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FRAMEWORK FOR ASSESSING DYNAMIC NUCLEAR ENERGY SYSTEMS FOR SUSTAINABILITY: FINAL REPORT OF THE INPRO COLLABORATIVE PROJECT GAINS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA NUCLEAR ENERGY SERIES No. NP-T-1.14

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FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property". The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

As an integral part of Phase 2 of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), several collaborative projects (CPs) were established by INPRO members. The CP, 'Global Architecture of Innovative Nuclear Energy Systems Based on Thermal and Fast Reactors Including a Closed Fuel Cycle (GAINS)', was one of them. This CP was jointly implemented in 2008–2011 by Belgium, Canada, China, the Czech Republic, France, India, Italy, Japan, the Republic of Korea, the Russian Federation, Slovakia, Ukraine, the United States of America and the European Commission, with Argentina as an observer. The objective of the CP was to develop a standard framework — including a methodological platform, assumptions and boundary conditions — for assessing future nuclear energy systems (NESs), taking into account sustainable development, and to validate the simulation results through sample analyses.

In the first stage of the project's implementation (2008–2009), nuclear energy needs during the twenty-first century were estimated, basic scenarios for the study were defined, essential data on current and future reactor systems were compiled, and a heterogeneous multigroup model of a global NES was developed. In the second stage (2010–2011), the results of calculations performed in the first stage using national and IAEA tools were cross-checked, and the sustainability of sample global nuclear energy architectures differing by the level of technical and institutional innovations were analysed, compared and assessed in the light of the INPRO methodology.

Interim results of the study were submitted to INPRO Steering Committee meetings held in Vienna in the course of 2008–2011 and at several international conferences and meetings. The overall results and findings of the project are summed up in this report and supporting material is included on the attached CD-ROM.

The IAEA officers responsible for this publication were V. Usanov and H. Hayashi of the Division of Nuclear Power.

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SUMMARY

BACKGROUND

Nuclear energy has the potential to be an important sustainable energy source for many countries and for the world as a whole. A methodology for assessing capabilities of innovative nuclear energy systems (NESs) to meet sustainability requirements at the national level was developed in the first phase of the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). A significant task of the current (second) phase is to find ways for optimum introduction of innovative NESs into national energy systems, taking into account regional and global trends of NES development that are connected with the fact that nuclear energy has a global character (safety, nuclear material resources, non-proliferation), and also taking into account that multilateral solutions of nuclear energy problems in the areas of non-proliferation and spent nuclear fuel (SNF) management may be attractive from an economic viewpoint.

In this context, INPRO's Programme Area B, Global Vision, Scenarios and Pathways to Sustainable Nuclear Development, is aimed at developing a global vision on nuclear energy sustainability in the twenty-first century. Several prior studies in this area have been performed by different countries and international organizations. The primary limitation associated with the results of these studies is that NES development is considered in the context of a homogeneous world model without taking into account the essential differences between various countries' approaches to NES development as a whole and to innovative nuclear technologies in particular. Analyses using the homogeneous model lead to identical solutions for all countries and preclude identification of possible synergies due to different NES development strategies.

Therefore, a new model of global NES development is necessary, which can take into account significant differences in the NES development strategies of various countries and cooperation between them. A collaborative project (CP) entitled Global Architecture of Innovative Nuclear Energy Systems Based on Thermal and Fast Reactors Including a Closed Fuel Cycle (GAINS) was launched to develop this NES model at the global level and to validate the results of its application through sample analyses. Sample base cases conducted to illustrate its application assumed an innovative NES using thermal and fast reactors (FRs) including closed fuel cycles. However, other innovative NES technologies were also considered as part of the project.

Fourteen countries and the European Commission joined the project, which has witnessed growing interest and an understanding of the necessity of international cooperation for the creation of safe and sustainable NE.

GAINS OUTCOMES

The primary outcomes of the GAINS project are: (i) development of a standard analytical framework (methodological platform, assumptions and boundary conditions) for assessing global NES architectures; and (ii) results from sample analyses for scenarios involving transition from the present NES architecture to future architectures including innovative NESs.

DEVELOPMENT OF AN ANALYTICAL FRAMEWORK FOR ASSESSING NUCLEAR ENERGY SYSTEMS REGARDING SUSTAINABLE DEVELOPMENT AS PART OF GLOBAL ARCHITECTURE

As a result of efforts carried out within the GAINS project, a unique analytical framework has been developed, including a heterogeneous world model to include consideration of specific NES development strategies which countries may pursue. This model simulates important realities of global NESs and allows identification and assessment of areas for potentially beneficial cooperation between country groups and exploration of multilateral approaches to solve NES development issues and achieve long term sustainability.

The analytical framework includes:

(a) In addition to a homogeneous model, an agreed-upon model of the heterogeneous world comprising countries of different nuclear strategy groups (NGs) based on the SNF management strategy being pursued for the back end of the fuel cycle at the end of the simulation period. For example, in the sample base case analysis, three NGs are modelled: NG1 recycles SNF; NG2 directly disposes of SNF; and NG3 sends SNF to NG1 or NG2. The methodology applied in the analysis does not assign individual countries to groups, but allocates a fraction of future global nuclear energy generation to each group as a function of time to explore 'what if' scenarios. The number of groups and the composition of each group can be altered to meet a particular analysis need.
(b) The ability to model three story lines for global NES development: a convergent homogeneous world without any differences in NES development strategies; a heterogeneous separate world based on self-reliance and preservation of local identities; and a heterogeneous synergistic world with rapid changes towards global solutions. The synergistic model is a key model for the study.
(c) A method for assessment of sustainability of an NES in dynamic transition from the current global architecture to a future state of innovation on the basis of ten key indicators (KIs) plus some related evaluation parameters (EPs). This simplified set of KIs and EPs were identified for assessments of global system architectures after considering more than a hundred indicators comprising all areas of evaluation of the INPRO methodology.
(d) A database for existing and conceptual reactor technologies and related nuclear fuel cycle (NFC) technologies that extends the existing IAEA's database and takes into account preferences of different countries.
(e) Two long term NES demand scenarios based on IAEA Member States' high and low estimations on nuclear power deployment until 2030, expected trends until 2050 and on forecasts of competent energy organizations. These scenarios provide a valuable contribution to modelling global nuclear futures and can serve as reference growth scenarios in preparation of analyses for the global system.

SAMPLE ANALYSIS RESULTS FOR SELECTED SCENARIOS

Sample analysis of base case NES scenarios using the GAINS framework have shown that a synergistic approach based on technological and institutional innovations could provide a significant potential for win–win collaboration between all NGs. In addition to base case scenarios, sample analyses applying the homogeneous world model and growth profile conditions to other innovative NES scenarios were also completed, and cross-checks showed similar trends for the most part between analytical tools used in the project.

Four types of NES architecture were defined and then analysed to estimate the effect of implementation of innovative technologies and their influence on the KIs:

(a) A homogeneous 'business-as-usual' (BAU) scenario based on pressurized water reactors (PWRs) (94% of power generation) and heavy water reactors (HWRs) (6%) operated in a once-through fuel cycle in which the world was modelled as a single NG. A variant of this scenario included the introduction of an advanced PWR replacing conventional PWR technology (termed the 'BAU+' scenario).
(b) Homogeneous (single group) scenarios for a closed cycle using thermal and FRs for comparison with the above scenarios. Some of these fuel recycle scenarios included HWRs (6%) operated in a once-through mode.
(c) A hybrid heterogeneous architecture scenario comprising a once-through fuel cycle strategy in NG2, a closed fuel cycle strategy in NG1 and use of thermal reactors in a once-through mode in NG3. Both synergistic and non-synergistic cases were analysed for this scenario. In the synergistic case, NG3 receives fresh fuel from NG2 and NG1 and returns associated SNF to those groups.
(d) Other innovative NES scenarios using the homogeneous world model including:
   (i) Construction of fast-spectrum reactors or thermal-spectrum HWRs using the thorium fuel cycle for reduction of natural uranium requirements;
   (ii) Reduction of minor actinides (MAs) using accelerator driven systems (ADSs) or molten salt reactors (MSRs) and other innovative NES scenarios.

The main results from these sample scenarios are as follows:

(a) There are significant differences in the values of KIs assessed with the use of the homogeneous and heterogeneous approaches to simulation of the global NES.

(b) The synergistic approach based on multilateral arrangement of the global NES architecture indicates opportunities to gain potential benefits in terms of resource utilization, SNF management, non-proliferation and economics:

  (i) *Resource utilization*: The synergistic approach can result in more effective utilization of resources in the global system by making fuel resources available for use which otherwise might not be used, and by additional energy extraction from recycling SNF (also referred to as used fuel or discharged fuel) shipped to NG1. SNF, a waste to NG3, could serve as a resource to NG1 by allowing possible fissile material deficits to be addressed during a transition to large scale nuclear energy generation, perhaps avoiding a need to develop fast-spectrum reactors with high breeding ratios (BRs).

  (ii) *SNF management*: The synergistic approach could facilitate a solution to the problem of accumulating SNF inventories and associated waste disposal in the world, in particular for the countries in NG3. Within the developed model of group interaction applied to sample scenarios, NG3 returns SNF to NG1 and NG2, averting the need to develop its own nuclear waste management infrastructure. NG1 would recycle the returned SNF for use in reactors, along with other SNF generated within NG1. A high level waste facility would still be required, however, for disposal of fission products (FPs).

  (iii) *Non-proliferation and plutonium inventories*: There are potential non-proliferation benefits to a synergistic approach associated with reducing both inventories of direct use material and sites using sensitive nuclear technology. Since plutonium is a long lived hazardous radioactive element and, at the same time, it is direct use material which can be suitable for manufacturing a nuclear explosive device, managing the plutonium inventory is very important both for nuclear waste management and for the reduction of potential proliferation risks. The heterogeneous synergistic model indicates that innovative nuclear technology based on the closed NFC with fast-spectrum reactors is a technical instrument for managing the plutonium balance in a global system, with reduced plutonium inventories achieved compared with the non-synergistic case. Additionally, the siting of facilities using sensitive nuclear technologies such as enrichment and reprocessing can be reduced via a multilateral approach. However, international transport of fresh fuel and SNF would be required to achieve these benefits and the associated costs and risks also need to be taken into consideration. Proliferation risk and physical protection assessment results are expected to depend on the particulars of a given scenario and more detailed assessments beyond the scope of GAINS can be used to identify the benefits and drawbacks of different approaches.

  (iv) *Economics and investment risks*: The synergistic or multilateral arrangement of the global NFC could help nuclear power producers in NG3 to significantly reduce their investments in research, development and demonstration (RD&D) and associated economic risks. Fuel cycle service providers could expand their markets and, thus, provide an early return of high RD&D costs. Both can benefit from the economies of scale and from extending conventional uranium resources. Multilateral approaches could also assist in reducing IAEA safeguards costs.

(c) Other innovative NES technologies, such as thorium NFC and MA reduction technologies, can provide potential benefits in terms of further extending nuclear fuel resources and facilitating waste management.

(d) Cross-check studies using IAEA tools and national tools are an essential step in harmonizing the analytical tools of IAEA Member States in support of decision making related to long term nuclear strategy and energy planning. Sample analyses for a set of homogeneous cases were executed and compared, and, for the most part, exhibited similar trends. Variations in the above heterogeneous scenario were examined with two different codes.

FUTURE APPLICATION AND ENHANCEMENT TO FRAMEWORK

The GAINS project has developed a framework for analysing global NES architecture and has shown — through sample analyses — that sustainability is more difficult to achieve on a global scale without collaboration.

The study highlighted some benefits that could be achieved through collaboration between countries, and also identified some challenges. Individual Member States can use this framework and customize it to evaluate particular NES approaches in a global context to assist them with long term planning. A follow-on CP could also help identify and evaluate in more detail mutually beneficial collaborative scenarios and propose pathways for achieving sustainable NESs by applying and enhancing the analytical framework developed in GAINS.

Further development of the framework could be focused on specific segments of a global architecture such as regional nuclear fuel centres under the auspices of the IAEA and take into account local benefits and challenges, such as fuel transport, minimization of proliferation risks, possible limitations from nuclear and non-nuclear industry, acceptable cost of nuclear services, desirable links to global infrastructure and some institutional barriers.

The analytical and methodological framework developed in GAINS could be extended and improved by the following efforts performed by individual Member States or through joint effort:

(a) Extend the reference database on reactors and their NFC to what is needed in support of strategic decision making for specific innovative NES scenarios, including additional detail on nuclear waste characteristics, safety and cost information.
(b) Further develop the KIs for global vision studies, such as the CP GAINS, for assessing economics, proliferation resistance, safety and other areas as needed for a specific NES of interest.
(c) Improve the IAEA computer tools for more precise evaluation of advanced fuel cycle system behaviour. Specific suggestions are provided in Section 10. Improve national/IAEA code capabilities for analysing interactions between multiple NGs and providing output information consistent with KIs and EPs. Cross-checks for more complex heterogeneous scenarios using different analytical codes are also recommended.
(d) Extend the results and graphics template developed in the project to address customization of KIs or EPs. This may entail a wider scope covering more dimensions of sustainability defined in the INPRO methodology. Enhance the template for heterogeneous scenarios through inclusion of side-by-side comparison charts for individual groups and the global system.

The IAEA/INPRO GAINS project has provided a foundational framework for analysis of global NES architecture which can be applied, customized and enhanced to support national and international collaborative assessments of NES technologies and scenarios. The diversity in approaches and perspectives offered by Member States participating in the project proved to be valuable in constructing a framework which is flexible and can be used to analyse a wide range of possible global nuclear energy futures.

1. INTRODUCTION

1.1. BACKGROUND

In 2001, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was initiated on the basis of a resolution of the IAEA's 44th General Conference in 2000. INPRO covers a broad range of missions and activities, including investigation and formulation of visions for opportunities and challenges of nuclear energy in the twenty-first century [1.1].

In order to perform a feasibility study for NFC installations, quantification of the key aspects characterizing their development and deployment is essential, including estimations of technical parameters, economic performance, and infrastructural and institutional arrangements. Studies of this kind have been implemented in many countries and by international groups. However, the need for more efforts on harmonization of national views on possible decisions of technical, institutional and political issues being raised by the transition to the global NESs with enhanced sustainability features is increasingly being recognized.

In INPRO, the implementation of CPs has been an important opportunity for effective multinational cooperation of IAEA Member States on areas of common interest contributing to establishing a global vision, scenarios and pathways for sustainable nuclear development [1.2]. This report sums up the results and findings of the study implemented by the participants of the CP Global Architecture[1] of Innovative Nuclear Energy Systems Based on Thermal and Fast Reactors Including a Closed Fuel Cycle (GAINS). The CP, established in 2008, deals with the development of an analytical framework to support scenario simulations of a future NES and assessment of how nuclear technology meets requirements of sustainability.

INPRO members recognize the importance of joint work on key issues of innovative NES deployment. In the first phase of the INPRO project (2005–2007), Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation and Ukraine had already made a common consideration (INPRO Joint Study [1.3]) of the closed nuclear fuel cycle (CNFC) with fast reactors (CNFC–FR) through the application of the INPRO methodology. The focus of this study was to examine FRs and associated fuel cycle technologies in equilibrium with thermal reactor systems. The study judged that the expected medium term CNFC–FR characteristics, to be achieved through R&D activities and construction in series, were in accord with the criteria of sustainability indicated by the INPRO methodology. A scaled use of the innovative nuclear energy system (INS) of this type was found to be critical for increasing sustainability features of nuclear power in the future and was, therefore, confirmed as an important target for technology development in some countries of the world.

However, existing components of NESs, almost entirely based on thermal reactors operating in an open fuel cycle, will continue to be the main contributor to nuclear energy production, at least for several more decades. The participants of the CNFC–FR agreed that research was needed to extend the frame of the CNFC–FR Joint Study and explore the transition scenarios multi-nationally from the current global NES to a future one which would have the synergy of thermal and FRs operating in a closed NFC (CNFC). The resulting CP GAINS was initially promoted by the Russian Federation, and at the 12th INPRO Steering Committee meeting held on 3–5 December 2007, interest in implementation of the project was expressed by Belgium, Canada, China, the Czech Republic, France, India, Italy (as an observer initially), Japan, the Republic of Korea, the Russian Federation, Slovakia, Spain, Ukraine, the United States of America (USA) and the European Commission. Argentina and Italy were observers for part of the project's duration[2]. The structure and scope of the CP were discussed at the GAINS kick-off meeting which was held on 3–4 October 2007 in Vienna. The terms of reference (ToR) containing the Implementation Plan of the CP was signed by the INPRO Steering Committee representatives on 10 July 2008.

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[1] 'Global architecture' refers to the configuration of a global NES which may comprise different types of reactor along with the corresponding fuel cycle installations located in different groups of countries and the dynamics/interactions between them, which can support common sustainability objectives.

[2] In 2009, Argentina became an observer and Italy joined GAINS.

1.2. OBJECTIVES AND PURPOSE

The overall objective of GAINS is to develop a standard framework — including a methodological platform, assumptions and boundary conditions — for assessing future NESs, taking into account sustainable development, and to validate the results through sample analyses.

This includes:

- Estimation, as a basis for modelling work, of nuclear energy demand during the twenty-first century globally and for groups of countries with similar fuel cycle strategies (NGs);
- Development of models of a global NES;
- Identification of a representative set of NESs (reactors and fuel cycles) characteristic of the systems necessary for implementation of the study, and scenarios of their deployment during the century;
- Determination of the most relevant indicators to assess the sustainability of identified nuclear energy scenarios (the INPRO methodology [1.4] was taken into account when considering these KIs);
- Simulation of the scenarios developed with the selected tools and evaluation of the results obtained from modelling based on relevant agreed-upon indicators;
- Evaluation capability of available analytical and modelling tools for modelling NESs and identification of areas to improve such analytical tools.

The set of reactor and fuel types to be taken into account and associated technology deployment time frames are presented in Fig. 1.1 and Table 1.1. Future technologies were assumed to be introduced in 2030, 2050 or 2075 when developing this table and figure, but not in the years between.

Table 1.1 includes NES innovations expected to have a major impact on the indicators selected for GAINS.

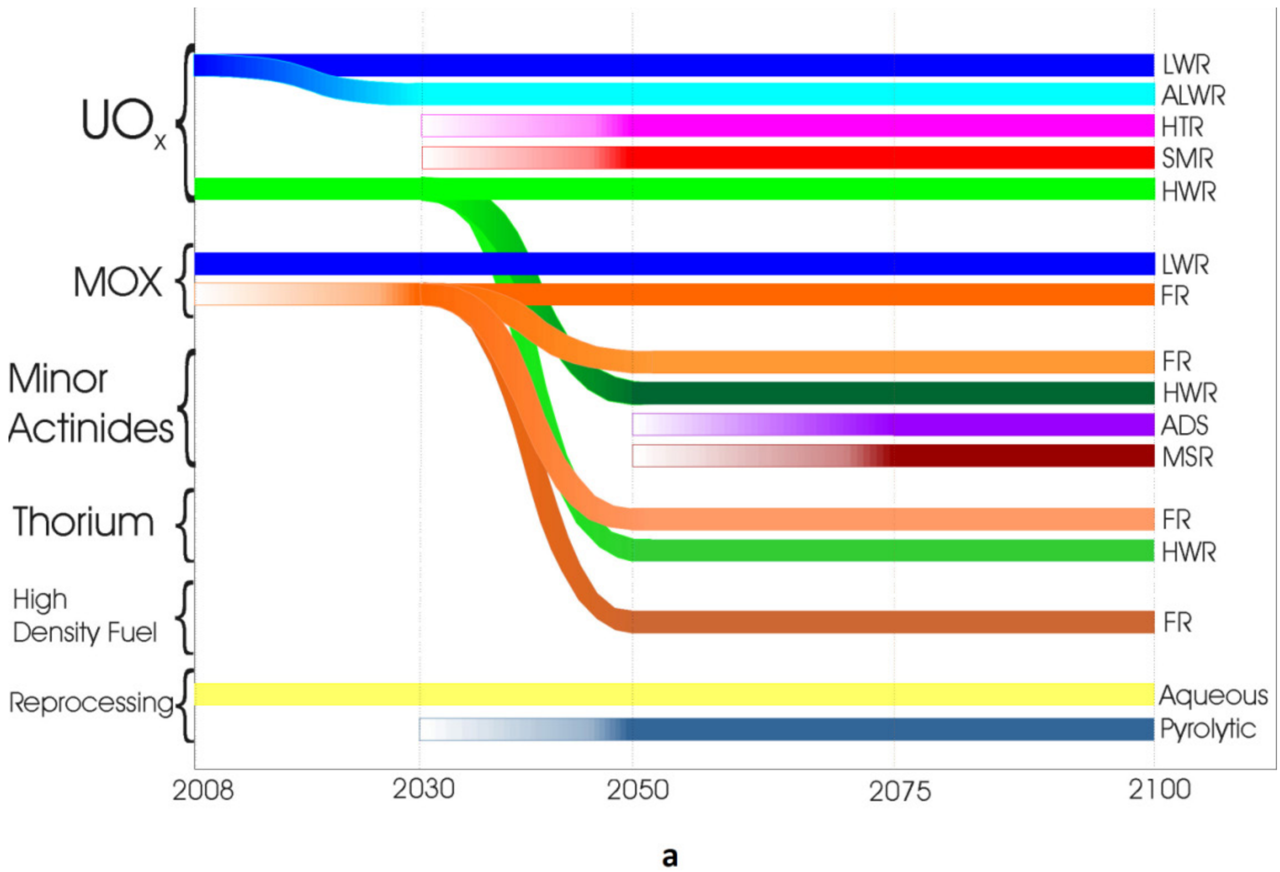

FIG. 1.1. Preliminary set of reactor and fuel types and deployment time frames.

