States*

Small reactors without on-site refuelling are reactors of 300 MW(e) or less power rating that are designed for infrequent replacement of well-contained fuel cassette(s) in a manner that impedes clandestine diversion of nuclear fuel material [1, 2]. Small reactors without on-site refuelling incorporate long refuelling interval (from 5 to 30 years) consistent with plant economy and considerations of energy security. Small reactors without onsite refuelling could be:

- Factory fabricated and fuelled transportable reactors; or
- Reactors with infrequent whole-core refuelling at the site performed by a special team that brings and takes away the refuelling equipment along with the used fuel.

Small reactors without on-site refuelling are intended to meet the needs of new classes of customers for nuclear energy – customers for which a smaller power rating and shared fuel cycle and waste management arrangements are better tailored to their energy supply needs[1] than are traditional large-scale plants with indigenous supporting fuel cycle and waste management infrastructure. Specifically, small reactors without on-site refuelling may essentially reduce or even eliminate the obligations of the user for dealing with fuel manufacture and with spent fuel and radioactive waste.

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[1] Which may include a variety of non-electrical applications.

In 2009, more than 25 design concepts of such reactors were analyzed or developed in IAEA Member States representing both developed and developing countries. Small reactors without on-site refuelling are being developed for several reactor lines, including water cooled reactors, sodium cooled fast reactors, lead and lead bismuth cooled reactors, and also include some non-conventional concepts [2].

Most of the concepts of small reactors without on-site refuelling reactors are at early design stages. To make such reactors viable, further research and development (R&D) is necessary to validate long-life core operation, define and validate new robust types of fuel and examine possible niches that such reactors could fill in future energy systems.

In many cases, small reactors without on-site refuelling are being developed for operation in remote off-grid locations. Most of the concepts foresee flexible non-electrical applications. Such conditions of operation may require proximity to the users, which puts forward the task of reducing emergency planning requirements for such reactors. Such reduction could be possible with the new, more robust types of fuel or via making a transfer to chemically inert ambient pressure coolants. A methodology to justify such revision needs to be developed and accepted by the regulators.

To further R&D in areas mentioned above and to achieve progress in design and technology development for small reactors without on-site refuelling IAEA has conducted a Coordinated Research Project (CRP) entitled 'Small Reactors without On-site Refuelling' (CRPi25001). The project has been started late in 2004 and, after a review in 2008, was extended for one more year to be ended in 2009.

The project has created a network of 18 research institutions from 10 Member States, representing both developed and developing countries. The participating research institutions were Eletronuclear and Federal University of Rio Grande do Sul (Brazil), Bhabha Atomic Research Centre (India), Bandung Institute of Technology (Indonesia), Politecnico di Milano (Italy), Hokkaido University and Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology (Japan), Lithuanian Energy Institute (Lithuania), Mohammed V University (Morocco), All Russian Institute of Atomic Machine Building, Russian Research Centre 'Kurchatov Institute', EDO 'Gidropress' and Institute of Physics and Power Engineering (Russian Federation), Argonne National Laboratory, Pacific Northwest National Laboratory and Westinghouse Electric Company (USA). At certain stages of the project, also participating were the Institute of Nuclear Science and Technology (Vietnam) and the Faculty of Electrical Engineering of the University of Zagreb (Croatia).

The structure of the project and the scope of its research activities were defined at the first Research Coordination Meeting convened at IAEA late in 2005. At that meeting it was decided that the project would have 4 research groups:

- Group 1'Revising the need of relocation and evacuation measures unique to all nuclear power plants (NPPs) with innovative small and medium sized reactors[2] (SMRs)'
- Group 2 'Feasibility studies for small water cooled reactors with new, robust types of fuel'
- Group 3 'Core neutronics and thermal-hydraulics of small lead and lead bismuth cooled reactors'

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[2] According to the classification adopted by IAEA, small and medium sized reactors (SMRs) are reactors with the equivalent electric output of less than 700 MW.

- Group 4 ‘Intra-regional and inter-regional scenario studies for energy systems involving small reactors without on-site refuelling’.

Over the CRP period, collaborative results were achieved for many of the abovementioned research areas. Some studies highlighted new directions of research to be furthered after the CRP completion. Some studies remained the efforts of particular research groups but produced results of common interest.

The outputs of this CRP may foster further R&D and increase the capability in Member States to achieve progress in development and deployment of small reactors without on-site refuelling. The final Research Coordination Meeting convened in November 2008 recommended that results of this CRP are published. Because of the preliminary nature of the results, it was decided that the CRP report will be published as an IAEA-TECDOC.

### *1.1.2. Previous IAEA publications*

With support from the participants of the CRP, the IAEA produced a status report on the rationales for, and the development status of, concepts of small reactors without on-site refuelling being actively developed or analyzed in Member States. Thirty concepts of small reactors without on-site refuelling were described using a consistent reporting format. The resulting compendium, IAEA-TECDOC-1536 on ‘Status of Small Reactor Designs Without Onsite Refuelling’[3], was released in January, 2007 [2].

The thirty concepts described in IAEA-TECDOC-1536 are nearly evenly divided among thermal spectrum/water cooled designs and fast spectrum/liquid metal cooled designs. Three of the designs are very high temperature reactor concepts intended to support hydrogen production based on water decomposition using nuclear heat and/or nuclear produced electricity. Figure 1 provides a graphical synopsis of the distinct groups of concepts of small reactors without on-site refuelling. The indicated deployment dates are on the optimistic side, assuming the financing needed to develop the technology and deploy the reactor is continuously available. In reality, the time needed to develop and deploy many of the indicated concepts is likely to be longer, depending on the circumstances for each particular project.

In July 2009, the IAEA published a Nuclear Energy Series report NP-T-2.2 entitled ‘Design Features to Achieve Defence in Depth in Small and Medium Sized Reactors’ [4]. The objective of that report was to assist developers of SMRs in Member States in defining consistent strategies regarding (i) elimination of accident initiators/ prevention of accident consequences by design, and (ii) incorporation of inherent and passive safety features and passive systems in safety design concepts of such reactors. Another objective was to assist potential users in Member States in their evaluation of the overall technical potential of SMRs with passive safety design features, including possible implications in areas other than safety. Among the 11 SMR concepts addressed, four represent small reactors without on-site refuelling.

[3] The report can be accessed through a link at the public website: http://www-pub.iaea.org/MTCD/publications/PDF/te_1536_web.pdf

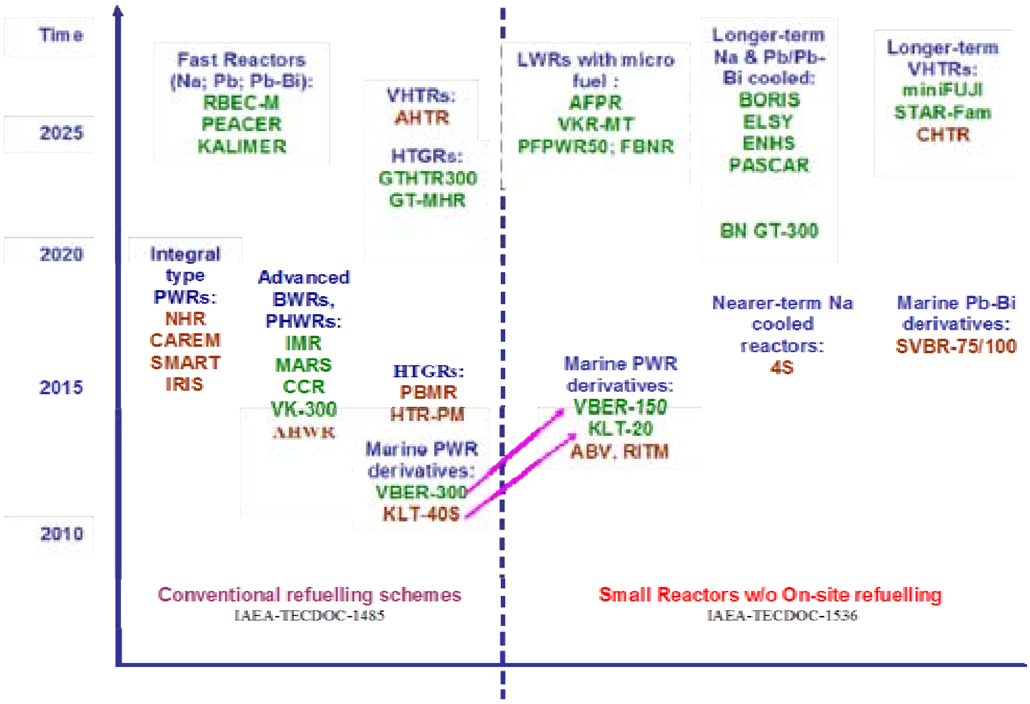



*FIG. 1. Deployment potential of innovative SMRs; right part [2], left part [3] (PWR - pressurized water reactor, LWR – light water reactor, PHWR – pressurized heavy water reactor, HTGR – high temperature gas cooled reactor, TRISO - tri-isotropic, VHTR – very high temperature reactor .*

### 1.2. Objective

The objectives of this report are:

- To document reference points and conclusions achieved through coordinated research conducted within the CRP ‘Small Reactors Without On-site Refuelling’;
- To suggest R&D activities to be furthered after the CRP completion.

The report is intended for research organizations in developed and developing countries involved in programmes on design and technology development for innovative nuclear reactors; energy planners in Member States considering longer term nuclear energy programmes; regulators considering risk-informed approaches to NPP qualification and licensing; students of nuclear engineering specialities in universities.

### 1.3. Scope of the coordinated research project

The overall objective of this CRP was to increase capability in Member States to develop and deploy small reactors without on-site refuelling.

The specific objectives were:

(1) To develop a vision statement for small reactors without on-site refuelling

(2) To carry out data and information exchange regarding the progress in design and technology development for the concepts of small reactors without on-site refuelling being developed by the participants (SVBR 100, SVBR 10, SPINNOR and VSPINNOR, CANDLE, AFPR-100, PFPWR50, FBNR, VKR-MT and others, see [2]);

(3) To develop a methodology to revise the need of evacuation and relocation measures beyond the plant boundary unique to NPPs with innovative SMRs

(4) To review the approaches to ensure long-life core operation without refuelling and to perform a comprehensive coordinated study of long-life cores for small reactors of various types with a focus on neutronics, thermal-hydraulics and new robust types of fuel

(5) To identity possible niches and applications for small reactors without on-site refuelling and to outline pathways for commercialisation of plants with such reactors.

The project was carried out through research agreements or research contracts with participating institutions and included three research coordination meetings RCMs held on 21-25 November 2005, 4-8 June 2007 and 3-6 November 2008, and the consultants' sessions held in Vienna in December 2003 and March 2004 to help define the structure of the project. Based on the discussions at these sessions, the participants organized themselves into four groups to jointly address issues of their particular interest (most institutions participated in more than one group).

The first group pursued the institutional strategy of reduced emergency planning zone – generally applicable to all innovative SMRs. The second and third groups pursued in depth the technology – specific aspects of a reduction of stored energy hazard. The fourth group investigated sequencing issues encountered in aggressive capacity growth scenarios for small reactors without on-site refuelling in the face of constraints on fissile inventory availability and of capitalization availability. The groups and their membership were:

Group 1: Revising the need for relocation and evacuation measures unique to NPPs with innovative SMRs

Eletronuclear (Brazil), (Faculty of Electrical Engineering of the University of Zagreb (Croatia), Politecnico di Milano (Italy), Lithuanian Energy Institute (Lithuania), Westinghouse Electric Company (USA)

Group 2: Design and technology development for light water reactors with particulate based fuel

Federal University of Rio Grande Do Sul (Brazil), Hokkaido University (Japan), Mohammed V University (Morocco), All Russian Institute of Atomic Machine Building (Russian Federation), (Institute of Nuclear Science and Technology) Vietnam, and Pacific Northwest National Laboratory (USA)

Group 3: Design and technology development for Pb, Pb-Bi and molten salt cooled reactors

Bhabha Atomic Research Centre (India), Bandung Institute of Technology (Indonesia), Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology (Japan), Russian Research Centre 'Kurchatov Institute', Institute of Physics and Power Engineering, EDO 'Gidropress' (Russian Federation), Argonne National Laboratory (USA)

Group 4: Capacity growth scenarios for small reactors without on-site refuelling

Institute of Physics and Power Engineering (Russian Federation), Argonne National Laboratory (USA)

At the project meetings it was decided that:

- All groups will elaborate a vision statement for small reactors without on-site refuelling;
- Group 1 will critically examine previous experience and develop a methodology to justify reduced emergency planning requirements for innovative reactors, and will perform trial application of the methodology to a SMR case;
- Group 2 will study new, robust types of fuel for water cooled reactors with long-life cores and also develop cell and fuel assembly depletion models of light water reactors with new, robust types of fuel and perform comparative neutronic calculations of these models to identify discrepancies between codes and data libraries and the causes of such discrepancies;
- Group 3 will develop whole core depletion model of a lead-bismuth cooled reactor and perform comparative neutronic calculations of these models to identify discrepancies between codes and data libraries and the causes of such discrepancies, and also develop a method for fission product simulation in neutronic calculations of long-life reactor cores and compare it to conventional calculation methods;
- Group 4 will use scenario codes to examine material and cash flows in energy systems including conventional water cooled reactors, fast breeder reactors, and small reactors without on-site refuelling with a fast neutron spectrum, to define possible niches for small reactors in future energy systems.

In 2007, collaboration with OECD-NEA was established in benchmark calculations of forced and natural convection modes of lead-bismuth simulating the tests performed in the HELIOS loop at the Seoul National University of the Republic of Korea. Four participants of Group 3 contributed to these activities, via the CRP ‘Small Reactors without On-Site Refuelling’.

### 1.4. Scope of the report

The report includes 8 Sections and 6 Annexes. The Sections present summaries of the results of particular R&D achieved within the project. The Annexes, referenced from corresponding chapters, present detailed results obtained for selected particular topics of the research. They are located on a CD-ROM attached to the back cover of this report.

Organized in this way, the report covers most of, but not all results produced during the project lifetime. Some preliminary or unfinished individual studies, such as those related to the development of a method for fission product simulation in neutronic calculations of long-life reactor cores, are not included. It is also assumed that the results of HELIOS benchmark exercise and relevant contributions of the participants of the present CRP will be duly presented in the future report by OECD-NEA.

In addition to this, the report presents the updates on certain concepts of small reactors without on-site refueling or even some new concepts that have not been previously addressed in reference [2].

Section 1 is an introduction; it describes the background, introduces the rationale, and defines the objectives and targeted users of the report. This chapter also explains the structure of the project and the report, and the approach to report preparation.

Section 2 presents the vision statement for small reactors without on-site refuelling as jointly developed by all participants of the project.

Section 3 identifies and discusses common themes among the diverse concepts of small reactors without on-site refuelling. The provided discussion is based on the outputs of the information exchange among the participants of the project and acts as a navigation tool for all subsequent chapters of the report.

Section 4 summarizes the developed methodology for justification of a reduction of emergency planning requirements, complete with an example of application of such methodology. More details of the methodology and its applications are provided in Annex I.

Section 5 describes the outputs of studies of different advanced types of particle-based fuel for small light water reactors, and also presents the results of the neutronic benchmarking carried out for cells and fuel assemblies of several concepts of such reactors. References are made to more detailed information provided in Annexes II and III.

Section 6 summarizes the results of depletion calculations performed for the original whole-core benchmark model of a Pb-Bi cooled fast reactor and presents some design updates on new concepts of small reactors with heavy liquid metal coolant.

Section 7 presents the examples of deployment approaches for small reactors without on-site refuelling under financings constraints and fissile mass availability constraints.

Section 8 presents the conclusions and recommendations. It is elaborated as an executive summary of the report and includes the suggestions for further research as identified on a consensus basis by all participants of the CRP.

Annex I presents details of the developed methodology for justification of a reduction of emergency planning requirements and its trial application.

Annex II presents an update for the concept of a small boiling water reactor with particle-based fuel, relevant to the discussion in Section 5.

Annex III presents the results of neutronic depletion code benchmarking performed for several concepts of water cooled small reactors with particle based fuel. This Annex is referenced from Section 5.

Annex IV presents full results of depletion calculations performed for the original whole-core benchmark model of a Pb-Bi cooled fast reactor. This Annex is referenced from Section 6.

Annex V presents brief description of a concept of a 10 MW(e) lead-bismuth cooled reactor SVBR-10. This design developed by EDO ‘Gidropress’ was not presented in the status report of small reactor designs without on-site refuelling [2].

Annex VI presents the designs status of a multipurpose power pack for satisfying energy related needs in remote Indian villages. The power pack is being developed by the Bhabha Atomic Research Centre (India) and is a design alternative to the Compact High Temperature Reactor previously described in reference [2]. The power pack operates at lower temperatures and the type of fuel employed is different from that described in [2]. Annexes V and VI are referenced from Section 6.

Finally, Annex VII presents details of the CANDLE and modified CANDLE burnup concepts for heavy liquid metal cooled reactors. The summary of these concepts is provided in Section 6, from which Annex VII is referenced.

Contributors to drafting and review of this report are listed on the last page.

### 1.5. Approach to preparation of the report

The report is based on the deliverables of the CRP participants submitted to IAEA in the period from 2004 till 2009. Annexes to the report are direct contributions from the project participants. Main part of the report was jointly developed by the participants and the secretariat. Conclusions and recommendations to the report were elaborated during the final research coordination meeting held in Vienna on 3-6 November 2008 and later updated via direct correspondence with all participants. The complete report was reviewed by all participants of the project.

## 2. VISION STATEMENT FOR SMALL REACTORS WITHOUT ON-SITE REFUELLING

The CRP participants have developed a vision statement for small reactors without on-site refuelling as presented below.

*What are the small reactors without on-site refuelling?*

Small reactors without on-site refuelling are reactors of 300 MW(e) or less designed for infrequent replacement of well-contained fuel cassette(s) in a manner that impedes clandestine diversion of nuclear fuel material. Small reactors without on-site refuelling could be:

- Factory fabricated and fuelled transportable reactors; or
- Reactors with a once-at-a-time core refuelling at a site performed by a special team that brings and takes away the core load and refuelling equipment

Specifically, small reactors without on-site refuelling reduce the obligations of the user for spent fuel and radioactive waste management

*What are the features of small reactors without on-site refuelling?*

- A key feature is absence of the refuelling equipment present permanently in the reactor or at the site
- Another key feature may be that no fresh or spent fuel is being stored on the site during reactor operation
- Small reactors without on-site refuelling incorporate increased refuelling interval (from 5 to 30 years) consistent with plant economy and considerations of energy security
- Small reactors without on-site refuelling rely strongly on passive safety design options and have a potential for simplified operational control, such as passive load follow operation, see [4]
- Small reactor does not necessarily mean a small nuclear power station; many small reactors without on-site refuelling are designed as modules capable of operation within a power plant of medium, large or even very large (1600 MW(e)) capacity

*What could be the applications for small reactors without on-site refuelling?*

- Cities in developing countries with small electricity grids and insufficient infrastructure
- Autonomous small settlements
- Power sources for district heating and desalination plants

- Power sources for advanced non-electric applications, such as hydrogen production, coal liquefaction, etc.
- Remote areas in the North with complicated access and high cost of energy from fossil fuel
- Dispersed or remote islands; mountainous areas, etc.

*What are the special features of small reactors without on-site refuelling?*

- Through adding a certain degree of independence on fuel supplier, small reactors without on-site refuelling could, perhaps, secure a motivation for a state to skip the development of the indigenous fuel cycle
- Small reactors without on-site refuelling may provide an attractive domain for fuel, reactor module or nuclear power plant leasing
- Small reactors without on-site refuelling could facilitate implementation of adequate safeguards in a scenario of expanded deployment of nuclear power, through:
    - Operation with weld-sealed reactor vessel and remote monitoring
    - Absence of refuelling equipment and fuel storages at the site,
    - Streamlined safeguard missions, etc.

*What could be energy systems with small reactors without on-site refuelling?*

- The proposers of small reactors without on-site refuelling often consider them in conjunction with centralized, perhaps, regional fuel cycle centres, probably, operated under an international control:
    - When systems with small reactors without on-site refuelling and regional fuel cycle centres are added to an overall energy system, this would reduce the degree of worldwide dispersal of the enrichment, reprocessing, fuel fabrication and, possibly, waste repository facilities
    - An integrated closed fuel cycle with light water reactors (LWR) and small reactors without on-site refuelling offers a potential for the reduced management of LWR spent nuclear, via relatively (compared to LWR mixed oxide fuel) long fuel sequestration in power-producing small reactors without on-site refuelling.

*What are the design approaches to ensure long-life core operation?*

- Reduced core power density
- Burnable absorbers (in thermal reactors)
- High conversion ratio in the core (in fast reactors)
- Refuelling performed without opening the reactor vessel cover (which offers a potential to develop a large reactor without on-site refuelling).

Most, although not all of small reactors without on-site refuelling end at the same or less values of fuel burnup and irradiation on the structures as achieved in conventional reactors based on the same basic technology.

*What is the design status of small reactors without on-site refuelling?*

- Nearly 25 concepts and designs of small reactors without on-site refuelling are being developed worldwide; the majority are at a pre-conceptual or conceptual design stage, but some have reached basic or even detailed design stage (the examples could be ABV and SVBR 100 (Russian Federation), 4S (Japan))

- The targeted dates for prototype deployment range from ~2010 to ~2030.

*What are the R&D needs for small reactors without on-site refuelling?*

- Most of the concepts of small reactors without on-site refuelling need validation, testing and demonstration of safety and reliability under long-life core operation. The available (shared) experience of marine reactors is limited by 7-8 years of continuous operation; while some concepts of civil small reactors without on-site refuelling target a 25-year or even a longer core lifetime
- Technologies of remote refuelling need to be developed; safety of a spent fuel load/reactor transportation needs to be proven for short cooling periods after operation
- An approach to licensing of long-life core operation needs to be elaborated and established (e.g. a 'license-by-test' approach, see [4])
- Construction and operation of a prototype plant appears to be a must for all concepts of small reactors without on-site refuelling
- Economic competitiveness of small reactors without on-site refuelling needs to be proven for anticipated conditions of their application. It may well be a diesel generator or candle lighting rather that a combined gas-turbine cycle plant that small reactors will compete with; however, the economies of factory fabrication of multiple units and advantages related to incremental capacity increase equally need to be examined
- Legal, institutional and infrastructure provisions for transportation of factory-assembled reactor modules/ plants and operation with regional fuel cycle centres need to be elaborated.

## 3. COMMON THEMES AMONG DIVERSE CONCEPTS OF SMALL REACTORS WITHOUT ON-SITE REFUELLING

The topics for discussions among the CRP participants were extremely diverse as might be expected for so diverse range of small reactor concepts. But even in the face of the diversity, an underlying theme could have been discerned as providing the impetus for much of the coordinated research. Having given up traditional economy of scale in order to meet the emerging market needs for power sources of smaller rating, each concept of a small reactor without on-site refuelling is required to find compensating strategies to retain an economically-competitive position. The diversity of approaches that have been taken revealed a range of innovations – both technical and institutional. They can be broadly categorized into the following seven common approaches.

### 3.1. Technical approaches

a) Economy of serial factory production of multiple standardized small reactor replicates, combined in some cases with the economy of scale for the supporting fuel cycle facility infrastructure;

b) Simplification and/or elimination of systems through application of passive safety features;

c) Reduction of safety hazard and the costs of resultant mitigating measures to be attained by reduction of the potential energy stored inside the reactor vessel – thermal energy, energy of high pressure, and/or chemical potential energy.

### 3.2. Institutional approaches

a) Reduction of the extent and associated costs of the emergency planning zone (EPZ) by crediting the reduced radioactive source term attendant to smaller power rating and by using a risk-informed methodology for determining off-site risk that credits both reduced source term hazard and reduced release frequency owing to the application of passive safety features

b) Reduction of financial risk to the buyers through factory serial fabrication of modularized standardized reactors followed by rapid on-site assembly that shortens the period of on-site construction and both reduces its associated cost of interest during construction and removes the uncertainty as to construction duration

c) Providing flexibility by more closely tracking demand growth with asset deployment conducted in smaller increments – thereby delaying commitment for financial outlay until demand growth can be more precisely gauged

These strategies are briefly described below, and discussions of their applications to specific cases provide more detail in later chapters of this report.

### 3.3. Economy of serial factory fabrication

The enormous energy density of nuclear fuel, and the ability to ship it on world encompassing distances with no spoilage, leakage, or other loss of energy content, when combined with the reactor designs that permit refuelling intervals to be extended up to several decades, offers the potential for a transformative energy supply architecture wherein economy of scale is employed for the fuel cycle infrastructure, but traditional economy of scale is replaced with the economy of mass production for the reactors of small power rating used for the local energy delivery itself. This architectural strategy allows changing the traditional approach of multi-year field construction of very large nuclear power plants to the one of factory fabrication of small modularized reactors suitable for easy shipment and rapid assembly at user sites. The factory fabrication may invest reactor manufacture with the very same economies of serial mass production and learning curve benefits that are found in the automobile, airplane, and other mass production industries, see [2]. These economies are unavailable to large nuclear power plants that require construction in the field. If large power output from a single site is required, then a cluster of smaller standardized units can be provided sequentially.

All concepts and designs of small reactors without on-site refuelling considered in the CRP are targeting this strategy of factory production of smaller units or modules.

### 3.4. Simplification and/or elimination of systems through application of passive safety features

Traditional reactor designs place heavy reliance on engineered safety features to maintain the reactor in a safe operating envelope and/or to mitigate the consequences of off-normal situations, should they arise. These engineered systems require assured power sources for sensing, actuation, and performance of their function. They add to design complexity, to capital cost and moreover, their lifetime maintenance and continual need to confirm their readiness to perform their function adds to operation and maintenance (O & M) costs.

In recent years, reliance on passive safety features that operate under laws of nature - requiring no assured power sources nor active sensing and switching equipment – has attained significant application in reactor design. Such passive features can always be used to back up active safety systems and in some cases can even replace or at least simplify active systems, see [4].

All small reactors considered in the CRP rely to a greater or lesser extent on the use of passive safety features to simplify their design and reduce their capital cost.

### 3.5. Reduction of stored potential energy hazard

Reduction of stored energy hazard helps to mitigate the strength required of, and/or the protection measures for the defence in depth barriers erected to preclude radioactivity release. As one example from this CRP, the use of tri-isotropic (TRISO) and/or cermet particulates rather than fuel pins in water cooled reactors vastly increases surface to volume ratio of the fuel and, thereby, vastly decreases the fuel temperature rise above the coolant temperature and practically eliminates the possibility of a heat exchange crisis, for more details see Section 5.

Similarly, the use of nitride or metal alloy fuel rods in fast spectrum heavy liquid metal cooled reactors with near-ambient pressure of the primary coolant system achieves the same goal (as compared to oxide fuel which has a 10-times smaller thermal conductivity value and to reactors with high primary circuit pressure which is a force capable of driving radioactivity out of the primary pressure boundary, potentially increasing the source term), for more details see Section 6.

In both these cases, the stored thermal energy that would have to be dissipated in order to reach a safe shutdown state from the operating state is vastly reduced. Additionally, the amount of positive Doppler reactivity that is released in cooling the fuel to the safe shutdown state is vastly reduced as well. As a result, the coolant temperature rise before reaching safe shutdown conditions can be reduced to remain in the safe range.

As an example of reduction in stored chemical energy, some water-cooled small reactors without on-site refuelling have considered TRISO particle fuel to remove potential for hydrogen production from Zr-water chemical reactions at the high temperatures reached in severe accident situations, see Section 5 and Annex II. For fast neutron spectrum, numerous Pb and Pb/Bi – eutectic cooled reactor designs remove the hazard of Na-water chemical reactions, thereby, offering the simplification advantage attendant to eliminating the intermediate coolant system (in plants using the Rankine steam cycle). Alternately, replacing the steam cycle with a S-$CO_2$ Brayton cycle offers the same simplification opportunity and has been exploited in some heavy liquid metal cooled reactor designs, for more details see Section 6.

Examples of the approaches to reduce the stored pressure energy hazard include the use of low vapour pressure liquid metal coolants in fast reactors (Section 6) and use of integral layouts (with in-vessel steam generators reducing high pressure primary piping runs) for water cooled SMRs (Section 5). Design pressure requirements on the containment may be mitigated – with attendant reductions of the construction costs.

### 3.6. Reduction in the spatial extent of the emergency planning zone

The spatial extents of regulatory-mandated emergency planning zones (EPZ) have historically been set based on conservative approaches to calculating bounding individual dose rates subsequent to a postulated accident sequence. The zones are not small – ranging up to 10 miles in radius. Moreover, regulations often require the reactor owner to provide for emplacement of infrastructure such as roads and bridges throughout the EPZ to facilitate public evacuation – as well as to periodic training and equipment supply to first responders. Current practice has been developed over many years specifically for the historical and current situation of large water-cooled reactor installations generating electricity for a regional grid.

Alternately, small reactors without on-site refuelling are being designed for local grids and some are even designed for cogeneration missions wherein the reactor must of necessity be placed very near the cogeneration application due to short heat transport distances. EPZ defined for large reactors on a one-size-fits-all basis can place a severe economic disadvantage on small reactors without on-site refuelling. For this reason, the CRP has conducted a review of the basis for the current regulations and has proposed a risk-informed methodology which could justify a reduced emergency planning zone extent on the basis of a smaller source term and a reduced probability of release for advanced small reactors, accounting for their passive safety and other risk reduction features. The methodology is not limited to small reactors without on-site refuelling, but is unique to many NPPs with innovative SMRs and larger reactors.

The study conducted in the CRP includes a sample application of the developed methodology for the IRIS-like SMR design [3] under conditions of a particular site. This application indicates a potential for remarkable reduction of EPZ radius without increase in the public risk. However, to achieve this practically the proposed methodology first needs to be embraced by regulatory authorities. The developed approach, which is applicable and recommended for all types of small reactors without on-site refuelling, is summarized in Section 4. More details of the methodology and its trial application are provided in Annex I.

### 3.7. Reduction of financial risk to the buyer

Small reactors without on-site refuelling offer three features which are significant for financial risk reduction for the buyer. First, because the power rating is smaller, the overall capital outlay is correspondingly smaller as well. The debt burden assumed for a new deployment becomes a smaller fraction of the buyer company's overall capitalization.

Second, because of factory fabrication and rapid site assembly of modules, the duration of site construction is expected to be only one or two years – down from 4-5 years for the traditional economy of scale site construction campaigns. Shortening the site construction period by assembling standard modules rather than carrying out field construction not only reduces the duration of site construction and the cost for interest during construction, but also decreases the uncertainty on construction schedule and its associated risk premium attached to the construction loan.

Finally, by adding capacity in smaller increments, with 1 or 2 year site construction duration, the capacity growth can better track demand growth – leading to less uncertainty in demand forecast than is possible using larger capacity increments but longer emplacement durations. Some preliminary considerations of this last feature have been investigated in the CRP for the specific case of Lithuania (implication for co-generation applications) and, more generically, a methodology has been developed and tested for the IRIS-like SMR, for details see Section 4.

# 4. REDUCTION OF EMERGENCY PLANNING ZONE

The spatial extents of regulatory-mandated emergency planning zones (EPZ) around nuclear plants have historically been set based on conservative approaches to calculating bounding individual dose rates subsequent to a postulated accident sequence. The zones are not small – ranging from 5 km to 10 miles in radius in different Member States[4]. Moreover, regulations often require emplacement of evacuation infrastructure such as roads and bridges throughout the EPZ, as well as periodic training and equipment supply to first responders. In some Member States, it is a utility's obligation to cover the insurance of all inhabitants of the emergency area.