TABLE 1.1. ELEMENTS AND TRANSITION POINTS OF THE TECHNOLOGICAL EVOLUTION

	2008	2030	2050	2075	2100
Uranium oxide (UOX)	LWR HWR	LWR HWR (HTR) (SMR)	LWR HWR HTR SMR	LWR HWR HTR SMR	LWR HWR HTR SMR
Reprocessing	Aqueous	Aqueous (Pyro)	Aqueous Pyro	Aqueous Pyro	Aqueous Pyro
Mixed oxide (MOX)	LWR (FR)	LWR FR	LWR FR	LWR FR	LWR FR
High density fuel		(FR)	FR	FR	FR
Minor actinides		(FR) (HWR)	FR HWR (ADS) (MSR)	FR HWR ADS MSR	FR HWR ADS MSR
Th fuel cycle		(HWR)	HWR (FR)	— FR	— FR

Note: Components written in brackets, e.g. (FR), indicate limited availability (only in 2 or 3 countries) at commercial level at that time. LWR: light water reactor; HWR: heavy water reactor; HTR: high temperature reactor; SMR: small and medium sized reactor; FR: fast reactor; ADS: accelerator-driven system.

The most relevant parameters and corresponding values for the different reactor types were agreed on by the participants on the basis of the available experience. By now, several comprehensive studies with objectives similar to GAINS have been performed. Some of them are briefly overviewed in this report (Section 2). Results of the studies were taken into account in GAINS to identify available inputs and avoid unnecessary duplication.

GAINS has several specific features that may provide an added value to the area of NFC and global nuclear architecture studies. These are listed below:

(1) Representative participation and joint contribution of IAEA Member States

The INPRO status does not limit participation of any IAEA Member State in its activities, including CPs. Different countries have different abilities and views on the arrangement of the national nuclear power providing a basis for complementation and harmonization. Broad multinational consensus and action are essential conditions for transition to sustainable development, since global sustainability cannot be achieved locally, by one country or region. The joint work in GAINS accumulates contributions from countries in different parts of the world with a fairly representative extract of the current nuclear options, and opens an opportunity to discuss views on the prospects of nuclear power and to consolidate approaches to increase its contribution to global energy supply.

(2) IAEA auspices and expertise

GAINS, as other INPRO projects, is being implemented under IAEA auspices and, thus, benefits from cooperation with the regular IAEA units. From its inception, the project garnered: expertise from the Divisions of Nuclear Power and Nuclear Fuel Cycle and Waste Technology on the recent technological and institutional developments in the areas of interest; advice from the Planning and Economic Studies Section on the selection of global, regional and national nuclear energy scenarios and some aspects of economic analysis; judgement from the Department of Safeguards on specific issues of the NES proliferation resistance; and competent assistance from structural units of the IAEA on many other issues.

(3) Application of INPRO methodology and cross-checked analytical tools

The assessment of the different NES options in GAINS was performed using the INPRO methodology [1.4], which covers safety, economics, environment, infrastructure, waste management, proliferation resistance and physical protection, and gives the criteria for nuclear power sustainability in each of these areas. The scope of GAINS does not cover all of these areas but focuses on the role of the NFC infrastructure (architecture) in the rational use of fissile material, improvement of waste storage and disposal management, and the economics and proliferation resistance of NES. Relevant analytical tools developed at the IAEA and in some countries participating in the project were cross-checked and provided to interested GAINS participants as an output of the project.

(4) Heterogeneous multigroup model of global nuclear system

Most studies on the future of global nuclear energy from France, Japan, the Russian Federation, the USA and other countries as well as from international organizations are based on a homogeneous model of the NES worldwide. This approach does not take into account national preferences and possibilities. Assuming simultaneous introduction of nuclear innovations all over the world, the homogeneous model gives a very idealistic, sometimes misleading, picture of the nuclear future. GAINS developed a heterogeneous model comprising groups of countries with differing fuel cycle strategies, extending earlier approaches [1.5, 1.6] to allow for global NES analysis. This model is capable of providing more realistic analysis of transition scenarios to a global innovative NES architecture by the end of the century. It can also be used to illustrate the global benefits of introducing innovative nuclear technologies without exposing the majority of countries to the financial risks and other burdens associated with the development of these technologies.

Eventually, all activities under INPRO are intended to provide practical decision making support to IAEA Member States. Insights from scenario studies such as GAINS may facilitate practical steps to developing specific multilateral and regional options and mechanisms for mutually beneficial collaboration.

1.3. PARTICIPANTS AND ORGANIZATION

Member States affiliated with GAINS together have more than half of the world’s population and include some large users of energy (Fig. 1.2).

The CP was implemented by the participants as a ‘joint initiative’ according to the framework described in the framework document, issued by the IAEA/INPRO Secretariat and endorsed by the INPRO Steering Committee [1.2]. Each participant was nominated by its responsible organization to perform the activities assigned in the ToR of the project. Each country met its own expenses, including those for human resources and travel deployed for this purpose.

Participants contributed data and information available to the public domain and necessary to perform the activities of the CP according to the ToR. The data and information included some elements of national nuclear energy policy and available parameters of current and innovative NESs. Participants were not requested to deliver confidential or proprietary data and information. Participants performed the activities assigned in the ToR, and those identified during the implementation of the CP necessary to achieve the goals, specific objectives and outcome of the CP according to available means.

Qualified experts from the countries affiliated with GAINS took part in several GAINS consultancy meetings and related activities (Fig. 1.3): O.E. Azpitarte, S.M. Jensen (CNEA, Argentina); G. Van den Eynde, D. Jaluvka (SCK•CEN, Belgium); G. Edwards, B. Hyland (AECL, Canada); J. Zhao (CIAE, China); R. Vocka (NRI, Czech Republic); J.P. Grouiller, L. Boucher, A. Vasile, C. Loaec (CEA, France); R.K. Sinha, P. Kelkar, A. Basak, V. Shivakumar (BARC, India), V. Rao (IGCAR, India); A. Lantiery, R. Calabrese (ENEA, Italy); K. Ono (JAEA, Japan); Y.-I. Kim, C. Jeong (KAERI, Republic of Korea); L. Andreeva-Andrievskaya (Rosatom, Russian Federation), V. Kagramanyan, E.E. Poplavskaya (IPPE, Russian Federation), G. Fesenko (MEPhI, Russian Federation); P. Darilek (VÚJE, Slovakia); F. Martin-Fuertes (CIEMAT, Spain); O. Godun, A. Gunaza (ENERGOATOM, Ukraine); J. Wheeler (DOE, USA), B. Dixon (INL, USA); J. Carlsson (JRC, European Commission). E. Bertel, P. Kovacs and Y.-J. Choi (OECD/NEA) participated in GAINS meetings as observers.

FIG. 1.2. GAINS participants: Belgium, Canada, China, the Czech Republic, France, India, Italy, Japan, the Republic of Korea, the Russian Federation, Slovakia, Spain, Ukraine, the USA and the European Commission, with Argentina as an observer.

FIG. 1.3. Participants of the second consultancy meeting of the collaborative project GAINS, O-arai, Japan, 29–31 October 2008.

The IAEA/INPRO provided support to the participants by facilitating coordination and meetings, through the scientific secretary of the GAINS CP (V. Usanov) and members of the INPRO group H. Hayashi and P. Villalibre, who made significant contributions to the project.

A basic implementation plan covered the activities to be performed, the participants involved and the corresponding schedule. An operational plan of work was developed and implemented between the consultancy meetings. A flow of information was provided by email correspondence and phone communication, thus avoiding travel expenses. Nevertheless, two meetings per year were found to be expedient for steering the CP and coordinating activities. To gain regular advice from IAEA experts and at the same time to keep direct communication with the members of national teams, it was agreed to alternate the meetings between the IAEA Headquarters and the nuclear centres of GAINS participants.

1.4 REPORT OUTLINE

Section 2 collects basic information on a number of international studies where there might be an interaction with the goals set out for the GAINS CP. This overview helps to avoid duplication of work and keep the focus in the GAINS project.

Section 3 describes principal assumptions and boundary conditions for the primary global models used in GAINS. These models are based on supporting two story lines for a global NES deployment: a separated world based on self-reliance and preservation of local identities; and a convergent world with rapid changes towards global solutions for economic, social and environmental challenges. Simulation of the global NES architecture is based not on geography but on the nuclear energy strategy pursued by different groups of countries.

Section 4 presents a set of high level indicators and EPs for use in evaluating results and comparing the different options for the global NES architecture. The ideas presented in the section should be considered in light of the definitions and concepts listed in the INPRO methodology; however, the scope of the sustainability indicators with regard to GAINS is focused on those aspects of an NES which have a broad and general context.

Section 5 introduces, within the wide range of uncertainty of available projections, a basis for the energy demand curves to be used in the framework. High and moderate nuclear energy demand scenarios for the GAINS study circumscribe an area of interest and make a basis for analysis and comparison of global NES architectures under assumed boundary conditions.

Section 6 provides basic input data and assumptions related to the NESs used to supply power to meet nuclear energy demand in the GAINS scenarios. Concerning nuclear reactor systems, the features including burnup performances and refuelling data linked to mass flow analysis are described. As for the NFC, important conditions which affect mass flow analysis results are also described. Some additional input data and assumptions are introduced in Sections 7–9 related to the specific systems addressed in those sections.

Section 7 defines the framework base cases and documents their sustainability performance so that they may serve as a reference by framework users. This includes a full description of the input parameters and outputs for each base case, as well as sustainability results for the framework base cases using the sustainability indicators.

Section 8 describes the results of sensitivity analyses, in which the impacts of changes in analysis conditions, including assumptions regarding potential reactor and fuel performance advances in the future upon various indices are studied, e.g. natural uranium consumption, uranium enrichment separative work, amount of spent fuel (SF) discharge, reprocessing load and plutonium balance.

Section 9 addresses additional innovative NES cases, the results of which are likely to have a higher uncertainty due to a lower technical maturity level than NESs assumed in the base cases and studies discussed in Sections 7 and 8. The first part of the section analyses the fuel and waste material flows in several different new NES types which may be implemented later in this century. The second part of the section is dedicated to some economic issues connected with development and deployment of a global NES.

Section 10 describes the result of GAINS activities to evaluate and select suitable computer codes that can be used to explore scenarios relevant to the deployment of the global NES in a sustainable manner. In the GAINS project, members have studied scenarios by using their own analysis codes. Results of cross-check calculations performed by some members with the use of different codes are presented and discussed in the section in order to quantitatively evaluate the credibility of the analysis.

Section 11 summarizes the main results of the study and provides an outlook on what can be expected to be achieved in the following phases of the INPRO vision and scenario studies.

Annex I illustrates a set of choices related to development of a long range national nuclear energy strategy in Ukraine. This country has significant experience in the use of nuclear energy and most of the necessary infrastructure is available, but there is not so clear readiness to incorporate rapidly the most advanced NESs from the moment of commercial availability. Typical issues of strategic planning and balancing of national and multinational opportunities are discussed in this annex.

Annex II comprises, in table form, fuel composition data for each reactor system which is used in various scenario studies in GAINS.

Annex III describes the nomenclature for scenario cases and provides a list of cases analysed by members for which the results are included on the attached CD-ROM[3].

Annex IV contains the annual data used for the nuclear power growth in framework base cases.

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[1.3] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of Nuclear Energy Systems Based on a Closed Nuclear Fuel Cycle with Fast Reactors, IAEA-TECDOC-1639, IAEA, Vienna (2010).

[1.4] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems, INPRO Manual — Overview of the Methodology, IAEA-TECDOC-1575 Rev. 1, IAEA, Vienna (2009).

[1.5] EUROPEAN COMMISSION, Coordination Action of the European Commission Sixth Framework Programme, PATEROS: Partitioning and Transmutation European Roadmap for Sustainable Nuclear Energy, FI6W-036418 SKC CEN, Belgium (2008).

[1.6] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, NUCLEAR ENERGY AGENCY, Trends in the Nuclear Fuel Cycle, OECD, Paris (2001).

[3] The material on the attached CD-ROM has been prepared from the original material as submitted for publication and has not been edited by the editorial staff of the IAEA.

2. PREVIOUS AND ONGOING STUDIES

2.1. BACKGROUND

At both national and international levels, fuel cycle studies and scenario studies have been and are performed on a regular basis. These studies differ in the fuel cycles (and corresponding reactor systems) studied and in their scope. Some studies give a broad, general picture; others focus on one specific issue. In order not to duplicate work and keep focus in the GAINS initiative, the participants thought it wise to collect basic information on a number of international studies where there might be an interaction with the goals set out for the GAINS CP. In this section, an overview is given of the studies most related to GAINS.

2.2. GENERAL PURPOSE STUDIES

2.2.1. General fuel cycle and IAEA scenario studies

The International Nuclear Fuel Cycle Evaluation (INFCE) [2.1] was a very large fuel cycle study conducted in 1977–1980. The effort was organized into eight working groups involving 519 experts from 46 countries and five international organizations, producing over 20 000 pages of documentation.

INFCE was conducted at a time when the first round of nuclear power deployment was at its zenith, so predicted growth and associated impacts were much higher or sooner. However, both the approach and the general INFCE findings were very similar to many more recent studies. The approach included future growth projections (low and high cases) and the assessment of a range of reactor types (light water reactor (LWR), HWR, high temperature reactor (HTR), fast breeder reactor (FBR)) and fuel cycles (once-through, plutonium recycle, thorium) and a range of sub-cases addressing potential technology improvements. The impacts of the reactor/fuel cycle combinations were then compared on a range of metrics including uranium/thorium and separative work unit (SWU) requirements, proliferation considerations, and SNF and waste management. Consideration of MA recycle was very limited compared to today. INFCE did not compare nuclear power to other energy sources (neither does GAINS) and did not consider significant non-electrical application of nuclear power (neither does GAINS).

Four basic country-level SNF management concepts were noted — decision to reprocess, deferred decision, fuel transfer (to another country) and no-reprocessing decision. Technical feasibility assessments noted that basic technology for enrichment, SNF transport, reprocessing (plutonium and uranium recovery by extraction) and LWR U–Pu MOX were all well established, while FBRs and their fuel and waste management and disposal were development areas.

While uncertainty concerning future doses was noted for geological repositories, “disposal can be carried out without undue risk to man or the environment” and “the cost of waste management and disposal is only a few per cent of the value of the electric energy generated and does not vary greatly between fuel cycles”. Radiological impacts “should be lower for FBRs than LWRs”, while for thorium cycles “daughter products of ^{232}U will call for additional radiological protection”.

> “The availability of small nuclear power plants would increase the number of developing countries that could introduce nuclear power in their electricity grids… Such smaller plant sizes would meet a potentially large market need, but they are not likely to be economically competitive except in special conditions.”

Proliferation was a major focus and concern of the study, and sensitive points in the fuel cycle were noted. Means for minimizing dangers of fuel cycle facility misuse included technical measures, improved safeguards and institutional arrangements.

The standard technical measures were listed for reducing theft and proliferation risks — reducing separated weapons-usable material and providing radiation and physical barriers. Specific methods to reduce separated material vulnerabilities included co-location of facilities, co-processing plutonium with uranium and denaturing of ^{233}U in the thorium cycle. Radiation barrier methods included pre-irradiation of fresh fuel, spiking and incomplete separations, but these measures “involve considerable environmental, radiological, economic and resource utilization penalties and tend, in some cases, to diminish the effectiveness of safeguards”.

Various types of institutional arrangements including those addressing both fuel cycle front end and back end were identified. The working group responsible for long term supply considerations asserted a general principle that “assurance of supply and assurance of non-proliferation are complementary”. Issues and recommendations related to institutional arrangements pertinent to the fuel cycle back end (e.g. SF management, reprocessing and repositories) were also identified. Particular emphasis was given to the special needs and conditions in developing countries throughout the study.

A study entitled Nuclear Energy Development in the 21st Century: Global Scenarios and Regional Trends [2.2] was carried out under the auspices of INPRO with the objectives to:

— Illustrate, using idealized examples, the potential contribution that innovative NESs using FRs and closed fuel cycles could make in meeting the global and regional demand for nuclear energy, for a range of possible demand scenarios and, in parallel, also illustrate the potential roles of different reactor types operating in combination in an evolving and growing global NES;
— Illustrate, for a range of possible demand scenarios, how nuclear material might flow between different regions of the world in the twenty-first century;
— Identify some of the issues that might need to be addressed to implement the global vision of nuclear energy development presented in the study, and, in particular, the vision for the two higher demand scenarios.

This study uses several reactor and fuel cycle types which are available today and are likely to be developed in the future to illustrate a modelling approach to the global growth of nuclear energy with the aim to demonstrate, within certain assumptions, the role of interregional transfer of nuclear fuel resources.

The INPRO methodology sets out criteria in seven areas — economics, infrastructure, safety, waste management, proliferation resistance, physical protection and the environment — that need to be satisfied for nuclear energy to be sustainable. A number of assumptions have been made to facilitate within the study that NES would meet those criteria.

The study then illustrates how FRs and closed fuel cycles can limit the demand on uranium resources. They have to do mainly with the need for a substantial growth of nuclear energy and its global development, continuation of thermal reactor use and the need for FBR deployment, availability of uranium resources, establishment of an international framework for non-proliferation and nuclear fuel supply, and the market development for small and medium size reactors, high temperature reactors and some others.

The study considered three scenarios for capacity growth with different target installed capacities by the end of the twenty-first century: low — 2500 GW(e), moderate — 5000 GW(e) and high — 10 000 GW(e).

One of the objectives of the study is to illustrate, for a range of possible nuclear energy demand scenarios, how nuclear material might flow between different regions of the world. For this purpose, the world has been divided into different regions to allow a global view consistent with that obtained by aggregating regional views. Regions were defined on the basis of geographical proximity, so the specific attributes of countries in a region need to be considered to arrive at the attributes of that region.

Since the study examines possible evolutions of the global NESs, the starting point is the current mix of reactors.

A set of reactor types was chosen that represents those listed above except that, rather than considering the deployment of high conversion thermal reactors, the study takes a low BR FR and assumes that they start their operation using fissile plutonium recovered from the SF of thermal reactors or from other FRs. Therefore, the study examines how a mixed set of reactors might evolve if the consumption of natural uranium in the twenty-first century were constrained to 20 million tonnes (Mt), while recognizing that additional resources may well become available.

As an alternative to deploying FRs with high BRs, the use of thermonuclear fusion systems for breeding Pu/^{233}U fuel for use in fission reactors could be considered. While such a development is, at present, conceptual, computer analysis can be used to demonstrate the impact of deploying a hybrid fusion–fission system, where the fusion reactor would be used primarily for the production of neutrons rather than conversion of energy per se and so its design and operation would not be determined by energy efficiency requirements but, instead, by its capability to generate fresh fuel (breed fissile material) for use in fission reactors.

As an option, an early uranium start for fast breeders was also shortly considered.

In the low growth scenario, a global nuclear power capacity increase from 370 GW(e) in 2009 to 2500 GW(e) by the end of the century was chosen. Assuming the free transfer of fuel services between regions, thermal reactors are shown to be sufficient to satisfy demand until 2100 while not exceeding the postulated limit of natural uranium consumption (20 Mt). The establishment of the necessary SNF disposal facilities for such an NES may, however, be problematic.

In the moderate and high growth scenarios, global nuclear power capacity was assumed to reach 5000 and 10 000 GW(e), respectively, by the end of the century. The study demonstrates that closing the fuel cycle and deploying FRs in combination with the continued deployment of thermal reactors could accommodate a large expansion of the role of nuclear power while keeping the uranium consumption within the limit of 20 Mt. Thus, the study has shown that reasonably priced fissile material is not a limiting factor for the sustainability of nuclear power.

While technical problems could be solved with subsequent investment and R&D efforts within the given time, non-technical challenges could present a serious barrier to the rapid growth of nuclear power and may require appropriate international efforts.

Given the long lead times required to develop and bring new nuclear energy facilities into commercial operation and the costs involved, it is clear that substantial investment by governments will be required. In seeking to justify such investment, a common understanding of the need for innovative NESs and of the evolution/transition of NESs as innovative technologies are introduced needs to be developed.

2.2.2. Organisation for Economic Co-operation and Development/Nuclear Energy Agency general purpose studies

Under the auspices of the Nuclear Science Committee of the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA), an expert group on fuel cycle transition scenarios studies was created in October 2004. This expert group has compiled and reviewed information on issues involved in transitioning from current fuel cycles to long term sustainable fuel cycles or a phase-out of nuclear energy. The scope of the expert group covers existing and future technologies available for the transition period, including transmutation and storage of SF, development and assessment of transition scenarios, and evaluation of the impact of the transition on reactors and fuel cycle facilities. The expert group has completed a status report, published in 2009 [2.3], which covers country specific scenarios for Belgium, Canada, France, Germany, Japan, the Republic of Korea, Spain, the United Kingdom and the USA, as well as a list of key technologies that were identified as crucial for the implementation of advanced fuel cycles. It is also investigating global and regional (European) transition scenarios to analyse the impacts of different strategies and policies and the role and characteristics of regional facilities. Finally, a benchmark study is underway to compare the results of scenario analysis codes developed by Member States.

The implementation of transition scenarios raises a number of key strategic and policy issues which have not yet been investigated in depth. The main objective of the present study on Strategic and Policy Issues Raised by Transition from Thermal to Fast Reactors [2.4], undertaken under the umbrella of the Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle (NDC) of the OECD/NEA, is to identify and analyse those policy issues and to record findings, draw conclusions and make recommendations for policy makers. The study is based on a review of transition scenarios developed at the national, regional and international levels which has provided the scientific and technical background material for identifying the opportunities and challenges associated with those scenarios and the policy issues to be addressed by governments and the industry when considering the transition from thermal to fast nuclear systems. The findings from the study highlight the need to evaluate the advantages and drawbacks of transition scenarios in a holistic approach, taking into consideration short term and long term aspects, and assessing environmental and social criteria as well as economics. Its conclusions show that the viability of transition scenarios and their successful implementation will require long term commitments and comprehensive and consistent planning. The study also illustrates the potential role of international cooperation and multinational endeavours in facilitating the implementation of transition scenarios.

Recently, a new ad hoc expert group has been assembled to update the OECD/NEA study, Trends in the Nuclear Fuel Cycle [2.5]. This study assembled a set of criteria to which nuclear energy should adhere as a sustainable energy source. The main areas identified are resource sustainability, economics, environment and social aspects (risk perception, proliferation risk). The study also gives an interesting overview of current and future fuel cycle schemes. The final report of this expert group has been completed.

2.3. SPECIAL PURPOSE STUDIES

2.3.1. Red-Impact

The Red-Impact project [2.6], funded by the Sixth Framework Programme of the European Commission, studied the impact of partitioning, transmutation and waste reduction technologies on the final nuclear waste disposal. From Ref. [2.6]:

> "The partnership collects 23 organizations drawn from European nuclear industry, waste agencies, research centres and universities. Five representative scenarios, ranging from direct disposal of the spent fuel to fully closed cycles (including minor actinide recycling) with fast neutron reactors or accelerator driven systems (ADS), were chosen in the project to cover a wide range of representative waste streams, fuel cycle facilities and process performances. High and intermediate level waste streams have been evaluated for all of these scenarios with the aim of analysing the impact on geological disposal in different host formations such as granite, clay and salt. For each scenario and waste stream, specific waste package forms have been proposed and their main characteristics identified. Both equilibrium and transition analyses have been applied to those scenarios. The performed assessments have addressed parameters such as the total radioactive and radiotoxic inventory, discharges during reprocessing, thermal power and radiation emission of the waste packages, corrosion of matrices, transport of radioisotopes through the engineered and geological barriers or the resulting doses from the repository."

The scenarios that have been studied in the Red-Impact project are subdivided into 'industrial scenarios' and 'innovative scenarios'. For the first type, three scenarios are studied:

— A1: LWR reactors, UO_2, once-through (reference scenario).
— A2: LWR reactors, UO_2 + MOX (once Pu recycling).
— A3: introduction of fast spectrum only for Pu reuse.

For the innovative type, three scenarios are considered:

— B1: a Generation IV solution based on an integral FR.
— B2: comparable to A2, but with the introduction of ADS in a second stratum for transmutation of remaining Pu and MA.
— B3: double-strata: LWR (UO_2 + MOX) + FR in first stratum; ADS with MA burning in second stratum.

For the scenarios mentioned above, equilibrium situations have been studied. Three transition scenarios have also been investigated. These all start with A1 as the reference and have as a final scenario A3, B1 and B2.

During the Red-Impact project, the partners calculated a set of indicators. These indicators are grouped into four main categories: environmental indicators, long term consequences of radioactive waste disposal, economic indicators and societal indicators.

(a) Environmental indicators:
  (i) Radiotoxicity of releases;
  (ii) Fuel cycle related waste;
  (iii) Pre-disposal waste management aspect.
(b) Long term consequences of radioactive waste disposal:
  (i) Radiotoxicity in the repository (500, 10 000, 1 000 000 years);
  (ii) Radiotoxicity flux release in the biosphere;
  (iii) Human intrusion into the repository;

(iv) Doses received due to scenario operation.

(c) Economic indicators:
   (i) Research and development costs;
   (ii) Facility construction costs;
   (iii) Facility operational costs;
   (iv) Waste operational costs;
   (v) Decontamination and decommissioning costs;
   (vi) Total repository cost.

(d) Societal indicators:
   (i) Years of life lost due to power generation;
   (ii) Fatalities due to accidents in an NFC;
   (iii) Proliferation resistance.

To be able to compare the different scenarios, a multi-criteria analysis approach was used. Each indicator received a certain weight and a weighted sum was made over all indicators for each scenario. This ‘result’ has neither a physical nor an economic meaning nor an absolute meaning. It simply allows comparison of the different scenarios. The weights, of course, play a crucial role in the final outcome of the analysis. In the case of the Red-Impact study, they were determined based on an enquiry with the different partners of the project.

Details on the scenarios and their analysis can be found in the synthesis report and deliverables D1.4 ‘Detailed description of selected fuel cycle scenarios’ and D3.1 ‘Definition of technical data and detailed hypothesis’. Unfortunately, these two deliverables are not publicly available.

2.3.2. PATEROS

The PATEROS research project was also funded by the Sixth Framework Programme of the European Commission [2.7]. Its goal was to establish a global partitioning and transmutation (P&T) roadmap up to the industrial scale deployment indicating critical milestones, preferred options and backups, according to timescales and shared objectives at the European level. The project acknowledges that the number and size of ‘building blocks’ will depend on the strategy and the policy of a given European Union Member State. However, a common objective of all strategies using P&T is to reduce the burden of long term waste management, in terms of radiotoxicity, volume and heat load of high level nuclear waste which has to be put into final repositories. Possible strategies can range from using dedicated transmuters in a separate fuel cycle stratum in a stable or expanding nuclear energy scenario, in order to drastically reduce the amount of MAs sent to the repository, up to the scenario of a nuclear phase-out.

The consortium of this coordinated action has structured its work into six work packages (WPs) as indicated below:

- WP1: Rational and added value of P&T for waste management policies.
- WP2: Review and selection of relevant fuel cycle strategies in Europe supplemented by regional context for development and deployment.
- WP3: Fuel cycle facilities related to reprocessing.
- WP4: Fuel cycle facilities related to fuel fabrication demonstration.
- WP5: Fuel cycle facilities for transmutation and associated technology.
- WP6: Integration and evaluation of resources.

Within WP2, two deliverables are of principal interest to GAINS: D2.1 ‘Fuel cycle scenarios selection for a regional approach at European scale’ and D2.2 ‘Results of the regional scenario studies’. Since the objective of the PATEROS project is to provide a regional perspective, countries have been grouped as follows:

- Group A: stagnant or phase-out; focus on SF management.
- Group B: continuation scenario; focus on optimization of plutonium for future deployment of FRs or ADS.
- Group C: subset of group A, after stagnation, envisages a nuclear ‘renaissance’.
- Group D: initially no nuclear power, decides to go for nuclear energy in the future.

Within GAINS, a similar approach has been proposed (see Section 3.3).

Four scenarios are studied:

— Scenarios 1 and 2: deployment of ADS shared by groups of countries A and B. ADS will use the plutonium of group A and transmute MA of both groups. Plutonium of group B is either mono-recycled in PWRs and stored for future deployment of FRs (scenario 1) or continuously recycled in PWRs (scenario 2).
— Scenario 3: deployment of a group of FRs in group B using plutonium from groups A and B. Objective: to decrease stock of SF of group A.
— Scenario 4: group C decides to relaunch nuclear energy with FRs, other countries of group A continue with their objective of waste minimization.

In all of the scenario analyses, the total electricity production coming from nuclear sources was frozen.

2.3.3. Advanced nuclear fuel cycles and radioactive waste management

This study [2.8] was carried out by the ad hoc expert group on the impact of advanced NFCs on waste management policies convened under the auspices of the NEA NDC. From Ref. [2.8]:

> “This study focuses on the impact of advanced fuel cycles on waste management policies. The set of fuel cycles considered here covers a broader spectrum than previous studies, from present industrial practice to fully closed, FR cycles, via partially closed cycles. Elements of fuel cycles are considered primarily as sources of waste (both primary and secondary), the internal mass flows of each cycle being kept for the sake of mass conservation. The masses, compositions, activities and heat loads of all waste flows are tracked. Their impact is finally assessed on waste repository concepts located in four different host rocks: clay, granite, tuff and salt. A track of economic impacts has been kept as well.”

This study investigates three families of scenarios: schemes based on current industrial technology (with possible extensions), schemes based on a partially closed fuel cycle and schemes with fully closed fuel cycles.

In the first family, one can find the LWR/once-through cycle and schemes with a single recycle (MOX). Two variants of the single recycle scheme are also considered (one where the neptunium goes with the plutonium to reduce proliferation and another one where the oxidation and reduction of oxides processing avoids any chemical reprocessing).

In the second family, cycles are studied which are fully closed for plutonium and the neptunium is immediately considered as waste. The schemes differ in their treatment of americium and curium.

In the third family, all advanced reactor types have fully closed fuel cycles. All actinides are recycled continuously until they fission. Two major sub-schemes are considered: one based on the integral FR system based on critical transuranic (TRU) burners and the double-strata approach using dedicated ADS.

A striking conclusion of this study was that related to the field of waste management, the experts could not indicate a single, simple and universally agreed upon indicator which could put a measure to the different options in order to compare them. A number of indicators were identified, some interlinked. However, their relative worth compared to each other was different according to one expert or the other.

The main indicators evaluated are waste volumes, resource consumption, radiation levels, thermal decay heat and actinide content. As the title and scope of the study indicate, all are situated in the field of nuclear waste management.

The study’s general conclusion is that: “It is therefore possible to design for acceptable costs innovative nuclear reactor cycles, which at the same time spare resources and make the most efficient use of the foreseen geological repository sites”.

2.3.4. INPRO joint study on closed nuclear fuel cycle with fast reactors

This joint study on the joint assessment of an INS based on the CNFC with FRs was initiated by the Russian Federation in 2004. It was implemented by Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation and Ukraine between 2005 and 2007 [2.9]. The main objectives of the study were to assess the INS CNFC–FR for satisfying criteria of sustainability, determine milestones for its deployment, and establish frameworks and areas for collaborative R&D work.

The joint study was implemented in different steps. In its first step, discussions were focused on the analysis of country/region/world data along with possible national and global scenarios for the introduction of the INS CNFC–FR. Then, technologies suitable for the INS were identified and reviewed, and finally a common INS CNFC–FR was defined.

In the second step, the characteristics of the INS CNFC–FR were examined to assess its compliance with sustainability criteria developed in the INPRO methodology in the areas of economics, safety, environment, waste management, proliferation resistance and infrastructure.

It was agreed to perform the assessment on the basis of a near/medium term INS CNFC–FR using proven technologies, such as sodium coolant, MOX pellet fuel and aqueous reprocessing technology. The main results and findings of the study are summarized below.

The successful operation of several demo and demo/commercial FRs has demonstrated that FRs can meet current safety standards. The results of safety analyses have also shown that the requirement to reduce the risk of severe accidents for future FRs by at least one order of magnitude can be fulfilled, if safety features are further enhanced through identified R&D. Probabilistic analysis has also demonstrated the ability of the INS to prevent the need for relocation or evacuation measures outside the plant site in case of a major accident. Thus, safety characteristics of near/medium term INS CNFC-sodium cooled FRs (SFRs) are judged to be in compliance with safety requirements of sustainable energy supply. Nevertheless, in spite of the judgement on the high safety of SFRs, the need for development of an alternative FR technology is clearly stated by the joint study group because of the crucial importance of CNFC–FR for enhancing the sustainability features of the future of nuclear power.

The environmental effects of the demo and near/medium term INS CNFC-SFR are well within the performance envelope of current NESs delivering similar energy products, with the lowest greenhouse gas emissions among them. The feasibility of excellent environmental and health preserving features of CNFC–FRs has also been demonstrated by the operation of demo/industrial FRs with sodium coolant and the associated fuel cycle. The introduction of CNFC–FRs in some countries might help to make most efficient use of nuclear fuel resources by using denatured uranium fuel and plutonium fuel, which can be generated in the FR core and blankets, if needed. The CNFC–FR would extend the available nuclear fuel by a factor of several thousand and would be nearly inexhaustible. De facto it can be considered as a renewable energy resource suitable for large scale national and global deployment.

Safe conditioning of waste arising from plutonium recycling is industrial reality today and an important practical milestone in reaching the ultimate goals of the closed cycle strategy. The CNFC–FR has in practice shown its potential to meet all of today's requirements related to waste management. With development and introduction of novel technologies for optimal management of nuclear fissile products and MAs, a CNFC–FR would have break-through potential to meet the sustainability requirements related to waste management.

The proliferation resistance of the INS CNFC–FR due to realization of the intrinsic features could be comparable or higher than that of the open fuel cycle. The INS provides a key technology for optimal utilization of fissile material and elimination of its disposal in geological repositories, thus providing a reliable background for applying extrinsic institutional arrangements. The joint study has judged that intrinsic features of the INS offer a unique technological platform to meet basic principles of sustainability in the domain of proliferation resistance. More efforts in further development of extrinsic measures have to be made to provide conditions for using these opportunities in transition to a new and higher level of nuclear power proliferation resistance.

A legal nuclear framework in accordance with international standards has already been established in all countries participating in the joint study as well as an appropriate economic/industrial infrastructure. Efforts are being undertaken to enhance public acceptance, political support and inflow of human resources. The study came to the conclusion that the CNFC–FR is a suitable technology for realization of a regional or multilateral approach to the assurance of the front and back end of fuel cycle services and transition to a global nuclear architecture that will provide new perspectives for growth of mature nuclear industries and at the same time facilitate the use of nuclear power by newcomers.

First of a kind INS CNFC–FRs did not fully meet the economic requirements. This is the area where the INS cannot fully comply with the INPRO demand. In accordance with the INPRO methodology recommendations, assessors have addressed the examination of possible improvements in the INS design and technology to meet the economic acceptance limits. Analysis of the study has shown that design simplification, increase of fuel burnup, and cost improvements via R&D along with construction in small series would result in competitive costs of FRs with thermal reactor nuclear power plants and fossil fuelled power plants. The joint study concludes that commercial INS CNFC–FRs based on available technologies will be affordable in 10–20 years in countries mastering the technology.

The overall judgement on the INS sustainability features was made in the study based on both the assessment to meet the INPRO acceptance limits and the expertise of the participants regarding the INS potential to reach target values beyond acceptance limits. The analysis has indicated that the capability of the INS to meet the INPRO criteria in the different areas of assessment, apart from economy, would be better, or much better, than current nuclear systems. Thus, the near term INS CNFC–FR will confidently meet the INPRO requirements of sustainable energy supply provided that the identified R&D (with a focus on economics and safety) were carried out.

2.4. CONCLUSIONS

In the field of fuel cycle analysis and long term vision of NES deployment, a number of studies have been performed and are ongoing. Some of the ones touching on the scope of the GAINS project were summarized above.

Some statements can be made in conclusion:

— Fuel cycle studies are common practice, both at the national and international level. The goal of enhancing sustainability features of NESs stimulates intensification of multinational studies on the future arrangement of the global fuel cycle.
— Collaborative studies (in the framework of the OECD/NEA, European Commission and IAEA) tend to focus on one specific aspect, such as waste or resources.
— In all studies, different indicators or metrics are identified with which different options can be compared.
— Studying all indicators in the INPRO methodology is nearly impossible for all different scenarios including different innovative reactor systems. Nevertheless, a trend to implementing the holistic approach based on assessing all areas of sustainable energy supply is gaining strength.
— In some studies, it is explicitely mentioned that is hard to combine different indicators in one global ‘score’.
— In the Red-Impact study, the results for the different indicators were aggregated in a single score using a weighting procedure. The weights are defined by interrogating different experts. The study explicitly mentions that the final score has no physical meaning whatsoever, but makes it possible only to compare different options.

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3. HOMOGENEOUS AND HETEROGENEOUS MODELS OF A GLOBAL NUCLEAR SYSTEM

3.1. INTRODUCTION AND PURPOSE

The overall objective for the CP GAINS, according to the ToR, was to:

> "develop a standard framework (methodological platform, assumptions and boundary conditions) for assessing NES regarding sustainable development, covering from the existing systems to the innovative ones potentially deployed in the 21st Century, and validate its results via sample analyses".

This section describes the basis, assumptions and boundary conditions for the primary global models used in GAINS. These models were based on supporting two 'story lines' for nuclear energy demand, as described in the detailed implementation plan for GAINS:

> "The first story line will describe a convergent (homogeneous) world with rapid changes towards global solutions for economic, social and environmental challenges. The opportunities facilitating creation of the global and regional nuclear architecture such as unification of reactor fleet, sharing infrastructure, arrangement of multinational regional fuel cycle centres, and innovative approaches to financing and licensing have to be taken into account under construction of this story line."

> "The second story line will reflect a heterogeneous world based on self-reliance and preservation of local identities. Nuclear services across regions converge very slowly, regional differences in availability of material resources, energy growth rate and nuclear energy deployment options remain significant. Definition of the regions should be based not on a geographical principle but on the typical conditions in different groups of countries and the modes of the nuclear power development. When summing up, key indices of these conventional regions related to growth of nuclear power, resource consumption, etc. should be in compliance with global expectations for the reviewed time horizon."

Superimposed on these two primary story lines are two growth rate scenarios, as described in Section 5. When applied to the two story lines, these scenarios allow exploration of different global architectures of innovative NESs in contributing to sustainable energy supply, while also revealing major constraints which apply during or after the transition to a future innovative nuclear system.

A practical constraint for CP GAINS was that the story lines and scenarios be structured so that they could be assessed without requiring major extensions to existing analytical tools. One of the specific objectives in the ToR was an "Evaluation of the existing analytical tools for modelling NES and establishment of a specification to improve such tools." It was felt that minor extensions could be accomplished in time to support the CP GAINS effort, while major extensions could not. Specific analysis considerations are discussed throughout this section.

According to the implementation plan, the time horizon for assessment was the remainder of the century, with transition points occurring at three moments: "Starting from 2008, transitions will be considered in 2030, 2050 and 2075. Assumptions made for the latter will cover the period until 2100." Analyses are run beyond 2100 (to at least 2110) to demonstrate that the scenario is stable with no sudden lack of material or other system upset appearing just beyond the end of the century.

3.2. GLOBAL AND REGIONAL CONSIDERATIONS

The first expected output from the GAINS ToR was a "framework process for analysing NFC options from a sustainable point of view, taking into account its implementation at global, regional and national levels".

The homogeneous story line is most easily assessed by modelling the world as a single global entity with a composite history and development pathway. In this global model, the nuclear system architecture is modelled as a composite fleet and material as composite inventories. The infrastructure includes composite conversion, enrichment, fuel fabrication, storage, separations and waste disposal capacity and also composite reactor fleets based on general reactor/transmuter types (e.g. LWR, HWR, FR, ADS). The inventories include raw ore deposits, depleted and enriched uranium, fuel in reactors and in wet and dry storage, recycled material inventories and wastes.

In contrast, the heterogeneous world story line requires different histories and development pathways for different parts of the world. This requires modelling the world as comprising different groups. Two approaches were established in the GAINS implementation plan:

> “(a) Non-geographic groups constituted by countries from all geographic areas in the world, having similarities from the point of view of nuclear technology development and deployment and status of necessary infrastructure.”

These groups were later renamed NGs, which is the terminology used in this report.

> “(b) Geographic groups established according to regional considerations.”

According to the implementation plan, the geographic grouping was not covered by the CP GAINS. Geographic group development and analysis may be included in follow-on studies.

The nuclear strategy grouping was based on composite values reflecting distinctions in the kinds of national nuclear systems that provide a basis for potential collaborations. An attempt to have specific listings of countries included in each group was not made, as many Member States do not have definite plans for nuclear power introduction/deployment and are sensitive about being grouped in this way. In addition, national nuclear strategies continue to evolve and a country specific approach could quickly be out of date.

3.3. NUCLEAR STRATEGY GROUP DEFINITIONS

The CP GAINS had considerable discussion concerning how to organize and model the NGs.