To make smaller reactors economically viable, a risk-informed methodology to define emergency planning zone radius on a case-by-case basis for each individual plant and plant site has been developed within the project. This methodology and the results of its trial application are summarized in brief in the following sections. More details about the methodology are available from Annex I.

## 4.1. A proposed EPZ redefinition methodology

The first part of the study comprised a review of the EPZ regulations in several Member States including Belgium, Czech Republic, Finland, France, Japan, Lithuania, Slovakia, Switzerland, the United Kingdom, and the USA. Substantial variability exists, but for the most part:

- Zone size is site dependant
- Zone size is determined by specified limits on dose received by an individual receptor at the zone boundary over a specified period of time;
- Zone size is calculated assuming the most severe design basis accident (or even a more severe beyond design basis accident)

Deterministic rather than probabilistic analyses are used. Generally speaking, a multi-reactor site and a single reactor site would have the same EPZ boundary (presumably based on the assumption of statistical independence of reactor malfunctions between reactors). The thus defined EPZ radiuses range from 1 to 20 km. The second part of the study critically reviewed the history of EPZ-related regulation - focusing on recent (past 15 years) attempts aimed at the redefinition of EPZ determination in light of developments in the LWR design (Generation-III, Generation-III$^+$) and in the regulatory philosophy (increasing reliance on risk-informed regulation). In the USA, previous reassessments have been undertaken by licensees, by the Nuclear Regulatory Commission (NRC) staff, and by independent organizations (EPRI, NEI). Internationally, the issue of revision of EPZ approach has been reviewed by the IAEA and the European utility requirements (EUR) initiative. While differing in scope and timing, a recurring theme in these reassessments has been the notion to introduce risk into the evaluation.

The third part of the study proposed and developed an integrated methodology for EPZ redefinition. The proposed methodology makes use of accepted concepts such as probabilistic risk assessment (PRA) techniques and deterministic dose evaluation as used in current practice; it suggests a more complete definition of the current and accepted criteria for the EPZ by focusing on the frequency of exceeding a given dose at a given distance accounting for the full spectrum of occurrences. The EPZ radius is redefined while still maintaining the

[4] It is noted that in some Member States, such as the Russian Federation, it is possible to justify smaller (e.g. 1 km or even less) EPZ radiuses within the currently enforced regulations.

same dose (explicitly defined in the current protective action guide (PAG)) and the same frequency (implicitly defined by the choice of a fixed distance) defined by the NRC.

The proposed methodology addresses the two conceptual weaknesses highlighted from previous efforts for the redefinition of the EPZ defining criteria:

- In the deterministic component of the methodology all the foreseen sequences are evaluated with no exclusion of severe accidents, which are obviously expected to be the limiting scenarios and cannot be removed from the analysis without infirming the completeness of the methodology (previous attempts in the EPZ redefinition were rejected because they lacked a satisfactory account of severe accidents);
- The probabilistic component is shifted from establishing a cut-off frequency to being a screening criterion of accident sequences by evaluating the frequency to overcome the dose limit at a certain distance. By means of the data provided by PRA, such a distance can be evaluated rather then pre-set (arbitrary selection of a value for the cut-off frequency represented the major objection against the earlier probabilistic approaches to EPZ redefinition).

The probabilistic starting point of this methodology (i.e. step 1) essentially covers the choice of the set of release scenarios to be addressed by a deterministic evaluation of the consequences. In order to obtain this outcome, the entire spectrum of accident sequences defined through the PRA of the plant must be reviewed and re-categorized. No additional cut-off frequencies are introduced, but the same truncation level applied and accepted for the PRA development must be maintained and should reasonably guarantee to cover all unlikely sequences. Similar accidents (in term of the release) could, of course, be lumped together to limit the analytical burden to a manageable level.

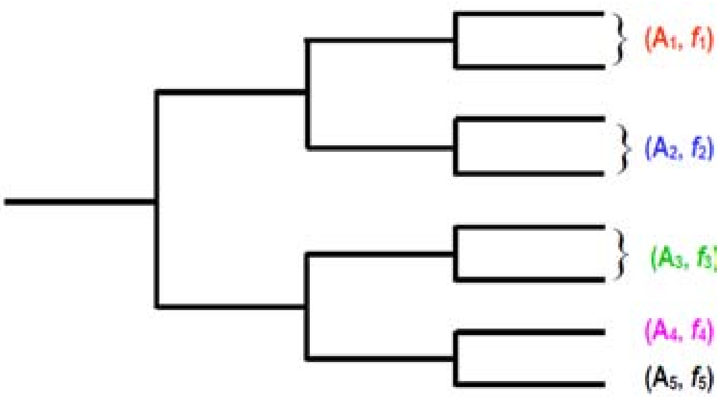



*FIG. 2. Step 1 of the methodology: Accident sequence re-categorization.*

A set of release scenarios ($A_i$) with their related frequency $f_i$ of occurrence is, therefore, the outcome of this first step of the methodology. Figure 2 shows a schematic representation of the results of this step for the case where five release scenarios are obtained after the re-categorization.

Once the set of release scenarios has been identified, the second step is a deterministic evaluation of the consequences. Appropriate assumptions must be made in order to outline the scenario phenomenology; such assumptions should be based in a wider extent on best estimate, realistic models rather than on large and over-conservative safety coefficients. Using appropriate codes, the dose absorbed by a hypothetical individual located at various distances from the reactor, during the days (especially the first hours) after the onset of the accident is calculated. This calculation should be performed considering a complete set of meteorological conditions.

The final outcome of this step is a set of curves of dose equivalent (D) versus distance (x), one curve for each release scenario, $A_i$, regardless of the frequency of the selected accident. Figure 3 provides an example of the results of this second step applied to the five release scenarios hypothetically identified in Figure 2.

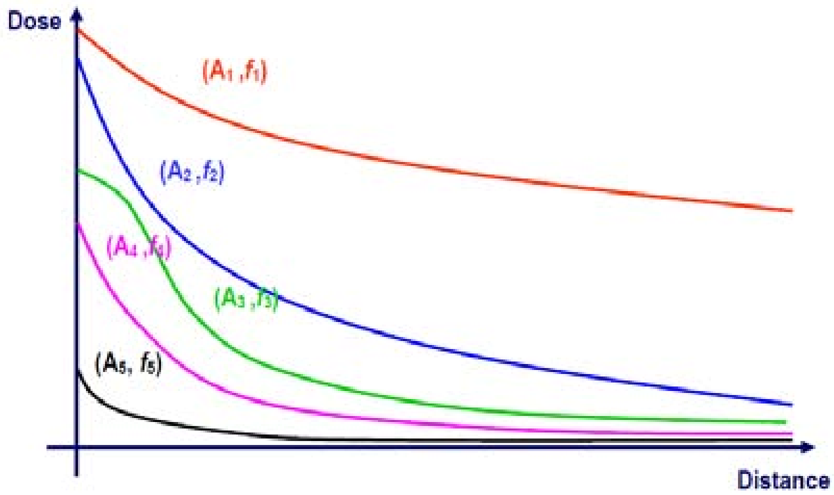



*FIG. 3. Step 2 of the methodology: Dose versus distance evaluation.*

To be able to combine the probabilistic and the deterministic contributions, a limiting dose D* and a limiting frequency f* are identified in steps 3 and 4, respectively. These are the two steps that require licensing considerations, as the limiting values should ideally be suggested by and/or agreed with a regulatory body.

Even though these two steps will require further investigation, as far as the limiting dose (D*) is concerned, the current mainly consequence-oriented approach for the selection of the EPZ defining criteria is felt to be able to concur in an easy identification of a value of general consensus (e.g. the PAG suggested by the US Environmental Protection Agency (EPA)).

The identification of a limiting frequency, f*, on the other hand, could be more controversial. However, a value of $10^{-7}$ recurrently appears in various documents, and it can be used as the f* value for a first approximation of the methodology.

The aim of the direct application of this methodology is the evaluation of the frequency of exceeding a limiting dose, rather than the frequency of occurrence of some accidents. Such a frequency, as described below, is not imposed but can be evaluated by applying the methodology to currently operating nuclear power plants (reverse application) so as to discern what frequency was implied by the regulation. In the framework of the risk-informed nature of this methodology, the reverse application is described later to illustrate its use to back-solve the problem for the limiting frequency from the current EPZ for current NPPs.

The fifth and final step is the combination of the probabilistic and deterministic contributions previously mentioned to determine the size of the EPZ. The methodology is as follows: each of the curves of dose versus distance (evaluated for each $A_i$ release scenario) is solved for the limit dose D* (see Fig. 4).

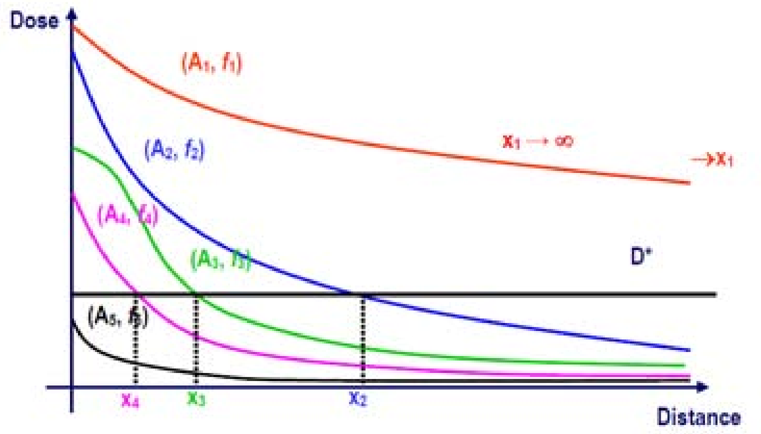



*FIG. 4. Step 5 of the methodology: Probabilistic/deterministic combination (part 1).*

That is, from the dose versus distance curves, the distance $x_i$ for which the calculated dose is equal to D* is easily identified. This is repeated for each scenario until a family of values of $x_i$ (with i=1 to n, where n is the number of considered scenarios) is generated. For better clarity, it is here assumed that the accidents are ranked according to decreasing consequences (i.e. in the example, $A_1$ is the event with the highest associated dose and $A_5$ is the event with the lowest associated dose). Curve $A_1$ can be considered as an example of a beyond design basis accident inducing a release of high doses at virtually all distances within the considered range. Curve $A_5$ can be considered as an example of accident with a low consequential release (for example, due to the design improvements incorporated in advanced and innovative reactors) and that, therefore, does not play any role in the definition of the EPZ distance.

By the definition of $x_i$ (distance at which the limit dose occurs), for each scenario $A_i$ there would be a probability 1.0 of exceeding D* at a distance smaller than $x_i$, and a probability 0.0 of exceeding D* at a distance larger than $x_i$. These probabilities should then be multiplied by the PRA calculated frequencies of the occurrence of each accident so that the frequency of exceeding the D* at a distance smaller than $x_i$ would have a value of $f_i$, for each scenario.

Note, that it can be expected that the larger $x_i$ will be associated with the more severe accidents, which should in turn have the lower frequencies (this is not a strict requirement of this application, but it represents a reasonable expectation). With this information collected, the $x_i$ are then ordered by decreasing values so that the frequencies of exceeding the dose limit as a function of distance can be calculated by simply considering, for each distance $x_i$, the contributions (i.e. the $f_i$) of all scenarios $A_i$ that at the selected distance induce a released dose higher than the limiting dose. The combination is therefore as follows:

$$\begin{aligned} &f_{D*}(x_1) = f_1 \\ &f_{D*}(x_2) = f_1 + f_2 \\ &\ldots \\ &f_{D*}(x_i) = \sum_{i=1}^{n} f_i \ \forall i / D(x_i) > D* \end{aligned} \tag{1}$$

where $f_{D*}(x)$ is the frequency of exceeding dose limit D* at the distance x.

Thus, a histogram of $f_{D*}$ versus distance can be completed. The last remaining input to the methodology, and a critical one, is the previously identified limit frequency (f*) of exceeding the dose limit (D*) that should be used to define the associated distance determining the EPZ requirements, see Fig. 5.

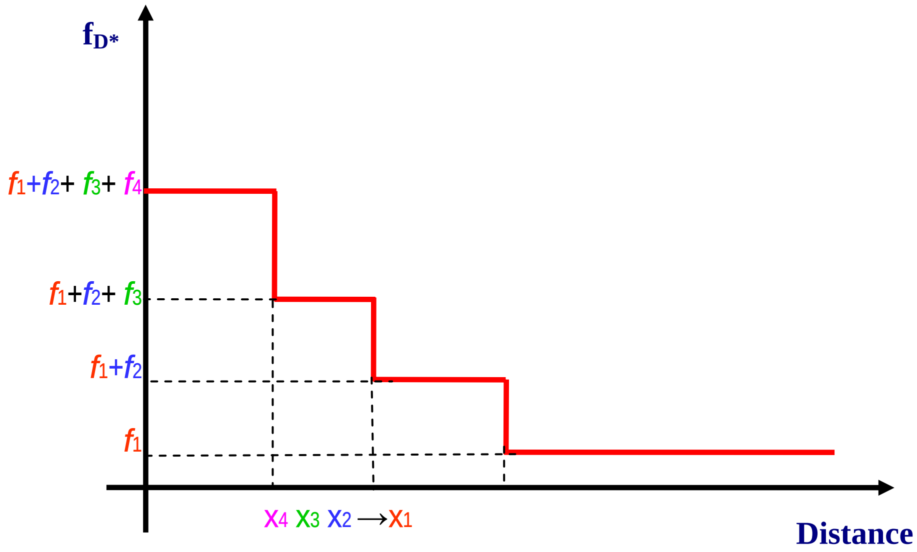



*FIG. 5. Step 5 of the methodology: Probabilistic/deterministic combination (part 2).*

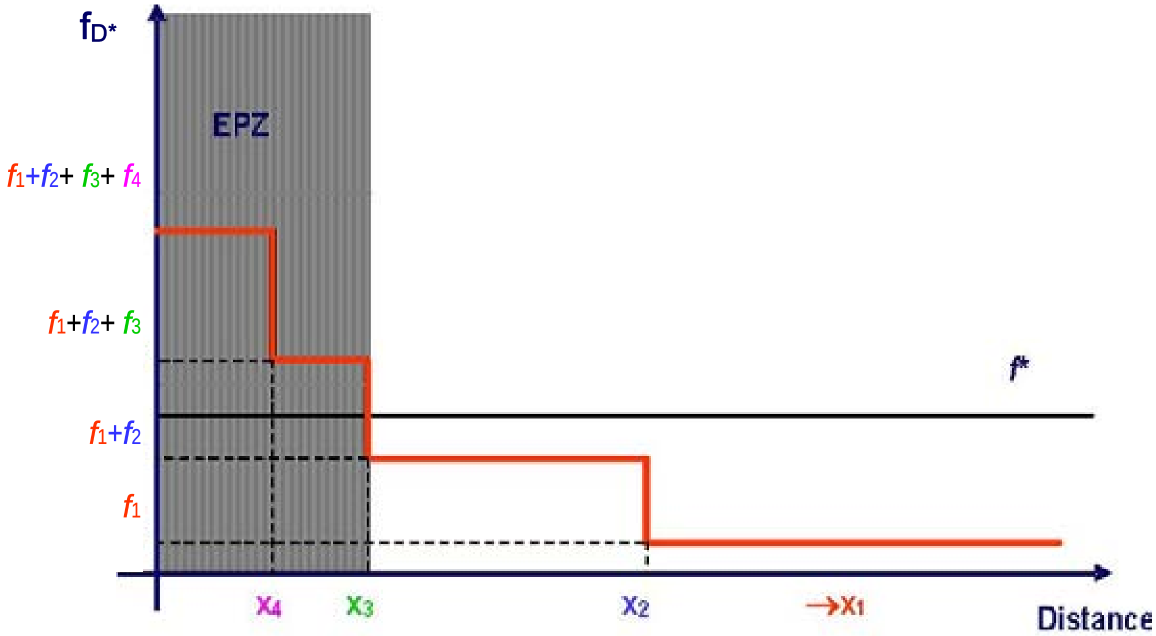



*FIG. 6. Step 5 - final result: Risk-informed EPZ definition.*

The EPZ distance will, in fact, be defined as the distance with a frequency equal to, or greater than the given limiting frequency (e.g. $x_3$ in Fig. 6, being ($f_1$+$f_2$+$f_3$) the lowest summation of frequencies which is greater than the given f*).

The methodology described above can provide a risk-informed definition of the EPZ, once the basic acceptance criteria in terms of limiting dose and limiting frequency have been provided (i.e. agreed upon with regulatory bodies); – but it remains to define the limiting frequency.

A reverse application of the methodology is proposed as a way to determine the level of risk associated with currently accepted EPZ size for the existing generation of nuclear power plants. Even though risk was not retained as the main defining basis for the EPZ size in the

US NUREG-0396 [5], a level of risk can actually be retrieved by estimating the frequency at which a pre-defined consequence is manifested at the distance from the plant which is currently adopted as the EPZ size. If estimated with this approach and on the basis of the rationale that was selected for the EPZ size, the level of risk associated with the currently accepted EPZ size will also factor in the additional margin associated with the unique emotional perception of the nuclear risk. Such a risk value could then be used as the reference baseline for the definition of an EPZ for a new NPP design. The methodology suggested herein, supported by a performance-based licensing approach, would in this way allow a new NPP design to maintain the implied acceptable risk, while reducing the EPZ size.

The reverse application of the methodology, as well as the methodology itself, are described in more detail in Annex I.

### 4.2. Case study on application of the methodology

A component of the study was to apply the methodology described above to test its efficacy in a SMR siting evaluation. This test case was conducted jointly by the Politecnico di Milano (Department of Energy, Nuclear Engineering Division) and the Westinghouse Electric Company (Science and Technology Department). Considered was the case of hypothetically siting an IRIS-like SMR at the Caorso NPP site in northern Italy (the original plant at this site is currently undergoing decommissioning). The IRIS reactor is a small PWR with integral design of the primary circuit described in detail in [3]. The extensive PRA and the deterministic results of analysis of design basis and beyond design basis accidents previously produced in support of the IRIS design activity were utilized to conduct the EPZ methodology application as indicated in Fig. 7. The application resulted in a required relationship among dose, frequency and the EPZ radius.

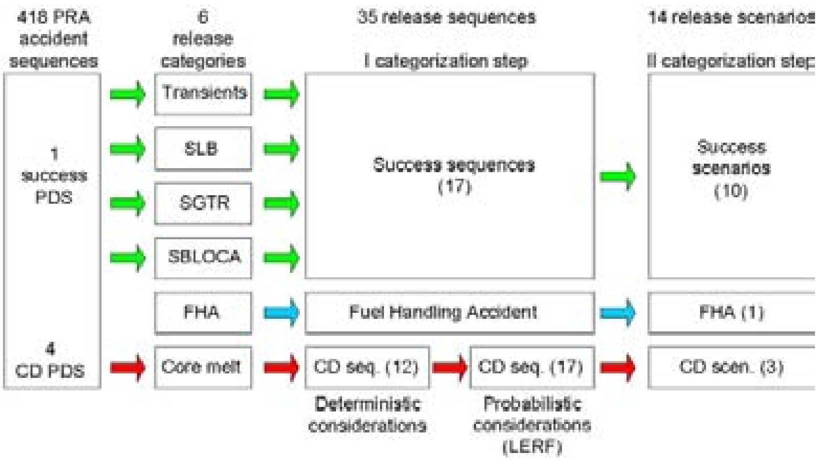



*FIG. 7. Schematic summary of accident sequence re-categorization (PRA – probabilistic risk assessment, PDS – plant damage state, CD – core damage, SLB – secondary line break, SGTR – steam generator tube rupture, SBLOCA – small break loss of coolant accident, FHA- fuel handling accident, seq. –sequence, scen. – scenario).*

The identification of a limiting dose and frequency are steps for which guidance from the regulatory body is obviously necessary. Given that this was a feasibility assessment and first application of the methodology, the limiting dose D* and the limiting frequency f* were postulated starting from the available, pertinent literature.

Consistent with considerations from the US literature (reviewed in Annex I), a limiting dose value D* of 1 rem (0.01 Sv) appears as a reasonable choice.

As far as the identification of the limiting frequency f* is concerned, which can be a more controversial matter, reliance is given on the EPRI literature study [6] that identifies $1 \cdot 10^{-7}$/reactor-year as a value of general consensus for a meaningful decision-making process.

It must also be observed that the selected values are not specific and unique to the USA reality; the IAEA indications [7] are of the same order of magnitude.

As described above, the definition of the EPZ size is done by investigating each dose versus distance curve, evaluated for the identified release scenarios from the PRA results, in order to establish a curve giving the overall frequency of overcoming the limiting dose.

Figure 8 shows the IRIS-like reactor dose versus distance curves with the limiting dose D* superimposed; this allows the identification of a set of crossing distances $x_i$, which are summarized in Table 1.

The information summarized in Table 1 is combined to obtain the diagram of Fig. 9, that identifies the overall frequency of overcoming the limiting dose D* as a function of the distance from the plant. When this curve is investigated with the limiting frequency f*, the IRIS EPZ can be identified as an area with a radius of 1800m.

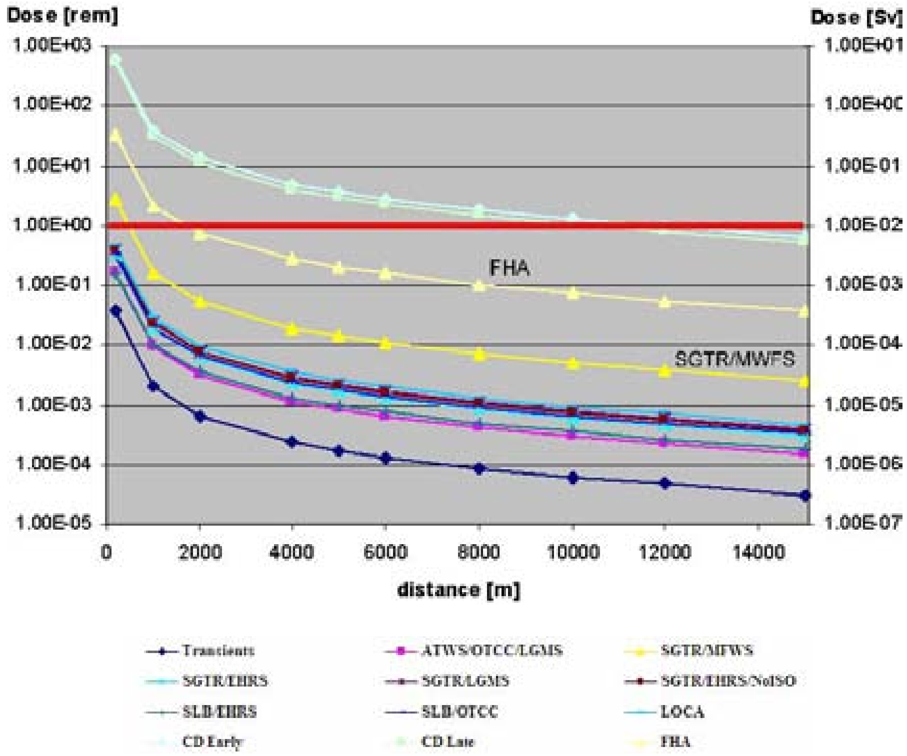



*FIG. 8. Dose versus distance evaluation for IRIS-like reactor release scenarios with superimposed D* (STGR – steam generator tube rupture, EHRS – emergency heat removal system, SLB – secondary line break, CD – core damage, ATWS – anticipated transient without scram, OTCC – once through core cooling, LGMS – long term gravity makeup system, MFWS – main feedwater system, NoISO – containment isolation failure, LOCA – loss of coolant accident, FHA – fuel handling accident).*

TABLE 1. CROSSING DISTANCE SET

| # | Release scenario description | Overall frequency [1/reactor-year] | D* crossing distance [m] |
|---|---|---|---|
| 1 | Transients successfully mitigated via MFWS | 1.14E+00 | 200 |
| 2 | ATWS successfully mitigated with OTCC | 8.00E-07 | 200 |
| 3 | SGTR successfully mitigated via MFWS | 1.77E-04 | 600 |
| 4 | SGTR successfully mitigated via EHRS | 1.10E-05 | 200 |
| 5 | SGTR successfully mitigated via OTCC | 1.68E-11 | 200 |
| 6 | Not isolated SGTR successfully mitigated via EHRS | 1.00E-08 | 200 |
| 7 | Not isolated SGTR successfully mitigated via OTCC1 | 2.41E-13 | N/A |
| 8 | Steam line break successfully mitigated via EHRS | 9.23E-04 | 200 |
| 9 | Steam line break successfully mitigated via OTCC | 2.91E-08 | 200 |
| 10 | Small break LOCA successfully mitigated | 1.02E-03 | 200 |
| 11 | Early core melt with heat removal capability | 1.97E-08 | 11800 |
| 12 | Late core melt with heat removal capability | 4.52E-10 | 10600 |
| 13 | Core melt with containment failure[2] | 6.85E-09 | ∞ |
| 14 | Fuel handling accidents | 1.00E-04 | 1800 |

Notes: 1. This scenario has not been evaluated and is merged with case 6.
2. This scenario has not been evaluated and an infinite distance is assumed.
3. Abbreviations are explained in the caption of Fig. 8.

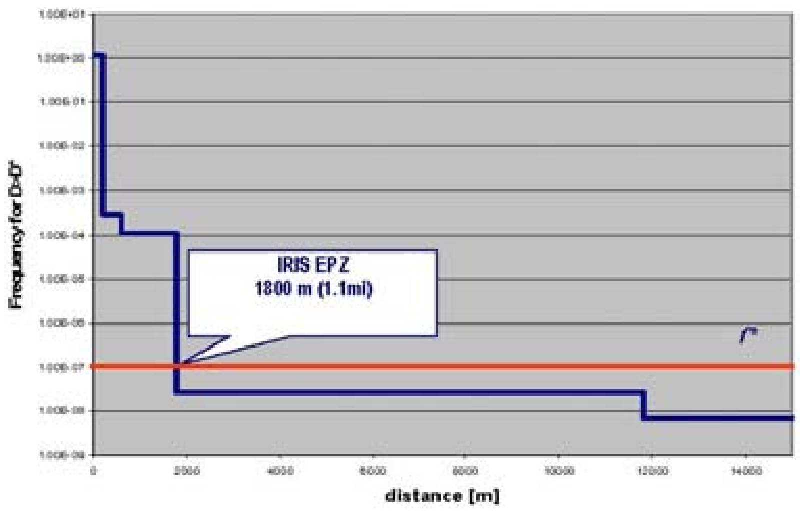



*FIG. 9. EPZ identification for IRIS-like reactor.*

Similarly, various limiting frequencies f* are considered; the interval between $1\cdot10^{-8}$/year and $1\cdot10^{-7}$/year was investigated in steps of $0.1\cdot10^{-7}$/reactor-year. The higher frequencies space has also been investigated, with a coarser approach, up to $1\cdot10^{-3}$/year. Figure 10 summarizes the results of the sensitivity study, providing a surface that identifies an EPZ size for each combination of limiting dose and limiting frequency.

The results herein presented for a possible EPZ definition of an IRIS-like reactor must be considered preliminary and will be updated as the design and methodology development progresses. These results must also be considered in the framework of the high degree of conservatism adopted for some of the most significant assumptions.

More details of the case study on methodology application are given in Annex I.

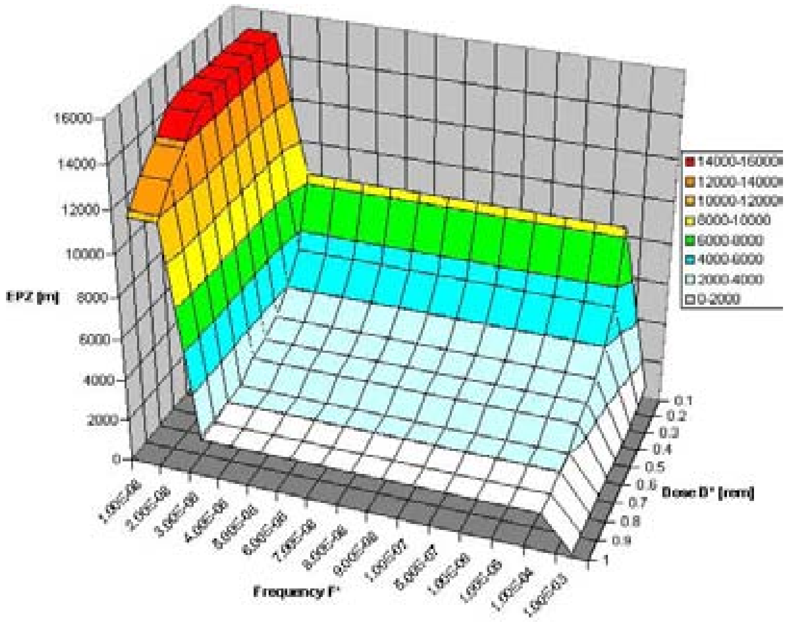



*FIG. 10. Results of sensitivity analysis on D* and f*.*

### 4.3. Case study for Lithuania – Implication for co-generation applications

The Ignalina Unit 2 RBMK-1500 was closed in the end of 2009 and Lithuania is considering both nuclear and fossil options for its replacement. Among the nuclear options is the 330 MW(e) IRIS reactor. It could be operated in either the electricity only or the co-generation mode. District heating is widely used in Lithuania, and the cities of Vilnius and Kaunas comprise the two largest consumers of district heat supply (see Fig. 11).

A case study was conducted to determine the best way to provide for the electricity and district heat needs of Lithuania out to year 2025 and to assess the tactical implications that a reduced-radius emergency planning zone might have on least cost planning with the IRIS-like reactor operating in the electricity only versus the electricity/district heat (co-generation) mode [8].

The length of any newly required hot water/steam pipelines into the cities of Vilnius and Kaunas will depend on the radius of the emergency planning zone emplaced around the IRIS-like reactor site; these pipelines represent a cost in construction dollars and a cost in lost heat which both increase with the pipeline length and, thereby, affect the viability of the co-generation mode of operation. The study was conducted parametrically for pipeline lengths of 0.5, 5, 15, and 30 km.

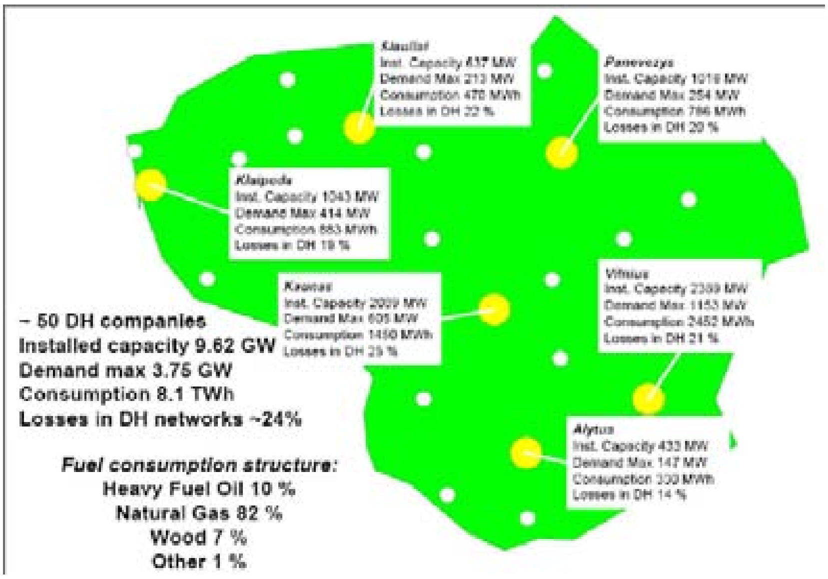



*FIG. 11. District heating sector in Lithuania (DH – district heating).*

The IAEA's energy planning tool, MESSAGE, was used to model several alternate scenarios for Lithuania for a time horizon of 2005 to 2025. MESSAGE is an optimization model which from the set of existing and possible new technologies selects the optimal, in terms of selection criterion, mix of technologies capable to cover given country demand for various energy forms during the whole study period. Table 2 lists the scenario options that were considered.