The GAINS objective of assessing sustainability required, as a minimum, assessment of the fielding of new technologies, either as an initial appearance of advanced technologies or as a wider use of an existing technology to achieve the evolution required by sustainable development. The definition of transition points supported this requirement. All of the fuel cycle models in use at the IAEA and by IAEA Member States supported the transition from one fuel cycle technology to another, so modelling of these transitions from the perspective of new technology emergence was straightforward for the homogeneous world story line.

To fully understand issues related to sustainability, it was also important to assess the role of non-uniformity of nuclear technology development in different groups of countries and to assess synergetic potentials between them, as depicted by the heterogeneous world story line. There was general agreement that the division should be based on the nuclear technologies deployed in each group. Technologies could evolve within a group throughout the modelled time frame, while maintaining the principal features of the technology basis for that group. This generated a requirement that different groups be able to have different technology bases and/or evolve at different transition points in the modelled time horizon. Several of the fuel cycle models had capability limitations relative to this requirement.

In a real situation, nuclear technology development and policy changes can move a country from a certain technology basis to a new innovative level. From the user perspective, it would be desirable for analytical or computer models to provide the capability to simulate transitions from one group to another over the course of the modelled time frame. However, this issue is rather complicated from the perspective of the modeller.

From the modelling perspective, it is very difficult to have the membership of a group change during a simulation as it impacts the infrastructure and inventories of each group. Most of the existing system’s modelling tools are not set up to handle the associated accounting adjustments, especially with regard to ‘look-ahead’ and predictive algorithms associated with ordering of new capacity.

For example, if a country with a large SF inventory moves from one group to another, the ordering of used fuel disposal capacity or separation capacity and calculation of plutonium availability for new reactors in both groups could be impacted. Reporting of results could also be impacted, since many of the indicators may be plotted as a function of time to show how they evolve as a system transitions. Changes in group membership would result in step changes in plots of inventories, capacities and performance indicators, requiring distracting explanations.

It was decided to adopt an approach where membership in each group was constant, but each group could develop independently based on the group's strategy and pace of technology evolution. Group composition was based on the technologies and strategies in use at the end of the scenario, and membership in the groups remained constant for the duration of a scenario. As group composition was based on status at the end of the scenario, a country currently having no nuclear generating capacity could be a member of any group. Sensitivity studies on the division of nuclear growth rates between groups provided a generic way of assessing shifts in the number of countries adopting one fuel cycle strategy versus another.

3.4. SYNERGISTIC AND NON-SYNERGISTIC GROUPS

In their simplest forms, the homogeneous world story line involves full cooperation between different parts of the world and uniform technology application, while the heterogeneous world story line involves either no cooperation or different degrees of cooperation between groups and application of different technologies and fuel cycle strategies. The homogeneous story line was modelled as a single world group while the heterogeneous story line was modelled as multiple groups operating separately from each other, or operating synergistically with interactions between the different groups as illustrated in Fig. 3.1. It should be noted that from a modelling perspective, the analyses of non-synergistic groups could be performed independently, since the groups operate in isolation from each other. In contrast, the heterogeneous synergistic case requires communication and material flows between groups. This enables modelling of limited shared nuclear services, which is an important story line option.

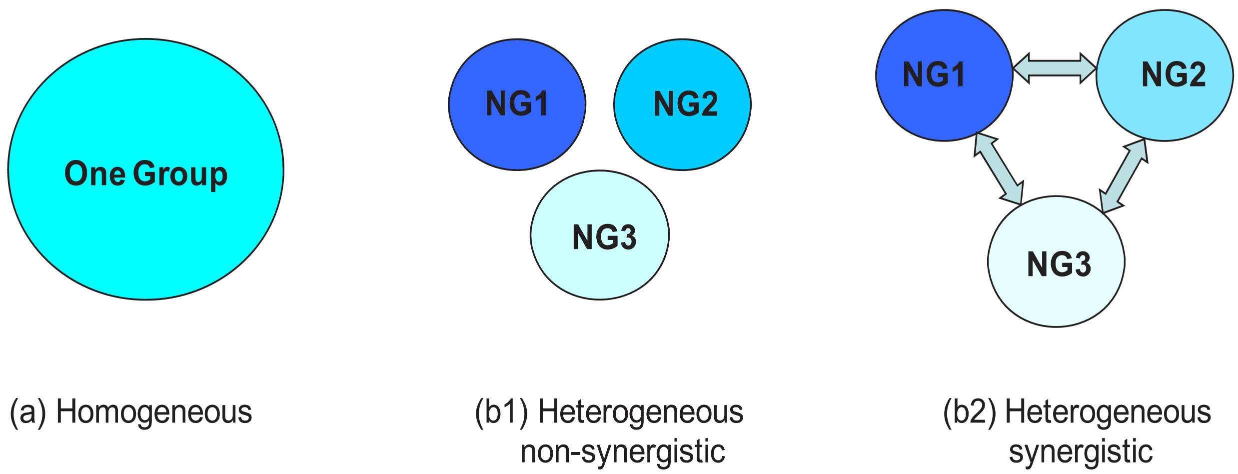

FIG. 3.1. Possible world models for fuel cycle analysis.

This middle ground between the two extremes of a homogeneous world and a non-synergistic heterogeneous world is an interesting area for investigation, where some level of interaction develops and evolves between groups while each group still maintains a separate identity based on the technologies and fuel cycle strategy in use. To support this heterogeneous synergistic story line, each group must be modelled separately but these model images must interact. The approach taken depended on the capabilities of the fuel cycle modelling code used. It was achieved either by keeping track of separate groups of reactors and their associated inventories, capacity and performance indicators all within one model 'image' or by running multiple simultaneous model images while including a mechanism for interaction between the images such as a shared 'blackboard' file which contains time dependent information on imports and exports and other interactions between groups.

It is important to note that all of the existing fuel cycle codes had some limitations with regard to modelling of multiple synergistic groups. This resulted in some scenarios only being assessable on a subset of the fuel cycle codes and some code extensions being needed before synergistic scenarios could be assessed. One way to make the scenarios assessable by a larger number of codes is to limit the number of synergistic groups and the number of reactor types within each group. Additional groups increase analysis options at the expense of increased modelling complexity.

3.5. RECOMMENDED MULTIGROUP MODEL

The implementation plan recommended the use of three NGs for modelling the heterogeneous world story line as the main option for the study. These same groups could be used to better understand the convergent story line and the potential system benefit and issues associated with multilateral arrangements to promote synergies.

The grouping described in the implementation plan was a matter of considerable discussion. This description is as follows:

— “Countries most involved in the development and deployment of the INS and consequently able to incorporate them as soon as commercially available are supposed to constitute the NG1.
— Countries having significant experience in the use of nuclear energy and most of the necessary infrastructure available, but not so clear readiness to incorporate rapidly the most advanced NES from the moment of its commercial availability would constitute the NG2.
— Finally, the countries supposed to incorporate nuclear energy in their energy mix, as newcomers, at some moments considered in the analysis would constitute the NG3.”

The primary limitation of this grouping approach was that it implied that all groups would reach the same final system, just at different rates. This did not allow for the assessment of some architectures in which countries may choose to operate reactors using fuel cycle services provided by another group. Such countries exist today and a projected expansion of nuclear energy may result in more in the future. Also, some countries may wish to remain with a once-through fuel cycle even while other countries adopt recycling. Thus, it was decided during the second GAINS consultancy meeting to adopt a grouping approach based on different approaches for managing the back end of the fuel cycle as also recognized in the implementation plan.

Three nuclear power producer groups (NG1, NG2 and NG3) were distinguished by the strategy for used fuel management on the back end of the fuel cycle adopted by the end of the simulation to develop the NG definitions. These three groups operate separately from or synergistically with each other to create a global architecture for innovative NESs:

— NG1: The general strategy is to recycle used fuel — the group plans to build, operate and manage used fuel recycling facilities and permanent geological disposal facilities for highly radioactive waste.
— NG2: The general strategy is to either directly dispose of used fuel, or reprocess used fuel abroad — the group plans to build, operate and manage permanent geological disposal facilities for highly radioactive waste (in the form of used fuel and/or reprocessing waste) and/or it works synergistically with another group to have its fuel recycled.
— NG3: The general strategy is to use fresh fuel, and send used fuel abroad for either recycling or disposal, or the back end strategy is undecided — the group has no plans to build, operate and manage used fuel recycling facilities or permanent geological disposal facilities for highly radioactive waste. They may obtain fabricated fuel from abroad and may arrange for export of their used fuel.

Depending on the specific scenario analysed, individual groups may or may not have front end fuel cycle facilities, including enrichment and/or fuel fabrication.

The following series of figures shows how these groups may be defined and how they may interact in different scenarios. Figure 3.2 shows graphically the fuel cycle functions each group would employ in a non-synergistic heterogeneous world model. Solid lines indicate required functions and actions, while dotted lines indicate additional options.

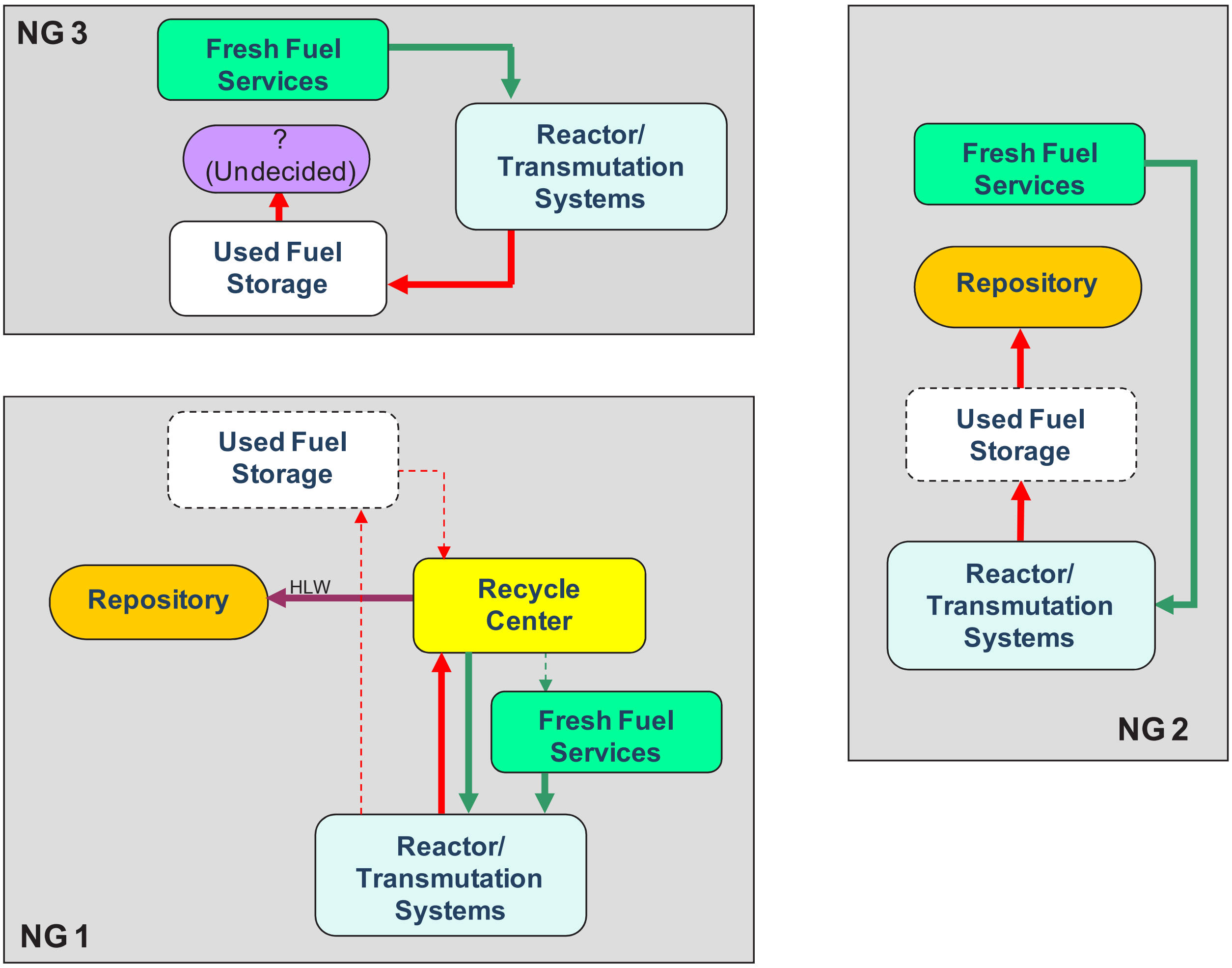

FIG. 3.2. Heterogeneous model with non-synergistic groups.

Figure 3.3 is a simplified synergistic model based on multilateral nuclear approaches (MNAs).

Figure 3.4 provides an expanded version of a synergistic model to illustrate various possible options and variants, which could become quite complex.

Finally, Fig. 3.5 depicts one example of a synergistic architecture in which a specific configuration of reactor types and fuel cycle services are represented for analysis purposes.

3.6. ADDITIONAL SCENARIO INFORMATION

For each scenario assessed, a set of parameters were needed. These included initial conditions, reactor characteristics, introduction rates for reactors and fuel cycle facilities, uranium limits and enrichment tail assays. A set of base case scenarios was defined which included a number of input and output parameters for both a once-through and a closed fuel cycle scenario in the homogeneous story line and both non-synergistic and synergistic scenarios in the heterogeneous story line. These parameters and base case scenarios are described in the next few sections, following a discussion of KIs and EPs.

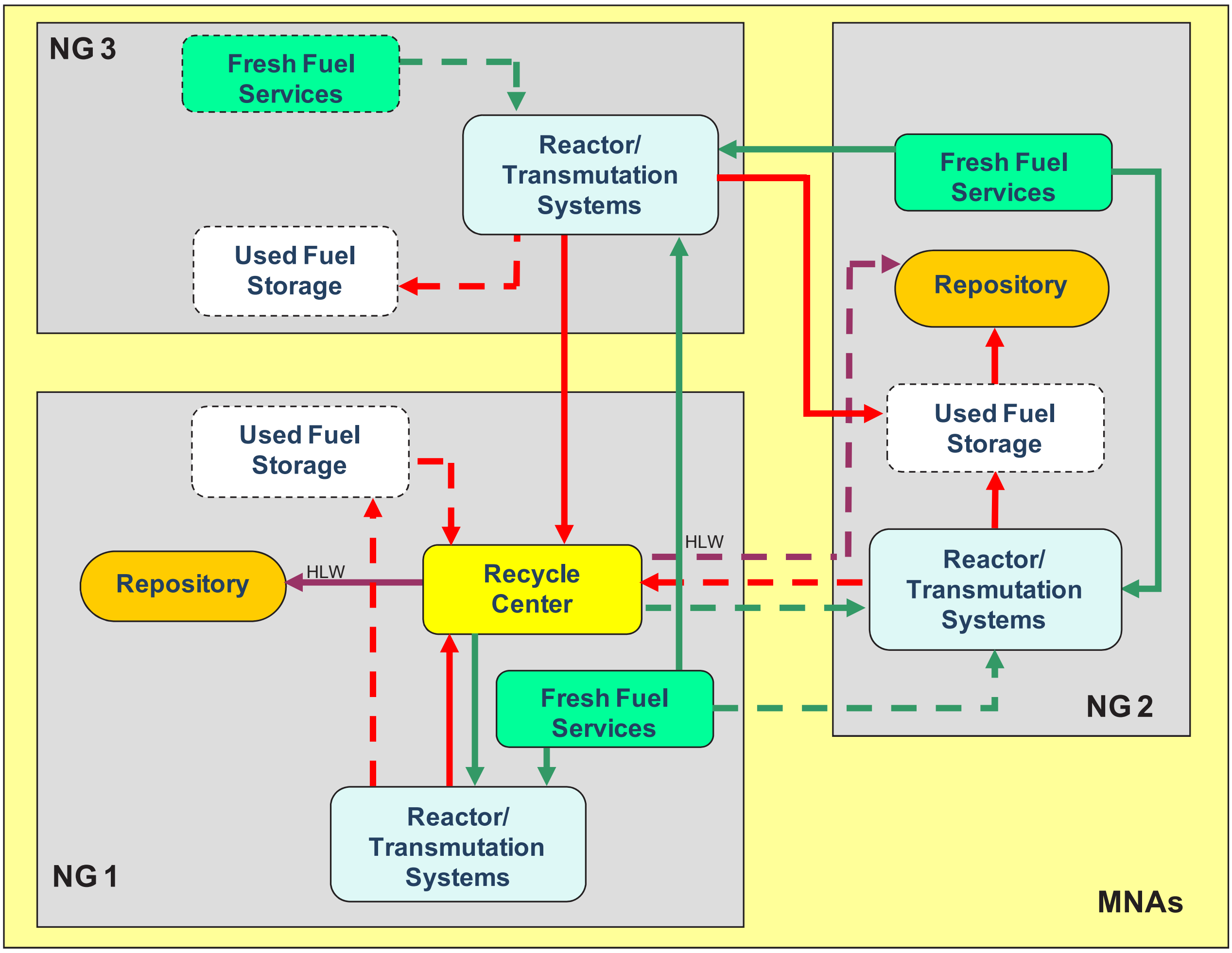

FIG. 3.3. Heterogeneous model with synergistic groups (simplified).

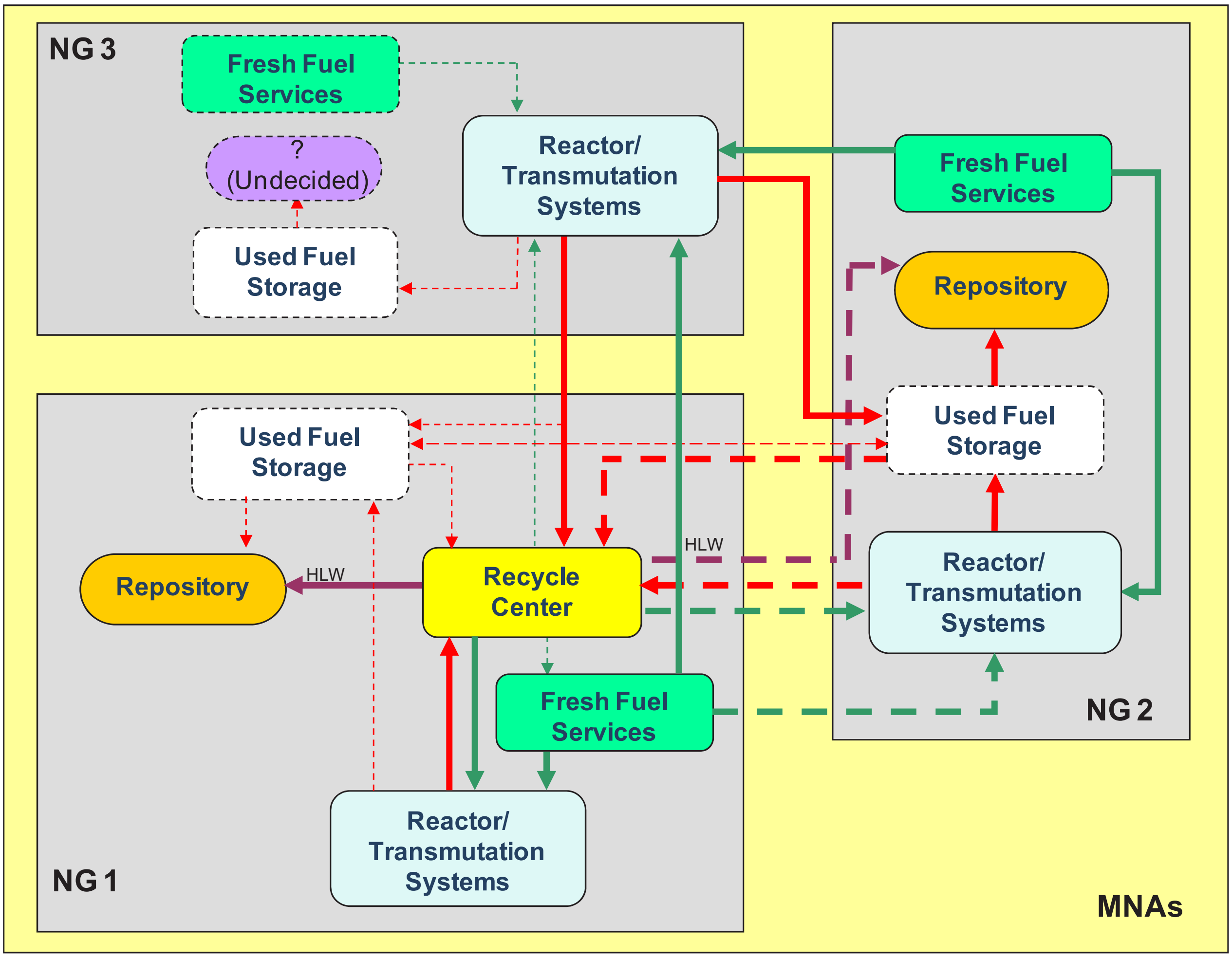

FIG. 3.4. Heterogeneous model with synergistic groups (expanded).

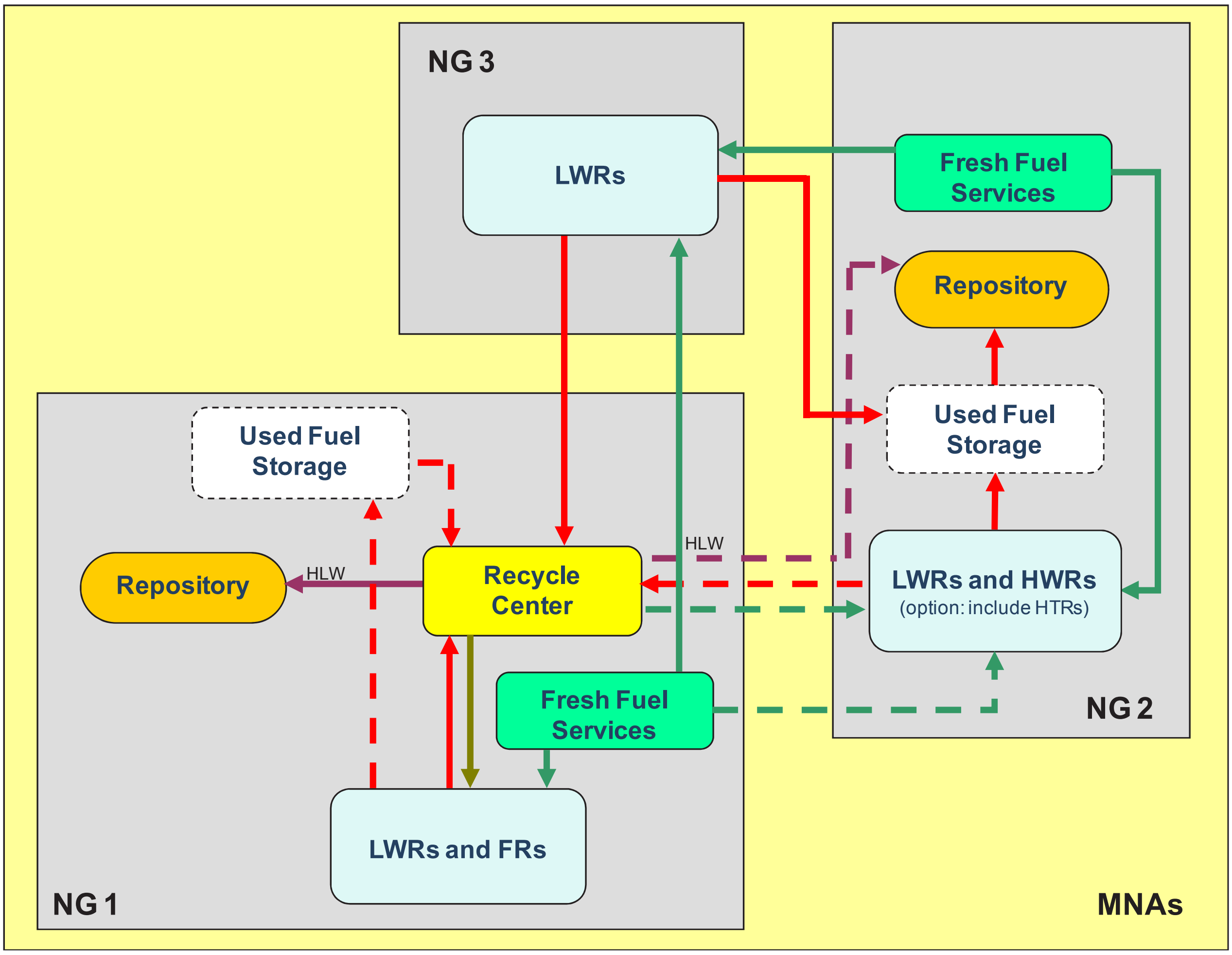

FIG. 3.5. Example of the heterogeneous synergistic world model with specific reactor types and fuel services identified.

4. INDICATORS OF SUSTAINABILITY FOR NUCLEAR ENERGY SYSTEMS USING THE GAINS FRAMEWORK

4.1. INTRODUCTION

Since the main objective of the CP GAINS is to develop a framework for assessing future NESs in regard to sustainable development and to validate the results via sample analyses, a set of KIs and EPs, calculated for each NES architecture, was developed and forms the main basis for comparing the different options and results. This section presents these high level indicators and parameters.

4.2. HIGHLIGHTS OF THE GAINS APPROACH TO DEFINING KEY INDICATORS AND EVALUATION PARAMETERS

While the full set of indicators listed in IAEA TECDOCs 1434 and 1575 [4.1, 4.2] aims to provide a tool for detailed evaluation of a given NES for potential use in an individual nation, the purpose in the GAINS framework is to provide a main basis for comparing the different options and the results of various scenarios on a global basis. Thus, one may realize that while INPRO methodology can be applied in the context of an NES with regard to specific local conditions, the scope of these indicators with regards to GAINS is rather limited to only those aspects of an NES which have a broader and more general context. The GAINS framework as presently constructed allows for the assessment of non-geographic NGs as described in Section 3, but if the framework is applied to geographic groups in the future, local conditions may need to be accounted for in more detail. An example of a quantity that is rather dependent on local condition is waste volume. The acceptable mode of waste disposal is subject to various factors, such as repository characteristics and local regulations. Assumptions for such parameters may need to be developed to promote a common basis for higher level assessments.

The INPRO methodology identifies seven assessment areas: economics, infrastructure, waste management, proliferation resistance, physical protection, environment and the safety of nuclear installations. For each of these assessment areas, the methodology provides a flow-down from basic principles to user requirements and then to criteria, the latter of which comprised a set of indicators and corresponding acceptance limits. The GAINS framework uses the concept of KIs introduced in the INPRO methodology [4.1]. KIs may be defined in specific, or in all, INPRO assessment areas, depending on the preferences of participants for a specific project. The idea is that a KI would have a distinctive capability for capturing the essence of a given area, and that they would provide a means to establish targets in a specific area to be reached via improving technical or infrastructure characteristics of the NES. KIs may be formulated by selecting a specific indicator, by grouping a few existing indicators or even by specifying a new indicator. Minimization of the number of KIs facilitates analysis based on implementation of a scenario-based approach. However, to get a quantitative value or further illumination of the indicators, it may be necessary to include some EPs [4.2]. In many cases, these parameters can serve as sub-indicators which give an additional depth to the estimation of the NES sustainability.

The whole set of INPRO indicators were screened by GAINS participants and those found to be of critical interest for the objectives defined in the GAINS ToR were put on the list of ‘GAINS KIs’. The choice made is based on the experience gained by the participants in national and/or international scenario studies and, to some extent, by the project resources available and the capabilities of computational tools currently at the disposal of the participants.

A further point to note is that GAINS does not intend to apply acceptance limits as part of the KIs, as required by INPRO methodology. GAINS applies a scenario-based ‘what if’ approach as part of its framework. Each KI is calculated in GAINS for a series of scenarios which depict incremental changes in the global NESs due to technological and infrastructural innovations. The potential for enhancing sustainability features of the reviewed NESs in the area characterized by a specific KI can also be estimated. The matrix ‘set of KIs versus set of scenarios’ also gives a comprehensive initial basis for making an overall judgement on the sustainability of the different alternatives of the global NESs. It is used in GAINS for discussion of strengths and weaknesses of the alternatives with regard to their sustainability. At the same time, GAINS does not address an integral estimation of the sustainability of the NES, which would involve weighting of the indicators, since Member States may have different views on the importance of specific KIs.

4.3. SUGGESTED KEY INDICATORS AND EVALUATION PARAMETERS

Table 4.1 provides the list of suggested KIs and EPs and shows that many of these are not rigidly bound to one assessment area but can be used for the evaluation of several areas of interest. However, KIs typically have a distinctive capability for capturing the essence of a particular area, and those areas are identified by a bold '**X**' in Table 4.1. As the GAINS framework examines NESs in a global context, some repackaging of the INPRO assessment areas from Ref. [4.2] is shown in the columns of the table. The user requirement (UR2) of the environment assessment area involving resource availability and use was called out separately to give additional emphasis on resource sustainability from a global perspective. The remaining user requirement for the environment (UR1), which addresses environmental stressors, was combined with the waste management area to promote consideration of waste-induced environmental stressors related to the global system and also to simplify the assessment process and reduce the number of KIs and parameters. Similarly, the assessment areas for proliferation resistance and physical protection were also grouped together in developing the initial GAINS framework.

It should be noted that this is in no way an exhaustive list. Individual Member States may wish to expand the GAINS framework to include additional parameters of interest to them when analysing scenarios. For the initial development of the framework, it was agreed that it would be best to limit the set to only a few KIs and related EPs for assessing NESs regarding sustainable development. These indicators and parameters would be evaluated for each GAINS group (NG1, NG2, NG3) where applicable, along with a set of total values for the global system.

An initial estimation of the relative uncertainty in calculating specific values for a future NES based upon currently available information is illustrated by the colour-coding in Table 4.1. For each given KI or EP, the uncertainty levels can vary considerably depending on the maturity level of the specific NES types under consideration. This may be due to a lack of detailed design information, experimental data, operating experience or other reasons.

Ultimately, assessments using the GAINS framework should seek to take such uncertainties into consideration (appendix B of Ref. [4.2]).

Most KIs and EPs shown in Table 4.1 can be evaluated using a combination of different analytical tools and methods, but the methods and assumptions used should lead to results which are consistent across KIs and EPs.

The KIs and EPs contained in Table 4.1 are discussed further below by assessment area:

- For resource sustainability, the KI is the average net energy produced per unit mass of natural fuel resource. This is a direct measure of the efficiency of use of a limited natural resource (natural uranium or thorium). It should be noted that while the energy is typically produced years after the material is mined (especially with fuel recycle), the indicator value is based on current year values for mining and energy production. To do otherwise would require bookkeeping well beyond the capabilities of most fuel cycle codes. The EP for cumulative demand of natural nuclear material is important for comparison to the OECD/NEA and IAEA 'Red Book' [4.3] values. In developing the KIs and EPs, 'cumulative' typically refers to summed values from the year of initial civilian nuclear power generation. Simplifying adjustments or user specified inputs may need to be made or introduced into analytical codes for consistency and comparison purposes, for example, for comparing total global uranium usage amounts to currently estimated remaining reserves or for the calculation of known present-day inventories.
- For waste management and environmental stressors, two KIs are the discharged fuel inventories and radioactive waste inventories per unit energy generation. Owing to different national waste classification schemes, an approach may need to be developed to provide a common basis for NES assessment, recognizing that application of waste management practices at a local level may differ. The KIs identified for this assessment area also serve to provide an indication of waste disposal infrastructure requirements.

TABLE 4.1. KEY INDICATORS AND EVALUATION PARAMETERS FOR USE IN GAINS FRAMEWORK

No.	Key indicators and evaluation parameters		INPRO assessment areas					
	Colour coding indicative of relative uncertainty level in estimating specific quantitative values for future NESs (can vary based on a particular scenario)	Low Medium–low Medium–high High	Resource sustainability	Waste management and environmental stressors	Safety	Proliferation resistance and physical protection	Economics	Infrastructure
Power production								
KI-1	Nuclear power production capacity by reactor type							**X**
EP-1.1	(a) Commissioning and (b) decommissioning rates			X				X
Nuclear material resources								
KI-2	Average net energy produced per unit mass of natural uranium		**X**	X				
EP-2.1	Cumulative demand of natural nuclear material, i.e. (a) natural uranium and (b) thorium		X	X				
KI-3	Direct use material inventories per unit energy generated (cumulative absolute quantities can be shown as EP-3.1)		X			**X**		X
Discharged fuel[a]								
KI-4	Discharged fuel inventories per unit energy generated (cumulative absolute quantities can be shown as EP-4.1)			**X**				**X**
Radioactive waste and minor actinides								
KI-5	Radioactive waste inventories per unit energy generated[b] (cumulative absolute quantities can be shown as EP-5.3)			**X**				X
EP-5.1	(a) Radiotoxicity and (b) decay heat of waste, including discharged fuel destined for disposal			X				X
EP-5.2	Minor actinide inventories per unit energy generated			X				X

TABLE 4.1. KEY INDICATORS AND EVALUATION PARAMETERS FOR USE IN GAINS FRAMEWORK (cont.)

No.	Key indicators and evaluation parameters		INPRO assessment areas					
	Colour coding indicative of relative uncertainty level in estimating specific quantitative values for future NESs (can vary based on a particular scenario)	Low Medium–low Medium–high High	Resource sustainability	Waste management and environmental stressors	Safety	Proliferation resistance and physical protection	Economics	Infrastructure
Fuel cycle services								
KI-6	(a) Uranium enrichment and (b) fuel reprocessing capacity, both normalized per unit of nuclear power production capacity					**X**		**X**
KI-7	Annual quantities of fuel and waste material transported between groups			X		X		**X**
EP-7.1	Category of nuclear material transported between groups					**X**		
System safety								
KI-8	Annual collective risk per unit energy generation				**X**			
Costs and investment								
KI-9	Levelized unit of electricity cost						**X**	
EP-9.1	Overnight cost for nth-of-a-kind reactor unit: (a) total and (b) specific (per unit capacity)						X	
KI-10	Estimated R&D investment in nth-of-a-kind deployment						**X**	X
EP-10.1	Additional functions or benefits[c]						X	

[a] Also referred to as 'used fuel' or 'spent fuel'.
[b] Excludes discharged fuel covered by KI-4.
[c] In addition to electrical power production, e.g. high temperature process heat production or transmutation.

— EPs for radiotoxicity and decay heat per unit energy generation can influence waste disposal considerations. Radiotoxicity provides a measure of potential hazard that needs to be isolated over time. It provides an indirect indicator of potential dose from releases from a repository or from intrusion into a repository. Release or intrusion could happen at different times for different repository designs, so radiotoxicity is used as a measurable parameter common to all repository designs. Time frames up to 1 000 000 years following fuel discharge or fuel separations are targeted for assessment. Decay heat can affect repository design and limitations on loading. MA inventories can affect the long term heat load burden, in addition to impacting waste radiotoxicity.
— An area of special significance to sustainability is that of safety. Ensuring safety under normal and design basis accident conditions is important for continued operation of existing plants as well as new plant deployment. Severe accident prevention and mitigation measures may also be expected to receive increased attention in the future. GAINS (as well as other studies) envisages a multifold increase (~16 times) in installed nuclear power plant capacity over the long term along with increases in associated fuel cycle facilities. Many of these installations are expected to be in countries with high population density and substantial pressure on land. Hence, a significant fraction of the projected long term nuclear power plant installations may need to be near population centres and, thus, the safety objectives may have to be more stringent and could require innovative approaches for their fulfillment. Ideally, it would be desirable to be able to quantify risk associated with the integrated NES operations, and compare NES safety approaches on that basis. However, because of the lack of sufficiently detailed probabilistic risk data for various NES concepts and configurations, a comparative qualitative discussion on potential risk contributors when comparing different NESs will likely need to suffice initially. For the current GAINS framework, it is assumed that regulators will ensure safety and related security issues are appropriately addressed in licensing and approvals for new NES facilities in scenarios of expanded nuclear generating capacity. Mass flows used in assessing future scenarios should be based on a set of reasonable assumptions for safe facility operation, if the detailed design data for innovative NESs are absent.
— Another area of special significance in sustainability is that of non-proliferation. Enrichment and reprocessing technology has been limited to a few countries, while reactors have yet to be deployed in many countries that do not have the requisite infrastructure. Issues of proliferation concern can be expected to rise with the global growth of nuclear power, absent off-setting advances or measures. For proliferation resistance and physical protection, two high level KIs are: (i) inventories of direct use material[4] per unit energy generation, and (ii) enrichment and reprocessing capacity. An EP related to physical protection requirements associated with international transport has also been identified.
— Although additional information on the location of direct use material in the fuel cycle (e.g. in-reactor, on-site storage at reactors or fuel cycle facilities, in-repository) could be shown for this parameter as available by the fuel cycle code, it is not necessary for the high level KI value.
— Enrichment and reprocessing capacity available per installed nuclear power plant capacity provides a secondary key proliferation resistance and physical protection indicator, and also provides insight into the required level of fuel cycle infrastructure to support particular NES scenarios.
— The categorization of nuclear material transported between groups, as defined in the Convention on the Physical Protection of Nuclear Material [4.5] provides an indication of the physical protection requirements associated with synergistic fuel cycle system concepts.
— A desirable high level economics indicator is the levelized cost to produce electricity; however, there is significant uncertainty in these values depending on the particular system concept. At this stage of the study, the focus is on the R&D development cost. Agreement on ranges of costs are needed to develop values for particular NES. This parameter is not an annual value to be reported by the codes but rather a separate estimation/discussion per scenario. Economic EPs relate to affordability of capital costs for nth-of-a-kind units (the cost per unit when the technology is fully mature), and supplementary functions, or benefits, of the NES in addition to electricity generation.

[4] Direct use material as defined by the IAEA Safeguards Glossary [4.4] includes any high enriched (20% or more) uranium, any plutonium containing less than 80% ^{238}Pu and ^{233}U.

— Four KIs for infrastructure were identified: (i) nuclear power production capacity, (ii) discharged fuel inventories per unit energy generated, (iii) uranium enrichment and fuel reprocessing capacity, both normalized per unit of nuclear power production capacity, and (iv) transport of fuel and waste material between groups.

The capacity for different types of nuclear power plant provides a key indication of the required infrastructure associated with power production and fuel cycle operations. In some cases, for mature systems, the capacity levels may be dictated by load factor and fraction of overall fleet power levels, but in other cases deployed capacity may depend on timing for completion of R&D or fissile material availability. Commissioning and decommissioning rates provide insight into the required industrial supply chain and level of industrial reactor decommissioning and waste management services needed to support a given power production profile.

Discharged fuel inventories provide an indirect indication of the level of infrastructure required for either above ground facilities or geological repositories for storage and/or disposal of discharged fuel. Similarly, uranium enrichment and fuel reprocessing capacity provide an indication of the level of infrastructure needed to support fuel cycle front end and back end activities for particular scenarios.

In heterogeneous synergistic scenarios for nuclear growth, fuel or waste material may be transported between regions. A KI is needed to provide some indication of the amount and types of material transported annually between groups. While this indicator does not account for transport within a group, it does provide some indication of the required infrastructure, both physically and institutionally, associated with international transport.

Table 4.2 lists proposed units for the KIs and EPs, and additional notes to guide calculation and preparation of the KIs and EPs by analysts. Lower-tier EPs, for example fuel discharge rates for specific reactor types, are needed to support calculation of KIs and primary EPs. Sections 5 and 6 provide data for power profiles and data used to calculate KIs for the GAINS example scenarios. Spreadsheet templates used to calculate, compile and plot the KIs and EPs for individual groups and the global system are included on the CD-ROM accompanying this report.

The scope of the GAINS CP focused on establishing a framework for assessing future NESs and providing illustrative sample analyses. Within the current scope of the CP, not all KIs and EPs are evaluated in such analyses provided in Sections 7–9, especially for those with a higher degree of uncertainty as noted in Table 4.1. As additional data are collected and supporting analyses conducted in the future, a more complete application of the framework for the assessment of global architectures involving innovative NES can be realized.

4.4. CONCLUSION AND RECOMMENDATIONS

A set of KIs and EPs for comparing global architectures of NESs has been established for the GAINS framework. For calculating some KIs or parameters, data may either not always be readily available and/or may have a higher degree of associated uncertainty, depending on the particular scenario and technologies employed. For a more complete application of the framework, economics/cost data and probabilistic risk data for advanced systems should be collected as concepts and technologies mature with time and data become available. It is also recommended that uncertainties be considered when performing comparative assessments, perhaps through the introduction of an indicator tied to technology maturity level. The set of GAINS KIs and EPs, although developed for global architectures, can also be adapted for a more localized application of the framework.

TABLE 4.2. UNITS AND NOTES CONCERNING KEY INDICATORS AND EVALUATION PARAMETERS

No.	Key indicator or evaluation parameter	Units	Notes
Power production			
KI-1	Nuclear power production capacity by reactor type	GW(e)	Stacked line chart by reactor type.
EP-1.1	(a) Commissioning and (b) decommissioning rates	GW(e)/a	Optional parameter to provide indication of level of infrastructure for supply chain and decommissioning.
Nuclear material resources			
KI-2	Average net energy produced per unit mass of natural uranium	GW·a/ktHM	Annual and cumulative values calculated. Calculated by taking the end use (net) energy delivered by the NES in a particular group and dividing by the mass of uranium extracted from nature, regardless of where front end fuel cycle activities may take place to prepare the fuel. Ideally, the net energy delivered would take into account energy demands for performing fuel cycle front end and back end activities, regardless of group location. However, if fuel cycle energy demand information is not available, KI values may be developed based on the cumulative net electrical power generation from the nuclear power plants using the fuel.
EP-2.1	Cumulative demand of natural nuclear material	ktHM	Calculate for cumulative demand for: (a) natural uranium and (b) thorium.
KI-3	Direct use material inventories per unit energy generated	kg/GW·a	Calculated by dividing the cumulative mass of direct use material within the group by the cumulative net electrical energy produced by nuclear power plants. Direct use material can be shown by (a) irradiated and (b) unirradiated portions if such output is available from codes.
Discharged fuel			
KI-4	Discharged fuel inventories per unit energy generated	tHM/GW·a and m3/GW·a	Calculated by dividing cumulative quantities of discharged fuel inventories in storage or disposal by the cumulative net electrical energy produced by nuclear power plants. Discharged fuel can be broken out to show: (a) fuel emplaced in a geological repository; and (b) fuel at non-repository locations, provided common assumptions regarding fuel disposal rates are established. Discharged fuel at non-repository locations may be provided as a single value and represent the sum of discharged fuel in SF cooling pools, dry-cask storage at reactor sites, a centralized storage location or other storage facility. Volume is unpackaged volume.