TABLE 2. DESCRIPTION OF SCENARIOS

| No. | Scenario name | Description |
|---|---|---|
| 1 | No IRIS-like NPP | Base scenario: construction of IRIS-like NPP is not allowed |
| 2 | Co-generation with IRIS-like reactor | Construction of IRIS-like NPP (with co-generation option) is allowed in Vilnius and Kaunas cities. No additional heat supply network must be constructed. (0.5 km pipeline) |
| 3 | 'IRIS EPZ' - IRIS-like NPP with larger emergency planning zone | Construction of IRIS-like NPP (with co-generation option) is allowed in Vilnius and Kaunas cities. The EPZ is parametrically assumed to be 5-30 km. Construction of IRIS-like units only for electricity production is also allowed in other locations. |
| 4 | IRIS-like NPP for electricity only | Construction of IRIS-like units used only for electricity generation is allowed (no co-generation option). |

Figure 12 shows the base case, where the Ignalina plant comes off line in 2009, and electricity and heat production is provided by the new fossil plants – some of which operate in electricity mode and some in co-generation mode. No nuclear plant is deployed in the base case. Figure 13 compares the total cost of this base case to the costs of the several options where IRIS-like NPP is deployed; clearly IRIS is a preferred option, no matter what configuration of its deployment.

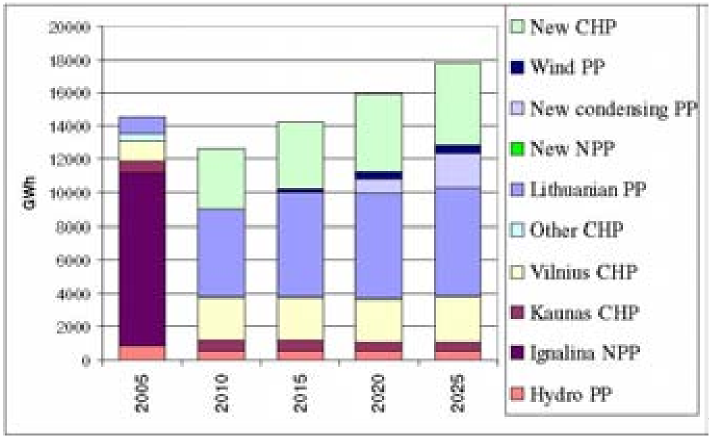



*FIG. 12. Dynamics of electricity production in the case of 'No IRIS-like NPP' scenario (CHP – combined heat and power plant, PP – power plant).*

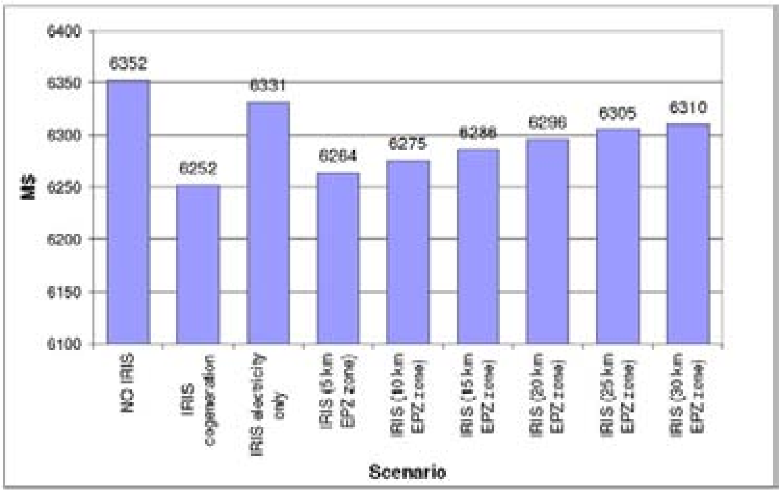



*FIG. 13. Discounted total cost of energy system operation and development in 2000-2025.*

At Lithuania's largest cities of Vilnius and Kaunas, legacy heat distribution pipelines already run through the neighbourhoods emanating from a massive heating plant sited at the outer edge of the city. Figures 14 and 15 show the evolution of electricity and district heat delivery by IRIS-like reactors operating in the cogeneration mode under the condition that the IRIS-like reactors can be sited at the city's edge within 0.5 km of the distribution header of the legacy district piping network. This option is labelled 'IRIS cogeneration' in Fig. 13 and is the overall lowest cost option. Figure 16 compares the 'base case' and the 'IRIS co-generation' scenario results in terms of primary energy shares of the Lithuania energy sector out to year 2025. The optimization shows that by 2025, three IRIS-like reactors would have been deployed and would be supplying 44% of Lithuania's electricity (Fig. 14) and 31% of Lithuania's district heat (Fig. 15) centred in the cities of Vilnius and Kaunas. The first two units are built in Vilnius and Kaunas before 2015. The third unit is built in Vilnius by 2020.

Alternately, if emergency planning zone requirements forced more remote siting of the IRIS-like reactors, such that new pipelines of 5, 15, or 30 km had to be emplaced in order to reach the headers for the district heat networks at Vilnius and Kaunas, then economic optimization reduces the IRIS-like reactor share of district heat delivery even though three reactors are still deployed.

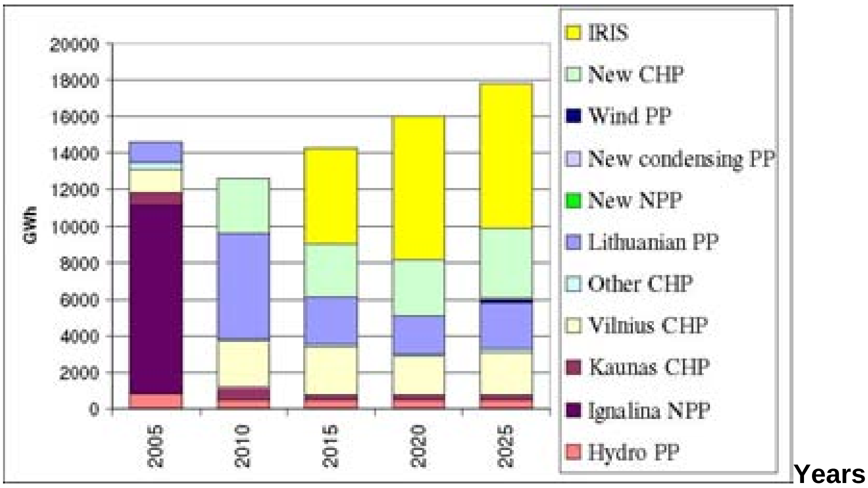



*FIG. 14. Dynamics of electricity production in the case of 'IRIS cogeneration' scenario.*

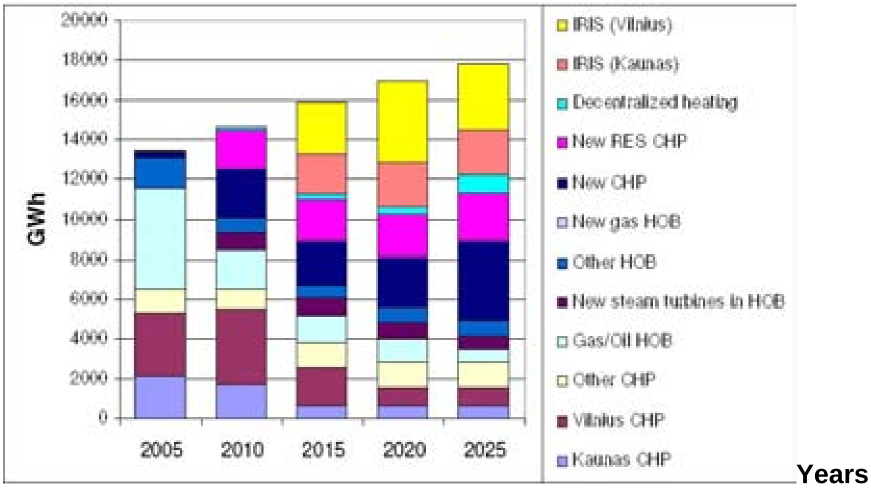
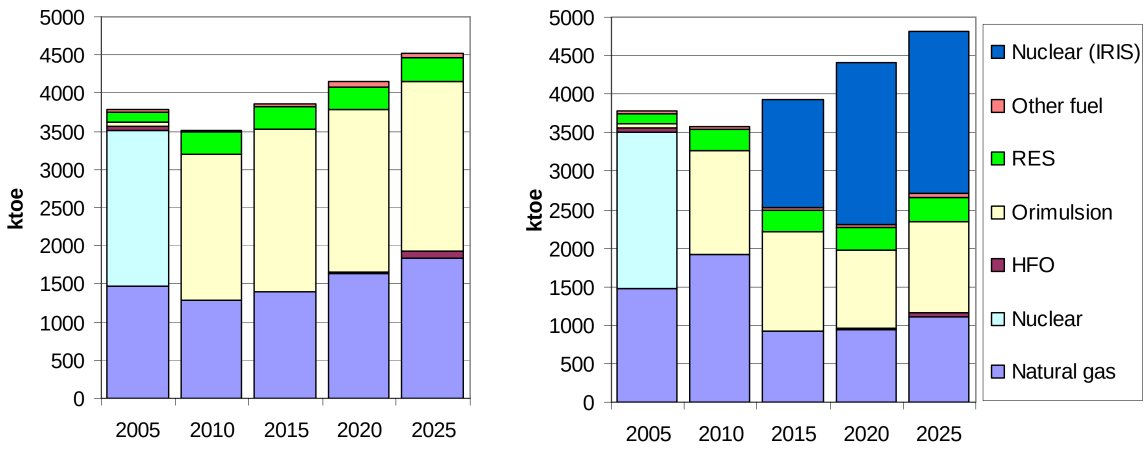



*FIG. 15. Heat production by technologies in Lithuania for 'IRIS cogeneration' scenario (HOB – heat only boilers, RES – renewable energy sources).*




a) 'No IRIS' scenario

b) 'IRIS cogeneration' scenario

*FIG. 16. Fuel consumption for electricity and heat generation (two scenarios).*

The results are summarized in Table 3 and Fig. 17. An increase in emergency planning zone radius makes only minor changes to the production of electricity at IRIS-like NPP units. The total electricity production at IRIS-like units is about 5.2-5.3 TW-hour in year 2015 and about 8 TW-hour in the period 2020-2025. Accordingly, it makes 37%, 49% and 45% of the total electricity generation in Lithuania in 2015, 2020 and 2025.

In the case that the IRIS-like reactors could be operated for electricity production only and district heat was provided by other means, then only two IRIS-like reactors would be built, see Fig. 18.

TABLE 3. THE MAIN RESULTS OF 'IRIS EPZ' SCENARIOS

| | **EPZ (length of heat pipeline), km** | | | |
|---|---|---|---|---|
| | **Co-generation scenario for IRIS-like NPP*** | **5** | **15** | **30** |
| Objective function | | 6263848 | 6286201 | 6310482 |
| Installed units, number (MW) | 3 (1005) | 3 (1005) | 3 (1005) | 3 (1005) |
| Total investment for heat supply pipes, M$ | 7.3 | 72.5 | 142.3 | 141.0 |
| Installed capacity of new heat pipes in Vilnius, MW | | | | |
| 2015 | 485 | 485 | 485 | 173 |
| 2020 | 970 | 858 | 485 | 179 |
| 2025 | 970 | 858 | 485 | 179 |
| Installed capacity of new heat pipes in Kaunas, MW | | | | |
| 2015 | 485 | 344 | 337 | 215 |
| 2020 | 485 | 485 | 393 | 256 |
| 2025 | 485 | 485 | 393 | 256 |
| Electricity generation | | | | |
| IRIS-like NPP with cogeneration, TW-hour | | | | |
| 2015 | 5.23 | 5.25 | 5.25 | 5.34 |
| 2020 | 7.87 | 7.88 | 5.23 | 5.3 |
| 2025 | 7.92 | 7.93 | 5.24 | 5.32 |
| IRIS-like NPP only for electricity, TW-hour | | | | |
| 2015 | 0 | 0 | 0 | 0 |
| 2020 | 0 | 0 | 2.73 | 2.73 |
| 2025 | 0 | 0 | 2.73 | 2.73 |
| Share from the total electricity generation in Lithuania,% | | | | |
| 2015 | 37 | 37 | 37 | 38 |
| 2020 | 49 | 49 | 50 | 50 |
| 2025 | 44 | 45 | 45 | 45 |
| Heat generation | | | | |
| IRIS-like NPP in Vilnius, TW-hour | | | | |
| 2015 | 2.61 | 2.61 | 2.61 | 1.29 |

| | **EPZ (length of heat pipeline), km** | | | |
|---|---|---|---|---|
| | **Co-generation scenario for IRIS-like NPP*** | **5** | **15** | **30** |
| 2020 | 4.1 | 4.1 | 2.74 | 1.57 |
| 2025 | 3.29 | 3.28 | 2.53 | 1.36 |
| Share from total heat generation in Vilnius,% | | | | |
| 2015 | 51 | 51 | 51 | 26 |
| 2020 | 74 | 74 | 50 | 29 |
| 2025 | 57 | 57 | 44 | 24 |
| IRIS-like NPP in Kaunas, TW-hour | | | | |
| 2015 | 2.05 | 1.64 | 1.64 | 1.16 |
| 2020 | 2.26 | 2.06 | 1.97 | 1.55 |
| 2025 | 2.25 | 2.06 | 1.89 | 1.47 |
| Share from the total heat generation in Kaunas,% | | | | |
| 2015 | 58 | 47 | 47 | 34 |
| 2020 | 59 | 55 | 52 | 42 |
| 2025 | 55 | 51 | 47 | 38 |

*0.5 km of the additional heat supply pipe was assumed

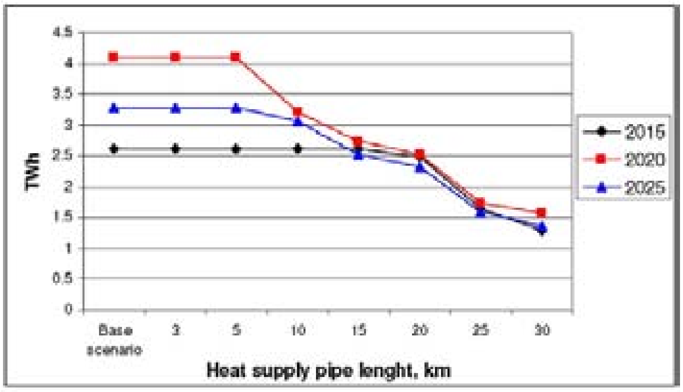



*FIG. 17. Heat generation at IRIS-like units in Vilnius.*

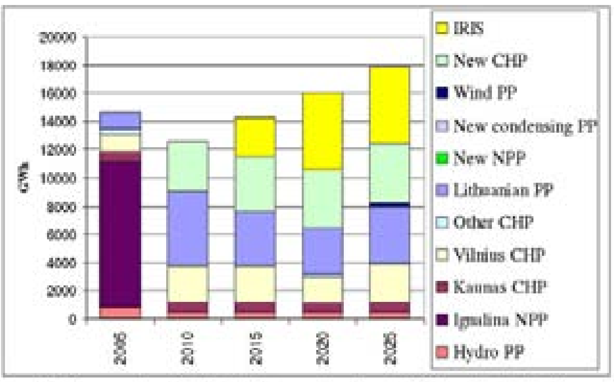



*FIG. 18. Dynamics of electricity production (‘IRIS electricity’ scenario).*

This case study, conducted for a real energy planning situation, illustrates several ways in which innovative small reactors can meet the needs of consumers. First is to add energy supply in small increments that are better matched to demand growth. Second is to broaden the slate of the energy services delivered – provided that institutional changes related to EPZ reduction can be made to facilitate that goal.

## 4.4. Conclusion to Section 4

### *4.4.1. Concept and application*

The quality of PRA techniques adopted as the main supporting tool for risk-informed applications has been continuously increasing over the past decades. A full power internal events PRA standard has been developed by the American Society of Mechanical Engineer (ASME) and endorsed by the US NRC as the basis for evaluating the quality of PRA models and assess their applicability for use in a broad spectrum of risk-informed applications (NRC Regulatory Guide 1.200). The second revision of the NRC Regulatory Guide 1.200 endorses a PRA standard that would merge the at-power internal events standard with an American Nuclear Society (ANS) standard on external events PRA and a fire PRA standard. Low power PRA and Level-2 PRA standards are currently being developed or planned. The maturity of the techniques involved in the risk-informed approach and concept suggests now the possibility of considering the extension of the range of potential risk-informed applications to the last level of the defence in depth philosophy, the off-site emergency preparedness.

In the present document the current bases for the definition of the emergency planning zone have been reviewed with the intent of re-introducing the concept of risk, previously ruled out due to technical considerations but especially due to public concern and distorted risk perception. A conceptual methodology is presented, which would allow relating the size of the emergency planning zone to the safety performance of a plant design, thus recognizing the enhancement in safety attained by new plant designs during the last thirty years, i.e. since when the basis for emergency planning have been defined.

The methodology, which allows for a bridge (i.e. applicable to a relatively early design phase) towards the use of a full scope Level-3 PRA as the reference supporting tool in the definition and sizing of the emergency planning around a nuclear power plant, builds on the fundamentals of the concept of risk, i.e. a potentially complete probabilistic approach to the entire spectrum of accident scenarios and the deterministic evaluation of consequences through dose and dispersion analysis. The simplified approach used for the test bed herein investigated (a NPP with the IRIS-like reactor) was geared towards a feasibility and conceptual test of the methodology rather than towards the details of the analysis implementation. Nevertheless, the very preliminary results show the potential for a significant reduction in the size of the EPZ for a small/medium sized nuclear power plant such as the IRIS.

Regarding further elaboration of the methodology, making a reference to IAEA publication [9], several participants of the CRP suggested that external events and reasonable combinations of the external and internal events need to be included in the Step 1 of the methodology (accident sequence re-categorization, illustrated by Fig. 2 and Fig. 7), as for smaller reactors with the enforced inherent and safety by design features it might be the impacts of external events that would dominate the risk of severe accidents with possible radioactivity release. Work in this direction has already been started and will be continued into the future, see [10].

### 4.4.2. *Qualitative impact of the EPZ redefinition*

Having underlined the potential for EPZ radius reduction at no increase in risk to the population (that is actually maintained equal to current PWR), a first test case was conducted as a semi-qualitative example in order to outline the potential benefit that the EPZ reduction could represent in terms of the burden that an utility has to carry for satisfying the current requirements. This example is applied to an Italian site for a general understanding of the potentiality of the methodology. Figure 19 identifies the EPZ size pertaining to an IRIS-like reactor hypothetically built on the site of the Caorso NPP (currently under decommissioning), in northern Italy. The outer circle identifies the EPZ size in accordance with the current US NRC requirements (i.e. 10 miles). The two inner circles indicate the IRIS-like reactor EPZ relative to the Caorso site as identified by the herein outlined methodology, which would be reduced to slightly less than 2 km (base case) or even 1 km (in case the effect of the fuel handling accident can be further reduced).

A detailed description of the practical aspects involved in the enforcement of the EPZ requirements is beyond the scope of the work performed within the CRP. Even without entering in the details, the beneficial impact on the economics of a hypothetical utility managing the Caorso IRIS-like NPP is easily understandable noticing the two relatively big population centers of Piacenza and Cremona (with up to 180 000 people in these two cities alone) being now excluded by the newly defined EPZ.

While the benefit for such a reduction for the utility and the nuclear industry is apparent, the main benefit for the final stakeholder (i.e. the public) is a reduced impact of the presence of the NPP from the economical and social points of view, due to the increase in safety and a corresponding reduction of the burden associated with outside emergency planning.

To illustrate this point, a second test case was conducted for the real energy planning situation in Lithuania. The impact that emergency planning zone radius would have on deployments of the IRIS-like reactors considered in an electricity/district heating co-generation mode was displayed parametrically in assumed EPZ radius.

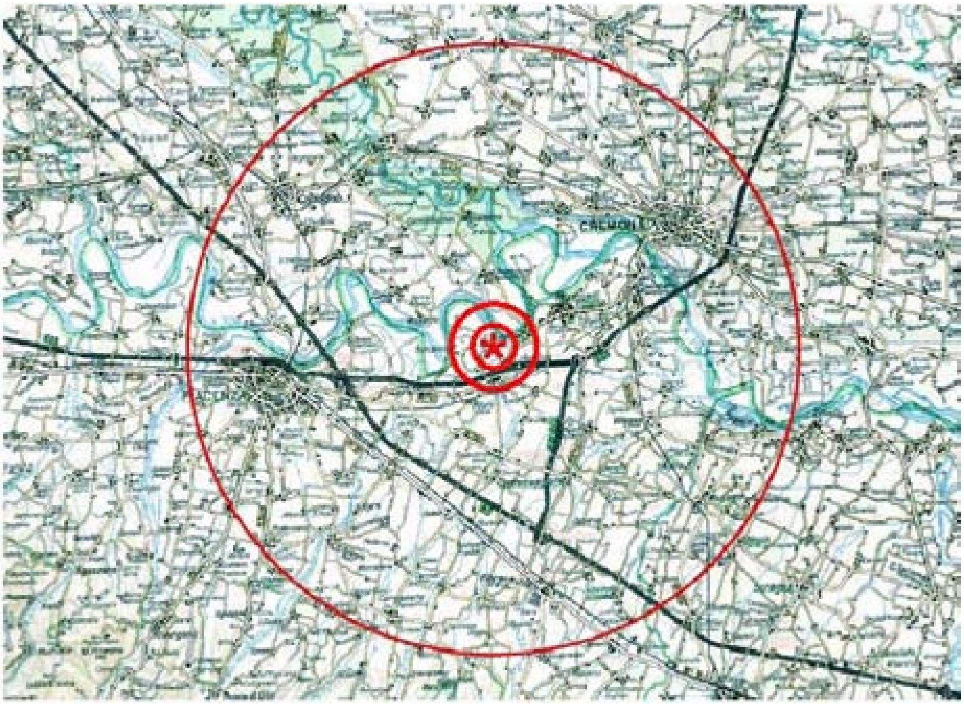


*FIG. 19. Hypothetical EPZ for an IRIS-like NPP located on the Caorso site: Potential reduction including or excluding fuel handling accident scenarios.*

# 5. WATER COOLED SMALL REACTORS WITH PARTICULATE FUEL

## 5.1. Introduction

One approach for increasing economic competitiveness of SRWORs could be to place increased reliance on passive safety features. This can facilitate simplification of active safety systems and perhaps even lead to an elimination of some safety grade equipment – favourably influencing cost competitiveness at no expense to safety.

Reduction of energy stored inside the system is one way to reduce hazard. Reducing stored heat energy; stored potential energy of chemical reactions; and stored mechanical (pressure) energy means that there is less to be dissipated should an off normal event occur, making it easier for passive (rather than active engineered) measures to handle the dissipation tasks.

In pursuit of this approach, Group 2 of the CRP conducted an extensive investigation of the implications of increasing the surface/volume ratio of the fuel form in small LWR reactors of long refuelling interval. The considered approach was to depart from traditional $UO_2$ fuel rods clad in zircaloy and to propose a contained particulate bed of fuel particles for fuel assembly design. By vastly increasing the surface area for heat transfer from the fuel to the coolant, numerous safety advantages can be attained:

- The core with coated particle fuel has a very large heat transfer surface and a relatively small amount of stored heat.
- The heat transfer surface of 3 mm diameter particle bed is 2000 $m^2/m^3$. For comparison the heat transfer of a BWR fuel assembly with 64 fuel rods of diameter ~10 mm is 150 $m^2/m^3$. Hence, for a fixed particle bed core, there are practically no limits related to the heat flux and fuel temperature.
- At the characteristic coated particle size (diameter 2.0 – 6.0 mm) the heat from coated particle fuel is transferred to the coolant with a delay time less than 0.1s. Hence, the reactor with coolant that directly cools coated fuel particles would provide a fast self-compensation or self-shutdown of practically any accident that happened not faster than in 0.1s, due to a negative temperature reactivity coefficient.

During any design basis transient (about 5 – 50s), the fuel and coolant temperatures remain nearly the same; the temperature difference for 2 mm particles is not bigger than 10 – 20°C. The reactor safety characteristics result from the strong negative temperature coefficient of reactivity and the short thermal lag time for the coolant temperature response to increases in the fuel temperature. These characteristics allow the reactor to shut itself down rapidly and passively – even without requiring the control rods to be scrammed during a loss of coolant accident, incurring no core damage.

- Robust safety characteristics are further enhanced during a postulated severe accident, including sabotage or any human actions of malevolent character, because of the capability of fuel particles to provide for high-temperature containment of their fission product inventory. The fuel and fission products will be kept contained within the fuel particles. It is also very attractive in that the core would be protected from reactivity accidents resulting from the introduction of positive reactivity increments.
- Finally, the multi-layer coating effectively confines fission products at temperatures up to 1600°C in the course of a long time and at 2100°C in the course of a few hours. At such temperature the removal of residual heat can be performed by means of natural convection, radiation and heat conductivity on a passive basis;

Several Member States had for several years been seeking to exploit these above-mentioned favourable features for small reactors without on-site refuelling, and Group 2 of the CRP was formed to share results. Over the course of the CRP, Group 2 continued further development on the several concepts of small and medium sized light water reactors without on-site refuelling, using particulate fuel elements of different types. The organizations from the USA, Japan, Russia, Morocco, Vietnam, and Brazil have participated in this project, and the reactor concepts included:

| | | | |
|---|---|---|---|
| VKR-MT | (BWR) | from | VNIIAM, Russian Federation (using particles) |
| AFPR | (BWR) | from | PNNL, USA (using particles and particle compact pebbles) |
| PFPWR | (PWR) | from | Hokkaido University, Japan (using particle compact rods) |
| FBNR | (PWR) | from | Federal University of Rio Grande do Sul, Brazil (using particle compact pebbles) |

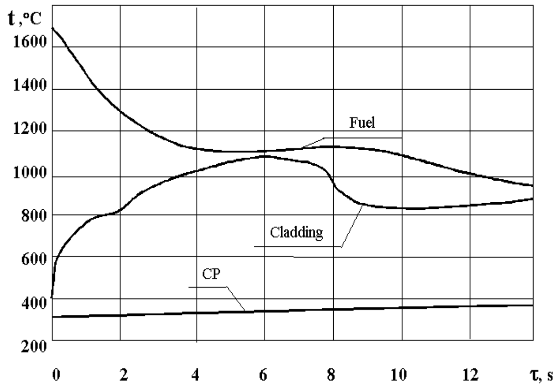
The CRP started after work on these four concepts was already in progress. The Boiling Water Reactor – Particle Bed concept (VKR-MT when abbreviated from Russian, Russian Federation) had the greatest amount of engineering development completed at the time (by RRC 'Kurchatov Institute', VNIIAM, and 'Luch'). It is a 300 MW(e) BWR in which TRISO particles are held in porous-walled cages and primary coolant flows crosswise through the particle bed. It drives a direct Rankine steam cycle. It also could provide 600 MW(th) of heat in combination with 180 MW(e) of electricity. The core operates for 10 years and whole core refuelling is performed without opening the reactor head using a hydraulic fuel particle transport system. Many tests and engineering and safety studies have been conducted to support the design. For example, Fig. 20 (borrowed from Annex X of reference [2]) compares fuel temperatures in the VKR-MT to those in WWER-1000 that are attained during a maximum diameter pipeline rupture accident. The approach of increasing fuel surface/volume ratio is seen to be remarkably effective. Annex II provides an update of the state of development of the VKR-MT subsequent to the work reported in reference [2] which extends refuelling interval to 26 years.

The Atoms for Peace Reactor (AFPR, from PNNL, USA) has certain similarities to the VKR-MT. It is a 100 MW(e) BWR using confined TRISO particle bed fuel assemblies with coolant cross flow. The AFPR incorporates fresh and used fuel storage tanks inside the reactor vessel and could operate for 40 years before recharging (and emptying) the fuel storage tanks.

The Particle Fuel PWR (PFPWR, from Hokkaido University, Japan) is a 50 MW(th) PWR for district heating. It's current version uses Th/Pu oxide TRISO particle compacts (in a graphite matrix) in standard Zircaloy clad fuel pin geometry and could operate for 10 years between refuelling.

The Fixed Bed Nuclear Reactor (FBNR, from Federal University of Rio Grande do Sul, Brazil) is a 40 MW(e) concept based on TRISO particle compacts (in graphite matrix) in a pebble bed configuration held suspended in place in a porous-walled core by upward-flowing primary coolant flow. Should pumping power be lost, the pebble bed would relocate downward under force of gravity to a subcritical, coolable configuration beneath the core cavity.

All four of these TRISO – fuelled water cooled reactor concepts are described in detail in reference [2] as they existed at the start of CRP activities. Only new work completed during the CRP will be discussed below.




CP - temperature of micro fuel elements in the VKR-MT core; two curves at the top show the temperatures of fuel and cladd fuel elements of a WWER-1000 reactor.

*FIG. 20. Accident with rupture of a maximum diameter pipeline.*

The present chapter describes the main results, the conclusions and recommendations from the four-year coordinated research effort. The work was motivated by the thought that if a small reactor can be designed to not release radioactivity under any conceivable conditions, then the nuclear power plant could be more easily sited. The concept of particulate fuel has been shown to be one of the candidates to achieve this objective for light water reactors, as an important feature of coated particle fuel is the proven good capability to confine fission products within a wide range of fuel temperatures. The proofs have been obtained in the development and operation of HTGR-type reactors with temperatures of fuel as high as 1600°C and above. Based on these known characteristics and on more recent developments, feasible concepts of small water cooled reactors with confined bed particulate fuel elements and/or particulate-based fuel rods and pebbles have been proposed and investigated.

Group 2 of the CRP organized itself to share the results of these conceptual design developments. Several inter-comparisons of the neutronics (benchmark) calculations were conducted. Most significantly, irradiation test results and fabrication technology were shared.

As the first candidate for coated particle fuel, TRISO fuel was investigated in the early phase of the CRP efforts. Out-of-pile corrosion testing of the TRISO particles with Si-C outer layer in hot water and in steam environments typical of reactor service conditions was producing mixed results. Testing at VNIIAM was showing excellent corrosion resistance but testing at PNNL suggested corrosion issues. This led the PNNL fuel developers to consider an alternative particulate fuel form – a cermet of $UO_2$ kernels in a Zr matrix coated with an outer Zr-1Nb layer impermeable to fission products.

Subsequently, during in-pile testing of the TRISO particles conducted by VNIIAM it was found that pyro-carbon and Si-C could experience integrity problems under low temperature irradiation, related to the accumulation of atomic displacements in the graphite lattice structure (Wigner energy), owing to insufficient annealing at temperatures below ~260°C. The open sharing of these testing results facilitated the designers to investigate whether or not their TRISO-fuelled designs could use the proposed cermet fuel form. The concepts of small light water reactors with micro fuel elements (MFE) were then re-designed using this new cermet type of particulate fuel.