TABLE 4.2. UNITS AND NOTES CONCERNING KEY INDICATORS AND EVALUATION PARAMETERS (cont.)

No.	Key indicator or evaluation parameter	Units	Notes
Radioactive waste and minor actinides			
KI-5	Radioactive waste inventories per unit energy generation	$m^3/GW{\cdot}a$ (or $kt/GW{\cdot}a$)	Calculated by dividing cumulative quantities of radioactive wastes generated (excluding discharged fuel, which is covered by KI-4 above) by the cumulative net electrical energy produced by nuclear power plants. Ideally, estimates would be provided by type for high level waste (HLW), intermediate level waste and low level waste, and be based on the primary canister volume (excluding overpacks packaging for additional shielding, shipment and/or final storage). However, if this information is not available, KI values may be provided for HLW in terms of mass (kt), where the mass value is calculated based on the mass of FPs and remaining actinides in the waste form originating from the initial heavy metal (HM) in the fuel.
EP-5.1	Radiotoxicity and decay heat of waste, including discharged fuel destined for disposal	$Sv/kW{\cdot}h$; or kW/t	Calculated for long lived isotopes of interest as a function of time from fuel discharge, based on their inventory in HLW or discharged fuel. For radiotoxicity ingestion or inhalation, dose conversion factors (ideally both types of values would be provided) and the most currently available ICRP dose conversion factors should be used. The radiotoxicity of the waste produced near the end of the scenarios (i.e. in the year 2100) is provided as a snapshot in time for simplicity.
EP-5.2	Minor actinide inventories per unit energy generated	$kg/GW{\cdot}a$	Calculated by dividing the cumulative mass of minor actinides by the cumulative net electrical energy produced by nuclear power plants.
Fuel cycle services			
KI-6	(a) Uranium enrichment and (b) fuel reprocessing capacity, per unit of nuclear power production capacity	(SWU/a)/GW(e); and (tDM/a)/GW(e)	(a) Calculated by dividing the total uranium enrichment capacity in separative work units (SWUs)/a within a given group by the total nuclear power plant capacity in GW(e) in that group. (b) Calculated by dividing the discharged fuel reprocessing capacity in tonnes of direct use material (tDM)/a within a given group by the total nuclear power plant capacity within that group.
KI-7	Annual quantities of fuel and waste material transported between groups	kt HM/a	Calculated for heterogeneous synergistic scenarios only based on fuel cycle codes. For this KI, imports to and exports from a particular group should be calculated for: (a) fresh fuel assemblies, (b) discharged fuel assemblies and (c) HLW in terms of initial HM mass. Transport of source material between groups is not required for this KI. For calculating KI values for the global system, total quantities shipped for the three types of material above is desired.
EP-7.1	Category of nuclear material transported between groups	Category (I, II or III)	Categorization of nuclear material based on the Convention on the Physical Protection of Nuclear Material [4.5].

TABLE 4.2. UNITS AND NOTES CONCERNING KEY INDICATORS AND EVALUATION PARAMETERS (cont.)

No.	Key indicator or evaluation parameter	Units	Notes
System safety			
KI-8	Annual collective risk per unit energy generation	Risk/MW·h; or qualitative discussion	Ideally, annual collective risk normalized per unit of net end use energy delivered would be provided. However, this information is not expected to be developed for initial synergistic scenarios involving future NESs, as it is very dependent upon assumptions related to the specific architectures and requires much more extensive analysis. For application of the GAINS framework, it is recommended that a qualitative discussion be provided at an overall systems level of potential risk contributors and unique risk mitigation measures when comparing NES concepts.
Costs and investment			
KI-9	Levelized unit of electricity cost	\$/MW·h	It is recognized that this cost will have significant uncertainties associated with it for the time frames being assessed, and sufficient information may not be available to support the development of estimates at this time, depending on the particular NES option.
EP-9.1	Overnight cost for nth-of-a-kind reactor	\$1 billion; \$/kW(e)	Based on best available estimates, recognizing uncertainties, such as those articulated above. Provide in terms of (a) total overnight cost and (b) specific overnight cost, i.e. per unit of installed capacity.
KI-10	Estimated R&D investment in nth-of-a-kind deployment	\$1 billion	Approximate order of magnitude of investment required for key infrastructure elements of NES architecture, including fuel cycle facility infrastructure as well as advanced nuclear power plants.
EP-10.1	Additional functions or benefits	Text providing qualitative description	Qualitative description of potential NES benefits or functions other than electrical power production. Some quantitative values to describe performance relative to potential alternatives may be used if sufficiently developed.

REFERENCES TO SECTION 4

[4.1] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Assessment of Innovative Nuclear Reactors and Fuel Cycles: Report of Phase 1B (First Part) of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1434, IAEA, Vienna (2004).

[4.2] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems, INPRO Manual — Overview of the Methodology, IAEA-TECDOC-1575 Rev. 1, IAEA, Vienna (2009).

[4.3] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, NUCLEAR ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Uranium 2009: Resources, Production and Demand, OECD, Paris (2010).

[4.4] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safeguards Glossary — 2001 Edition, International Nuclear Verification Series No. 3, IAEA, Vienna (2002).

[4.5] The Convention on the Physical Protection of Nuclear Material, INFCIRC/274/Rev. 1, IAEA, Vienna (1980).

5. ESTIMATIONS OF NUCLEAR ENERGY DEMAND

5.1. PURPOSE

Section 5 introduces a basis for the nuclear energy demand curves in the twenty-first century to be used in the GAINS framework. There is a wide range of uncertainty in the overviewed projections associated with long term estimates of the demand. It is not possible within the huge scope of these options to make any definite judgement on the issues related to the project's objectives. The first purpose of the section is to define high and moderate nuclear energy scenarios for the GAINS study that would circumscribe an area of interest of the project participants and allow a basis for performing analyses on various global NES architectures under assumed boundary conditions. Sections 5.2–5.3 describe the GAINS approach to the estimation of the global nuclear energy demand. In time, it may be expedient to consider global nuclear power deployment scenarios with higher or lower rates of nuclear power deployment than those selected in the study.

In accordance with the requirements of the heterogeneous model of the global NES developed by the participants of the project (Section 3), the nuclear power demand has to be estimated for each nuclear energy strategy group comprising the global system. Thus, the second purpose of the section is to present the procedure for estimating nuclear power demand profiles in these groups (Section 5.4).

5.2. ANALYSIS OF LONG TERM ENERGY AND ELECTRICITY DEMAND

Nuclear power is an integral part of the energy sector. Therefore, similar to other energy options, its deployment depends on factors such as primary energy demand, electricity demand, environmental constraints and progress in technological development. Analysis of these factors was performed by the Intergovernmental Panel on Climate Change (IPCC) [5.1], International Institute for Applied Systems Analysis [5.2], International Energy Agency [5.3], OECD [5.4], IAEA [5.5], United States Department of Energy/Energy Information Administration [5.6], and by many other organizations.

Primary energy consumption is one of the fundamental factors in projecting future nuclear energy demand. According to recent scenarios, the range of expected primary energy demand in the long term perspective varies considerably because of the uncertainty in future conditions and driving forces that define need in energy. The story lines based on different assumptions of demographic, social, economic, technological and environmental developments result in divergent trends of energy consumption, from exponential growth to stabilization near existing level.

The graph in Fig. 5.1 illustrates scenarios for global energy demand and supply in the twenty-first century provided by the IPCC in a Special Report on Emissions Scenarios (SRES) [5.1].

A variety of future socioeconomic conditions and technology development was reflected in the study by four story lines which are labelled A1, A2, B1 and B2. Economic objectives dominate in the 'A' story lines, while environmental objectives dominate in the 'B' story lines. As shown in Fig. 5.1, variation of values of the projected global primary energy consumption by 2100 under different assumptions is very significant. Nevertheless, a majority of the evaluations made by different energy organizations and experts predict that global primary energy use will continue to grow.

A large set of evaluations related to estimating long term energy demand, as one of the key factors in projecting future greenhouse gas emissions, was compiled in a comprehensive database [5.7]. A median of the global annual energy demand in the scenarios of the database increases in the mid-century from the present ~500 EJ (~16 TW·a) by a factor of two (up to ~1000 EJ or ~32 TW·a) and at the end of the century by a factor of four (up to ~2000 EJ or ~64 TW·a) (orange line in Fig. 5.2). This median scenario is considered as a reference trend in the report.

The estimators are unanimous that the rate of electricity demand will be higher than those of primary energy demand (roughly twice as large). This is an important conclusion for nuclear power, which is a well proven option for electricity production.

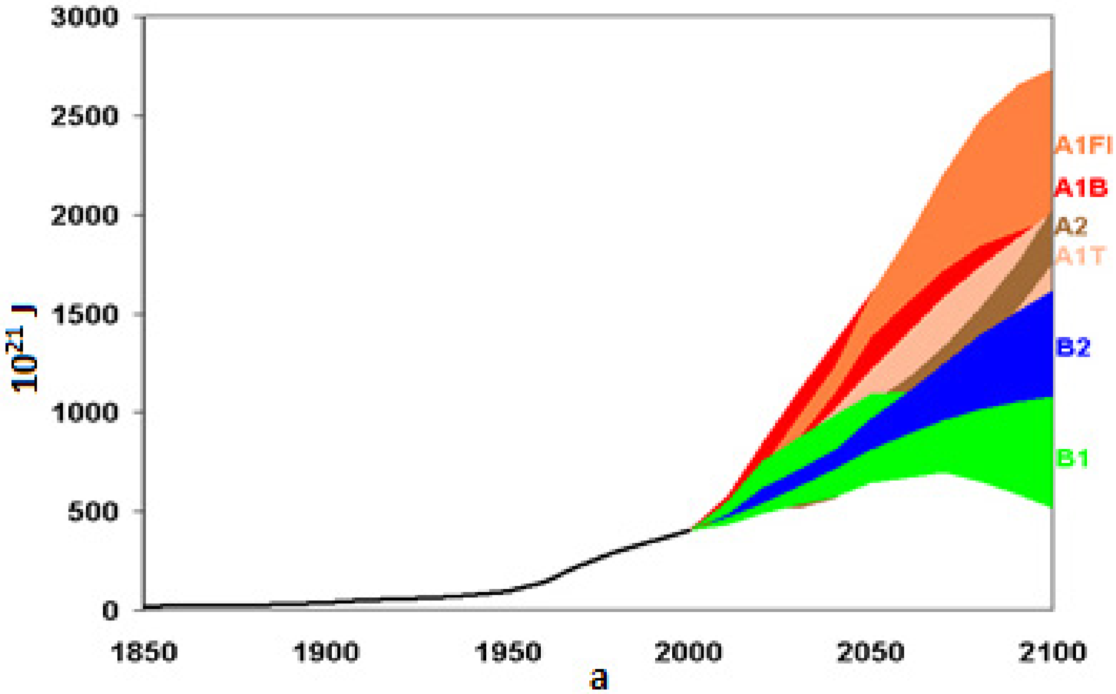

FIG. 5.1. Range of future global annual primary energy demand in SRES scenarios, 2000–2100, IPCC [5.1].

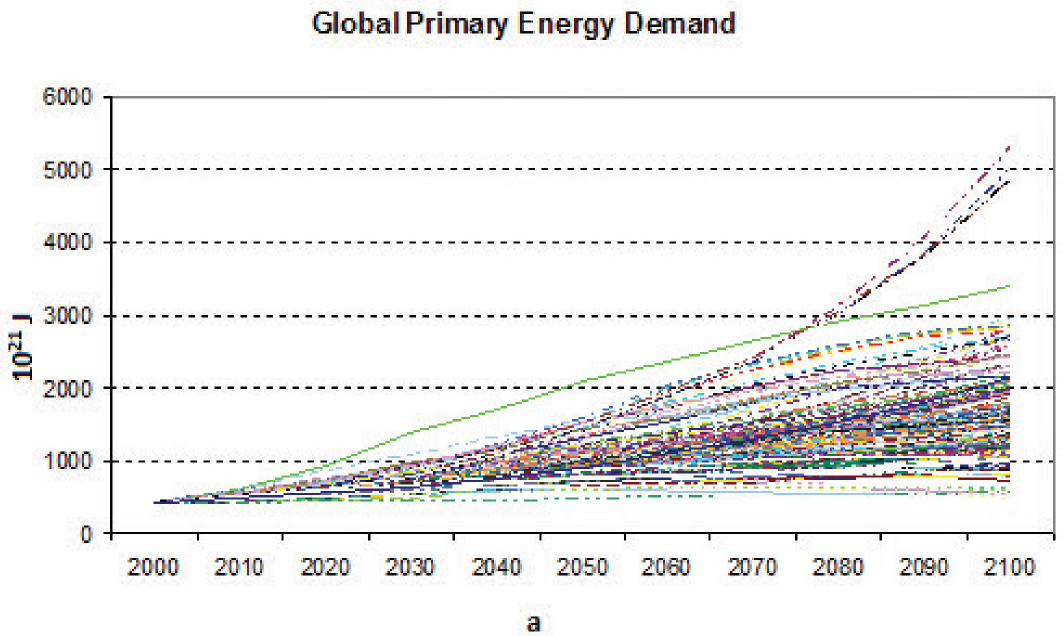

FIG. 5.2. Spread of the projections for future global annual primary energy demand from Ref. [5.7] and a reference median trend.

5.3. NUCLEAR POWER DEMAND ESTIMATIONS

A century forecast for nuclear energy demand is even more uncertain than demand for primary energy. High estimations predict a huge annual demand of about 1.9×10^{21} J by the end of the century, while in the lowest evaluations, nuclear power is expected to be phased out. It was decided to select, within the wide range of uncertainty of the overviewed projections, high and moderate scenarios for GAINS that would circumscribe an area of concern for the project participants and help them to draw specific conclusions on the arrangement of a nuclear architecture within this area.

Along with examination of nuclear projections based on macroeconomic studies of competent energy agencies and organizations, GAINS examined national middle term and long term nuclear strategies and programmes that provide another viewpoint on the nuclear energy future. The medium term period of nuclear power deployment is comprehensively covered by the IAEA projections [5.5].

Every year since 1981, the IAEA has published two updated medium term projections for the world's nuclear power generating capacity, a low projection and a high projection. The low projection is a down to earth, BAU projection. It assumes that nuclear investment projects currently underway or firmly in the pipeline are implemented, but not much more; that existing plants are retired as scheduled unless licence extensions have been granted or applied for; and that current policies are unchanged, such as the German and Belgian phase-outs of nuclear power. The high projection takes into account government and corporate announcements about longer term plans for nuclear investments as well as potential new national policies, for example, to combat climate change.

An example of the projections of this kind is shown in Fig. 5.3. In the low projection, the projected nuclear power capacity in 2030 is 473 GW(e), some 27% higher than today's 377 GW(e). In the high projection, nuclear capacity in 2030 is 748 GW(e), double today's capacity.

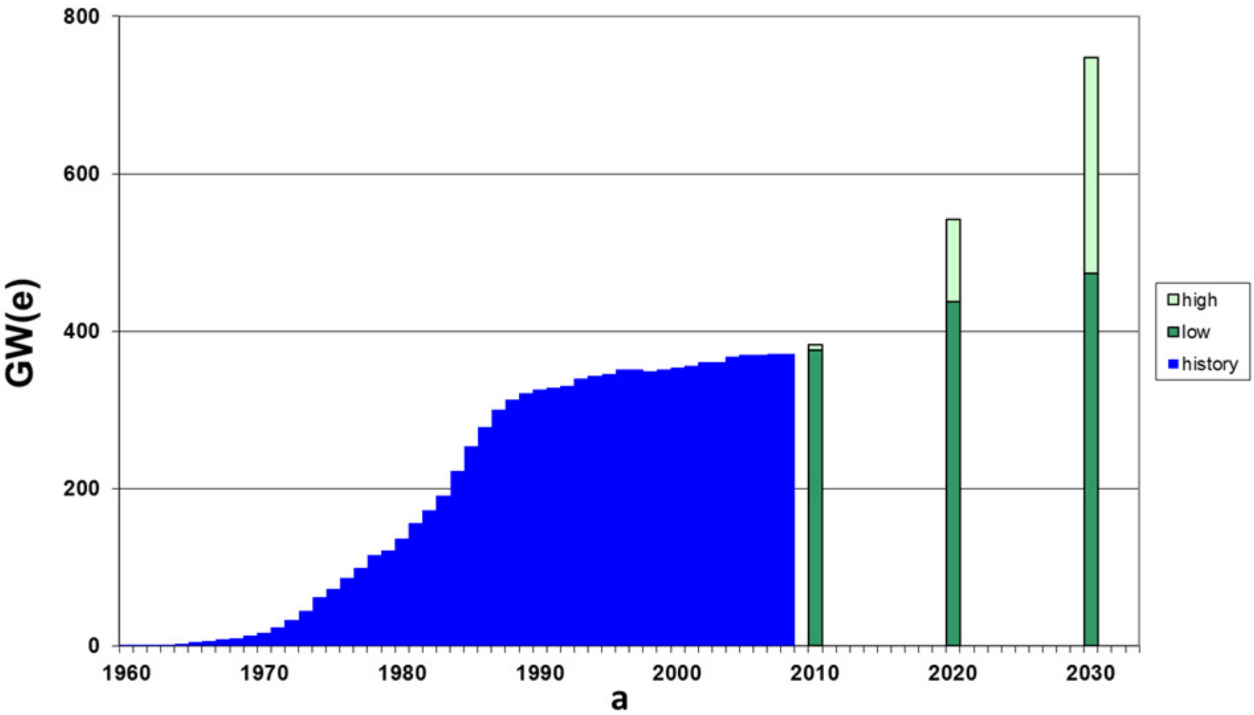

FIG. 5.3. IAEA projections for the world's nuclear power generating capacity to 2030.

The target of 700 GW(e) by 2030 was selected for nuclear power generating capacity as the high GAINS scenario, while 600 GW(e) should reflect expectations of the moderate GAINS scenario. Both of the values are within the IAEA projection envelope.

Some studies fulfilled in the framework of IAEA projects have provided a guide for construction of nuclear scenarios beyond the middle of the century. The survey undertaken by the participants of the INPRO joint study [5.8, 5.9] summarizes the long term energy strategies of Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation and Ukraine. Naturally, the social and economic conditions in these countries, that together account for more than half of the world's population and use large amounts of energy, differ to a great extent. There is not much similarity in either the history or current status of nuclear power. Specific local conditions result in significant differences in expected rates of nuclear growth. Nevertheless, there was a consensus between the representatives of IAEA Member States participating in this study on the prospects of nuclear power in the twenty-first century. Most Member States have long term strategies and scenarios of nuclear power deployment.

Figure 5.4 illustrates nuclear projections for France (stabilization of installed nuclear capacity at the level of ~60 GW(e) since the year 2000) and India (fast growth of nuclear capacity up to 275 GW(e) by the middle of the century).

(a) (b)

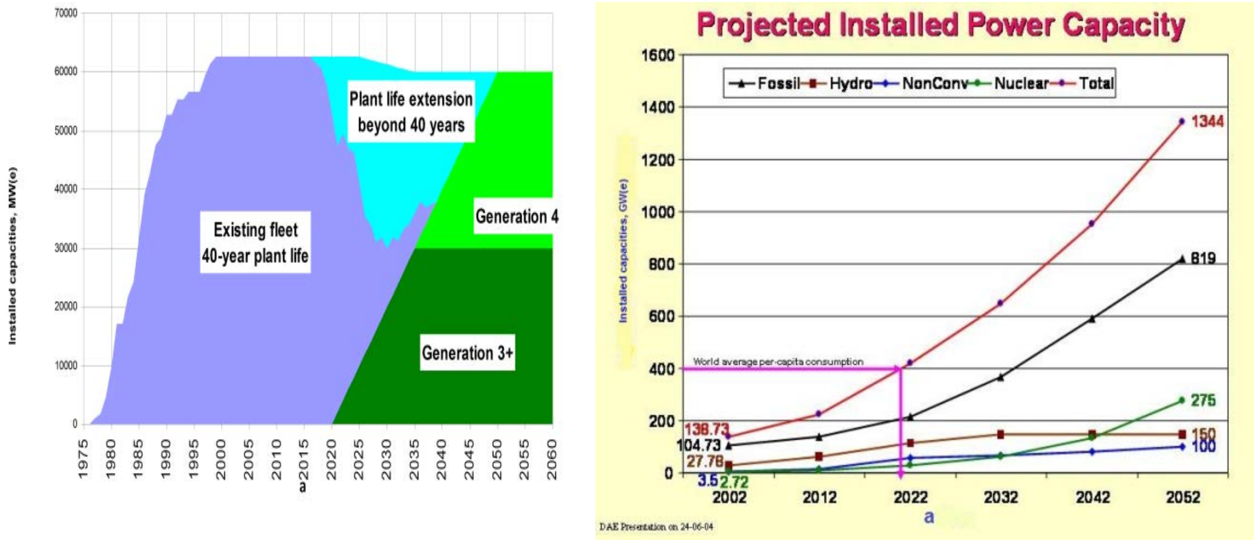

FIG. 5.4. Projected trend of installed nuclear capacity in (a) France and (b) India.

After comprehensive analysis and intensive discussions of the available projections on the nuclear demand in the twenty-first century, the GAINS participants selected two nuclear energy demand scenarios [5.10], as illustrated in Fig. 5.5:

— A high nuclear energy demand scenario, a variant of the medium expectation of the IPCC SRES. In this scenario, global annual nuclear energy generation reaches approximately 1500 GW(e)/a[5] by the mid-century and 5000 GW(e)/a by 2100.
— A moderate nuclear energy demand scenario, assuming approximately 1000 GW(e)/a by mid-century, and 2500 GW(e)/a by the end of the century.

If a constant thermal power profile is desired to analyse variations in contributions to non-electrical and electrical power demand, these scenarios can be used to construct a set of companion thermal power (GW(th)) demand profiles by applying an assumed thermal-to-electric efficiency conversion value.

5.4. ASSUMED NUCLEAR POWER DEMAND PROFILES IN GAINS NUCLEAR STRATEGY GROUPS

5.4.1. Main steps in modelling nuclear power demand profiles for GAINS groups

Modelling of nuclear power demand profiles for nuclear energy strategy groups (group nuclear demand scenarios) is a rather new and challenging task. The procedure developed by the GAINS community included the following steps:

— Composing a list of nuclear users for a century prospect;
— Group definition and splitting current/future nuclear countries into NGs;
— Analysis of long term projections of nuclear deployment in each country;

[5] GW(e)/a is equivalent to 1 GW(e) of capacity operated at 100% for a full year.

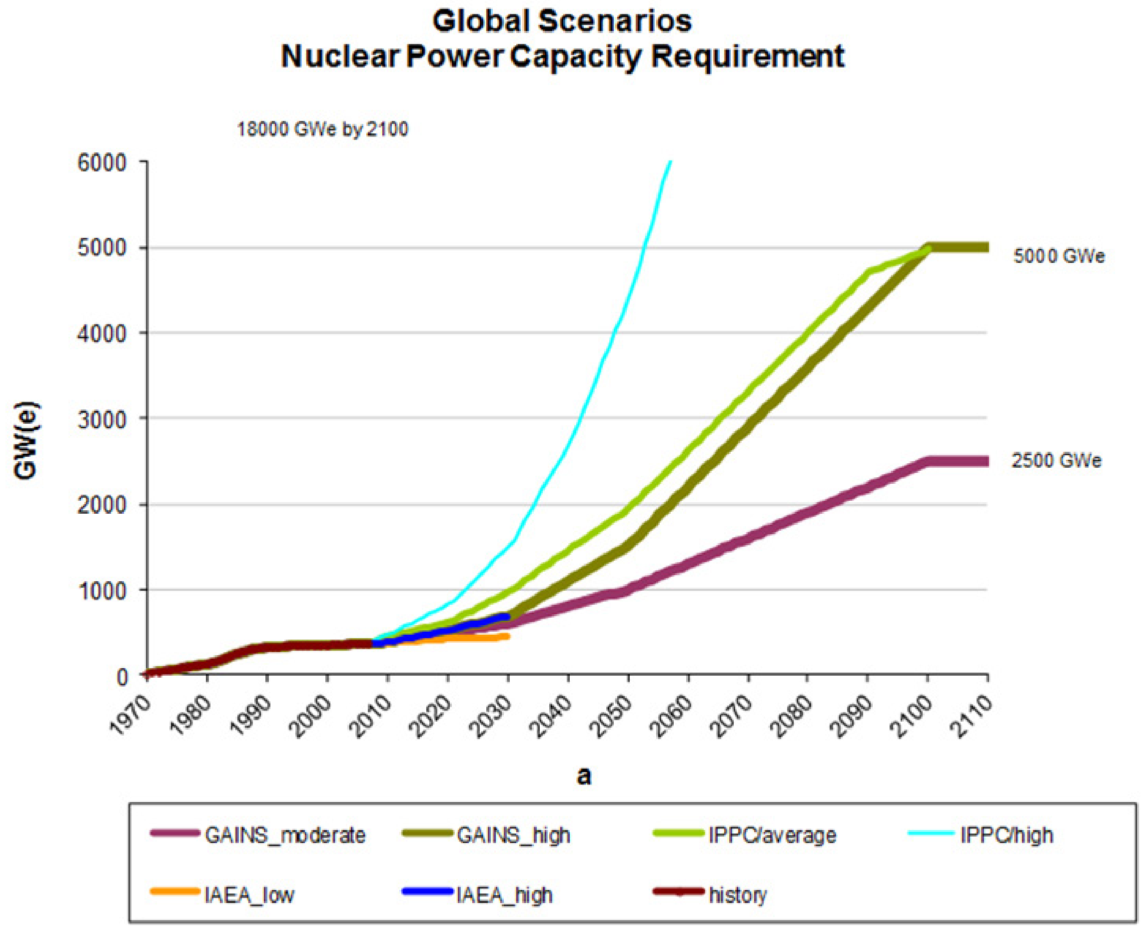

FIG. 5.5. GAINS scenarios for modelling nuclear power generation (values shown are actual power produced not capacity).

— Consolidation of the projections in GAINS groups;
— Definition of a nominal case: summing up of the countries' scenarios inside groups and adjusting results to global GAINS baselines of nuclear demand;
— Variation of nuclear demand scenarios in the nominal case groups for sensitivity analysis;
— Development of scenarios for introduction of the components for an innovative nuclear system.

The substance of each step and difficulties arising in realization of the approach are discussed below.

5.4.2. List of nuclear users for a century prospect

Composing a list of nuclear users for a century prospect is one of the most difficult and sensitive problems in assessing nuclear power demand profiles for NGs.

First of all, with an aim to prepare a comprehensive background for the analysis, the members of the INPRO international group in the IAEA which supports activities of GAINS, and the IAEA's experts which consulted them had to make an extensive search of documents and publications related to a century's projection for nuclear power demand in individual countries. The set of surveyed sources of information included:

— Government statements at the IAEA General Conferences;
— Estimations of energy, electricity and nuclear power capacities deployment on a national level for the period up to 2030 as provided by the IAEA;
— Consideration of long term plans, strategies and expectations on deployment/entering nuclear power worldwide provided by IAEA Member States and competent energy agencies;
— Examination of relative studies of competent groups.

The World Nuclear Association's (WNA) Nuclear Century Outlook gives [5.11] an example of a long term projection designed to gauge the prospects for the worldwide growth of nuclear power in the twenty-first century. An excerpt from this outlook is presented in Table 5.1. The full list in Ref. [5.11] consists of 65 countries. As can be seen from the table, the uncertainty of the projections for individual countries has a tendency to increase with time, resulting in an enormous range by the end of the century. Similar to the situation with global projections, it can be explained by the choice of different paradigms in realization of national social, economic and environmental goals. In many countries, nuclear power has become a subject of acute public discussion and political strife that also adds to the future uncertainty.

5.4.3. Group definition and consolidation of nuclear projections in groups

In accordance with requests of the heterogeneous model approach, presented in Section 3, the countries that use or intend to use nuclear energy are to be split into several groups by their approach to the strategy of NES development and deployment. The nuclear demand for each group has to be identified. This task was solved by taking into account information on national expectations of nuclear power development, the nuclear energy strategic group definition and the judgement of the GAINS participants and experts from the IAEA. The composition of NGs for a century's prospect made on an expert basis was only used as internal working material for the project since some countries have not defined plans for nuclear power introduction/deployment or for future nuclear infrastructure and they are very sensitive about this issue.

The results of consolidation of the nuclear projections for three groups in the case of the moderate and high GAINS scenarios of nuclear energy demand are summarized in Table 5.2.

In the nominal case, the shares of nuclear energy generation in groups related to the total nuclear energy generation by the year 2100 are:

— 40% in NG1 (recycle based architecture);
— 40% in NG2 (mature once-through nuclear cycle architecture);
— 20% in NG3 (elements of once-through nuclear cycle architecture).

Variations of these shares were also developed for possible use in sensitivity studies.

5.4.4. Profiles for introduction of nuclear energy system components

To a certain extent, development of long term profiles for introduction of NES components in a multigroup splitting of the world is not an established procedure. When entering this area, GAINS was faced with a lack of necessary data and reliable projections. As was noted earlier, uncertainty of long term plans, strategies and political decisions on development and deployment of innovative NESs is one of the main obstacles in this area. Assumptions made in GAINS for building of NESs in different nuclear energy strategy groups are discussed below.

The milestones for the introduction of the main innovative components of the GAINS NESs are defined in Table 1.1. In some cases, there is no need for much additional information to build a scenario for their deployment after startup. For instance, assuming that operating commercial reactors of a certain type, upon reaching their end of life, are replaced by advanced commercial reactors of the same type, it is quite easy to build a scenario of commissioning, operation and decommissioning of the reactors of the type that would meet given demand.

For the substantially innovative components of the NESs, it is expedient to define not only the time of their commissioning but also their rate of deployment for the first period of operation when their introduction is limited by unavailability of appropriate material and infrastructure/industry capacity. The assumptions for the introduction of innovative components of NESs selected for this study and related issues are discussed in Sections 6–9.

TABLE 5.1. EXCERPT FROM THE WNA NUCLEAR CENTURY OUTLOOK ON THE PROSPECTS FOR THE WORLDWIDE GROWTH OF NUCLEAR POWER IN THE TWENTY-FIRST CENTURY [5.11]

	Capacity by year (GW(e))						
Major nuclear programmes	2008	2030 Low	2030 High	2060 Low	2060 High	2100 Low	2100 High
China	9	50	150	150	750	500	2800
France	63	65	75	80	110	80	130
India	4	20	70	60	350	200	2750
Japan	48	55	70	80	140	80	200
Russian Federation	22	45	80	75	180	100	200
United Kingdom	11	20	30	30	80	40	140
USA	99	120	180	150	400	250	1200
SUBTOTAL	363	531	951	887	2538	1627	8443
Smaller nuclear programmes	2008	2030 Low	2030 High	2060 Low	2060 High	2100 Low	2100 High
Argentina	1	4	11	5	30	10	90
Armenia	0	1	0	1	1	2	4
South Africa	2	10	25	30	50	30	55
SUBTOTAL	4	30	86	64	251	102	694
Nations planning nuclear	2008	2030 Low	2030 High	2060 Low	2060 High	2100 Low	2100 High
Egypt	0	3	10	6	40	10	90
Indonesia	0	2	6	3	35	5	175
Kazakhstan	0	0	2	3	5	5	20
Turkey	0	5	15	10	50	20	160
Vietnam	0	2	4	4	30	6	120
SUBTOTAL	0	30	112	78	300	126	910
Potential entrants	2008	2030 Low	2030 High	2060 Low	2060 High	2100 Low	2100 High
Italy	0	7	20	10	40	25	70
Portugal	0	0	5	5	10	5	14
Other	0	0	8	4	40	20	200
WORLD TOTAL	367	604	1289	1140	3538	2062	11 046

TABLE 5.2. NOMINAL SCENARIO OF NUCLEAR ENERGY GENERATION IN THE GAINS NUCLEAR ENERGY STRATEGY GROUPS FOR HIGH AND MODERATE GAINS DEMAND

Nuclear energy strategy groups (NGs)	GW(e)/a						
	2008	2030 Moderate	2030 High	2050 Moderate	2050 High	2100 Moderate	2100 High
NG1	149	285	333	455	682	1000	2000
NG2	149	285	333	455	682	1000	2000
NG3	0	30	34	90	136	500	1000
WORLD TOTAL	298	600	700	1000	1500	2500	5000

5.5. CONCLUSION

Analysis of a number of international scenario studies with goals similar to those set out for the GAINS CP (Section 2) and the experience of the GAINS participants in modelling of various NESs indicate a dependence of an NES architecture on forecasts of nuclear energy demand. Thus, estimation of nuclear energy demand during the twenty-first century globally and for groups of countries with similar fuel cycle approaches (NGs) was an essential task within the objectives of the project. A special procedure was developed in close cooperation with the IAEA to define nuclear power demand profiles for NGs and globally. High and moderate nuclear energy scenarios developed for the GAINS study circumscribed an area of interest of the project and allows a basis for performing analyses on various global NES architectures under assumed boundary conditions. The estimation of nuclear power demand profiles is part of the GAINS framework and should be considered only as an illustrative case based on expert opinions.

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6. NUCLEAR ENERGY SYSTEMS ADOPTED FOR ANALYSIS IN GAINS

6.1. BACKGROUND

This section provides data and assumptions related to the NESs used to supply energy demand in the GAINS framework. Different types of systems are considered, such as LWRs and FRs. Variations within a given system type are also considered, for example LWR variations based on fuel burnup, or FR variations based on fuel burnup or BR. Section 6 provides the data of different reactor systems and the conditions of fuel cycle systems, which can be used in the scenario studies or sensitivity analyses in Sections 7–9.

Concerning nuclear reactor systems, the features of fuel burnup performances and refuelling data of each reactor concept linked to global mass flow analysis are described. As for the NFC systems, the basic flows of typical systems and some important conditions which affect mass flow analysis results are also described.

6.2. REACTOR SYSTEMS

In this section, major specifications which are linked to the mass flow analysis for scenario studies are described for each reactor system. For advanced or innovative concepts yet to be developed fully, the maturity level of the technology is outlined.

6.2.1. Light water reactors

Data and assumptions for three variations of LWRs are provided: a low burnup (45 GW·d/t (HM)) LWR designated as ‘L1’, a high burnup (60 GW·d/t) LWR designated as ‘L2’ and a medium burnup (51 GW·d/t) LWR designated as ‘L3’. The low burnup and high burnup LWR models represent, respectively, historical and future LWR systems along the analysis period for nuclear power. When performing scenario analyses in Sections 7–9, the historical period of nuclear power is typically calculated to establish the initial conditions for analysing future scenarios. However, analysts may at times prefer to use a single average burnup LWR to simplify modelling, particularly when examining effects that are not that dependent upon LWR input assumptions. Preliminary analyses were conducted in GAINS using the low burnup LWR throughout the analysis period, the results of which are presented and discussed in the next section. The medium burnup LWR provides another option for use over the analysis period.

6.2.1.1. Low burnup light water reactor (‘L1’)

A typical PWR design is assumed with an average burnup of 45 GW·d/t, a fuel enrichment of 4%, a specific power density of 38.5 MW/t and a load factor of 85%.

The major specifications, based on current PWR operation data, of ‘L1’ are shown in Table 6.1. The specific power density was matched with the typical fuel residence time and average discharged burnup typical of PWRs operated to this point.

The composition data of fresh fuel and discharged fuel were calculated using the nuclear fuel cycle simulation system (NFCSS) code burnup calculation routine. The composition data are shown in Table II–1 of Annex II. As shown in Annex II, the compositions are given for the initial core loading, equilibrium core fresh fuel loading and discharged fuel, and retirement full core discharge. The composition data of discharged fuel is immediately following discharge from the reactor. Thus, the composition change during cooling, storage and processing periods should be suitably calculated using analytical tools.

TABLE 6.1. MAJOR SPECIFICATIONS OF LOW BURNUP LWR (‘L1’)

Reactor net electric output	MW	1000
Reactor thermal output	MW	3030.3
Thermal efficiency	%	33.000
Average load factor	%	85
Operation cycle length	EFPD	292
No. of refuelling batches[a]		4
Fuel residence time[a]	EFPD	1168
Specific power density[a]	MW/t	38.527
Average discharged burnup[a]	MW·d/t	45 000
Initial core inventory	tHM	78.653
Equilibrium loading	tHM/a	20.892

[a] Equilibrium cycle averaged data.
Note: Average load factor is an input assumption that may be adjusted by analysis when performing scenario analyses. EFPD: effective full power day.

6.2.1.2. High burnup light water reactor (‘L2’)

High burnup LWR (advanced LWR, also referred to as advanced light water reactor (ALWR)) fuel designs have been proposed for economic considerations. The reactor parameters of the ALWR investigated in GAINS are based on design data provided by the representatives of France.

Table 6.2 shows the major specifications of the ALWR. In the table, the specifications correspond to a 1500 MW(e) reactor. In comparison with the specifications of the low burnup LWR (Table 6.1) for the same total reactor power, the equilibrium loading per year of the ALWR is 30% smaller than that of L1 due to the longer fuel residence time and higher average discharged burnup, although the fuel inventory of the ALWR is 10% larger (corresponding to a 10% lower specific power density) than that of L1. The reduction of fuel loading in the equilibrium cycle directly is why the ALWR is more economical than L1.

The composition data for refuelling are shown in Table II–2 of Annex II. Initial core fuel enrichment is 3.40% and the equilibrium core fuel enrichment is 4.95%.

6.2.1.3. Medium burnup light water reactor (‘L3’)

The medium burnup LWR represents a typical PWR in the USA operating on UOX fuel and using recent average burnup levels. The data are based on a three-batch PWR, but have been recalculated to reflect five batches, each one calendar year in length, to better match with simulation models. This reactor/fuel can be used in sensitivity analyses, providing a burnup between that of L1 and L2.

Table 6.3 shows the major specifications of a medium burnup LWR. Although the average load factor of 90% is typical for the LWR fleet in the USA over the last decade, the table assumes the same load factor of 85% as the other reactor types. The composition data for refuelling are attached in Table II–3 of Annex II.

TABLE 6.2. MAJOR SPECIFICATIONS OF HIGH BURNUP LWR (‘L2’) (ALSO REFERRED TO AS ‘ALWR’)

Reactor net electric output	MW	1500
Reactor thermal output	MW	4410
Thermal efficiency	%	34.014
Average load factor	%	85
Operation cycle length	EFPD	440
No. of refuelling batches[a]		4
Fuel residence time[a]	EFPD	1760
Specific power density[a]	MW/t	34.091
Average discharged burnup[a]	MW·d/t	60 000
Initial core inventory	tHM	129.360
Equilibrium loading	tHM/a	22.803

[a] Equilibrium cycle average.
Note: EFPD: effective full power days.

TABLE 6.3. MAJOR SPECIFICATIONS OF MEDIUM BURNUP LWR (‘L3’)

Reactor net electric output	MW	1000
Reactor thermal output	MW	2941
Thermal efficiency	%	34.000
Average load factor	%	85
Operation cycle length	EFPD	505
No. of refuelling batches[a]		3
Fuel residence time[a]	EFPD	1513.8
Specific power density[a]	MW/t	33.69
Average discharged burnup[a]	MW·d/t	51 000
Initial core inventory	tHM	87.301
Equilibrium loading	tHM/a	17.892

[a] Equilibrium cycle average.
Note: EFPD: effective full power days.

6.2.2. Conventional heavy water reactor (‘H1’)

A nominal HWR design, which has an averaged burnup of 7 GW·d/t, natural uranium loading, a specific power density of 24.0 MW/t and a load factor of 85%, was selected as the representative of all HWRs. The major specifications of the reactor are shown in Table 6.4. Current CANDU-6 reactors routinely achieve a burnup of 7.5 GW·d/t and ~33% electrical efficiency. The composition data of the fresh fuel and discharged fuel of HWR are shown in Table II–4 of Annex II. The composition data were calculated in the same manner as for the LWR by using the NFCSS code.

TABLE 6.4. MAJOR SPECIFICATIONS OF CONVENTIONAL HWR (‘H1’)

Reactor net electric output	MW	600
Reactor thermal output	MW	2000
Thermal efficiency	%	30.000
Average load factor	%	85
Operation cycle length	EFPD	292
No. of refuelling batches[a]		Continuous refuelling
Fuel residence timea	EFPD	292
Specific power density[a]	MW/t	23.973
Average discharged burnup[a]	MW·d/t	7000
Initial core inventory	tHM	83.429
Equilibrium loading	tHM/a	88.643

[a] Equilibrium cycle average.
Note: EFPD: effective full power days.