Preliminary studies demonstrate suitability of such fuel for small water cooled reactors without on-site refuelling. Moreover, cermet fuel in some cases appears to be more adequate than the TRISO fuel to achieve the desired features of such reactors.

The LWR concepts with cermet fuel comprise a change in fuel form but rely on existing LWR technology – which is one advantage over other small reactor concepts currently being pursued. Even the innovative new spherical cermet fuel element relies on the use of common fuels materials, and fabrication techniques, although in a somewhat novel way.

### 5.2. Irradiation test results for TRISO fuel in LWR conditions

In support of the development of TRISO fuelled LWRs , irradiation tests were conducted (on particles intended for high temperature gas cooled reactors (HTGRs)) in the research reactor, IVV-2M located in Zarechny, Russian Federation. Although the test particles had been optimized for HTGR conditions (with a thick inside layer of dense

pyrolitic graphite) whereas optimization for LWR service calls for only a thin PyC layer, the availability of the HTGR particles facilitated an early testing campaign in an experimental mode. The original outer coating layer of pyrolitic graphite was removed from all samples, to meet the design conditions of a particulate-bed fuelled LWR (see Annex II), so that it was the SiC force-bearing coating layer that was in direct contact with water or steam coolant.

About 28 000 HTGR coated particles (11 lots of different background) were tested in a water loop at a pressure of 11 MPa and water coolant temperatures varying from 200 to 280°C. The particles were exposed to six irradiation cycles of 14 days each cycle. At the end of six cycles, the maximum burnup was 3.4% and the fast neutron fluence had reached 3.4 $10^{20}$ n/$cm^2$ (E >0.1 MEV), This was an accelerated burnup campaign – the particles were operated at ~3600 W/cm3 which is 10 times that in WWER fuel. No corrosion of the particles was observed and the cumulative fission gas release was <$10^{-5}$.

The intent was to continue irradiation in cycle 7 with periodic examination until a burnup of 5-10% had been obtained. Then the plan was to expose the irradiated particles to a temperature of more than 1600°C.

For cycle 7, the inlet water temperature to the test loop was raised from the previous value of 200°C up to 250°C. Part way through the $7^{th}$ cycle, the coolant outlet temperature from the loop began to fluctuate, with swings of up to 30°C, and extensive fission product release into the coolant was detected (see Fig. 21). The irradiation was terminated, and a comprehensive study was initiated to identify and understand the causes of the unexpected results.

It was found that a miscalculation of heat transfer had caused some of the particles to be irradiated at a temperature <200°C during cycles 1 through 6. During this low temperature irradiation, a significant level of Wigner energy had accumulated in the graphite comprising the particles. Wigner energy is the potential energy of graphite atomic lattice distortion that is caused by exposure to fast neutron fluence. It is safely annealed out continuously during high temperature (>200°C) irradiation, but, to the contrary, it accumulates during irradiation at <200°C. Table 4 shows the character of Wigner energy accumulation versus temperature. Saturation requires a fluence of more than $10^{21}$ n/cm.

Wigner energy accumulation can be safety annealed out using a slow temperature rise, but instantaneous release will be triggered by exposure to temperature of 20 – 30°C above irradiation temperature. That is what happened as cycle 7 was initiated with 50°C higher test loop coolant inlet temperature. Although the time constant for heat release from the particle to the water is 0.01 to 0.03 s, the time constant for heat transport between TRISO particle layers is ten times less. Therefore, the released heat did not go immediately to the coolant, but instead remained in the particle. It caused the thick PyC layer to thermally expand which then ruptured the outer SiC layer – thus releasing fission products into the coolant. Figure 22 shows the character of the failed TRISO particles.

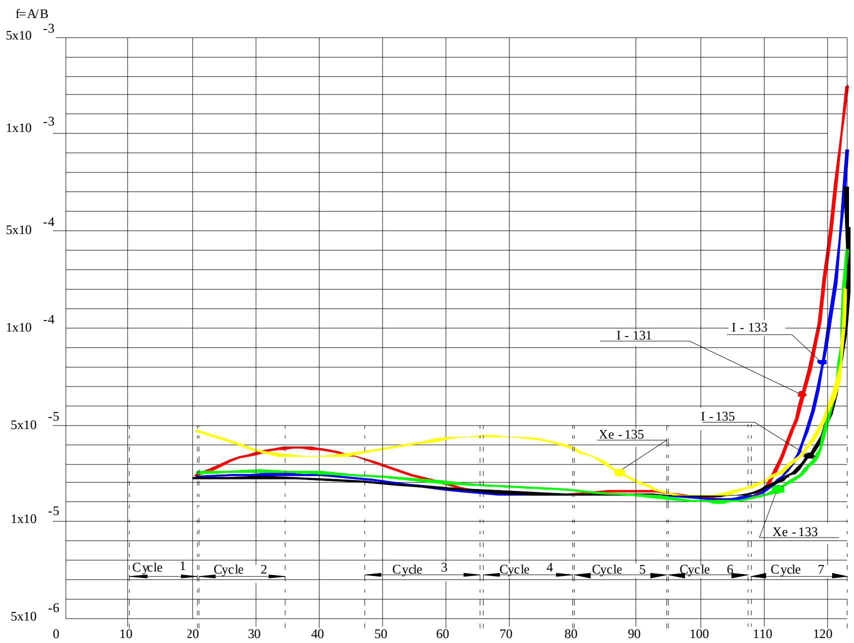



*FIG. 21. Average relative releases of fission products into water coolant.*

TABLE 4. DEPENDENCE OF WIGNER ENERGY ON TEMPERATURE FOR GRAPHITE AT THE $10^{20}$ n/cm$^2$ FLUENCE

| **Temperature, $^o$C** | **Wigner energy, Cal/g** | **Temperature jump, $^o$C** |
|---|---|---|
| 30 | 65 | 273 |
| 60 | 45 | 190 |
| 100 | 24 | 101 |
| 130 | 17 | 71 |
| 160 | 15 | 63 |
| 190 | 11.5 | 48 |
| 230 | 10 | 52 |
| 250 | 9 | 38 |
| 340 | 5 | 21 |

The Wigner energy phenomenon does not of itself preclude use of TRISO particles in LWRs. The British graphite-moderated MAGNOX reactors have operated safely for many decades simply by assuring that coolant inlet temperature always remains above 250°C during operation. Such would be the case for a TRISO-fuelled LWR whose traditional coolant operating conditions are 280°C inlet and 315°C outlet.

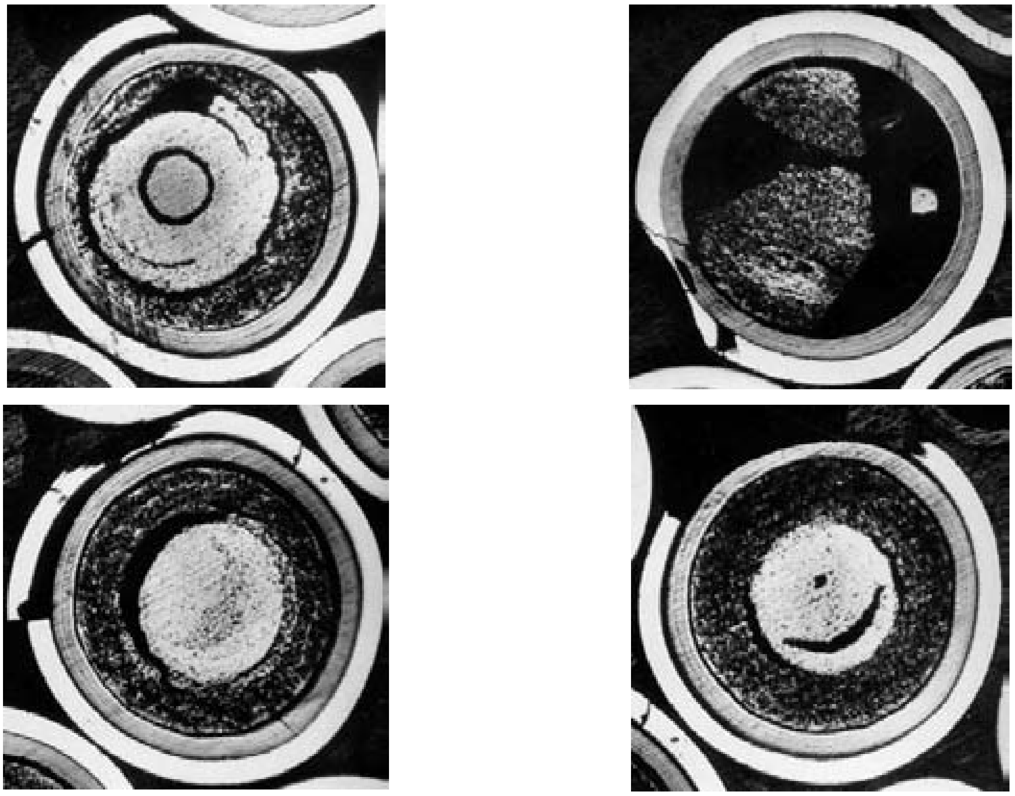


*FIG. 22. Structure of the coated particle lot 11, irradiated in a capsule No.4 (increase 50$^{o}$C).*

The researchers comprising Group2 of the CRP, all except for the designers of the VKR-MT, decided to re-engineer their earlier TRISO-based small reactor design from TRISO particle fuel to cermet particle fuel in order to provide themselves with future flexibility, because at this early stage of development, neither fuel form is fully proven for LWR use:

- The TRISO fuel has a more mature fabrication database and an earlier start on in-pile performance testing, but has experienced ambiguity in corrosion testing results from different laboratories
- The cermet fuel has a less mature fabrication database and currently no in-pile or out of core performance testing, but would not have a Wigner energy issue

The re-design efforts using cermet fuel and the original design using TRISO fuel have been shown to be acceptable in both cases; they, therefore, provide for future options as more technology emerges from development programmes.

### 5.3. Spherical cermet fuel concept

Figure 23 shows the proposed cermet particulate fuel form.

This new fuel has four barriers for fission product retention:

- Low temperature retention of fission products in $UO_2$ matrix
- Zr based coating layers of each tiny fuel particle
- Gas fission products (<1% of all fission products produced) will be trapped in Zr matrix
- Zr-1Nb alloy outer coating provides an additional fission product retention and corrosion protection.

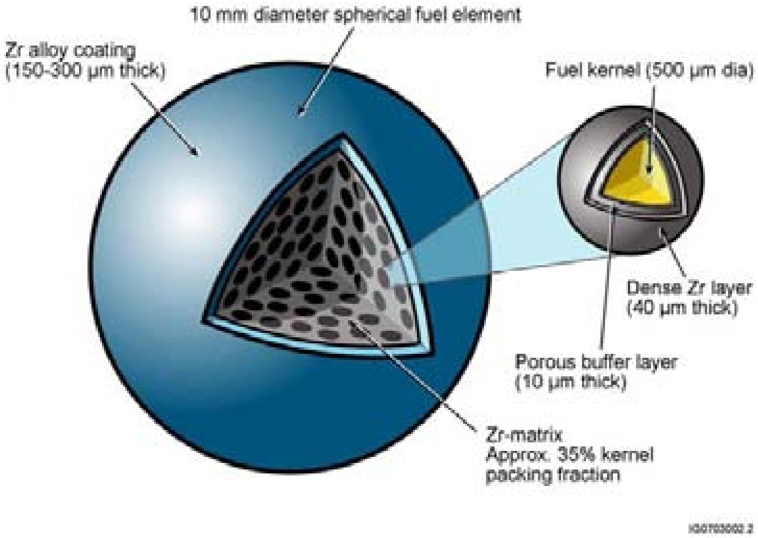



*FIG. 23. Spherical cermet fuel concept.*

Neutronic and thermal-hydraulic evaluations have demonstrated that not only are the cermet fuel elements feasible, but, in fact, may offer many advantages, such as:

- High thermal conductivity of fuel
- Rapid thermal response times (0.1 s versus 5 s for conventional fuel)
- Low stored non-nuclear energy
- Low fuel peak temperature (less than 400°C)
- Metallic matrix is robust barrier against fission-product release

These new spherical cermet fuel elements could be operated to 100 GW·d/MTU burnup over 20 years without re-fuelling in small proliferation resistant light water reactors, such as (see reference [2]) the AFPR, the FBNR, the PFPWR50, and others. Otherwise, these concepts are based on proven pressurized water reactor technology.

#### *5.3.1. Fabrication process*

The technology of fabrication of the new cermet fuel for PWRs was developed by a group of scientists (D. Senor, C. Painter, K. Geelhood, and others) at the Battelle Pacific Northwest National Laboratory (PNNL), USA [11, 12]. The fuel fabrication process consists of five main operations:

- Gel precipitation process to produce $UO_2$ fuel kernels
- Chemical Vapour Deposition (CVD) process, using Zr as coating on $UO_2$ fuel kernels
- Mixing of Zr powder and Zr-coated fuel kernels
- Hot press in a die to a desired diameter of sphere
- Spray-coating of pebbles with metallic Zr1%Nb and sintering of coated pebbles to produce leak tight metallic shell.

The fuel manufacturing for the Pebble Bed Modular Reactor (PBMR), currently under development in South Africa, uses a sol-gel process to produce kernels. Essentially the same process is used by nuclear fuel industries in Japan to manufacture fuel kernels for the high temperature test reactor [13]. The technology consists of two main steps. First, a uranyl nitrate solution is prepared by dissolving $UO_2$ powder in nitric acid. Next, the solution is conditioned by the addition of organic compounds. Gelled particles are then produced by reacting droplets of the solution with ammonia vapour and ammonium hydroxide solution. The gelled particles are aged, washed, dried, and then reduced at 400°C for one hour. Finally, the particles are sintered at 1700°C for two hours. Dimensional and density controls are achieved by varying process parameters, such as the sintering conditions.

The proposed approach for producing Zr-coated $UO_2$ fuel kernels utilizes a $ZrBr_4$ precursor in a fluidized bed CVD reactor. A fluidized bed of fuel kernels is inductively heated to the appropriate decomposition temperature and $ZrBr_4$ vapour would be passed through the fluidized bed by an inert carrier gas. Fluidizing the $UO_2$ kernels is an established process used during CVD coating of TRISO fuel particles, and should pose no significant technical problems.

However, establishing an effective induction couple with the fluidized particles will require some demonstration, as will controlling the process to ensure uniform kernel temperatures and, therefore, uniform Zr deposition. Process development will be performed using surrogate kernels of the appropriate size fabricated from an alternative ceramic material (e.g. $ZrO_2$ or yttrium-stabilized zirconia (YSZ) beads). The primary issues related to the production of Zr coatings using the proposed approach will be the high temperatures needed for reaction of the precursor to a fully metallic film, requiring a core particle having thermal stability to temperatures in excess of 1600°C (this should not be an issue for $UO_2$) and CVD reactor walls capable of handling such temperatures.

A variation of the gel precipitation process has recently been employed at Battelle, PNNL. D. Senor, C. Painter, K. Geelhood and others [11, 12] performed the feasibility demonstration of capability to fabricate fuel pebbles using the CVD method. As a surrogate kernel they used 300 μm diameter YSZ beads. Two different kernel coating processes were evaluated: the first $ZrI_4$ and $ZrBr_4$ CVD reaction in a fluidized bed, and the second - Zr sputtering in a fluidized bed. The sputtering coating in fluidized bed was proposed for the pebble bed coating process, and liquid phase was added for sintering. Pebbles for the PBMR and other similar pebble bed reactors are produced by hot pressing. The TRISO particles are mixed with an appropriate quantity of graphite powder, and the spheres are hot pressed in a die. After the fuel-bearing portion of the pebble is pressed, a 5 mm fuel-free layer of graphite is added by hot pressing the graphite powder around the pebble in a larger die.

Finally, the pressed sphere is heat treated at 1950°C for 9 hours to produce the desired properties in the pebble matrix and the fuel-free graphite coating on the exterior of the pebble. For Zr-matrix pebbles, Zr powder could be mixed with the Zr-coated fuel kernels and hot pressed in a die to the desired dimensions. The resulting pebble would likely have some porosity, and this would need to be characterized along with other micro-structural features for the prototypic dimensions and processing conditions. The most likely candidate stoichiometry for the $ZrH_x$ matrix is $x = 1.6$, which produces the face-centred cubic δ-phase operating temperatures (300-400°C). At these temperatures, the $ZrH_{1.6}$ should be thermodynamically stable.

The pebble will require an outer coating to serve as an additional fission product barrier and as a protective layer to protect the pebble and fuel kernels from the primary coolant. Although a Zr pebble matrix would be reasonably corrosion resistant in the PWR water conditions, the matrix will likely have some porosity. To prevent interaction between the primary coolant water and the fuel kernels, a protective coating will be required. If a $ZrH_x$ material is used as the pebble matrix, a protective outer coating will be needed to prevent rapid corrosion of the matrix material itself. The thickness of the outer protective coating is therefore a function of its expected corrosion rate and the projected lifetime of the pebble in the primary coolant. For maximum compatibility of the materials and coolant, a Zr1Nb alloy outer protective layer is the most desirable material. A thickness of 150-300 μm should provide up to 30 years of corrosion protection in 300°C water, based on literature data for Zr1Nb corrosion rates. A low temperature spray process combined with a liquid phase sintering step to fully densify the resulting pebble coating will provide an acceptable approach to production of the outer protective layer. Coatings will be applied to the pebbles in a fluidized or vibratory bed to ensure initial uniform coverage prior to the sintering step. This method offers many advantages for application of the outer coating, including the ability to apply thick and/or multiple layers of the coating material and the ability to produce intricate alloys.

### *5.3.2. Thermal properties of cermet spherical fuel*

The effective specific heat for the spherical cermet fuel element is found by combining the specific heat for the individual materials based on their mass fraction in the pebble. This is acceptable since none of the materials are expected to form new compounds, so the overall heat capacity of the kernel should be the sum of its parts. Likewise, the density of the spherical cermet fuel element is found by combining the densities for the individual materials based on their volume fraction in the pebble. Table 5 shows the effective thermal conductivity, heat capacity, and density for a pebble with a zirconium and zirconium hydride matrix for normal operating conditions, and for accident conditions. The effective thermal conductivity of the spherical cermet material (16 W/m-K at 300°C with Zr matrix) compares very favourably to the thermal conductivity of traditional $UO_2$ pellets used in commercial LWRs (approximately 2-3 W/m-K at LWR operating temperatures).

TABLE 5. EFFECTIVE THERMAL PROPERTIES FOR SPHERICAL CERMET FUEL ELEMENT AT NORMAL AND ACCIDENT CONDITIONS

| | | **Normal conditions, 300°C** | **Normal conditions, 300°C** | **Accident conditions, 600°C** | **Accident conditions, 600°C** |
|---|---|---|---|---|---|
| Property | | Zirconium Matrix | Zirconium Hydride Matrix | Zirconium Matrix | Zirconium Hydride Matrix |
| Thermal conductivity | W/m-K | 15.6 | 28.2 | 16.1 | 21.9 |
| Specific heat | J/kg-K | 296 | 431 | 320 | 520 |
| Density | $Kg/m^3$ | 7470 | 6940 | 7470 | 6940 |

Understanding the temperature transient response time associated with nuclear fuel as a result of the postulated reactivity insertion accidents, potential loss of coolant accidents, loss of flow conditions, or other transient conditions is important in designing fuel that is inherently safe. The time it takes for fuel to reach steady state temperature from a cold start if the neutron flux is suddenly raised, the time it takes to reach fuel melting temperatures after a complete loss of coolant accident, and the time it takes for the coolant to reach equilibrium temperatures when the fuel temperature is suddenly raised are equally important to understanding the behaviour of a reactor. Time constants associated with the thermal response time of a fuel element should be kept low—on the order of 0.1 to 1 s—to ensure an inherently safe fuel design.

The limiting factor in determining maximum pebble size is the thermal time constant, which should be below 1 s. The parametric study was performed for the normal operating coolant temperatures of 300°C, and a 500 μm diameter kernel, that is consistent with TRISO fuel manufacturing practice, a 0.3 kernel packing fraction that is consistent with past cermet fuel practice, a 300 μm outer Zr coating, and the upper limit of 20% enrichment. Table 6 shows the results of the calculations for the pebble time constant and also presents the temperature drop, ΔT, between pebble centre and coolant. The results indicate that a maximum pebble diameter of 10 mm for the Zr matrix and 15 mm for the $ZrH_{1.6}$ matrix would achieve time constants less than 1 s and peak fuel temperatures less than 350°C.

These values were taken as the reference pebble dimensions for more detailed assessments of the time constant under both normal and accident conditions. Table 6 shows these time constants and the more limiting time constant for the normal and accident conditions for a zirconium matrix and a zirconium hydride matrix. From this table it can be seen that for both materials at normal and accident conditions, the time constant is less than 1 s.

TABLE 6. TIME CONSTANTS AND TEMPERATURES FOR SPHERICAL CERMET FUEL ELEMENT AT NORMAL AND ACCIDENT CONDITIONS

<table>
<tr><th colspan="2"></th><th>Normal conditions, 300°C</th><th>Normal conditions, 300°C</th><th>Accident conditions, 600°C</th><th>Accident conditions, 600°C</th></tr>
<tr><td colspan="2">Property</td><td>Zirconium matrix</td><td>Zirconium hydride matrix</td><td>Zirconium matrix</td><td>Zirconium hydride matrix</td></tr>
<tr><td colspan="2">Time constant, s</td><td>0.52</td><td>0.39</td><td>0.55</td><td>0.61</td></tr>
<tr><td>ΔT (Coolant to centre)</td><td rowspan="2">°C</td><td>15.6</td><td>28.2</td><td>16.1</td><td>21.9</td></tr>
<tr><td>Centre temperature</td><td>329</td><td>320</td><td>628.5</td><td>623.3</td></tr>
</table>

The centre temperature was calculated for the spherical CERMET fuel elements using the effective thermal properties for the fuel element and assuming the heat production in the fuel is uniform. Table 6 shows the centre temperature values for both normal and accident conditions for a zirconium matrix and a zirconium hydride matrix.

The results indicate that the fuel will indeed have a fast thermal response time during temperature transients. This may translate into a reactor which will have more inherent safety features compared to existing LWRs which have thermal response times on the order of 20 s, normal centreline temperatures on the order of 1500°C, peak fuel temperatures on the order of 2100°C during anticipated operational occurrences, and can hypothetically reach fuel melting temperatures (2840°C) during a loss of coolant accident.

### 5.4. Long-life water cooled nuclear reactor concepts based on the cermet fuel form

For many developing countries with small electricity grids, and less developed infrastructure, the traditional economy of scale approach for LWR construction may be inappropriate [14]. As it has already been mentioned, to meet the needs of such customers, SMR designers are pursuing new design approaches such as reactor concepts without on-site refuelling. These reactors are designed for one time replacement of well-contained fuel cassettes that impedes clandestine diversion of nuclear fuel material. Small reactors without on-site refuelling are capable to incorporate increased refuelling interval (from 5 to 30 years and more). The reactors are either factory fabricated and fuelled by vendors or fuelled on the site by a dedicated service team. Such team is assumed to bring in and take away the fresh and spent fuel load and refuelling equipment.

Group 2 of the CRP has considered small light water reactors based on CERMET spherical fuel as re-designs of the former TRISO particle based designs reported in reference [2]. The pebble cermet fuel can simplify the LWR engineering safety systems due to the robustness of the CERMET spherical fuel to avoid loss of fuel integrity even in beyond design basis events. The water cooled concepts with CERMET particulate-based fuel that have been considered are listed in Table 7.

These four concepts of small reactors without on-site refuelling formerly based on TRISO spherical particles fuel were fully described in some detail in reference [2]. Brief descriptions are presented here for the re-engineered designs based on cermet particulate fuel.

TABLE 7. PEBBLE BED LIGHT WATER REACTORS CONCEPTS

| **Name (organization, country)** | **MW(e)** | **MW(th)** | **kW(th)/litre** | **Fuel type** | **Enrichment** | **Refuel Inter. years** | **Burnup MWd/t** |
|---|---|---|---|---|---|---|---|
| VKR-MT (VNIIAM-RRC KI), Russia | 300 | 890 | 140 | $UO_2$ | 10% | 9.6 | 48 000 |
| AFPR PNNL, USA | 100 | 300 | 13.25 | $UO_2$ | 14% | 20 | 80 000 |
| FBNR, Federal University of Rio Grande do Sul, Brazil | 70 | 218 | 45.6 | $UO_2$ | 5% | 2.2 | 26 000 |
| PFPWR50, Hokkaido University, Japan | | 50 | 87.2 | $UO_2$ | 5% | 10 | 50 000 |

#### *5.4.1. The Particle Fuel Pressurized Water Reactor (PFPWR50) concept*

The concept of PFPWR50 reactor, a 50 MW(th) PWR for district heating applications, is under development at the Hokkaido University’s Department of Nuclear Engineering (Japan). The design was originally based on $ThO_2$ – $PuO_2$ TRISO fuel particles embedded in a carbon matrix to form a cylindrical fuel compact clad in zircaloy in a pin lattice [15].

The re-design retained the original (TRISO) particle packing fraction, the Th/Pu ratio, the assembly geometry and overall core layout (see Figures 24 and 25 and Tables 8 through 17). A parameter study first selected Zr as preferable among the fuel kernel coating candidates (Zr,

$ZrO_2$ and $ZrH_2$) on the basis of burnup reactivity loss, although Zr and $ZrO_2$ produced nearly identical neutronics results. Next, a fuel compact matrix of Zr (rather than graphite) was chosen to altogether avoid any potential Wigner energy issues in the fuel pin. Then, seven distinct hexagonal fuel assembly layouts were chosen, each with 37 pins but each with differing allocations of pin positions to the fuel pins, gadolinia loaded fuel pins of differing gadolinium content, and guide tubes (see Table 17 and Fig. 25).

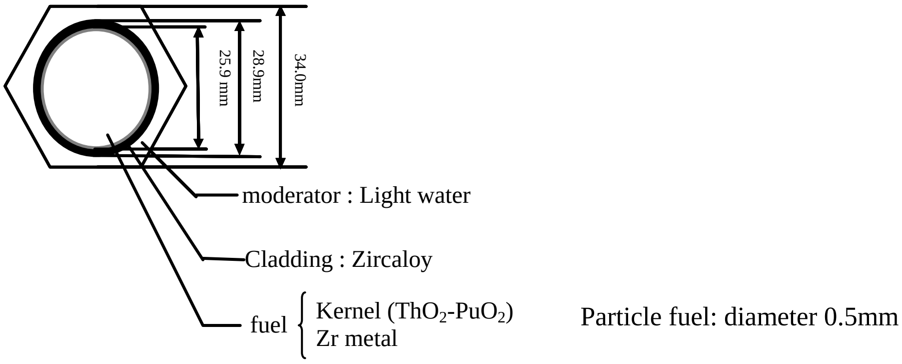



*FIG. 24. Plane view and concept of particle fuel.*

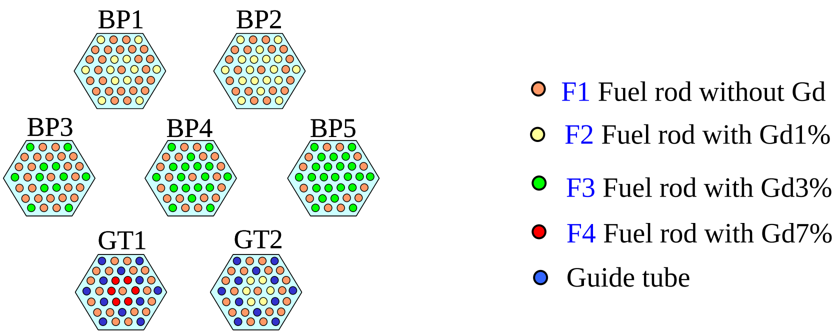



*FIG. 25. Configurations of fuel assemblies.*

TABLE 8. GEOMETRY

| Geometry | Hexagonal |
|---|---|
| Fuel rod pitch | 34 mm |
| Fuel diameter | 25.9 mm |
| Cladding thickness | 1.5 mm |

TABLE 9. OPERATION CONDITIONS

| | |
|---|---|
| Average fuel temperature | 280°C |
| Average cladding temperature | 265°C |
| Average moderator temperature | 250°C |
| Average linear heat rate | 9.1 kW/m |

TABLE 10. FUEL COMPONENTS

| Material | Density ($g/cm^3$) | Atomic mass number | Weight fraction (weight%) | Volume fraction (volume%) |
|---|---|---|---|---|
| $ThO_2$ | 10 | 264.037 | 90 | 20 |
| $PuO_2$ | 11.46 | 271.530 | 10 | |
| Zr | 6.511 | 91.224 | — | 80 |

Th isotope composition: Thorium-232 100%.

TABLE 11. PLUTONIUM ISOTOPIC VECTOR

| Isotope | Weight fraction (weight%) |
|---|---|
| Pu-238 | 2 |
| Pu-239 | 63 |
| Pu-240 | 19 |
| Pu-241 | 12 |
| Pu-242 | 4 |
| Total | 100 |

TABLE 12. GADOLINIA ISOTOPIC VECTOR

| Isotope | Weight fraction (weight%) |
|---|---|
| Gd-152 | 0.2 |
| Gd-154 | 2.1 |
| Gd-155 | 14.8 |
| Gd-156 | 20.5 |
| Gd-157 | 15.7 |
| Gd-158 | 24.8 |
| Gd-160 | 21.9 |
| Total | 100 |

TABLE 13. NUMBER DENSITY OF FUEL

| Material | Gadolinia concentration | | | |
|---|---|---|---|---|
| | 0% weight | 1% weight | 3% weight | 7% weight |
| | Number density ($\times 10^{24}/cm^3$) | | | |
| Th-232 | 4.1556E-03 | 4.0983E-03 | 3.9847E-03 | 3.7609E-03 |
| Pu-238 | 1.0245E-05 | 1.0217E-05 | 1.0162E-05 | 1.0054E-05 |
| Pu-239 | 3.2137E-04 | 3.2050E-04 | 3.1877E-04 | 3.1537E-04 |
| Pu-240 | 9.6517E-05 | 9.6256E-05 | 9.5737E-05 | 9.4717E-05 |
| Pu-241 | 6.0705E-05 | 6.0541E-05 | 6.0215E-05 | 5.9573E-05 |
| Pu-242 | 2.0151E-05 | 2.0097E-05 | 1.9989E-05 | 1.9776E-05 |
| O | 9.3293E-03 | 9.3125E-03 | 9.2793E-03 | 9.2140E-03 |
| Zr | 3.4391E-02 | 3.4391E-02 | 3.4391E-02 | 3.4391E-02 |
| Gd-152 | - | 1.3416E-07 | 4.0032E-07 | 9.2413E-07 |
| Gd-154 | - | 1.4087E-06 | 4.2034E-06 | 9.7034E-06 |
| Gd-155 | - | 9.9281E-06 | 2.9624E-05 | 6.8386E-05 |
| Gd-156 | - | 1.3752E-05 | 4.1033E-05 | 9.4724E-05 |
| Gd-157 | - | 1.0532E-05 | 3.1425E-05 | 7.2544E-05 |
| Gd-158 | - | 1.6636E-05 | 4.9640E-05 | 1.1459E-04 |
| Gd-160 | - | 1.4691E-05 | 4.3835E-05 | 1.0119E-04 |

TABLE 14. NUMBER DENSITY OF CLADDING

| Material | Number density ($\times 10^{24}/cm^3$) |
|---|---|
| Zr | 4.2672E-02 |
| Fe | 1.5450E-04 |
| Cr | 9.0126E-05 |

TABLE 15. WATER PARAMETERS

| | |
|---|---|
| Temperature | 250℃ |
| Pressure | 8.6MPa |

TABLE 16. NUMBER DENSITY OF WATER

| Material | Number density ($\times 10^{24}/cm^3$) |
|---|---|
| H | 5.3786E-02 |
| O | 2.6893E-02 |

TABLE 17. DESCRIPTION OF ASSEMBLIES

| Assembly name | Fuel rod (without gadolinia) | Gadolinia concentration | | | Guide tube | Assembly number in the core (total: 85) |
|---|---|---|---|---|---|---|
| | | 1wt% | 3wt% | 7wt% | | |
| BP1 | 25 | 12 | | | | 18 |
| BP2 | 19 | 18 | | | | 12 |
| BP3 | 25 | | 12 | | | 7 |
| BP4 | 19 | | 18 | | | 18 |
| BP5 | 12 | | 25 | | | 6 |
| GT1 | 19 | | | 6 | 12 | 12 |
| GT2 | 19 | 6 | | | 12 | 12 |

A core layout was found (see Fig. 26), having acceptable power peaking throughout life and achieving 7.1 equivalent full power years before refuelling. It attained 28 000 MW(th)·d/t core average discharge burnup. A very flat $k_{eff}$ versus burnup profile was attained with maximal excess reactivity confined to <8% Δk/k, see Fig. 27.

Suitably negative reactivity coefficients were attained and satisfactory stuck-rod shutdown margins and associated stuck-rod power peaking performance were confirmed. In summary, the re-design yielded reactor performance that is quite satisfactory [16].

In parallel to the design activities, a confirmatory analysis using alternative computer codes and alternative basic nuclear data libraries was performed at the Nuclear Physics Laboratory of the Mohammed V University in Rabat (Morocco). Whereas the Hokkaido University used JAEA's SRAC2006 computer code package and the JENDL-3.3 basic data library (Japan), the confirmatory analyses were conducted with the APOLLO and CRONOS codes based on the JEF library (France). The trends of the cermet-based design vis-à-vis the earlier TRISO design were confirmed, but the design predictions for the cermet-based design were not completely identical. Brief comparisons are summarized as follows:

1. Reactivity difference between TRISO fuels and cermet fuels from cell calculations: the Hokkaido University predicted that the reactivity decreased by 1000 pcm for the cermet fuel. On the other hand, the Mohammed V University predicted the decrease to be 6000 pcm. The discrepancy appears too large
2. Burnup characteristics were almost identical in the results of both participants.
3. The predicted burnup characteristics of the whole core were as follows. Core lifetime of the TRISO fuelled core without burnable poison and control rod guide tubes (GT) was predicted to be 8.8 effective full power years by the

Hokkaido University. The Mohammed V University predicted the lifetime of the core with the TRISO based fuel with burnable poison and GT as 8.6 effective full power years. In the calculations performed by the Hokkaido University, when burnable poison and GT are taken into account, about 9.4 effective full power years of life time is predicted. Thus, the total lifetime difference between the results of the Hokkaido University and the Mohammed V University could be evaluated as approximately 0.8 effective full power years for cores with TRISO based fuel. For the cermet fuelled core, the predictions were as follows. The Hokkaido University predicted the lifetime to be 6.7 effective full power years and the Mohammed V University - 7.7 effective full power years.

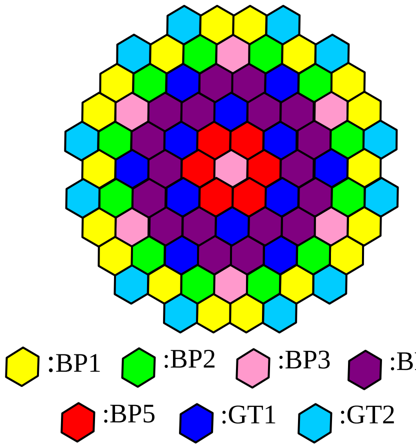



*FIG. 26. Loading pattern of the whole core.*

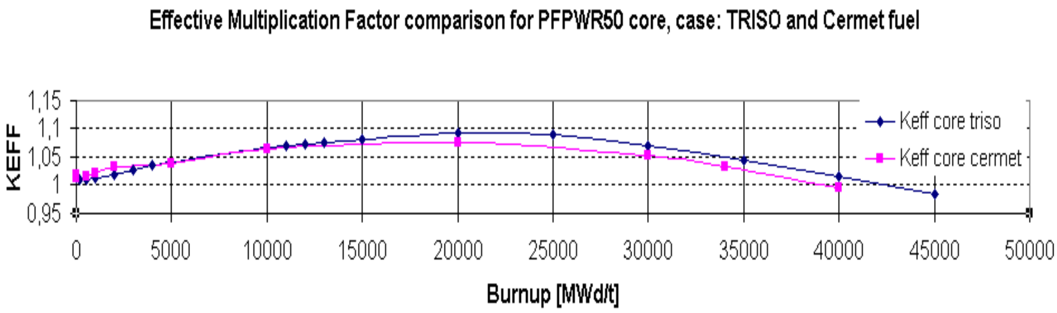



*FIG. 27. Burnup characteristics of the PFPWR50 core loaded respectively with TRISO and cermet fuel*

While some difference can be seen between the two sets of benchmark calculations, the fundamental feasibility of a long operation period without refuelling and low excess reactivity has been verified by both parties. The reasons for the discrepancies are being investigated.

The re-design of the PFPWR50 to employ a $THO_2$ – $PUO_2$ cermet fuel form in place of a TRISO fuel form has shown that equivalent neutronics performance can be retained, and the neutronics predictions have been broadly confirmed by the independent analyses. Future work on PFPWR safety and economics will continue based on the cermet fuel form.

### 5.4.2. *The Atoms for Peace Reactor (AFPR-100) concept*

In 2005 Battelle Pacific Northwest Laboratory applied for an international patent on a proliferation resistant nuclear reactor.[5] Later this reactor was named as Atoms For Peace Reactor (AFPR-100) [17, 18].

The conceptual design for the AFPR at a power of 100 MW(e) is depicted in Fig. 28. The thermal power of the reactor is 300 MW. The reactor shares similar features with a traditional PWR; control rods are inserted from the top. The core height is 3.0 m. The core diameter is 3.1 m and the inner diameter of the vessel is in the range of 5.0 m. The reactor core contains a confined bed of micro-fuel elements, immobilized in four concentric cylindrical zones of the core, as shown in the figure. The feed water enters the reactor vessel through the annular nozzle and flows around the vessel then downward between the core barrel and vessel. The coolant then flows upward to the core and traverses the pebble bed of the micro-fuel elements in a cross-flow direction as explained below.

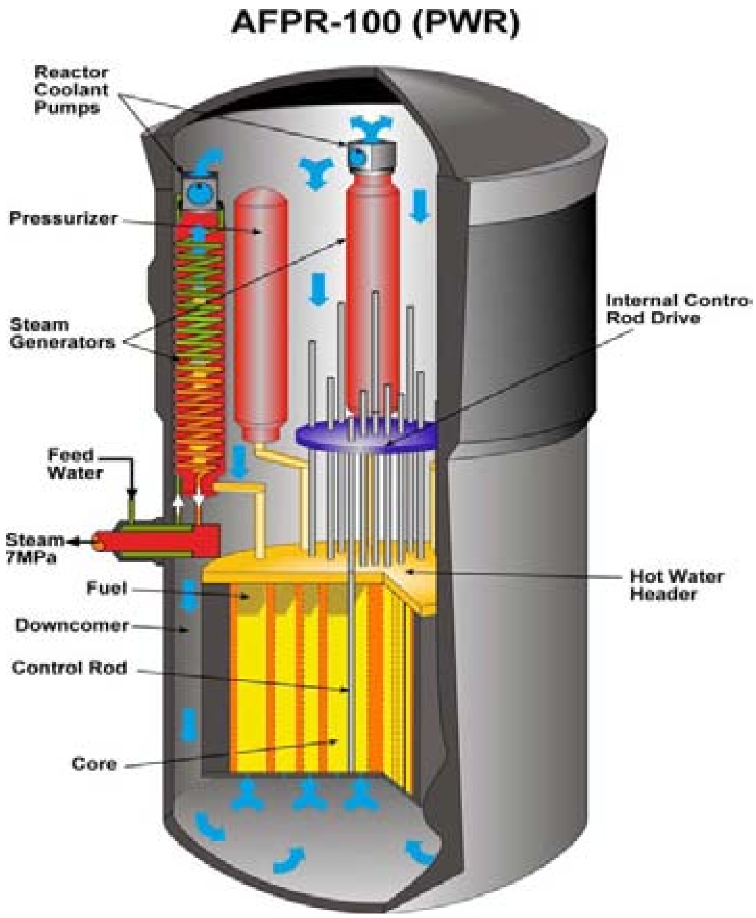



*FIG. 28. Vertical view of the AFPR-100.*

[5] The inventors are Georgi Tsiklauri, Robert Talbert, Alan Waltar, Thomas Shea and Darrell Newman [17, 18] The invention was made with the Government support under contract DE-AC0576RLO1830, awarded by the US Department of Energy.

The control rods and their drives are located on the top inside of the reactor vessel and are normally partially inserted into the core during full power operation. Perforated hot vents in the annular channels constrain the micro-fuel element beds. The coolant moves in the micro fuel element beds in a radial cross-flow direction. An upward water flow comes from the bottom water plenum into three annular perforated water channels. The coolant comes through perforations in the wall and penetrates into the micro-fuel element bed in a cross flow direction. The packed bed of micro-fuel particles with a porosity of about 35% is located between the water channels and the steam channels, as shown in the figure. A spring-loaded upper plate restrains the micro-fuel element beds, in the annular fuel channels. The core structure can be made of ferritic/martensitic stainless steels.

The technical and design bases for the AFPR core are as follows:

- Multiple coatings of the cermet based pebble fuel effectively confine fission products at normal operation and accident conditions in the course of a long time. At such temperature the removal of residual heat can be performed by natural convection, radiation and heat conductivity on a passive basis.

- Given the characteristic pebble size (diameter 10 mm), the heat from the Zr-coated particle fuel is transferred to the coolant with a delay less of less than 1 s. Hence, the core of a reactor with pressurized water coolant that directly cools the cermet based pebble fuel elements would provide a very rapid self-compensation of practically any positive reactivity if it is introduced not faster than in 0.5- 1 s.

- The core with cermet based pebble fuel has low stored heat, as the temperature of fuel is only 20-30°C higher that a coolant temperature due to the large heat exchange surface and small thermal resistance of coated particles. Hence, for such a core there are practically no limits related to critical heat flux or the departure from nucleate boiling (DNB).

- The low accumulated heat allows elimination of the high pressure stage in the emergency core cooling system.

The primary objective of the neutronics analysis was to determine the feasibility of the new spherical cermet fuel element as it relates to AFPR reactor physics performance parameters. New analyses were needed because the current concept of zirconium metal matrix fuel pebbles with sub-cooled water coolant is significantly different from the previous AFPR physics studies [2] with the SiC coated micro-fuel elements and two–phase coolant flow.

The assessment of reactor physics parameters is necessarily an iterative process as the physics parameters, thermal-hydraulics parameters, and fuel element materials properties are all interrelated. The approach taken for this study was to first perform some initial parametric studies, followed by more detailed analysis of reference core configurations, and then an evaluation of the impact of a few key variations of core parameters on the reference cases. The parametric studies fed into the preliminary choice of fuel properties for the reference cases, such as fuel pebble matrix material, fuel enrichment, fuel kernel packing fraction, and fuel pebble size. Preliminary choices of basic core parameters such as core size, geometry, configuration, power level, etc., were also made. The results of the parametric studies were used to define two reference cases with different fuel pebble matrix materials, zirconium and zirconium hydride. These reference cases were then evaluated in terms of the reactor physics performance of core lifetime, burnup reactivity, power distributions, and spent fuel isotopic composition. Several reactivity coefficients related to safety performance were also evaluated. Finally, the effects of selected variations in core parameters on the core physics parameters were evaluated, including the use of burnable poisons, adding radial reflectors, increasing core size, adding enrichment zoning, and adding moderator rods.

Figure 29 shows the burnup characteristics of particle based fuels using Zr as covering layer. The multiplication factors for Zr covering layer are lower than those of TRISO fuel during burnup because of the decrease of the amount of pyro-carbon moderator. Figure 29 shows the change in $k_{eff}$ due to the reactivity loss related to burnup for each case. The fresh fuel enrichment of 12% provided $k_{eff}$ of 1.4 at the beginning of life, which was a sufficient initial reactivity to maintain the criticality for 20 years of operation. The burnable absorbers Ga (1.5% by weight) and Eu (0.5% by weight) were used to flatten reactivity during lifetime. The average burnup for the core for a 20 year irradiation period would be ~50 GW day/t U. The burnup increases almost linearly after the first year and at 20 years the burnup for the various regions range from 40 to 1000 GW day/t U. In comparison, the US NRC has currently licensed commercial nuclear fuel to a 62 GW day/t U rod average burnup level.

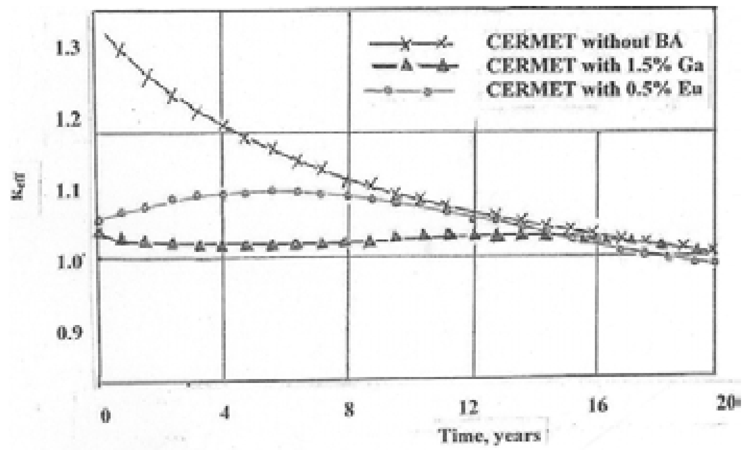



*FIG. 29. Burnup characteristics for the AFPR-100 whole core.*

The design goal was to achieve average fuel power densities well below existing commercial fuel power densities. The achieved average fuel power density for AFPR-100 is ~142 W/cm$^3$ compared to ~290 W/cm$^3$ for a BWR and ~320 W/cm$^3$ for a PWR.

Initially it was found that at the beginning of life (BOL) there is a factor of almost ten between power densities in the core centre and those in the core periphery, while at the end of life (EOL) this range is reduced to about a factor of two. Hence, it would be necessary to flatten the radial power distribution, which could be accomplished through enrichment zoning or burnable poisons. The conclusions from the performed parametric studies were that:

- At BOL $k_{eff}$ is not very sensitive to kernel packing fraction, kernel size, or pebble size
- Average fuel enrichment in the core was changed from 12 to 14%
- Europium burnable absorber (0.5% by weight of $Eu_2O_3$ in $UO_2$ kernels) was found preferable to control excess reactivity at BOL.

Table 15 summarizes the AFPR-100 core parameters.

TABLE 15. AFPR-100 CORE PARAMETERS

| General parameters of the plant | | Coolant parameters | |
|---|---|---|---|
| Electric power, MW(e) | 100 | Type of coolant | Water |
| Thermal power, MW(th) | 300 | Coolant flow direction | Cross-flow |
| Type of reactor | PWR | Feedwater pressure, MPa | 17 |
| | | Average temperature, ˚C | 266 |
| **Reactor core parameters** | | | |
| Core configuration | Annular core | Mass of $UO_2$ in fresh fuel storage, t | 40 |
| Core inner diameter, m | 3.1 | Average fuel enrichment by $^{235}U$,% | 14 |
| Core height, m | 3.0 | Spent fuel burnup, GW day/t U (for steady-state core) | 100 |
| Core volume, $m^3$ | 25.6 | Average core power density, $MW/m^3$ | 14.2 |
| Fuel bearing core volume, $m^3$ | 12.8 | Years of continuous operation without on-site refuelling | 20 |
| Pebble bed porosity | 0.35 | Number of fuel zones | 4 |
| Mass of micro fuel elements in the core, t | 48 | Number of water inlet headers | 3, see Fig. 28 |
| Mass of $UO_2$ in the core, t | 33 | | |

### *5.4.3. The Fixed Bed Nuclear Reactor (FBNR) concept*

The fixed bed nuclear reactor (FBNR) is a small reactor of 70 MW(e) without the need of on-site refuelling. It utilizes PWR technology and the pebble bed cermet fuel concept. It has the characteristics of simplicity of design, incorporates inherent safety features, passive decay heat removal, and measures to enhance proliferation resistance and secure a reduced environmental impact.

### *5.4.4. Pneumatically suspended core*

Cermet fuel pebbles are being considered for the FBNR reactor, configured in a pneumatically suspended core by force of the primary coolant flow. The fuel consists of coated $UO_2$ kernels embedded in a zirconium matrix which is then coated with a protective outer zirconium layer. The 15 mm diameter spherical fuel particles are held in the reactor core by the upward flow of coolant water creating a suspended particle bed core in the reactor. In the operating condition, the fuel elements are pressed together with a pressure of about 0.2 bar and the force on them is more than 27 times the force of gravity; this guarantees the bed to remain as a fixed bed during the reactor operation so long as forced coolant flow exists. Upon loss of pumping power, the fuel relocates under force of gravity into a criticality safe, coolable configuration.

The reactor is schematically shown in Fig. 30; the reactor core and a steam generator occupy its upper part, and a transportable fuel chamber occupies its lower part. The other components of the reactor are essentially the same as in a conventional pressurized water reactor. Table 16 summarizes the FBNR parameters.

*Refuelling*

The FBNR criticality-safe, transportable fuel chamber is fuelled with fresh cermet fuel particles at the factory. The sealed fuel chamber is then transported to the site to fuel the FBNR. The FBNR may have reasonably long refuelling cycle time and there is no need for onsite refuelling. At the end of life, the fuel chamber is transported back to the factory.

The fuel chamber is a 60 cm diameter tube made of high neutron absorbing alloy, which is directly connected underneath the core tube, see Fig. 31. The fuel chamber consists of a helical 40 cm diameter tube flanged to the reserve fuel chamber that is assumed to be sealable by the national and international authorities. A grid is provided at the lower part of the tube to hold the fuel elements within it.

*Primary heat transport system*

The core consists of an annular, ~200 cm high particle bed held between two concentric perforated zircaloy tubes of 31cm and 171cm in diameters. During the reactor operation, the spherical fuel elements are held together by the coolant flow in a fixed bed configuration, forming a suspended core. The coolant flows vertically up into the inner perforated tube and then, passing horizontally through the fuel particle bed and the outer perforated tube, it enters the outer shell where it flows vertically up to the steam generator.

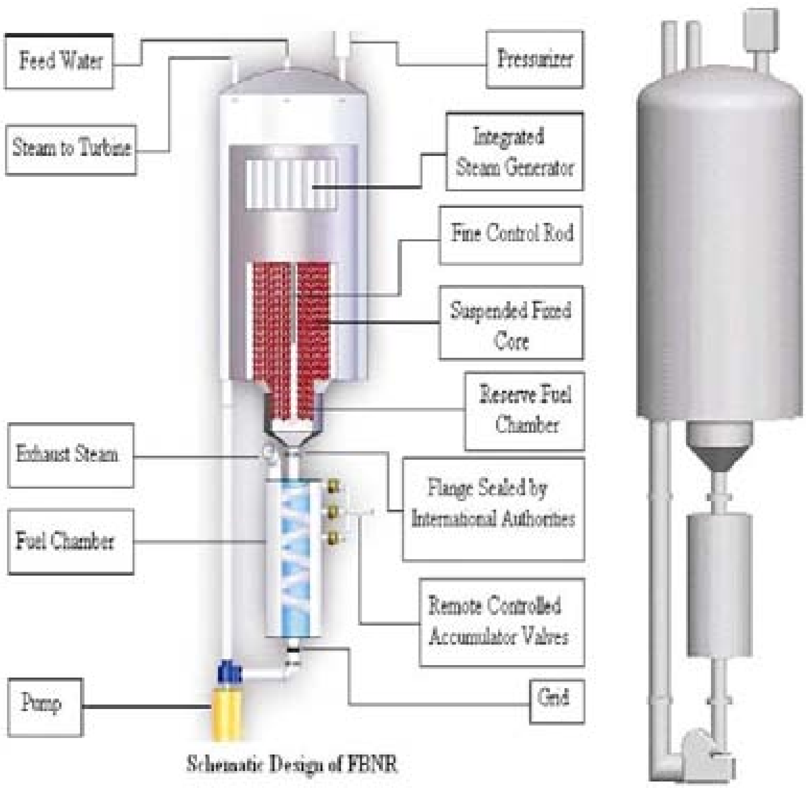



*FIG. 30. Schematic view of the FBNR.*

TABLE 16. TECHNICAL DATA FOR THE FIXED BED NUCLEAR REACTOR (FBNR) WITH PEBBLE CERMET FUEL

| Parameter | Value | Parameter | Value |
|---|---|---|---|
| *Power* | | *Thermal parameters* | |
| Net power generation (MW(e)) | 70 | Coolant inlet temperature (ºC) | 290 |
| Thermal power generation (MW(th)) | 218 | Coolant outlet temperature (°C) | 326 |
| Core power density (KW(th)/litre) | 45.6 | Coolant average temperature (°C) | 308 |
| Pump power (MW(e)) | 2 | Fuel operating temperature (°C) | 354 |
| Pump power fraction (%) | 2.8 | Coolant inlet enthalpy (kJ/kg) | 1284 |
| *Fuel element parameters* | | Coolant inlet density (kg/m$^3$) | 747 |
| Fuel element outer diameter (cm) | 1.5 | Coolant average density (Kg/m$^3$) | 710 |
| Zircaloy clad thickness (cm) | 0.03 | Enthalpy rise in the core (kJ/kg) | 1490 |
| $UO_2$ in each fuel element (% volume) | 23.9 | Film boiling convective heat transfer coefficient at 300 ºC (W/m$^2$ ºC) | 454 |
| $UO_2$ density (g/cm$^3$) | 10.5 | Fuel element average thermal conductivity (W/mºC) | 12.5 |
| Zirconium density (g/cm$^3$) | 6.5 | Thermal conductivity of Zirconium (W/m°C) | 18 |
| Fuel element average density (g/cm$^3$) | 8.09 | *Neutronic characteristics* | |
| *Core parameters* | | Moderator coefficient (mk[6]/°C)-BOL | $-3\times10^{-4}$ |
| Core height (cm) | 200 | Moderator coefficient (mk/°C)-EOL | $-8\times10^{-4}$ |
| Core inner diameter (cm) | 31 | Doppler coefficient (mK/°C) - BOL | $-6\times10^{-5}$ |
| Core outer diameter (cm) | 171 | Doppler coefficient (mK/°C) - EOL | $-7\times10^{-5}$ |
| Core volume (m$^3$) | 4.78 | Core height level limiter (CHLL) Sensitivity (mk/cm) - BOL | 0.37 |
| Number of fuel elements in the core. | $1.62\times10^{6}$ | Core height level limiter (CHLL) Sensitivity (mk/cm) - EOL | 0.059 |
| Weight of fuel elements in the core (tons) | 23.2 | Boron sensitivity (mk/ppm) – BOL | 0.039 |
| Weight of $UO_2$ in the core (tons) | 11.5 | Boron sensitivity (mk/ppm) – EOL | 0.080 |
| *Hydraulic parameters* | | *Fuel burnup characteristics* | |
| Coolant volume (m$^3$) | 10 | Fuel burnup (MW day/t U)/refuelling interval (years) | 26 000/ 2.2 |
| Coolant mass flow (kg/s) | 1060 | Plutonium production (Kg) | 62 |
| Coolant pressure (bar) | 160 | Remaining $^{235}U$ (Kg) | 340 |
| Pressure loss in the loop (bar) | 12.3 | *Reactor performance in accidents* | |
| Pressure loss in the bed (bar) | 1.3 | Maximum fuel temperature after a LOCA (ºC) | 542 |
| Terminal velocity (m/s) | 1.5 | Coolant temperature rise after a LOFA after 10 days (ºC) | < 1 |
| Operating coolant velocity (m/s) | 7.23 | Water needed to cool the reactor during for 10 days after LOCA (m$^3$) | 0.9 |

[6] mk stands for $10^{-3}$.

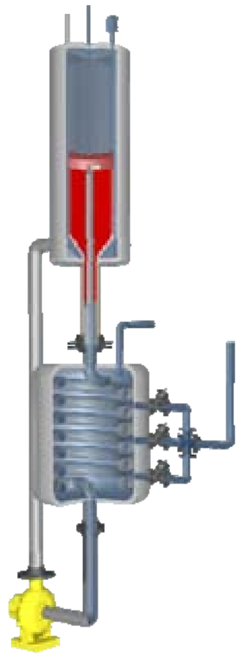


*FIG. 31. Design of the FBNR fuel chamber.*

A steam generator of the shell-and-tube type is integrated in the upper part of the reactor module. The reactor is provided with a pressurizer system to keep the coolant at a constant pressure. The inlet and outlet temperature of coolant in the core are 290°C and 326°C corresponding to the enthalpies of 1283 and 1489 KJ/kg and giving an enthalpy rise of 206 KJ/kg. The coolant mass flow rate at the operating condition is 1060 Kg/s corresponding to a coolant velocity of 7.2 m/s. Thus, the reactor produces a thermal power of 218.4 MW(th) corresponding to an electric power of 70MW(e).

The pump circulates the coolant inside the reactor moving it up through the fuel chamber, the core, and the steam generator. Thereafter, the coolant flows back down to the pump through the concentric annular passage. At a flow velocity called terminal velocity, the coolant holds the 15 mm diameter spherical fuel elements pneumatically suspended in the core. A fixed suspended core is maintained in the reactor so long as the pump is running. In the shut down condition, the suspended core breaks down and the fuel elements leave the core by the force of gravity and fall back into a subcritical configuration in the fuel chamber.

The pump circulates the water coolant in the loop and at the mass flow rate of about 220 kg/s, corresponding to the terminal velocity of 1.50 m/s in the reserve fuel chamber, carries the fuel elements into the core and forms a fixed bed. At the operating flow velocity of 7.23 m/s, corresponding to the mass flow rate of 1060 kg/s, the fuel spheres are firmly held together by a pressure of 0.188 bars that exerts a force of 27.1 times their weight, thus forming a stable fixed bed. The fixed bed is compacted by a pressure of 1.3 bars. The coolant flows in the core in radial direction and, after absorbing heat from the fuel elements, it enters the integrated heat exchanger of tube and shell type. Thereafter, it circulates back into the pump and the fuel chamber.

*Passive scram action and decay heat removal*

The operating condition of the 70 MW(e) reactor corresponds to the coolant flow velocity of 7.2 m/s. The terminal velocity (being the minimum coolant velocity to carry the fuel elements into the core) is 1.5 m/s. The maximum flow velocity above which the reactor operation becomes impractical is of 25 m/s owing to excessive pumping power requirement.

A trip signal from any of the detectors, due to defined initiating events, will cut-off power to the pump, causing the fuel elements to fall back into the fuel chamber where they remain in a highly subcritical and passively cooled condition.

The water flowing from an accumulator, which is controlled by a multi redundancy valve system, cools the fuel chamber – functioning as the emergency core cooling system. The fuel chamber is also cooled by natural convection, transferring heat to the water in the tank housing the fuel chamber.

*Control and protection systems*

The long-term reactivity is provided by fresh fuel addition via increasing the height of the particle bed and possibly aided by a fine control rod that moves in the centre of the core controlling short-term reactivity. A piston type core limiter adjusts the core height and controls the amount of fuel particles that are permitted to enter the core from the reserve fuel chamber.

The critical core height is about 200 cm. The core height can be changed by the core height level limiter (CHLL). The largest effect of CHLL is 0.37 mk/cm at BOL and decreased down to 0.059 mk/cm at EOL. The effect of soluble boron in the moderator is 0.039 mk/ppmB[7] at BOL.

The plant protection system is conceived to operate ‘fail safe’ wherein the pump can only operate when all the signals coming from the control detectors simultaneously indicate safe operation. Under any possible inadequate functioning of the reactor, electrical power does not reach the pump and the coolant flow stops, causing the fuel to relocate out of the core by the force of gravity and become stored in the criticality-safe, passively cooled fuel chamber.

The fuel operating temperature is 354° – only 46°C above the coolant average temperature and only 28°C above the coolant outlet temperature. Maximum fuel temperature in a loss of coolant accident (LOCA) is 542°C, providing at least 1000°C margin to fuel damage.

In addition to standard low enrichment uranium fuelling, the FBNR has been evaluated for U/Pu oxide fuel particles manufactured from a FBNR spent fuel recycle and also for the mixed content of $UO_2$/$ThO_2$ particles where only the uranium particles are enriched.

*Future work*

The innovative pneumatic core support approach requires delicate force balances for maintaining the steady state configuration and, especially, for assembling the core at start-up and after a scram. It is planned to conduct experiments in a scaled test rig to further develop the engineering details of the FBNR concept.

The FBNR has been evaluated using the IAEA INPRO methodology in respect to its safety and non-proliferation features.

### 5.5. Conclusions to Section 5

Within the activities of CRP Group 2, several technology development issues for small water cooled reactors without on –site refuelling with particulate based fuel were addressed. First, the issues of fuel performance were reviewed. As the first candidate for particulate-bed fuel,

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[7] ppmB is for percent per million ($10^{-6}$) of boron.

TRISO fuel was investigated in the early phase of the CRP efforts. Out-of-pile corrosion testing of the TRISO particles with Si-C outer layer in hot water and in steam environments typical of reactor service conditions was shown to produce mixed results. Testing at VNIIAM was showing excellent corrosion resistance but testing at PNNL suggested corrosion issues. This led the PNNL fuel developers to consider an alternative particulate fuel form – a cermet spheres made of $UO_2$ kernels in a Zr matrix coated with an outer Zr-1Nb layer impermeable to fission products.

Subsequently, during in-pile testing of the TRISO particles conducted by VNIIAM it was found that pyro-carbon and Si-C could experience integrity problems under low temperature irradiation, related to the accumulation of atomic displacements in the graphite lattice structure (Wigner energy), owing to insufficient annealing at temperatures below ~260°C. The open sharing of these testing results facilitated the designers to investigate whether or not their TRISO-fuelled designs could use the newly proposed cermet fuel form. The concepts of small light water reactors with micro fuel elements (MFE) were then re-designed using this new cermet type of particulate fuel. Preliminary studies performed during the project demonstrated suitability of such fuel for small water cooled reactors without on-site refuelling. Moreover, cermet fuel in some cases appears to be more adequate than the TRISO fuel to achieve the desired features of such reactors.

Second, benchmarking of the neutronic depletion codes on cell and fuel assembly models of small water cooled reactors with particulate-based fuel was performed. First, TRISO particulate-based unit cell evaluations of $k_\infty$ versus burnup were conducted for all reactor concepts (AFPR, VKR-MT, PFPWR-50, and FBNR) using the computer code packages and databases in use at the several design organizations (APOLLO, SPAC95, and MCNP). Burn-up reactivity letdown curves and lifetime predictions were similar between APOLLO and SRAC95 but BOL $k_\infty$ had a nontrivial variability between concepts. Using MCNP as the standard, each code displayed a 2% $\Delta k/k$ swing in bias from one reactor to another. Additionally, effects of double heterogeneity could not be well addressed because the option to address them properly was not available in all codes. The results of these first benchmarking works for TRISO fuel were published and are available in reference [19].

The above issues for TRISO fuelled design benchmarks were set aside unresolved so as to address the differences between neutronics performance of TRISO versus cermet fuelling of the several concepts. A one-for-one replacement of cermet for TRISO particle fuelling of the various reactor configurations was evaluated at the unit cell level at the Mohammed V University in Rabat (Morocco). These results are presented in Annex IV. Generally speaking, the unit cell results showed a higher $k_\infty$ and larger attainable discharge burnup owing to a much larger thermal component in the neutron spectrum for the TRISO fuelling as compared with the cermet. This problem can be addressed via changes in fuel enrichment.

# 6. FAST NEUTRON SPECTRUM REACTORS WITH CHEMICALLY INERT COOLANT

## 6.1. Introduction

Nuclear energy would have to significantly increase in market share of world energy supply in order to offer a significant contribution to reduction in global greenhouse gas emissions. The participants of the CRP believe that small reactors without on-site refuelling could facilitate such increases in market share by virtue of their targeted applicability to the needs in developing countries where future energy demand growth will be increasing. Given a large and ever-growing nuclear deployment, some fraction of the small reactors without on-site refuelling could be fast reactors for reasons of resource extension and of nuclear waste management. The fast spectrum small reactors without on-site refuelling can be deployed along with light water reactors (LWR) in a symbiotic fuel cycle which incorporates the spent fuel of LWRs into the working inventories of newly deployed fast spectrum small reactors without on-site refuelling, as a waste management strategy.

At the same time, fast spectrum small reactors without on-site refuelling could be deployed along with dedicated fast breeder reactors to transform the world's massive ore reserves of fertile material ($^{238}U$ and/or $^{232}Th$) into fissile materials to fuel a growing global nuclear energy park, see Section 7. These small reactors could maintain a conversion ratio of about unity as the way to avoid large loss of reactivity over long refuelling intervals. While not breeding excess fissile mass, they at least are fissile self-sufficient once deployed.

The fissile self-sufficient fast spectrum small reactors without on-site refuelling could be either sodium cooled fast reactors or the reactors cooled by Pb-Bi alloy or just by Pb coolant. While the traditional fast breeder reactors, which operate at high power density, are best cooled by sodium because it can be pumped at high velocities to remove the heat, small reactors without on-site refuelling, to the contrary, operate at low power density and can easily be cooled by the more dense and difficult to pump heavy liquid metals such as Pb-Bi eutectic or pure Pb. The heavy liquid metal coolants eliminate a stored chemical energy hazard because, unlike sodium, these coolants don't react vigorously with air or water. Moreover, they retain the ambient pressure primary coolant circuit advantage shared by all liquid metal cooled systems and dramatically extend the margin to coolant boiling which is already large for sodium cooled systems. These features offer the potential for plant simplification, enhanced passive safety features, and cost reduction[8].

In light of the above mentioned features, significant worldwide interest in heavy liquid metal cooled small reactors without on-site refuelling exists, see 11 concept descriptions of such reactors in reference [2]. Four participants of the CRP coming from 4 IAEA Member States member were developers of such concepts:

- Russian Federation (IPPE) — SVBR-75/100, SVBR-10
- USA (ANL) — SSTAR, STAR-LM
- Indonesia (Bandung Institute of Technology — SPINNOR/VSPINOR, CANDLE
- Japan (Research Laboratory for Nuclear Reactors, Tokyo Institute of Technology) — LSPR, CANDLE
- India (Bhabha Atomic Research Centre) — Design alternative for CHTR[9]

[8] For balance, a $^{210}Po$ problem should be noted in conjunction with Pb-Bi cooled reactors. The volatile, $\alpha$-active Polonium-210 is produced from Bi under irradiation and has a half-life of 138.39 days. It is deadly radiotoxic when inhaled; therefore, special measures need to be taken to trap it and prevent its release into the environment.

[9] Compact High Temperature Reactor (CHTR) is lead-bismuth cooled thermal spectrum reactor with high-temperature particulate-based $^{233}U$-Th fuel. It has a conversion ratio well below 1.0 (0.7-0.8). The design alternative considered in this chapter incorporated alternative fuel design operating at a lower temperature. This non-conventional design concept should not be mixed with the rest of the concepts considered in this Section.

The activities of Group 3 of the CRP were focused on Pb-Bi or Pb cooled fast-spectrum small reactors without on-site refuelling. This chapter summarizes results for the three activities, with more detail provided in Annexes:

1. A neutronics benchmark exercise – Annex IV
2. Conceptual designs of two small relocateable Pb-Bi cooled power plants for remote settlements (not addressed in reference [2]) – Annex V and Annex VI
3. The CANDLE breed-and-burn reactor concept – Annex VII.

As it was already mentioned, collaboration with OECD-NEA was established in benchmark calculations of forced and natural convection modes of lead-bismuth simulating the tests performed in the HELIOS loop at the Seoul National University of the Republic of Korea. Several participants of Group 3 contributed to these activities, via the CRP 'Small Reactors without On-Site Refuelling'. The results obtained by them, along with results of other participants of this benchmark exercise, will be presented in a OECD-NEA report, once the benchmarking is completed (tentatively, in 2010), see [20].

### 6.2. Benchmarking on a depletion model of the whole core of a Pb-Bi cooled reactor

While the technology for Pb-Bi coolant has operational and testing experience in the Russian Federation (epi-thermal spectrum reactors for submarine service), the use of Pb-Bi coolant in a fast neutron spectrum reactor is new even in the Russian Federation.

In addition to this, many of the small heavy liquid metal cooled fast spectrum reactor concepts have a design objective to minimize the burnup reactivity swing during core lifetime, specifically, to keep it close to zero (less than one effective delayed neutron fraction) in order to exclude accidents with inadvertent control rod withdrawal. Accurate simulation of such small burnup reactivity swing requires a careful modelling of the balance of fissile material consumption and build-up of absorbers (fission products and fertile materials) versus in-core production of secondary fissile materials. Therefore, the CRP deemed it worthwhile to conduct a detailed neutronics (depletion) benchmark exercise to gage the degree of consistency that would be attained by the various design teams working on small reactors without on-site refuelling cooled by Pb-Bi alloy.

Accordingly, a benchmark geometry, composition, power level, and refuelling interval were specified by the Russian Research Centre 'Kurchatov Institute' (Moscow, Russian Federation). The benchmark exercise included the reporting of calculated beginning of cycle (BOC) and end of cycle (EOC) eigenvalues, power profiles, reactivity as a function of fuel burnup, and detailed neutron balances. The specification and results are detailed in Annex V. This chapter summarizes main outputs of the performed investigation.

Six design teams participated in the benchmarking exercise:

ANL – Argonne National Laboratory, Argonne, Illinois, USA;
BARC – Bhabha Atomic Research Centre, Mumbai, India;
Gidropress – EDO 'Gidropress', Podolsk, Moscow Region, Russia;
ITB – Bandung Institute of technology (ITB), Bandung, Indonesia;
RRC KI – Russian Research Centre 'Kurchatov Institute', Moscow, Russia'
TokyoTech – Tokyo Institute of Technology, Tokyo, Japan.

RRC KI (Russian Federation) took the lead in assembling and comparing the results.

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The designers of the power pack participated in benchmarking exercises within Group 3 for the purpose of verification of codes.

The calculations were carried out using different code systems and nuclear data derived from different libraries. Both deterministic and Monte Carlo methods have been used. Table 17 shows code systems and nuclear data libraries used by participants of the benchmark calculations.

The results of calculations displayed notable differences – especially in $k_{eff}$ – among the participants. The inter-comparison study has been aimed at identification of the sources of the discrepancies between the different methods and libraries.

First, as shown in Table 18, the eigenvalues calculated at BOC ranged from 0.995 to 1.010 – a range of ~1.5% $\Delta k/k$ – for the specified geometry and composition.

Next, as shown in Fig. 32, the changes in reactivity with burnup, while all manifesting slow growth and saturation, showed variation in the size of the burnup swing (between $k$ in the peak and $k$ at BOC).

TABLE 17. CODE SYSTEMS AND NUCLEAR DATA LIBRARIES USED

| Participant | ANL | BARC | Gidropress | ITB | RRC KI | Tokyo Tech |
|---|---|---|---|---|---|---|
| Code system for cell calculation | MC2-2 | | | | | SRAC |
| Code system for criticality calculation | DIF3D,<br>TWODANT | ERANOS2.0 | DIFRA,<br>KINRZ | SRAC<br>FI-ITB-CHI | MCNP5 | Original |
| Code system for burnup cycle analysis | REBUS-3 | | BURNUPRZD,<br>BURNUPRZK | | ISTAR-2 | Original |
| Nuclear data library | ENDF/B-V.2 | JEF2 | ENDF/B-VI.3,<br>ENDF/B-VI.5 | JENDL3.3 | ENDF/B-VI.8,<br>ENDF/B-VII* | JENDL3.3 |
| Number of energy groups | 33 or 230 | 33 | 30 | 107 | Pointwise cross-section data | 21 |

*-additional data.

TABLE 18. CALCULATED $K_{eff}$ AT BOC

| Participant | ANL (230 groups) | | BARC | Gidropress | | ITB | RRC KI | Tokyo Tech |
|---|---|---|---|---|---|---|---|---|
| Moment of time | DIF3D | TWODANT | ERANOS2.0 | DIFRZ | KINRZ | SRAC | MCNP5 | Original |
| BOC | 0.99781 | 0.99937 | 0.99498 | 1.0076 | 1.0084 | 1.00361 | 1.00383 | 1.0104 |

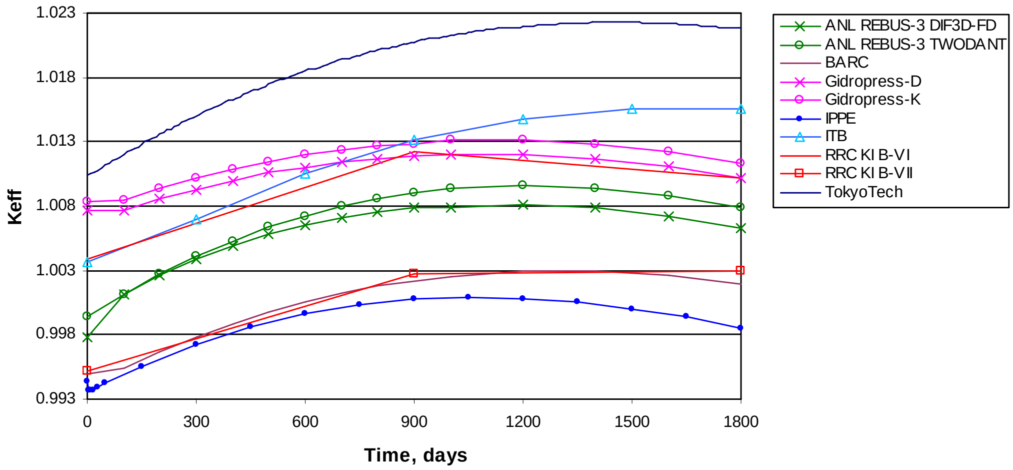


*FIG. 32. Evolution of $k_{eff}$ on 1800-day campaign at thermal power 900 MW.*

Extensive evaluations were undertaken at RRC KI to track down the underlying sources of the observed differences at BOC; those included the examination of:

- Basic nuclear data libraries;
- Data pre-processing, including
  - Spectral effects;
  - Isotopic neutron balance effects;
- Pb inelastic scattering effects.

It was possible to place the source for most of the variability on the differences among the basic data libraries used by the participants or on the use of multi-group cross section sets that had been generated using generic fast neutron spectra for compositions that are different from that of the benchmark compositions. For example, in all multi-group calculations for $^{238}U$, a value of ~2.7 was used for the total fission neutron production per fission whereas the MCNP5 calculates this value as ~2.5.

Pb inelastic scattering and absorption data were found to have smaller effects on variability of the outcomes than had been found in earlier studies. Indeed, for the benchmark composition, the inelastic scattering on $^{238}U$ is ~3 times that of Pb.

In addressing the variability in predicted burnup swing, the differences among EOC eigenvalues were larger than at BOC, specifically:

- The evolutions of $k_{eff}$ on a 1800-day campaign in the ANL, BARC, Gidropress, IBT, RRC KI and TokyoTech simulations are of the same type – slight increase due to breeding in blankets and then decrease due to fission product accumulation;
- The evolution of $^{238}U$ and Pu isotopic densities for Core-l is similar; similar also is the evolution of $k_{eff}$ during the cycle. In the BARC and TokyoTech simulations the growing of $^{239}Pu$ content shows that the core breeding ratio is greater than 1. For the lateral blanket, $^{239}Pu$ generation is maximal in the RRC KI calculation, with smaller values obtained in the BARC and TokyoTech calculations;

- The radial power density shifts during burnup were quite small (small swings in core leakage probabilities), and all participants calculated nearly the same region-wide and local power distributions during the cycle.

Simulations of the accumulation of fission products during the cycle show that the BARC and the RRC KI fission product parameters are very close. In the TokyoTech simulation, the atomic density of the main fission products in Core-1 is smaller than in the RRC KI simulation, but the neutron capture macroscopic cross section $\Sigma_a$ is larger. The reason may be in using the neutron spectra different (softer) from that of the benchmark composition to prepare the multi-group microscopic cross sections. In lateral blanket the neutron spectrum is softer then in Core-l, so greater fission product generation in the TokyoTech simulation causes greater neutron capture.

Multi-group libraries usually contain several combined fission products – this modelling option drastically reduces the number of fission products that must be followed in criticality calculations, but can produce some additional error especially for a new type of reactor and neutron spectra. Therefore, for comparison, RRC KI accounted for fission products in detail, using about 100 isotopes in criticality calculations and about 1000 in isotope kinetics calculations, but it made the calculations more complicated and required significantly more time.

The power fractions in radial zones at BOC, at the middle of the burnup cycle (900 days) and at EOC (1800 days) displayed rather good agreement, see Figures 33, 34 and 35.

Despite the noted differences, – especially in $k_{eff}$ – all-in-all, a gratifying level of consistency among different design teams was displayed on this first of a kind Pb-Bi alloy cooled fast spectrum depletion benchmark. While the degree of consistency lends some confidence to predictions of design performance at the conceptual and preliminary stages of design, the large spread in $k_{eff}$ predictions makes it clear that critical experiments would ultimately be needed as the concepts progress toward advanced design stages.

Annex V presents the benchmark specification and the inter-comparison of results in detail.

### 6.3. Design concepts of the two relocateable heavy liquid metal cooled small reactors without on-site refuelling

Many communities and industrial sites are so remote that they will never be considered for connection to a regional electrical grid. For example, the northern and far-eastern shores of Siberia are home to such sites. Similarly, high mountainous regions of north-eastern India and islands off India’s coasts are populated with ‘forever off grid’ sites. The thousands of islands of Indonesia provide similar examples of autonomous small local-grid situations. These are typical of the extensive opportunities for small reactors without on-site refuelling to meet the needs of off-grid populations and industrial facilities (e.g. mines) throughout the world by providing safe, secure, and clean energy services.

Two concepts for very small, relocateable small reactors without on-site refuelling designed to meet such needs were examined as a part of the CRP.

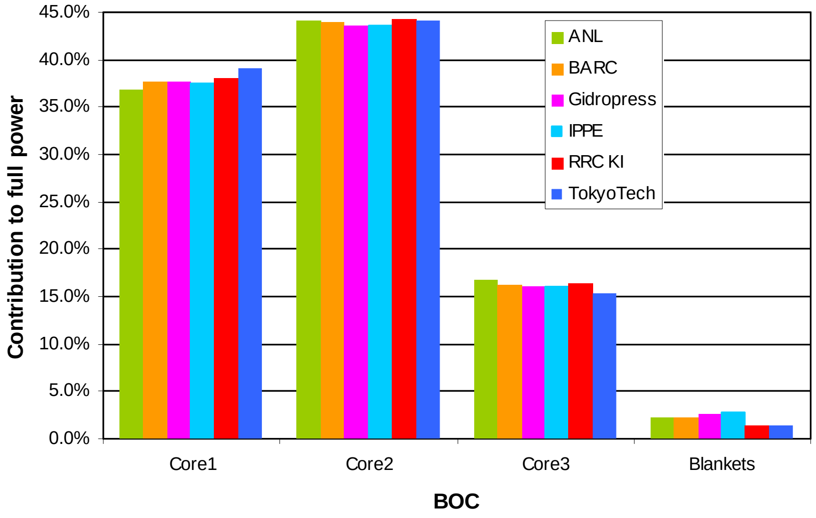


*FIG. 33. Power in zones at the beginning of the cycle.*

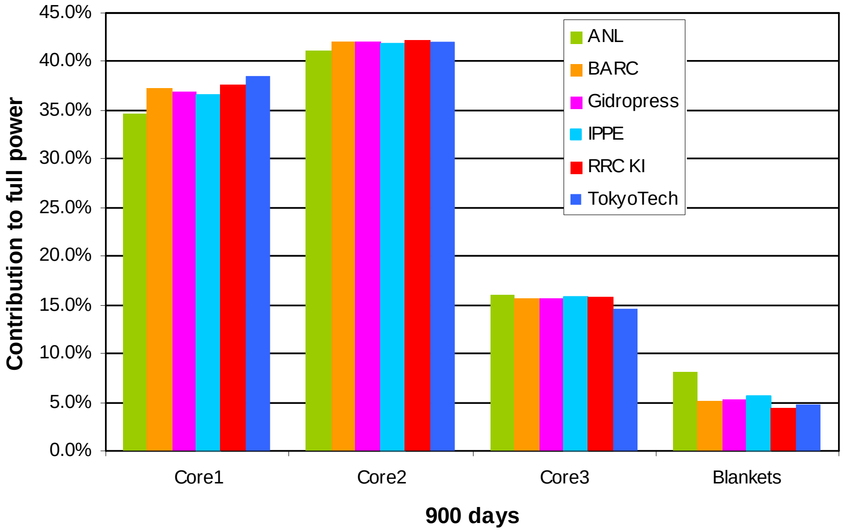


*FIG. 34. Power in zones at 900 days.*

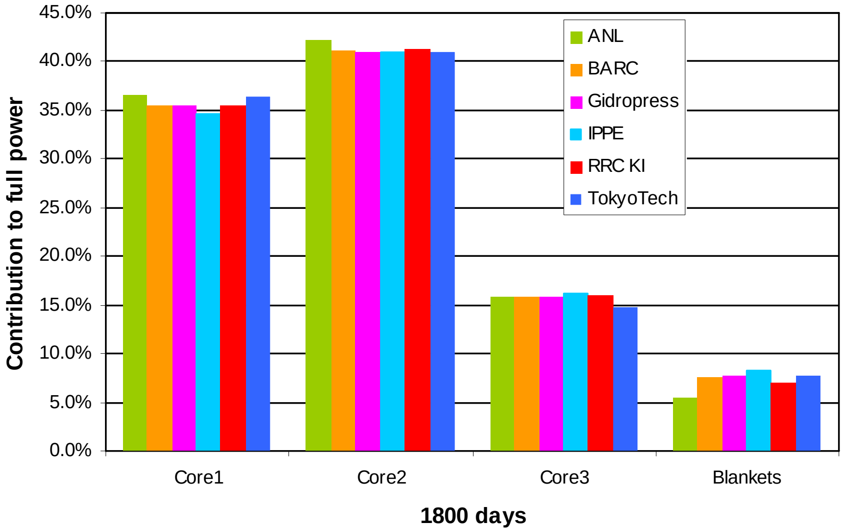


*FIG. 35. Power in zones at 1800 days.*

### 6.3.1. SVBR-10

Conceptual design of a barge-mounted power plant comprised of the two SVBR-10 Pb-Bi cooled reactors has been developed at the EDO ‘Gidropress’ in the Russian Federation. It can provide 20 years of energy supply before being towed back to the factory for refuelling. It delivers 12 MW(e) and up to 50 GCal/hour of co-generation heat to remote water-accessible sites. The twin power plants are mounted on an ocean-going barge of 8000 ton displacement (93 m length × 21.6 m beam with 4 m draft). The two SVBR-10 power plants are similar to the SVBR 100 concept; they also rely extensively on the experience of Russian Pb-Bi cooled reactors for submarines. The SVBR-10 concept is described in more detail in Annex V.

### 6.3.2. Multi-purpose nuclear power pack

The concept of a 5 MW(th), Pb-Bi-cooled, thermal spectrum small reactor without on-site refuelling of ~10-year refuelling interval is under development at the Bhabha Atomic Research Centre in India. The reactor is intended for off grid energy delivery. It employs a thorium based closed fuel cycle using $^{233}U/^{232}Th/Zr$ metallic alloy fuel. The high temperature capability of Pb-Bi coolant is being explored with a core outlet temperature of 600°C to drive (for example) passive thermoelectric power generation. The multi-purpose power pack is a design alternative to the compact high temperature reactor (CHTR) described in annex XXIX of reference [2]. Different from CHTR, the power pack has lower operating fuel temperature and incorporates an alternative fuel design.

The concept incorporates numerous innovations, including:

- Natural circulation flow of Pb-Bi primary coolant;
- Heat pipe transfer of heat from the primary coolant to diverse heat applications in the balance of plant;
- An innovative passive power self regulation system comprised of a boron carbide control rod floating in Pb-Bi in an arrangement that passively moves the rod in response to coolant inlet temperature sensed at the return condition of the heat pipes that carry heat to the balance of plant;
- A passive decay heat removal path whose heat transfer impedance is decreased when required by a siphon switch that fills gas gaps with Pb-Bi to increase heat transfer.

The concept development is proceeding at the intended commercial scale of 5 MW(th) – with the intent to subsequently demonstrate its many innovative features in a prototype test. The concept and the status of its development are summarized in Annex VI.

## 6.4. CANDLE breed-and-burn reactor concept

Most of the thermal neutron spectrum small reactors without on-site refuelling operate in a once-through fuel cycle. They use enrichment services for each reload but they don’t require reprocessing services. They manage burnup reactivity loss over extended refuelling interval with burnable poisons (or soluble poison) and their efficiency of ore resource use is similar to that of standard light water reactors.

Most of the fast neutron spectrum small reactors without on-site refuelling operate in a closed fuel cycle. They use enrichment services only once-and-for-all for the first (enriched-uranium) core loading but, thereafter, their refuelling needs are met by self-fuelled recycle with make-up feedstock comprised only of depleted uranium. They manage burnup reactivity loss with an internal breeding ratio of unity and their efficiency of ore resource use is extremely high – repeated recycle will eventually fission essentially 100% of uranium in a two step process – first converting it to fissile transuranics, then fissioning them.

The concept of a small reactor without on-site refuelling that lies between these two traditional approaches has been under consideration by the two participants of the CRP (TokyoTech, Japan, and Bandung Institute of Technology, Indonesia). It is a fast neutron spectrum reactor that employs enrichment services only once-and-for-all for the first core loading, operates in a once through cycle, but none-the-less achieves an ore resource use about half way between those of the two approaches mentioned in the previous paragraph. This conceptual approach called CANDLE (also known as a breed-and-burn, a nuclear deflagration wave, or a travelling wave approach) initially ignites a stable chain reaction using an enriched uranium zone located at the end of a depleted uranium core; thereafter, neutron leakage into the adjacent depleted uranium core will, over time, breed transuranics sufficient to sustain a chain reaction, and the burn zone will progressively move along the core as it breeds fuel in front of its travel and leaves spent 'ash' behind the wave front, see Fig. 36 and reference [1]. The 'ash' can be removed, and new depleted uranium can be placed in front of the burn zone to sustain power production without any further need for enrichment services or reprocessing of any kind. Consumption of the $^{238}U$ as high as 40% may be attainable.

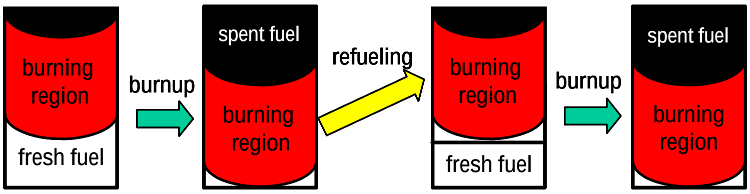


*FIG. 36. CANDLE burnup and refuelling scheme [21].*

In practice, it is found that fuel re-cladding would be required to deal with neutron fluence limits on the available materials. Fuel shuffling of the re-cladded fuel may also be required to correct wave front distortion or sideways drift of the wave front away from the axis. Results of the feasibility studies (focused primarily on neutronics issues) that have been conducted in Japan and Indonesia are presented in Annex VII.

### 6.5. Conclusions to Section 6

Collaborative activities of Group 3 of the CRP were focused on Pb-Bi or Pb cooled fast-spectrum small reactors without on-site refuelling; specifically, they included a depletion benchmark exercise. A numerical whole core depletion model of a Pb-Bi reactor was developed by RRC KI and used as a benchmark to perform verification of the neutronic codes and data libraries. The calculations were carried out using different code systems and nuclear data derived from different libraries. Both deterministic and Monte Carlo methods have been used.

The results of calculations displayed notable differences – especially in $k_{eff}$ – among the participants. The inter-comparison study has been aimed at identification of the sources of the discrepancies between the different methods and libraries.

Despite the noted differences, – especially in $k_{eff}$ – all-in-all, a gratifying level of consistency among different design teams was displayed on this first of a kind Pb-Bi alloy cooled fast spectrum depletion benchmark. While the degree of consistency lends some confidence to predictions of design performance at the conceptual and preliminary stages of design, the large spread in $k_{eff}$ predictions makes it clear that critical experiments would ultimately be needed as the concepts progress toward advanced design stages.

In addition to the benchmarking, individual participants of Group 3 carried on, and shared information about:

- Conceptual design development for two small relocateable Pb-Bi cooled power plants for remote settlements (not addressed in reference [2]) – see Annex V and Annex VI;
- Further development of the CANDLE breed-and-burn reactor concept – see Annex VII.

In addition to this, collaboration with OECD-NEA was established in benchmark calculations of forced and natural convection modes of lead-bismuth simulating the tests performed in the HELIOS loop at the Seoul National University of the Republic of Korea. Several participants of Group 3 contributed to these activities, via the CRP 'Small Reactors without On-Site Refuelling'. The report on this exercise will be produced by OECD-NEA.

# 7. STUDY OF DEPLOYMENT APPROACHES FOR SMALL REACTORS WITHOUT ON-SITE REFUELLING UNDER CONSTRAINTS

## 7.1. Introduction

Design concepts of small reactors without on-site refuelling exist in many forms (thermal reactors and fast reactors with various coolants, fuel forms, and power ratings). But as a class, many of them are being designed to meet the growing demand for secure, safe, and affordable energy supply in developing countries to help fuel the economic growth [2]. The long refuelling interval particularly makes such reactors attractive for the architecture based on distributed small reactors without on-site refuelling supported by the regional fuel cycle centres providing fuel supply and waste management services for the regional nuclear energy park as a whole. In this architecture, the small reactors without on-site refuelling ensure energy supply security to the users, because of their long refuelling interval, and the regional centres facilitate an integrated, symbiotic application of the constrained global fissile resource. In such an architecture, it is presumed that small reactors without on-site refuelling will occupy a significant market share of national nuclear energy parks and that the make-up and evolutionary growth of each national park will differ as each would seek how best to meet local needs and preferences.

Over the current century the growth of energy demand is projected to be quite substantial and, as a result, small reactors without on-site refuelling may have a chance to be deployed in significant numbers. One of the activities conducted within the CRP was to investigate what influence various constraints might place on achievable growth rates of the small reactor sector of an overall nuclear energy park. Two categories of constraints have been investigated, growth under a self-financing constraint (Section 7.2) and growth under fissile mass availability constraints (Section 7.3). Section 7.4 presents the study on deployment approaches to minimize external financing in a capitalization-constrained growth of the nuclear park. Finally, an approach to sequencing of the evolutionary steps toward a closed nuclear fuel cycle was proposed and examined (see Section 7.5).

## 7.2. Growth under a self-financing constraint

The first study addressed the early stage of national nuclear park growth under an assumption that the revenues generated by the nuclear park by themselves cover the cost both of replacing the existing capacity as it reaches end of life and of financing the incremental capacity growth with whatever financing remains. The Russian nuclear park was taken as an example and the question was what growth rate could be attained under the two plausible approaches:

a. Existing WWER type reactors, as they reach the end of life, would be replaced by the WWER-1500 reactors, and incremental deployments would also use the WWER-1500 reactors;
b. Existing WWER type plants, as they reach the end of life, would be re-powered with fast spectrum small reactors without on-site refuelling, and incremental deployments would again use the WWER-1500 reactors.

Option (b) applies the re-powering approach to the nuclear sector for the first time. Re-powering is an option considered for the replacement of power plants reaching the end of their life. In the case of coal plants, the furnace (boiler) may be decommissioned and replaced with a combustion gas turbine. A heat recovery heat exchanger is added to the gas turbine exhaust so as to generate steam to be fed to the existing Rankine steam cycle balance of plant equipment, thereby creating a combined cycle gas power plant at a fraction of the cost of a green field plant. The re-powering option also reduces cost by using existing civil structures, switch yard and the utilities and avoids the costs of permitting a green field site.

Small reactors without on-site refuelling offer a potential to adopt the re-powering option to large central-station nuclear power plants. In strategy (b) the refurbishment would emplace 100 MW(e) SVBR 100 small reactors [2] without on-site refuelling into vacated steam generator chambers of the WWER based power plants, as they reach their end of life. The SVBR 100 reactors would supply steam to the existing balance of plant Rankine steam cycle equipment which, along with most of the original civil construction, would not be replaced but rather would only be refurbished. This, of course, reduces the capital cost of refurbishment as compared to the cost of the replacement experienced in approach (a).

With a presumed replacement rate:

$$W = 1 GW_e/\text{year},$$

and with components of the tariff from sale of electricity from the nuclear park specified:

$C_r$ – component of tariff set aside for replacement, cents/$kW_e$ hour,

$C_d$ – component of tariff set aside for growth, cents/$kW_e$ hour,

along with specified capital costs:

$$K_r \frac{\$}{kW_e} \quad \text{for replacement reactors},$$

$$K_d \frac{\$}{kW_e} \quad \text{for growth reactors},$$

an equation for achievable park growth rate could be derived as the following:

$$\frac{dP(t)}{dt} = a\,P(t) - b \tag{2}$$

Where

$$a = LF * 87.6\ \frac{C_r + C_d}{K_d} \tag{3}$$

$$b = \frac{K_r}{K_d} W$$

Here LF is the nuclear park average load factor.

Using capital cost numbers reported in references [22, 23] and total re-investment ($C_{inv} = C_r + C_d$) of 0.49¢/kW(e)hour from the nuclear park as a whole, the achievable growth rate results are as shown in Table 19. The growth rate P(t) achievable under self-financing in approach (b) is 0.89 GW(e)/year or a reactor park doubling time of 25.1 years whereas under approach (a) the achievable growth rate is 0.35 GW(e)/year or a doubling time of 63.5 years.

TABLE 19. DYNAMICS OF NUCLEAR POWER DEVELOPMENT WHEN DIFFERENT DEPLOYMENT APPROACHES ARE USED

| **Parameter** / **Nuclear power technology** | **$C_{inv}$ ¢/kW•h Total reinvestment** | **$C_r(0)$ ¢/kW•h Reinvest-ment for renovation** | **$C_d(0)$ ¢/kW•h Reinvest-ment for new build** | **IBR Investment breeding ratio** | **$T_2$ Doubling time, years** | **P(W(th)) GW(e)/y Park growth rate** |
|---|---|---|---|---|---|---|
| According to approach (a) | 0.49 | 0.46 | 0.03 | 1.05 | 63.5 | 0.35 |
| According to approach (b) | 0.49 | 0.27 | 0.22 | 1.47 | 25.1 | 0.89 |

Additional benefits identified for approach (b) result from:

- The fact that, as the growth rate of the reactor park (P(t)) and the general profit increase ($C_r$) will decrease at a constant value of W=1 GW(e)/year, the $C_d$ will increase correspondingly, and

- The fact that such an approach will include an opportunity to expeditiously bring the SVBR 100 industrial infrastructure into place in Russia – holding the potential for ensuring international sales.

The details of these analyses are reported in reference [24].

### 7.3. Growth under fissile mass availability constraints

The second study addressed the long-term (100 years) growth potential of the world nuclear park when it contains a significant market share of small reactors without on-site refuelling and is constrained by fissile mass availability.

An idealized representation of the growth of a reactor park comprised of a specified mix of light water reactors (LWRs), fast breeder reactors (FBRs), and fast-spectrum small reactors without on-site refuelling was considered:

$$\frac{d}{dt} P(t) = \alpha P(t) \tag{4}$$

where α is the growth rate attainable under self-fuelling via net breeding of the reactor mix, i.e. the park's attainable growth rate under self-generated fissile mass production. P(t) is the thermal power of the nuclear park as a whole. Thermal (rather than electrical power) is used because it is directly tied to fissile mass consumption via a constant (1 g fissioned ≡ 1 MW(th) day) – and a capacity factor of unity is assumed for this idealized model.

Fissile consumption and production rates were assumed equal to those for typical LWRs and powerful FBRs, and the properties of the STAR Pb-cooled 400 MW(th) reactor [2] represented fast spectrum small reactors without on-site refuelling as a typical example. The reactor properties for each sector of the park are shown in Table 20.

TABLE 20. REACTOR PROPERTIES

| | **Initial fissile in-core inventory (tons fissile/GW(th)** | **TRU produced per GW(th)/year (tons transuranic)** | **Fissile fissioned per GW(th)/year (tons fissile)** | **Discharge burn-up MW(th)day/kg heavy metal** |
|---|---|---|---|---|
| LWR | 1.398 | 0.09271 | 0.34670 | 50 |
| STAR | 8.910 | 0.34675 | 0.34675 | 100 |
| FBR[t] | 3.310 | 0.59641 | 0.34675 | 100 |
| Symbol | I | FP* | FD | |

t This hypothetical FBR design has a BR = 1.72.
*The value of FP for LWRs is net production of transuranics (TRU) assuming $^{235}U$ fuelling. The value of FP for breeders and STARs is total production of TRU including that which are created, then subsequently burned in situ – assuming equivalence of TRU and $^{235}U$ in a fast reactor.

The LWR properties are typical of a ~1000 MW(e) PWR, and it is clear that it is a net consumer of fissile mass (FD > FP). The small reactor without on-site refuelling (STAR) is a small (400 MW(th)) lead cooled fast spectrum reactor with a breeding ratio of unity (FD ≡ FP). It has a very large in-core working inventory (I) and operates at a derated power density in order to attain 20 year whole-core refuelling interval. The breeder is a 1000 MW(e), metal alloy fuelled, sodium cooled breeder of very high breeding ratio (BR = 1.72) that was designed as part of the IAEA-sponsored INFCE study done in 1979.

The self-fuelled growth of the park (assuming 100% MOX fuelling for new-build of LWRs) with time-invariant mix of reactor types is modelled by the state transition equations:

$$\frac{d}{dt}\begin{Bmatrix} P_{park}(t) \\ TRU(t) \end{Bmatrix} = \begin{bmatrix} -f_t / \text{LWR life} & 1/\left(I_{park} * \text{holdup}\right) \\ FD * \left(\left(FP_{park}/FD\right) - 1\right) & -1/\text{holdup} \end{bmatrix} \begin{Bmatrix} P_{park}(t) \\ TRU(t) \end{Bmatrix} \qquad (5)$$

Where:

$$\left.\begin{aligned} FP_{park} &= f_t * FP_t + f_s * FP_s + f_b * FP_b \\ I_{park} &= f_t * I_t + f_s * I_s + f_b * I_b \\ f_t + f_s + f_b &= 1 \end{aligned}\right|,$$

holdup = 5 years (average lifetime in reprocessing); LWR life = 45 years (average lifetime before LWR decommissioning), and the *f*’s are the thermal power fractions by reactor type (LWR, STAR, and FBR, respectively). The fissile inventory self-generated and made available for new build (TRU (t)) is the annual net production ($FP_{park}$ – $FD_{park}$) * $P_{park}$ (t) minus the annual rate of withdrawal TRU/holdup. The new build deployment is the product, (TRU/holdup) * (inverse of park inventory/unit power, I). Only LWR decommissioning is modelled – fast reactors are assumed to merely transfer inventory to a replacement power plant when they reach end of life.

The upper bound on self-fuelled growth rate of the park, α, is the most positive eigenvalue of the state transition matrix. Feasible growth rates based on fissile self generation are low unless the breeder fraction in the park is high, because the excess fissile mass generated from breeding must fuel not only new breeders but new LWRs and new STARs as well. Moreover, the STAR in-core working inventory is high (to attain a 20-year refuelling interval) and this slows down the achievable growth rate.

Figure 37 shows the achievable growth rate, α, as a function of STAR fraction in a STAR/FBR park having no LWRs (and parametric in recycle turnaround time). The figure indicates that self-fuelled growth rates – even in an all-fast reactor park – are quite limited unless the park comprises a low market share for small reactors without on-site refuelling. Clearly, an exogenous source of fissile mass is required in order to attain rapid growth in a park having significant market share of small reactors without on-site refuelling. An exogenous source of fissile mass is available in the $^{235}U$ contained in the resource base of uranium ore, but the amount is constrained. Assuming 15 million tonnes of U contained in the ore resource base and 0.3% enrichment tails assay, only 65 000 tonnes of $^{235}U$ are available to supplement self-generated fissile mass production. In fact, as the reserves are consumed, uranium price will rise which will stimulate exploration and development of additional supply.

A value of 15 million tonnes has been used here as a conservative estimate based on the current Uranium Redbook estimates of 15 million tonnes U of known plus speculative reserves recoverable for <130 $/kg U. If significant growth of the world nuclear park can be established as feasible using the conservative resource base, then it will be even more feasible as additional ore is found. The price or uranium is but a small component of the cost of nuclear electricity production.

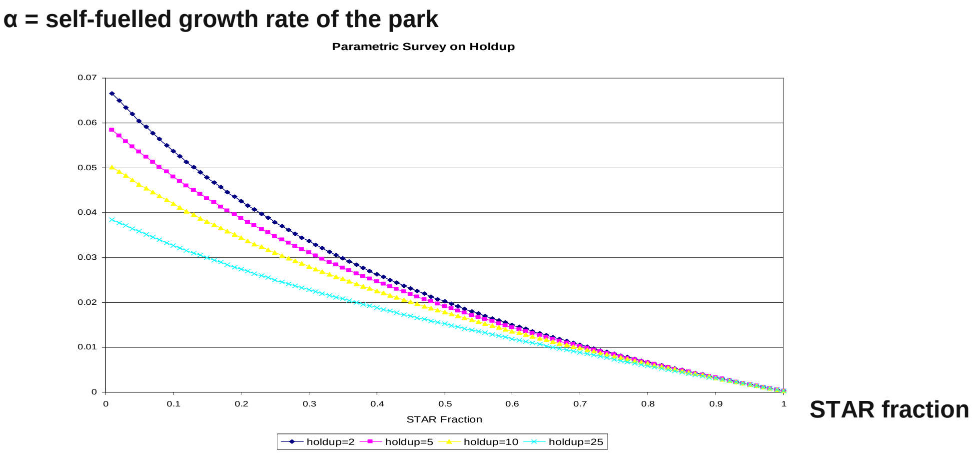


*FIG. 37. Growth rate as a function of STAR/FBR ratio, parametric in recycle holdup time.*

The architecture proposed for this study is one in which LWRs and STARs are assumed to be broadly distributed, delivering energy services throughout the world, but that the FBRs are confined to siting only at a dozen or so regional fuel cycle centres where their mission is to produce long distance energy carriers – nuclear fuel from breeding and synthetic chemical fuels from hydrogen production. The distributed (LWR+STAR) market share should be made as large as possible as compared to the centralized (FBR) market share, so the goal of the study was to maximize the market share of LWRs and STARs, while achieving high growth rates, under a fissile availability constraint.

To achieve a successful transition over a 100-year market penetration campaign, an optimal use has to be made of both constrained self-generated transuranic fissile mass and the limited (65 000 tonnes) exogenous source of fissile $^{235}U$ mass available from uranium ore. Equation 5 is modified to include the exogenous source of $^{235}U$, represented as $S_o * f(t)$, where $S_o$ is a scale factor and f(t) is the time allocation schedule for its usage. Thus, given a specified nuclear park make-up (i.e. shares of LWR, STAR, and FBR), a given initial condition of $P_o$ = 1 TW(th), and a target power, $P_1$, at the completion of a 100-year transition, the problem is to:

$$\text{minimize} \int_{o}^{100\,\text{years}} S_o f(t)\,dt \tag{6}$$

subject to the constraints:

$$\frac{d}{dt} P(t) = \alpha P(t) + S_o f(t) \tag{7}$$

This constrained optimization problem is simple enough to permit an analytic solution that can be obtained by application of the calculus of variations and Hamilton's Principle. It can be shown that the optimal transition from $P_o$ to $P_1$ over time interval ($t_o$, $t_1$) that minimizes the consumption of the $^{235}U$ resource base is given by the solution of the equation:

$$\begin{gathered} \frac{d}{dt} P(t) = \alpha P(t) + S_o \left[1 - e^{-\alpha(t_1 - t)}\right] \\ P(t_o) = P_o \quad ; \quad P(t_1) = P_1 \end{gathered} \tag{8}$$

The analytic solution of equation 8 is the sum of a homogenous component, $P_o e^{\alpha t}$, plus a convolution integral component, $\int_{t_o}^{t} e^{\alpha(t-\tau)} S_o f(\tau)\,d\tau$, such that:

$$P(t) = P_o e^{\alpha t} + S_o \left\{ \frac{1}{\alpha}\left(e^{\alpha(t-t_o)} - 1\right) - e^{\alpha(t_1 - t)}\left(t - t_o\right) \right\} \tag{9}$$

$S_o$ can be found be applying the boundary conditions and is:

$$S_o = \frac{P(t_1) - P(t_o) e^{\alpha t_1}}{\frac{1}{\alpha}\left\{ e^{\alpha(t_1 - t_o)} - 1 - \alpha\left(t_1 - t_o\right)\right\}} \tag{10}$$

and f(t) derives from the solution of the Euler equations associated with the calculus of variation requirements:

$$f(t) = 1 - e^{-\alpha(t_1 - t)} \tag{11}$$

*f(t)* prescribes the optimal allocation schedule versus time for the exogenous fissile mass injection into the park (see equation 6). It is seen that the injection starts immediately and is large at first; then it steadily decreases with time, reaching zero at the completion of the transition. Thus, as compared to exponential growth, the park growth rate is faster than exponential early in the transition and slows toward self-fuelled growth as time goes on.

The model is too idealized to be predictive. Its utility derives from easily gaining strategic insights to guide planning and higher fidelity modelling for market penetration of nuclear into the energy sector. The analytic solution has been applied to various postulated growth scenarios. For example, it was shown that for an annual growth rate of 1.2%/year, a 50% STAR/50% FBR reactor park could maintain current nuclear market share (6% of the world primary energy) to year 2109 based solely on self-production plus fissile mass made available by harvesting transuranics from the spent fuel from currently emplaced LWRs, i.e. no exogenous $^{235}U$ source for fast reactor deployment would be required.

However, by simply maintaining the current market share, nuclear would fail to alleviate the growth of greenhouse gas emissions. More aggressive growth of the nuclear park would be desirable. Assuming a world population of 10 billion people and an average energy use per capita of 4 toe per capita per year by 100 years from now, the world primary energy demand would grow from the current value of 16 TW(th) to 53 TW(th) by the year 2109. For this second example, the equations confirm that the growth of a nuclear park from 350 GW(e) (1 TW(th)) to 53 TW(th) over a 100 year transition on a park comprised of 60% FBRs + 40% STARs would be possible if fast reactors are fuelled not only with the transuranics self generated within the park, but also with $^{235}U$ from the world's ore reserves. Even though a million tonnes of fissile mass will need to be fissioned or put into reactor working inventory by the completion of the 100-year growth campaign, it was found that the 15 million tonnes of U in the known + speculative resources recoverable at the price ≤130 $/kg contained sufficient $^{235}U$ amount (at 0.003 tails assay) to enable this very aggressive transition. The solution showed that 10 TW(th) of the new build used $^{235}U$ as feedstock, while 43 TW(th) was self-fuelled by the net breeding of the park.

Figure 38 shows the optimal growth trajectory to 53 TW(th) (labelled 'optimal'). It is not exponential (labelled 'target'); rather, over most of the transition, it can be fit to a logistic transition (labelled 'logistic') – growing fast initially while the exogenous $^{235}U$ fissile mass is injected preferentially early in the transition, and slowing down to nearly self-fuelled growth late in the transient. Such an S-shaped 'logistic' pattern of growth is very common in historical market penetration trajectories and was in fact experienced in the build-up of the deployments of current LWRs, see Fig. 39.

It was found that if the trajectory is forced to be exponential at the rate $\alpha_1 = \frac{1}{100} \ell n \, {}^{P_1}\!/_{P_o}$, then the required exogenous $^{235}U$ addition for such (sub-optimal) growth increases by a factor of four, up to about 60 million tonnes or ore.

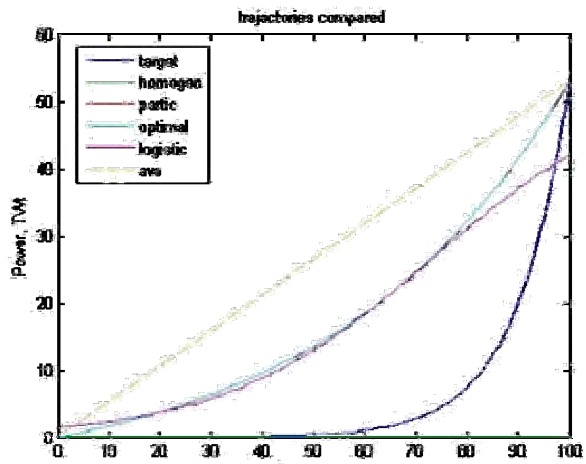


*FIG. 38. Optimal growth trajectory.*

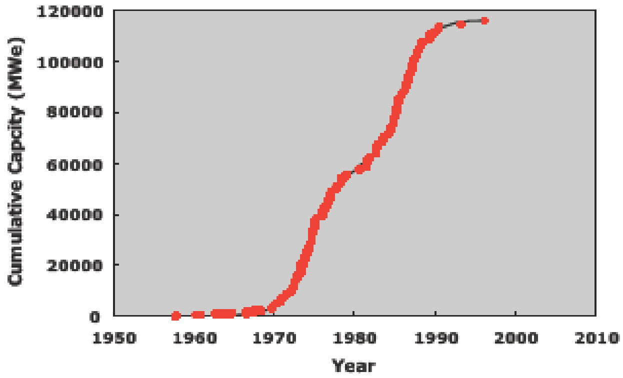


*FIG. 39. Historical data on LWR deployment [25].*

More complex modelling that allows market share of LWRs, FBRs and small reactors without on-site refuelling to evolve over time has also been performed numerically, using the calculus of variations technique. Transition results show that LWR market share that starts out at 100% is gradually replaced by breeders and fast spectrum small reactors without on-site refuelling over roughly the first 50 years of the transition. In that case the final FBR to small reactor without on-site refuelling mix is skewed more toward FBRs, unless ore reserves increase into the range of 25 million tonnes U of ore.

Based on these idealized results, it is possible to conclude that world nuclear architectures containing significant market share of small reactors without on-site refuelling and supported by regional fuel cycle centres will not be ruled out by fissile mass constraints, even if very rapid growth sufficient to attain substantial contributions to global sustainable development are required. It is very important, however, to use the ore wisely and to tailor the nuclear market shares of thermal and fast reactors carefully.

### 7.4. Deployment approach to minimize external financing in a capitalization-constrained growth of the nuclear park

The mathematical approach used in Section 7.3 to determine a fuel cycle approach with minimal uranium resource withdrawals in a fissile-constrained growth of a nuclear park can also be applied to modelling of the capitalization-constrained growth of a nuclear park under a mix of the external financing and self-financing. In fact, the analytic solution remains unchanged; all that is needed is a redefinition of the input parameters.

Consider, as in Section 7.3, a nuclear park comprised of an unchanging mix of reactor types that is growing on the basis of plowing back (reinvesting) a fraction *K* of the annual profits to finance new build. The dynamics equations would be:

$$\frac{d}{dt}\begin{Bmatrix} P(t) \\ C(t) \end{Bmatrix} = \begin{bmatrix} -\dfrac{1}{T} & \dfrac{1}{I*\text{holdup}} \\ \text{IRR}*I*K & -\dfrac{1}{\text{holdup}} \end{bmatrix} \begin{Bmatrix} P(t) \\ C(t) \end{Bmatrix}, \tag{12}$$

$$P(t = t_o) = P_o$$

where P(t) is the park power (TW(th)); C(t) is the capitalization escrow fund (Billion US$); *T* is the average asset lifetime before decommissioning (years); holdup is the construction period (years); *I* is the all-in capital cost of new build (Billion US$/TW(th)); and IRR is the financial internal rate of return of the nuclear park [(Billion US$ profit/TW(th)-year)/(Billion US$ investment cost/TW(th))].

The most positive eigenvalue, α, of the state transition matrix specifies the maximum achievable growth rate of the park under self financing derived from a specified reinvestment fraction, *K*, of the profit. If the growth rate, α1, desired by the society exceeds α, then an exogenous injection of financing, S(t), into the capitalization escrow will be required in order to attain the specified growth:

$$\frac{d}{dt}\begin{Bmatrix} P \\ C \end{Bmatrix} = [A]\begin{Bmatrix} P \\ C \end{Bmatrix} + \begin{Bmatrix} 0 \\ 1 \end{Bmatrix} S(t)$$
$$P(t_o) = P_o \tag{13}$$
$$P(t_1) = P_o\, e^{\alpha 1 (t_1 - t_o)}$$

As with the fissile-constrained growth model in Section 7.3, the minimum exogenous capitalization and its allocation pattern versus time can be determined using the calculus of variations and Hamilton's Principle – and the resulting analytic solution is the one already displayed (see equations 9, 10, and 11) for fissile constrained growth. Projections of economic parameters over periods of 100 years cannot be done accurately. The goal of this simplified analysis is to provide insights into trends and relative magnitudes. Several simple examples will be used to illustrate the model.

*Example 1: LWR park*

Suppose it is desired to grow a park of LWRs from the current deployment of 350 GW(e) or 1 TW(th) up to 5000 GW(e) or 15 TW(th) over the next 100 years. Assuming a world population of 10 billion people by 100 years from now and a world-average annual primary energy use per capita of 4 toe = 5.3 kW(th), then the world primary energy use in one hundred years will stand at 53 TW(th) and the nuclear at 15 TW(th) would supply 28.3% of the world primary energy, up from ~6% today[10]. Such a scenario would require a nuclear park growth rate of $\alpha_1 = \frac{1}{100}\ell n\left(15/1\right) = 0.0271/\text{year}$.

Suppose the all-in capitalization[11] cost is 6 Billion US$/GW(e), which is equivalent to 2000 Billion US$/TW(th), and the internal rate of return from the park is 14%. Assuming an average NPP lifetime, T = 60 years, and a construction period, holdup = 5 years, one can determine the attainable self-financed growth rate as a function of the reinvestment fraction, *K*, as shown in Table 21.

The data in Table 21 show that with unrealistically-high reinvestment fractions of the gross profit (reinvestment > 0.355), the park can more than self-finance the desired growth, but at more reasonable reinvestment fractions of 15 to 25%, external financing will be required as a supplement to self financing. As reinvestment fraction decreases, self financing is sufficient to capitalize ever smaller fractions of growth, and when reinvestment fraction is down to only 12%, then external financing will be required to finance 100% of the new build.

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[10] This would be enough for nearly 100% of the world electricity needs.

[11] The specific all-in cost (Billion US$/GW(e)) is an issue of current debate. A characteristic value of 6 Billion US$/GW(e) is used here to illustrate the methodology and to indicate the huge magnitude of capitalization that will be required to grow the world park to 5 000 GW(e) within 100 years. Using a plant efficiency of 0.33, 6 Billion US$/GW(e)~2000 Billion US$/TW(th).

TABLE 21. SELF-FINANCED GROWTH RATE VERSUS REINVESTMENT FRACTION

| Reinvestment fraction, K | Alpha | Situation |
|---|---|---|
| 0.5<br>0.4 | 0.0470<br>0.0460 | Self financing is more than sufficient for growth |
| 0.355 | 0.0269 | Self financing is just barely sufficient to support growth |
| 0.35<br>0.20<br>0.15 | 0.0266<br>0.010<br>0.0039 | Self financing is insufficient to attain desired growth rate; external investments are needed |
| 0.12 | 0.0000 | Self financing handles replacement of decommissioned units, but any growth will require 100% external funding |
| 0.10 | -0.0025 | Self financing can't even keep up with the decommissioning |

TABLE 22. SUMMARY OF FINANCING REQUIREMENTS FOR THREE SCENARIOS

| | 100% LWR park | 100% park of small reactors without on-site refuelling | 70% small reactors 30% LWR |
|---|---|---|---|
| Construction period, years | 5 | 2 | 2.9 |
| All-in cost, US$/kW(e) | 2000 | 1500 | 1650 |
| Internal rate of return on investment | 0.14 | 0.14 | 0.14 |
| Reinvestment fraction | 0.25 | 0.25 | 0.25 |
| Plant lifetime | 60 | 60 | 60 |
| Alpha | 0.0158 | 0.0172 | 0.0167 |
| TW(th) externally funded | 3.5 | 2.9 | 3.1 |
| TW(th) internally funded | 11.5 | 12.1 | 11.9 |
| Billion US$/year external investment in early years | ~115 | ~70 | ~83 |
| Trillion US$ total external investment over 100 years | 6.9 | 4.4 | 5.1 |

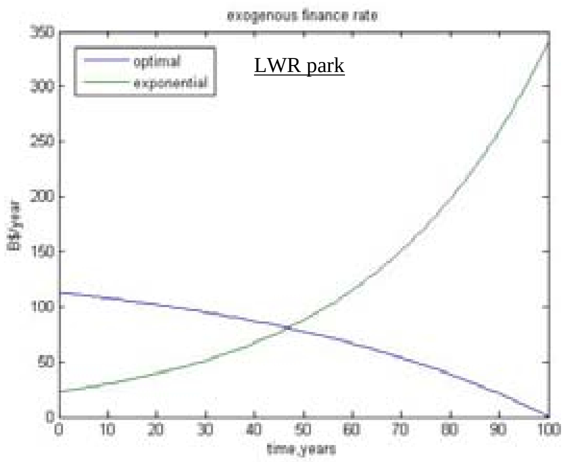


*FIG. 40. Optimal allocation of exogenous financing for all-LWR park.*

When external financing is required, the optimal strategy for its application is to preferentially inject it early in the transition so that self-financing will build up in scale early in the transition. To illustrate this, a case with reinvestment fraction K = 25% was solved using the optimal analytic solution.

Given a reinvestment fraction of 25%, the analytic solution shows that the minimum exogenous financing requirement is a total of 6.9 Trillion US$, see the first column of Table 22. It is distributed over the 100-year transition time as shown in Fig. 40. Initially, external financing injection rate is high; (~115 Billion US$/year), then it diminishes over time, showing the same optimal pattern as observed for the fissile-constrained growth addressed in Section 7.3. Also shown on Fig. 40 is the external financing required if the transition is forced to follow a (sub-optimal) exponential growth to the same final power of 5000 GW(e); it starts at a lower level, (~25 Billion US$/year) but grows rapidly and becomes unmanageably large during the last half of the transition.

The optimal trajectory for the build-up of deployment in the park is shown in Fig. 41. The growth rate is very rapid early in the transition, benefiting from front-loaded external financing, and it slows over time. The two components which comprise the optimal trajectory are also shown in Fig. 41 – the *particular solution* component is driven by external financing plus reinvestment, exclusive of the reinvestment from the initial legacy park. The *homogeneous solution* component accounts for reinvestment from the legacy (initial condition) park. It is seen that, absent the external financing, the legacy park could grow over 100 years to only less than 5 TW(th) on its own.

Compared to a constant average growth rate, the optimal trajectory starts slower and ends faster. Compared to a constant exponential growth, the optimal trajectory starts faster and ends slower. This is made more evident by re-plotting Fig. 41 on a semi-logarithmic axes, see Fig. 42.

The external investments to attain the required growth rate are astoundingly large – in the order of 115 Billion US$/year in the early years of the transition, see Fig. 40. While this is a remarkably large number, one can observe that the current deployment of 350 GW(e) ~ 1 TW(th) is already generating an annual profit of about two and a half times larger – based on the assumption of a 14% internal rate of return and all-in capital cost of 6 Billion US$/GW(e), $1\mathrm{TW_t}\frac{\mathrm{y}}{\mathrm{y}} * 0.14 * 2000\frac{\mathrm{B\$/y}}{\mathrm{TW_t}} = 280\,\mathrm{B\$/y}$.

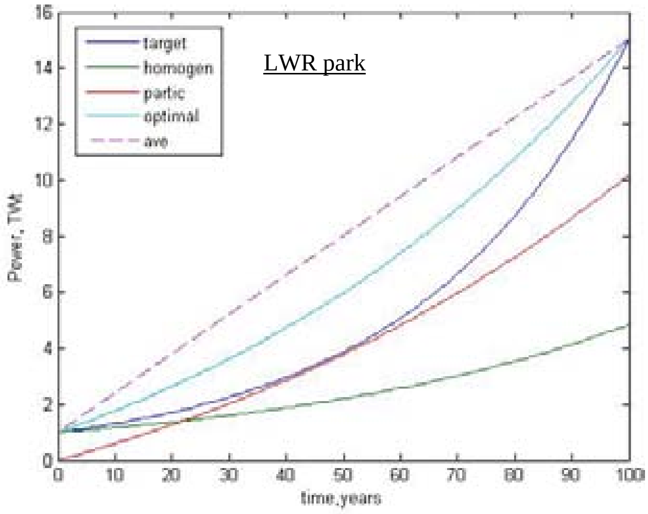


*FIG. 41. Growth of the all-LWR park and components of the optimal growth pattern – linear axes.*

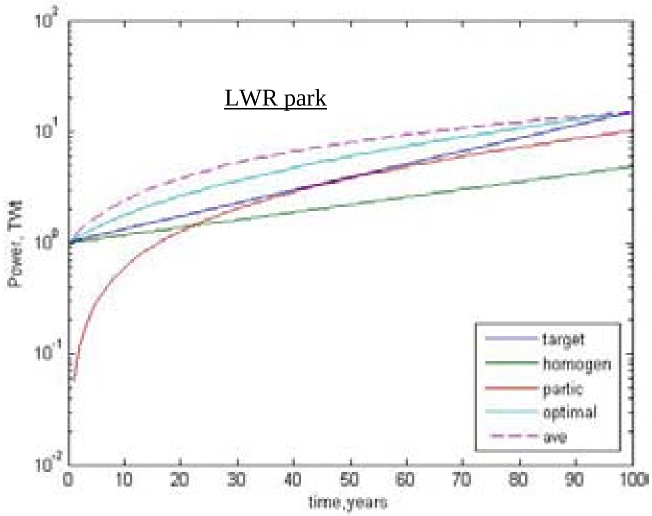


*FIG. 42. Growth of the all-LWR park and components of the optimal growth pattern – semi-logarithmic axes.*

Accounting for profit from the already existing (legacy) deployments, the net profit cash flow exceeds the influx of external financing already in year 1; in fact it exceeds the total (external + reinvestment) financing for new build already in year 1.[12]

Owners of the legacy assets might not wish to participate in self financing growth for new entrants into the market. Even if one ignores profit from the legacy 350 GW(e) park and considers financing reinvestment only from new build, then the new build net profit exceeds the external financing by year 7; it exceeds the external + internal financing by year 10, see Fig. 43.

[12] For this idealized modeling, use of the net present value financing methodology is overdoing the refinement on what is merely a scoping analysis. Straight cash flows are sufficient to illustrate the strategic insights.

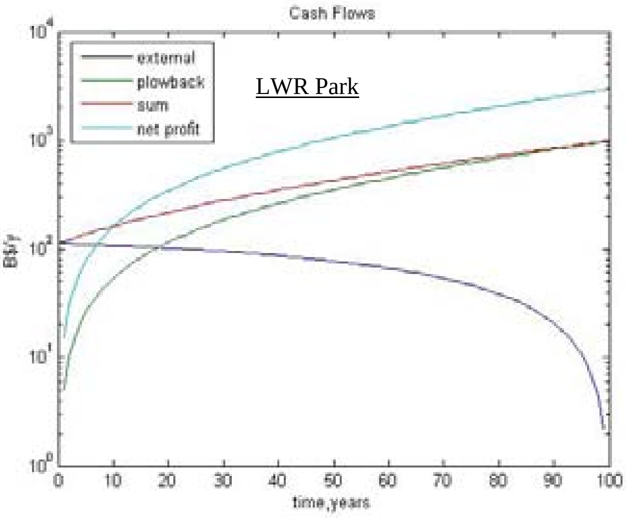


*FIG. 43. Cash flows from new build (plowback = reinvestment).*

By the end of the transition, 5000 GW(e) (15 000 GW(th)) of nuclear power is deployed and the park is generating 4.2 Trillion US$/year of gross profit annually by year 100. At that point, if further growth is not desired, the reinvestment fraction could be reduced from its 25% level to the level required for replacement only, and the net profit would increase to greater than 75% of gross profit.

External debt financing in the range of 115 Billion US$/year – even on a world wide level – is not likely to be realistic. In order to reduce the external financing requirements, the reinvestment fraction must be increased. For example, by increasing it from 25% to 30%, the initial external investments drop from the 115 Billion US$/year range to the 48 Billion US$/year range during the early years of the transition. On a world basis, this investment rate is on a par with those of annual petroleum industry investments (of several tens of Billion US$/year in the USA alone) for exploration and infrastructure emplacement.

The results for growth of a pure LWR park illustrate a dilemma facing feasibility for aggressive growth of the world nuclear park. For a goal of growth to 5000 GW(e) by century's end, at reasonable self-financing profit reinvestment fractions ($\lesssim$ 25%) the external financing requirements rise to over a hundred billion dollars/year – whereas in order to reduce external capitalization rates to a more reasonable range of tens of billion dollars per year, the reinvestment fraction of profit has to be raised to ~30%. And as illustrated by Fig. 40, if a (sub-optimal) pattern of growth is chosen to reduce the external financing demand initially, then the required external financing becomes untenable later in the transition. None of these options are attractive.

*Example 2: Small reactors without on-site refuelling*

Small reactors without onsite refuelling may help to resolve this dilemma by virtue of their short construction period. Attainable self-financed growth rates benefit from short construction periods because this allows a revenue stream to start early to fund the escrow for new build. An additional indirect benefit of short construction period is to reduce the interest during construction component of all-in deployment cost.

To illustrate the impacts of these features, assume the construction period of a small reactor without on-site refuelling of 2 years (down from 5 years assumed for an LWR). For a 100%

LWR park and IRR=0.14, the interest during construction sums to 70% of the overnight cost.[13]:

$$(\text{all in cost}) = 2000 \text{ Billion US\$/TW(th)} = (\text{overnight cost}) * (1 + 5 * 0.14)$$

To the contrary, for a 100% park of small reactors without on-site refuelling:

$$(\text{all in cost}) = (\text{overnight cost}) * (1 + 2 * 0.14)$$

Assuming overnight cost is the same[14], the park with 100% of small reactors without on-site refuelling would have an all-in cost per TW(th) that is only ¾ that of an economy of scale type of plant:

$$1500 \text{ Billion US\$/TW(th)} = 2000 * \frac{1.28}{1.70}$$

When the optimization formulas are applied to a park with 100% of small reactors without on-site refuelling with the construction time of 2 years (assuming the all-in cost = 1500 Billion US$/TW(th) and the reinvestment fraction of the profit = 25%), the external financing requirement can be dramatically reduced both in the total and in the external investment cash flows in the early years of the transition, see Table 22.

*Example 3: Mixed park of small reactors without on-site refuelling and LWRs*

The fissile constrained growth analyses displayed in the earlier Section 7.3 of this chapter showed that aggressive nuclear growth would require a fast reactor-heavy mix in the world nuclear park in light of the fissile mass constraint to growth. Nevertheless, LWRs will maintain a degree of market share, especially in the early decades of the transition. A case was run for a 70% small reactors without on-site refuelling/30% LWR park to grow to 5000 GW(e) in 100 years assuming a reinvestment fraction of 25% of the profit and an internal rate of return on investment of 14%. For this mixed park case, Fig. 44 shows the cash flows and Fig. 45 shows the growth trajectory of the mixed park power output. The last column in Table 22 tabulates the funding requirements.

*Example 4: Comparing fissile mass versus financing constraints*

In Section 7.3, the extremely aggressive growth to 53 TW(th) (more than three times larger than the cases considered here) in 100 years was shown to be possible within the constraint of 15 million tonnes of U ore and 0.3% enrichment tails. It required a park with 60% of FBR and 40% of small reactors without on-site refuelling. If the 70% STAR/30% LWR park considered here at a reinvestment fraction of 25% (of the gross profit) would attempt to grow up to 53 TW(th), such growth simply could not be financed – requiring ~400 Billion US$/year of external financing during each year of the early part of the transition. Alternately, at 35% reinvestment, the early external financing drops down to ~116 Billion US$/year. Or, as another alternative, at 25% reinvestment but a 150-year transition, the early debt financing drops down to ~125 Billion US$/year.

One can conclude that the financing constraint is a more limiting one than is the fissile constraint for extremely aggressive growth of the nuclear park.

[13] Again, net present value refinements are not used here.

[14] A major focus of this CPR and of closely associated IAEA activities has been to investigate approaches for achieving cost competitiveness of small reactors without on-site refuelling. A goal is to employ innovation and thereby to at least match the LWR capital cost. An assumption of success is employed here.

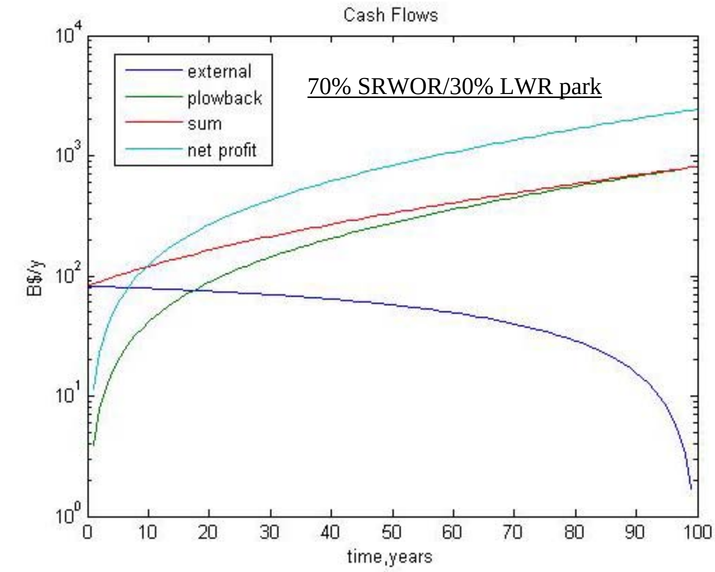


*FIG. 44. Cash flows from new build for the park with 70% of small reactors without on-site refuelling (SRWOR) and 30% of LWRs.*

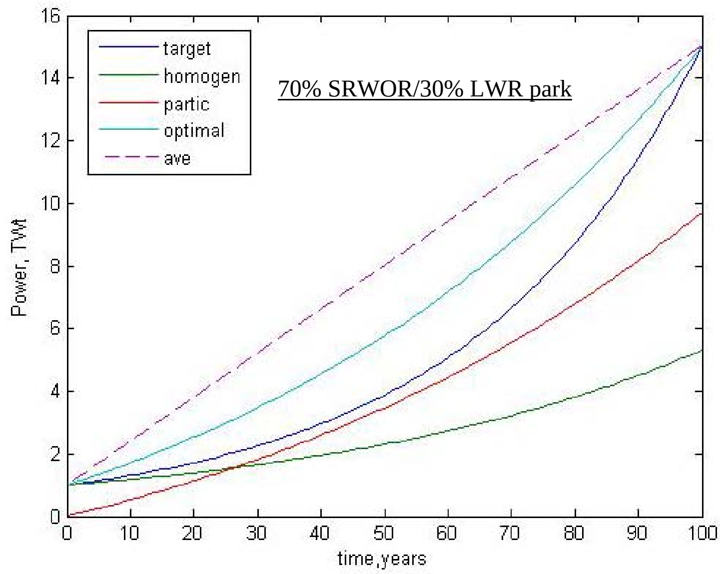


*FIG. 45. Growth of the park with 70% of small reactors without on-site refuelling (SRWOR) and 30% of LWRs, and components of the optimal growth pattern – linear axes.*

## 7.5. Sequencing of evolutionary steps toward the closing of the fuel cycle

The idealized model discussed in the previous section shows that $^{235}$U fuelling of fast spectrum small reactors without on-site refuelling will be essential early in any aggressive global nuclear growth scenario, due to fissile mass constraints. Such an approach makes sense from the financial investment and from the infrastructure timing considerations as well. A study based on the SVBR 100 design concept of a fast spectrum small reactor without on-site refuelling illustrates these practical considerations.

The design concept of the SVBR 100 [26] allows it to operate using different types of fuel and in various nuclear fuel cycles, without changing the nuclear island/ balance of plant design or deteriorating the safety characteristics [27, 28].

Once-at-a-time infrequent whole core refuelling that is adopted in the SVBR 100 makes it possible to change the fuel load characteristics in each subsequent refuelling and to use the type of fuel that is most economically effective at a given stage of nuclear power development.

Five variants of fuelling of the SVBR 100 core were considered, different in fuel types; they are as follows:

1. Uranium dioxide, $UO_2$, with an effective density of $\gamma_{eff}$ = 9.65 g/cm$^3$; hereinafter, the term *effective density* refers to a fuel composition homogenized over internal volume of the fuel element cladding;
2. Vibro-packed MOX fuel, $PuO_2$+$UO_2$, with the addition of depleted metal uranium (10% by weight); $\gamma_{eff}$ = 9.7 g/cm$^3$;
3. Another variant of MOX fuel, including minor actinides such as Np and Am; this composition is referred to as TRUOX fuel;
4. Uranium mono-nitride, UN, with the density $\gamma_{eff}$ = 12.5 g/cm$^3$;
5. A mixture of plutonium and depleted uranium mono-nitrides (PuN + UN); $\gamma_{eff}$ = 10.9 g/cm$^3$. Such fuel composition with a low effective density was selected as a result of reactivity versus burnup calculations, because it assured the smallest reactivity change during the lifetime.

The isotopic content of plutonium used in the calculations of the above mentioned variants 2 and 5 approximately corresponds to that in light water reactor (LWR) spent fuel. The total quantity of plutonium and minor actinides in variant 3 was taken in accordance with the data of [23], corresponding to LWR spent fuel after a long cooling (~15 years). The data on isotopic content of the plutonium fuel compositions are summarized in Table 23.

The power profile along core radius was, for each fuel type, shaped to flatten power distribution. The radial non-uniformity of power distribution is reduced by changing the content of fissile material in the fuel, which increases from the core centre to the periphery. The maximal radial power peaking factor $K_r^{max}$ was less than, or equal to, 1.25 in all calculations described below.

The lifetime calculations were performed for the five variants highlighted above. For variants with uranium fuel (1, 4), the lifetime duration was presumed to be 2200 effective full power days (EFPD); for variants with plutonium fuel (2, 3, 5), the lifetime duration was presumed to be 3200 EFPD. The $K_{eff}$ changes over the lifetime are shown in Fig. 46.

TABLE 23. ISOTOPIC CONTENT OF PU AND MINOR ACTINIDES IN FUEL COMPOSITIONS (ATOMIC%).

| **Isotope** | **$^{238}$Pu** | **$^{239}$Pu** | **$^{240}$Pu** | **$^{241}$Pu** | **$^{242}$Pu** | **$^{237}$Np** | **$^{241}$Am** | **$^{243}$Am** |
|---|---|---|---|---|---|---|---|---|
| Variants 2, 5 | 1 | 59 | 22 | 13 | 5 | - | - | - |
| Variant 3 | 1.6 | 51.5 | 21.4 | 6.6 | 5 | 5.5 | 7.4 | 1 |

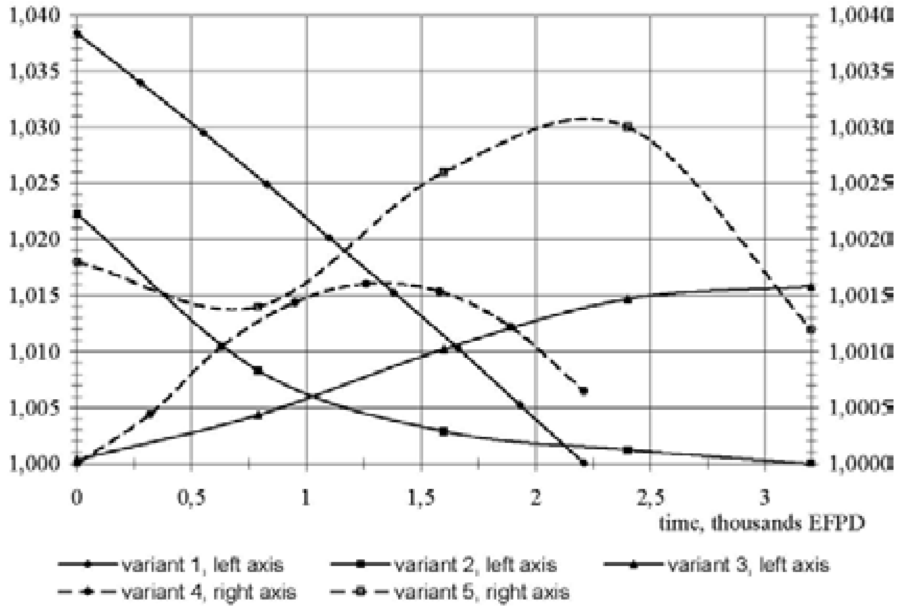


*FIG. 46. $K_{eff}$ as a function of time for different types of fuel load.*

Having established that acceptable performance can be attained for different core loadings, the question is what sequence of fuel choices would be preferable. In the nearest future, the use of enriched uranium oxide fuel operating in the open fuel cycle will be most economically effective. Changeover to the mixed uranium-plutonium fuel and to the closed fuel cycle, with core breeding ratio CBR≥1 will be economically effective after the cost of natural uranium increases. The switchover time will occur when the expenditures for constructing the factories for reprocessing of the spent nuclear fuel and re-fabrication of the fresh fuel with plutonium, as well as their operating costs, become less than the corresponding costs of natural uranium, its enrichment, the cost of manufacturing of the fresh uranium fuel, and the cost of long-term spent nuclear fuel storage.

Fast reactors operating in the open fuel cycle using uranium fuel consume much more natural uranium as compared with thermal reactors. This is because their initial working inventories and their enrichment levels are increased more in comparison to thermal reactors than their discharge burnup increases, compared to thermal reactors. For a very high pace of nuclear power development, the resources of cheap natural uranium may expire around to the middle of the current century. This will cause an increase of the uranium cost and, therefore, the period of fast reactors operating in the open nuclear fuel cycle will be naturally terminated by cost considerations. This would eventually happen even though the available resources of natural uranium will increase progressively (at increasing cost) by virtue of carrying out geological and exploration works.

Whenever the changeover from an open fast reactor cycle to a closed one takes place, it is clear that it will be cheaper if plutonium is extracted from the spent nuclear fuel of a fast reactor itself rather than from a LWR spent fuel, because the transuranic content per ton of spent nuclear fuel is ten times higher in the spent fuel of a fast reactor compared to a LWR.

When operating on oxide fuel, a comparatively high breeding ratio (BR) of the SVBR 100 (~0.84) sets the conditions for sufficiently large plutonium content in its spent nuclear fuel. This plutonium could then be used in the next fuel lifetimes while organizing the closed fuel cycle. Moreover, the spent nuclear fuel contains a substantial remaining amount of the unburned $^{235}U$ that may also be recycled for forming the next lifetime fuel load.

Such an approach to the evolution of the fuel cycle with sequential recycling of the own spent nuclear fuel of the SVBR reactors will considerably reduce the lifetime consumption of natural uranium. Operation in the closed fuel cycle prior to reaching the equilibrium refuelling mode will be realized practically without consumption of natural uranium. Comparison of natural uranium consumption by 10 SVBR 100 reactors at the proposed changeover to the closed cycle, as compared to that by one WWER-1000 reactor operating in the open cycle, is shown in Fig. 47.

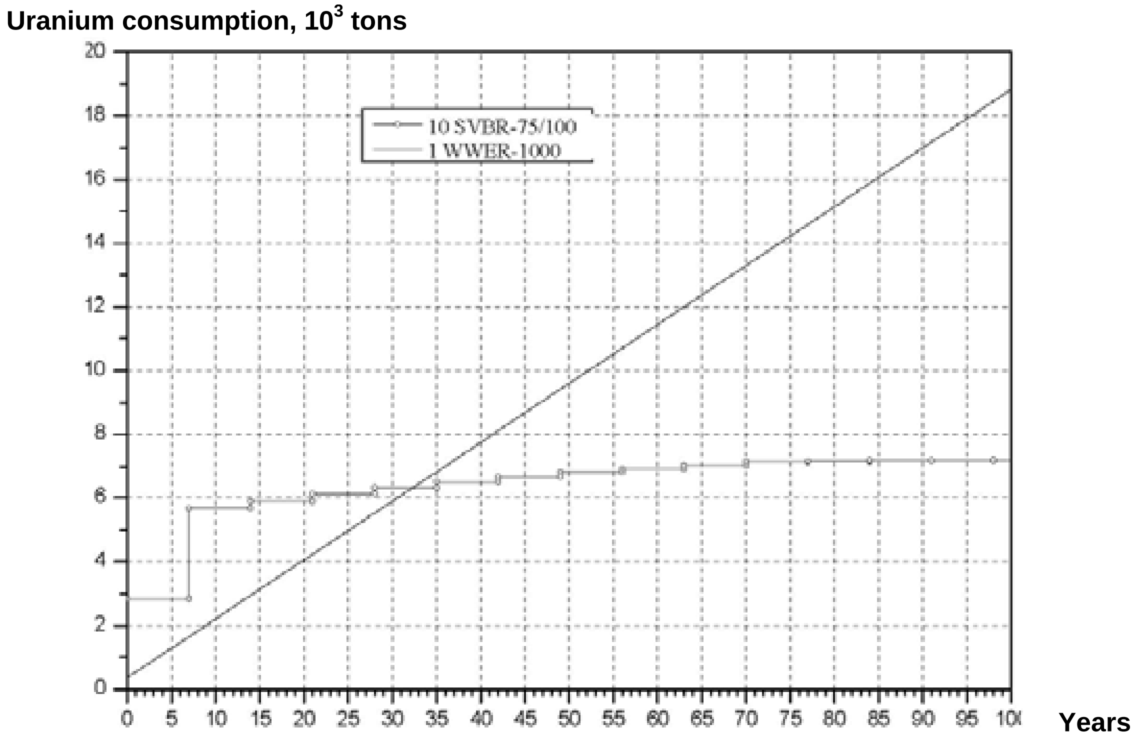


*FIG. 47. Integral consumption of natural uranium per 1 GW(e).*

Calculations have shown that a changeover to the completely self-fuelled closed cycle for the SVBR 100 is possible after the second lifetime, i.e. in 16 years. During the first 16 years, the total consumption of natural uranium calculated per 1 GW(e) will be ~5670 tons. During the 60 years of its service lifetime, the consumption of natural uranium by the SVBR 100 calculated per 1 GW(e) will be 40% lower than that by a WWER-1000 reactor delivering the same energy during the same period.

The preferable approach for the SVBR 100 is to recycle the self-produced plutonium, which will be less expensive as compared to the extraction and use of the transuranics from LWR spent fuel. As the cost is defined by the scope of spent fuel reprocessing calculated per 1 ton of plutonium, it will be up to ten times higher for LWR spent fuel reprocessing as compared to the reprocessing of the SVBR spent fuel, owing to a much smaller content of the transuranics.

In the recommended strategy, a changeover from an open to a closed fuel cycle would be postponed until after the second fuel campaign of the SVBR 100, which will spread in time the required upfront investments. Calculations have shown that the investments needed to build the fast reactor spent fuel reprocessing and refabrication facilities could be accumulated over approximately two years after recovery of the initial SVBR 100 upfront investment by retaining the amortization component in electricity cost while keeping the tariff for electricity to the customer at the same level. Thereafter, the same approach could be followed to raise the investments for continual introduction of more SVBR 100 capacities.

There is another reason to follow the approach suggested above. In the closed fuel cycle of a fast reactor and, specifically, when MOX fuel is manufactured for the SVBR 100, the spent nuclear fuel from LWRs may be blended into the feedstock directly, without separating uranium, minor actinides and fission products, e.g. by using the DUPIC technology developed for the CANDU reactors. In this case, after the gas and volatile fission products are eliminated, the LWR spent nuclear fuel can be used in place of depleted uranium when manufacturing the MOX fuel for the SVBR 100. The flowchart of this fuel cycle is shown in Fig. 48.

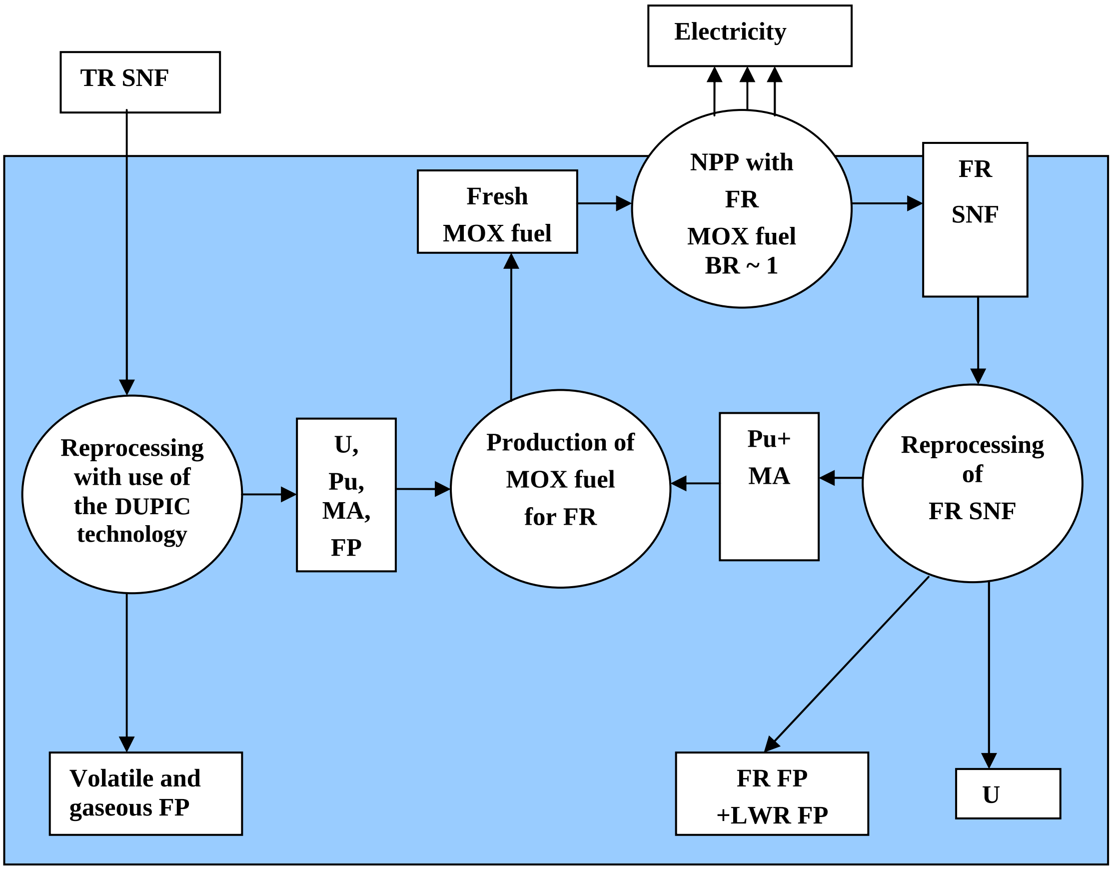


*FIG. 48. The flowchart of LWR spent fuel direct utilization in the SVBR 100 (FP- fission products, MA – minor actinides, FR – fast reactor (e.g. SVBR 100), T - thermal reactor (e.g. LWR), SNF – spent nuclear fuel).*

Adaptability of the SVBR 100 to different types of fuel and different fuel cycles makes it possible to realize a timely and gradual changeover from an open to a closed fuel cycle, when it becomes economically justified. It also helps solve a problem of LWR spent fuel utilization and secures that radioactive isotopes discharged from the fuel cycle for a final disposal will need to be monitored only in the course of several hundreds of years, taking into account that all minor actinides will be effectively burned in the SVBR-100.

The flexibility of the SVBR 100 in relation to fuel cycle technologies is realized in compliance with the principle 'to operate using the type of fuel and the fuel cycle that is most efficient at the time'. Such a philosophy makes it possible to postpone the construction of specialized closed fuel cycle facilities for several decades after the first NPP unit with the SVBR 100 modules is launched. Specifically, after the introduction of about 10 GW(e) using the SVBR 100 and repaying the NPP construction costs, a share of the profits could be spent to develop the facilities for spent fuel reprocessing and MOX fuel fabrication for fast reactors.

In reprocessing of the SVBR 100 spent nuclear fuel, it is presumed that the extracted fission products will be vitrified and after necessary cooling, being enclosed in special containers providing a multi-barrier shielding, will be transported for final disposal in a deep geological formation. Minor actinides would not be separated from plutonium and will be used in the reactor as a fuel component.

### 7.6. Conclusions to Section 7

Several studies of both near-term and of longer term nuclear park deployment approaches for parks containing significant market share of small reactors without on-site refuelling were conducted within the CRP. They addressed attainable growth under constraints on internally generated and external fissile mass availability, on internally-generated and external capital financing availability, on mix of reactor types in the nuclear park, and on timing considerations for closing the fuel cycle. Some of the studies were based on idealized models and were intended to gain strategic insights to guide future higher-fidelity modelling.

The conclusions from the deployment approach studies include the following insights:

1. Fissile mass availability should not constitute a limit on quite significant growth of a nuclear park so long as:
   - the park contains significant market share of fast breeder reactors and fast spectrum small reactors without on-site refuelling that are fissile self sufficient; and
   - $^{235}U$ fuelling of these reactors can be used to accelerate the early introduction of these reactor types.
2. On the other hand, capitalization for financing of an aggressive growth is much more confining than is fissile mass availability. Even given non-negligible reinvestment of profit (~25%) for self-financing of new deployments, massive external cash flows (~100 Billion US$/year) would be required for important but still only moderate (up to 5000 GW(e) within 100 years) growth.
3. Small reactors without on-site refuelling can be effective in mitigating this financing challenge if they offer shorter (than that of the economy of scale LWRs) on-site construction time which both hastens revenue generation and reduces interest during construction.
4. The ability of fast spectrum small reactors without on-site refuelling to accommodate fuels of various isotopic composition (and to eventually, upon repeated recycle, convert any feed into an asymptotic mix of transuranic isotopes) provides valuable flexibility for the timing of closing of the fuel cycle. In this, $^{235}U$ fuelling may be used until such time as a cost advantage accrues to closing the fast reactor fuel cycle.
5. A symbiotic fuel cycle for feeding LWR used fuel into the fast reactor closed fuel cycle need not require LWR fuel reprocessing. Instead, the LWR used fuel can be crushed and injected into fuel fabrication for fast-spectrum small reactors without on-site refuelling as is, e.g. in a DUPIC type process. Harvesting of fissile mass from fast reactor spent fuel is ~10 times more efficient than from LWR spent fuel because the used fast reactor fuels have a ~10 times higher fissile content per unit mass. So it pays to wait for recycle until fast reactor fuel recycle is required.

6. The strategy of repowering a legacy LWR nuclear power plant with small reactors without on-site refuelling at the time of WWER reactor decommissioning can be a cost effective way to grow the park overall. In the considered repowering strategy, the PWR Rankine steam cycle equipment is retained and driven by steam generated by newly deployed small reactors without on-site refuelling. For a given financing reinvestment fraction of the overall park profit, this strategy retains a larger capitalization fund for new build.

The optimization studies of capital-constrained growth presented in this Section are based on a highly idealized model and are therefore useful primarily for gaining strategic insights concerning alternative options. The illustrative examples shown are for significant growth of the world/nuclear park (to 5000 GW(e))[15] within 100 years. They indicate that exogenous capitalization flows of multiple tens of billions of US$/year would be sufficient to achieve such growth. Even more aggressive growth, although achievable within a fissile constraint, is not practical under a financing constraint unless the deployments are stretched out over a longer transition time (i.e. 150 years).

Small reactors without onsite refuelling facilitate reduction of the exogenous financing burden because of their anticipated short construction period which quickens the establishment of a revenue stream (and its associated reinvestment to finance a new build). A further and important indirect contributor to reduced financing is that interest during construction is dramatically reduced by shorter construction period – which reduces specific all-in cost of deployment of small reactors without on-site refuelling as compared with large economy of scale reactors.

## 8. CONCLUSIONS AND RECOMMENDATIONS

Small reactors without on-site refuelling are reactors of 300 MW(e) or less power rating that are designed for infrequent replacement of well-contained fuel cassette(s) in a manner that impedes clandestine diversion of nuclear fuel material. In 2009, more than 25 design concepts of such reactors were analyzed or developed in IAEA Member States representing both developed and developing countries. Small reactors without on-site refuelling are being developed for several reactor lines, including water cooled reactors, sodium cooled fast reactors, lead and lead bismuth cooled reactors, and also include some non-conventional concepts.

Most of the concepts of small reactors without on-site refuelling reactors are at early design stages. To make such reactors viable, further research and development (R&D) is necessary to validate long-life core operation, define and validate new robust types of fuel and examine possible niches that such reactors could fill in future energy systems.

To further R&D in areas mentioned above and to achieve progress in design and technology development for small reactors without on-site refuelling IAEA has conducted a coordinated research project (CRP) entitled ‘Small Reactors without On-site Refuelling’ (CRPi25001). The project has been started late in 2004 and, after a review in 2008, was extended for one more year to be ended in 2009.

[15] This would correspond to ~15 TW(th). The current world primary energy demand is 16 TW(th) but can be expected to quadruple within 100 years. At 15 000 GW(e), the nuclear was assumed to have grown from 6% to ~25% of the world primary energy supply.

The project has created a network of 18 research institutions from 10 Member States, representing both developed and developing countries. The participating research institutions were Eletronuclear and Federal University of Rio Grande do Sul (Brazil), Bhabha Atomic Research Centre (India), Bandung Institute of Technology (Indonesia), Politecnico di Milano (Italy), Hokkaido University and Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology (Japan), Lithuanian Energy Institute (Lithuania), Mohammed V University (Morocco), All Russian Institute of Atomic Machine Building, Russian Research Centre 'Kurchatov Institute', EDO 'Gidropress' and Institute of Physics and Power Engineering (Russian Federation), Argonne National Laboratory, Pacific Northwest National Laboratory and Westinghouse Electric Company (USA). At certain stages of the project, also participating were the Institute of Nuclear Science and Technology (Vietnam) and the Faculty of Electrical Engineering of the University of Zagreb (Croatia).

The overall objective of the CRP was to increase capability in Member States to develop and deploy small reactors without on-site refuelling.

The specific objectives were:

(1) To develop a vision statement for small reactors without on-site refuelling

(2) To develop a methodology to revise the need of evacuation and relocation measures beyond the plant boundary unique to NPPs with innovative SMRs and advanced reactors of larger capacity;

(4) To review the approaches to ensure long-life core operation without refuelling and to perform a comprehensive coordinated study of long-life cores for small reactors of various types with a focus on neutronics, thermal-hydraulics and new robust types of fuel;

(5) To identity possible niches and applications for small reactors without on-site refuelling and to outline pathways for commercialization of plants with such reactors.

The CRP established collaboration with OECD-NEA in benchmark thermal-hydraulic calculations of forced and natural convection modes of lead-bismuth simulating the tests performed in the HELIOS loop at the Seoul National University of the Republic of Korea. Several participants of the CRP contributed to these activities and continued their involvement after the completion of the CRP. It was agreed that the report on this exercise will be produced by OECD-NEA.

The CRP was effective in reaching the identified specific objectives related to vision of, and requirements to, small reactors without on-site refuelling, understanding of the variety of issues related to long-life core operation, and re-definition of emergency planning zone (EPZ) radius.

Over the CRP period, collaborative results were achieved for many of the abovementioned research areas. The project outputs are documented in this report to foster further R&D and increase the capability in Member States to achieve progress in development and deployment of small reactors without on-site refuelling.

At the final research coordination meeting held in Vienna on 3-6 November 2009, the CRP participants have reviewed the overall results achieved within the project and came up with the following suggestions for further work:

(1) Regarding the developed methodology for EPZ radius redefinition:

- efforts need to be continued to include external events and reasonable combination of the external and internal events in Step 1 of the methodology - accident sequence re-categorization, within the framework of the already developed approach;
- further progress could be achieved via discussion of this methodology with national regulatory authorities in those Member States that are considering performance-based and risk-informed licensing approaches for future NPPs;

(2) It may be interesting to continue studies on attainable nuclear park growth when constrained by capital financing availability – both self generated and coming from an external source.

(3) Participation in the HELIOS thermal-hydraulics benchmark exercise would be continued beyond the CRP to calculate the natural circulation case – because natural circulation is especially relevant to the transient performance in a loss of flow without scram events in heavy liquid metal cooled reactors.

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