6.2.3. Fast reactors

In order to study an FR introduction scenario, different FR designs were provided by the representatives of India, Japan, the Russian Federation and the USA, selected based on BR and burnup, and recompiled in a common data form for the NFC analyses in GAINS:

— ‘F1’: a ‘break-even’ (BR: ~1.0) FR.
— ‘F2’: a medium-BR (BR: ~1.2), medium burnup (~31 GW·d/t) FR.
— ‘F3’: a medium-BR (BR: ~1.2), high burnup (~54 GW·d/t) FR.
— ‘F4’: a Burner FR (CR: ~0.75), medium burnup (~100 GW·d/t) FR.

6.2.3.1. Break-even fast reactor (‘F1’) (demonstration type)

A break-even FR was to be investigated first in the FR introduction scenario study. For this purpose, a demonstration FR design based on the BN-800 of the Russian Federation was selected as a break-even FR. This reactor has a BR close to 1.0. Corresponding to the reprocessing strategy planned in many countries, in which core fuel and radial blanket subassemblies are dissolved together and reprocessed at the same time, and to make the input for fuel cycle calculation codes more convenient to calculate, the reactor parameters of the FR were to be homogenized to a one-region core such as LWRs or HWRs. Table 6.5 shows the major specifications of the break-even FR.

6.2.3.2. Medium breeding ratio, medium burnup, breeder fast reactor (‘F2’) (prototype)

As an alternative design for the break-even FR in the first stage of GAINS, a prototype breeder FR from India was investigated. The reactor parameters of the breeder FR were based on the prototype fast breeder reactor design of India, which has a BR of 1.16. Table 6.6 shows the major specifications of the breeder FR. As shown in the table, the whole reactor averaged parameters are obtained by homogenizing core, axial blanket and radial blanket in the same manner as the break-even FR previously mentioned.

TABLE 6.5. MAJOR SPECIFICATIONS OF BREAK-EVEN FR (‘F1’) (DEMONSTRATION TYPE)

Reactor net electric output	MW	870		
Reactor thermal output	MW	2100		
Thermal efficiency	%	41.43		
Average load factor	%	85		
Operation cycle length	EFPD	140		
		Core	Axial blanket	Radial blanket
Power share of each region[a]	%	94.5	3.0	2.5
No. of refuelling batches[b]		3	3	3.5
Fuel residence time[b]	EFPD	420	420	490
Specific power density[a]	MW/t	157.00	11.465	8.532
Average discharged burnup[a]	MW·d/t	65939	4815	4181
Thermal power of each region[a]	MW	1984.5	63.0	52.5
Heavy metal weight share				
Intial core and full core discharge	%	52.0	22.6	25.4
Equilibrium refuelling	%	54.0	23.5	22.5
Average burnup of whole core[a]	MW·d/t	37677		
Average residence time of whole core[a]	EFPD	435.771		
Average power density of whole core[a]	MW/t	86.462		
Initial core inventory	tHM	24.288		
Equilibrium loading	tHM/a	17.292		

[a] Equilibrium cycle average.
[b] Half of radial blanket fuel assemblies have three refuelling batches, the other half four refuelling batches.
Note: EFPD: effective full power days.

As shown in the table, the whole reactor averaged parameters were obtained by weighting with power share, fuel residence period, HM mass and achieved burnup of each region of core, axial blanket and radial blanket. Table II–5 in Annex II shows the homogenized composition data of fresh fuel and discharged fuel provided by the Russian Federation.

Table II–6 of Annex II shows the homogenized composition data of fresh fuel and discharged fuel provided by India.

6.2.3.3. Medium breeding ratio, high burnup, breeder fast reactor (‘F3’) (commercial type)

Table 6.7 shows the major specifications of commercial FR data provided by Japan. The breeder FR has a BR of 1.20. The design work assumes an advanced cladding material of ODS (oxide dispersion strengthened) steel being developed to achieve high irradiation resistance. This reactor has very good economics — a high average discharged burnup of 54 GW·d/t (core and blanket average) and a long fuel residence time. Due to the long residence time, this reactor would be most suitable for slowly growing FR fleets. The composition data for refuelling are shown in Annex II, Table II–7. Unlike ‘F1’ and ‘F2’, the fresh fuel contains around 1% MA, because MA recycling to FRs is one of the design conditions for the commercial FR.

TABLE 6.6. MAJOR SPECIFICATIONS OF MEDIUM BR, MEDIUM BURNUP, BREEDER FR (‘F2’) (PROTOTYPE)

Reactor net electric output	MW	500		
Reactor thermal output	MW	1250		
Thermal efficiency	%	40.00		
Average load factor	%	85		
Operation cycle length	EFPD	180		
		Core	Axial blanket	Radial blanket
Power share of each region[a]	%	92	3	5
No. of refuelling batches[a]		3	3	5
Fuel residence time[a]	EFPD	540	540	900
Specific power density[a]	MW/t	141.590	7.191	4.467
Average discharged burnup[a]	MW·d/t	76459	3883	4020
Thermal power of each region[a]	MW	1150	37.5	62.5
Heavy metal weight share				
Intial core and full core discharge	%	29.72	19.08	51.20
Equilibrium refuelling	%	37.37	24.00	38.63
Average burnup of whole core	MW·d/t	31061		
Average residence time of whole core	EFPD	679.06		
Average power density of whole core	MW/t	45.740		
Initial core inventory	tHM	27.328		
Equilibrium loading	tHM/a	12.486		

[a] Equilibrium cycle average.
Note: EFPD: effective full power days.

6.2.3.4. Burner fast reactor (‘F4’)

A burner FR complements the breeder FRs by providing the option where the fast spectrum system is used to consume transuranics, both plutonium and MAs, to support waste management and proliferation resistance objectives. Unlike the other FRs in this section, the burner FR has no blankets for breeding and the fresh fuel contains MAs. Table 6.8 shows the major specifications for a burner FR with a fissile conversion ratio (CR) (or BR) of BR = 0.86 and an overall TRU CR (fissile and fertile TRU isotopes) of CR = 0.75. The equilibrium composition data for refuelling are shown in Table II–8 of Annex II and are based on reprocessing of the FR fuel with additional make-up transuranics provided from reprocessing of LWR fuel. The fuel has a lower uranium content to suppress breeding, resulting in a smaller core inventory and a high average burnup of ~100 GW·d/t. Owing to the high TRU content of the fuel and associated decay heat, metal fuel is used to facilitate reprocessing at elevated temperatures using electrochemical processes involving molten salts. The data were provided by the USA [6.1].

TABLE 6.7. MAJOR SPECIFICATIONS OF MEDIUM BR, HIGH BURNUP, BREEDER FR ('F3') (COMMERCIAL TYPE)

Reactor net electric output	MW	1500		
Reactor thermal output	MW	3570		
Thermal efficiency	%	42.017		
Average load factor	%	85		
Operation cycle length	EFPD	540		
		Core	Axial blanket	Radial blanket
Power share of each region*	%	89.3	8.6	2.1
No. of refuelling batches[a]		4	4	4
Fuel residence time[a]	EFPD	2160	2160	2160
Specific power density[a]	MW/t	70.115	5.476	1.763
Average discharged burnup[a]	MW·d/t	151448	11828	3807
Thermal power of each region	MW	3188.01	307.02	74.97
Heavy metal weight share				
Intial core and full core discharge	%	31.56	38.92	29.52
Equilibrium refuelling	%	31.56	38.92	29.52
Average burnup of whole core	MW·d/t	53526		
Average residence time of whole core	EFPD	2160		
Average power density of whole core	MW/t	24.780		
Initial core inventory	tHM	144.07		
Equilibrium loading	tHM/a	20.693		

[a] Equilibrium cycle average.
Note: EFPD: effective full power days.

6.2.4. Other advanced reactor concepts in a uranium–plutonium system

In GAINS transition scenario studies, other than the transition from thermal reactors to FRs, the transition to or adding several other potential NESs has been investigated, such as to a molten salt MA burning reactor (MSR) or MA burning lead cooled ADSs. These alternative advanced concepts had their parameters provided by Member States and recompiled in the same common form as the other reactors in this section.

6.2.4.1. Lead cooled fast reactor ('F5')

The ELSY (European lead cooled system) design is a 600 MW(e) pool-type reactor cooled by pure lead. It was developed in the EURATOM (European Atomic Energy Community) Sixth Framework Programme. The ELSY project aims at the demonstration of designing a competitive and safe fast power reactor using simple technical engineered features. The design should also comply with Generation IV goals. The MA burning capabilities have received particular attention in this project. Core loads with up to 5% MAs and at the same time using burnable absorbers have been investigated. MOX fuel is the primary option. Table 6.9 shows the major specifications of the reactor. The composition data for refuelling are shown in Table II–9 of Annex II.

TABLE 6.8. MAJOR SPECIFICATIONS OF BURNER FR ('F4')

Reactor net electric output	MW	1000
Reactor thermal output	MW	2632
Thermal efficiency	%	38.000
Average load factor	%	85
Operation cycle length	EFPD	232
		Core[b]
Power share of each region	%	100
No. of refuelling batches[a]		6.25
Fuel residence time[a]	EFPD	1451
Specific power density[a]	MW/t	69
Average discharged burnup[a]	MW·d/t	99 604
Initial core inventory	tHM	38.328
Equilibrium loading	tHM/a	8.211

[a] Equilibrium cycle averaged data.
[b] Fast burner reactor with no blankets.
Note: EFPD: effective full power days.

TABLE 6.9. MAJOR SPECIFICATIONS OF LEAD COOLED FR ('F5')

Reactor net electric output	MW	600
Reactor thermal output	MW	1500
Thermal efficiency	%	40.000
Average load factor	%	85
Operation cycle length	EFPD	1825
No. of refuelling batches[a]		1
Fuel residence time[a]	EFPD	1825
Specific power density[a]	MW/t	29.176
Average discharged burnup[a]	MW·d/t	53245
Initial core inventory	tHM	51.413
Equilibrium loading	tHM/a	8.740

[a] Equilibrium cycle average.
Note: EFPD: effective full power days.

6.2.4.2. Minor actinide burning lead cooled accelerator driven system ('A1') (conceptual feasibility established)

The EFIT (European Facility for Industrial Transmutation) reactor is a 400 MW pool-type ADS cooled by pure lead. It was developed within the EURATOM Sixth Framework Programme. The core is sub-critical with a

k_{eff} of about 0.97. This reactor is aimed at burning MAs from reprocessed LWR SF. From LWR SF, plutonium and MAs are used in ceramic–ceramic fuel with about 50 vol.% MgO inert matrix and 50 vol.% oxide of the fissile phase of plutonium and MAs. The plutonium content is designed to remain about the same during burnup, and plutonium is only added during the first cycle. The MA burning rate is 42 kg/TW·h. Table 6.10 shows the major specifications of the reactor. The composition data for refuelling are shown in Table II–10 of Annex II.

TABLE 6.10. MA BURNING LEAD COOLED ADS ('A1')

Reactor net electric output	MW	160
Reactor thermal output	MW	400
Thermal efficiency	%	40.000
Average load factor	%	85
Operation cycle length	EFPD	1800
No. of refuelling batches[a]		1
Fuel residence time[a]	EFPD	1800
Specific power density[a]	MW/t	59.69
Average discharged burnup[a]	MW·d/t	107447
Initial core inventory	tHM	6.701
Equilibrium loading	tHM/a	1.155

[a] Equilibrium cycle average.
Note: EFPD: effective full power days.

6.2.4.3. Minor actinide burning molten salt reactor ('M1')

The MSR used in the GAINS scenario calculations is that of a high flux MSR with a fast-thermal spectrum. This reactor has homogeneous liquid fuel continuously circulating through the core. At the same time, the fuel is continuously fed and discharged (with reprocessing). The presented MSR design implies that half of the fuel is in the core at any one time, while the other half is out of the core. The MSR is fed with Np, Am and Cm from LWR, ALWR and FR SF. The average neutron flux is 10^{15} neutrons/cm^2·s. The major specifications of the reactor are shown in Table 6.11. The composition data for refuelling are shown in Table II–11 of Annex II.

6.2.5. Thorium cycle reactors

In order to save natural uranium resources, natural thorium utilization has been proposed. Irradiation of ^{232}Th breeds the fissile ^{233}U, which supports the chain reaction. To start the process, an initial fissile component has to be added to the thorium, typically ^{235}U, ^{239}Pu or ^{233}U. The last two are particularly attractive from a sustainability perspective. Plutonium-239 is bred initially in reactors using enriched uranium as fuel, and ^{233}U can be both recycled from the Th cycle reactor itself, or from natural thorium blankets placed around other reactors — in particular in FRs.

6.2.5.1. ThO_2 and PuO_2 fuelled CANDU reactors ('H2')

This reactor concept uses the reactor channel structure of a CANDU-6 reactor but is fuelled by a mixture of ThO_2 and PuO_2. The plutonium for this reactor has the isotopic structure of 45 000 MW·d/t LWR discharge fuel after five years of cooling. The bundle structure is a modified form of a low void reactivity fuel bundle having three fuelled rings of pins (3.75% plutonium and 96.25% thorium) and a larger central pin containing hafnium-poisoned zirconia. This central pin is designed to reduce the reactor coolant void coefficient, which is positive.

TABLE 6.11. MAJOR SPECIFICATIONS OF AN MA BURNING MSR ('M1')

Reactor net electric output	MW	1000
Reactor thermal output	MW	2860
Average load factor	%	85
Plant life time	a	60
Operation cycle length	EFPD	not limited
No. of refuelling batches		Continuous refuelling
Fuel residence time	EFPD	-
Specific power density	MW/t	80
Average discharged burnup[a]	MW·d/t	1044615
Initial core inventory	tHM	35.250
Equilibrium loading	tHM/a	0.895

[a] Equilibrium cycle averaged data excluding initial loading of HMs.
Note: EFPD: effective full power days.

The exit burnup of this fuel is 20 300 MW·d/t. Discharged fuel bundles contain approximately 0.9% ^{233}U in heavy elements, which may be used as a seed fissile material for another reactor type if reprocessed. The major specifications of the reactor are shown in Table 6.12. The composition data for refuelling are shown in Table II–12 of Annex II.

TABLE 6.12. MAJOR SPECIFICATIONS OF A ThO_2 AND PuO_2 FUELLED CANDU REACTOR ('H2')

Reactor net electric output	MW	668
Reactor thermal output	MW	2064
Thermal efficiency	%	32.364
Average load factor	%	85
Operation cycle length	EFPD	825
No. of refuelling batches[a]		Continuous refuelling
Fuel residence time[a]	EFPD	825
Specific power density[a]	MW/t	24.596
Average discharged burnup[a]	MW·d/t	20292
Initial core inventory	tHM	71.397
Equilibrium loading	tHM/a	31.557

[a] Equilibrium cycle averaged data.
Note: EFPD: effective full power days.

6.2.5.2. ThO_2, $^{233}UO_2$ and PuO_2 fuelled CANDU reactors ('H3')

This reactor concept uses the reactor channel structure of a CANDU-6 reactor but is fuelled by a mixture of ThO_2, $^{233}UO_2$ and PuO_2. The plutonium for this reactor has the isotopic structure of 45 000 MW·d/t LWR discharge fuel after five years of cooling. The bundle structure is a modified form of a low void reactivity fuel bundle having three fuelled rings of pins and a larger central pin with hafnium-poisoned zirconia designed to reduce the reactor coolant void coefficient, which is positive. Both the plutonium and ^{233}U content of the pins is graded in the radial direction, with the average enrichment being approximately 1.5% (^{233}U in heavy elements) and 1.1% (all plutonium in heavy elements). At exit burnup, the fuel contains, by design, exactly the same ^{233}U content (by weight) as the input fuel, allowing the fuel cycle to be self-sustaining on ^{233}U, assuming an initial ^{233}U core load and no recycling losses. The major specifications of the reactor are shown in Table 6.13. The composition data for refuelling are shown in Table II–13 of Annex II.

TABLE 6.13. MAJOR SPECIFICATIONS OF A ThO_2, $^{233}UO_2$ AND PuO_2 FUELLED CANDU REACTOR ('H3')

Reactor net electric output	MW	668
Reactor thermal output	MW	2064
Thermal efficiency	%	32.364
Average load factor	%	85
Operation cycle length	EFPD	810
No. of refuelling batches[a]		Continuous refuelling
Fuel residence time[a]	EFPD	810
Specific power density[a]	MW/t	24.506
Average discharged burnup[a]	MW·d/t	19850
Initial core inventory	tHM	71.397
Equilibrium loading	tHM/a	32.260

[a] Equilibrium cycle averaged data.
Note: EFPD: effective full power days.

6.3. NUCLEAR FUEL CYCLE SYSTEMS

The basic flows of typical NFC systems combined with nuclear reactor systems studied in the GAINS project are described in this section together with some important analysis conditions.

6.3.1. Once-through cycle system based on thermal reactors

A once-through (open) fuel cycle system is currently applied in most countries. Most of the reactors in the world are thermal-spectrum reactors, and of the various kinds of thermal reactors in the world — e.g. PWRs, boiling water reactors (BWRs), pressurized heavy water reactors (PHWRs), water cooled water moderated power reactors (WWERs) — only a small fraction use recovered plutonium as an MOX fuel. Quantitatively, the current global NFC system can be modelled, with good accuracy, as a once-through cycle system.

6.3.1.1. BAU scenario and its fuel cycle conditions

Figure 6.1 shows a typical diagram of a once-through fuel cycle system. As shown in the figure, the once-through cycle system consists of steps of uranium mining, conversion, enrichment, depleted uranium storage, fuel fabrication, nuclear power plant, SF nuclear power plant storage and SF long term storage. In the case of HWRs, the steps of conversion, enrichment and depleted uranium storage do not exist because HWRs use natural uranium as the fuel.

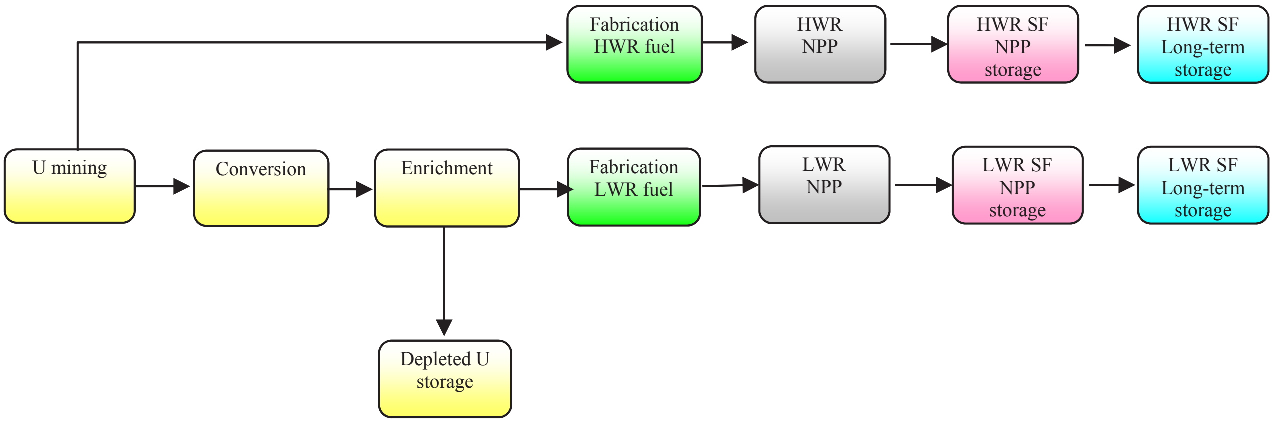


FIG. 6.1. Basic diagram of a once-through cycle system (BAU scenario).

In scenario studies, major parameters characterizing the once-through fuel cycle system are:

— Reactor types and their fuel burnup;
— Proportions of each reactor type and their change with time;
— Tails assay of uranium enrichment;
— Cooling time for SF in nuclear power plant storage.

In order to gain a perspective on the global mass flow to be expected in the next century without the introduction of advanced nuclear reactors, the BAU scenario was studied in GAINS. Figure 6.1 also shows the BAU scenario. Annual or cumulative natural uranium demand, SWU for uranium enrichment and the amount of long term storage for SF are useful indicators for the system.

The fuel cycle conditions for the BAU scenario are as follows:

— Reactor types and their fuel burnup: Only two kinds of reactor, LWR (L1) and HWR (H1), as previously mentioned, are operated.
— Proportions of each reactor type and their change with time: The proportion of two reactors is modelled based on historical data from 1970 to 2008, and from 2008 the share of LWR (L1) to HWR (H1) is assumed to remain as 94:6.
— Tails assay of uranium enrichment: In the BAU scenario, the tails assay is 0.3% and is constant during the whole period.
— Cooling time for SF in nuclear power plant storage: In the BAU scenario, the cooling time in nuclear power plant storage is 6 years.

6.3.1.2. BAU+ scenario and its fuel cycle conditions

High burnup ALWR (L2) fuel designs mentioned in Section 6.2 are to replace LWR (L1) from 2015 with the goal of improving economics and reducing the amount of discharged SF. At the same time, the tails assay of uranium enrichment facilities is to be changed to 0.2% from 0.3% to save natural uranium resources. Figure 6.2 shows the diagram of the BAU+ scenario.

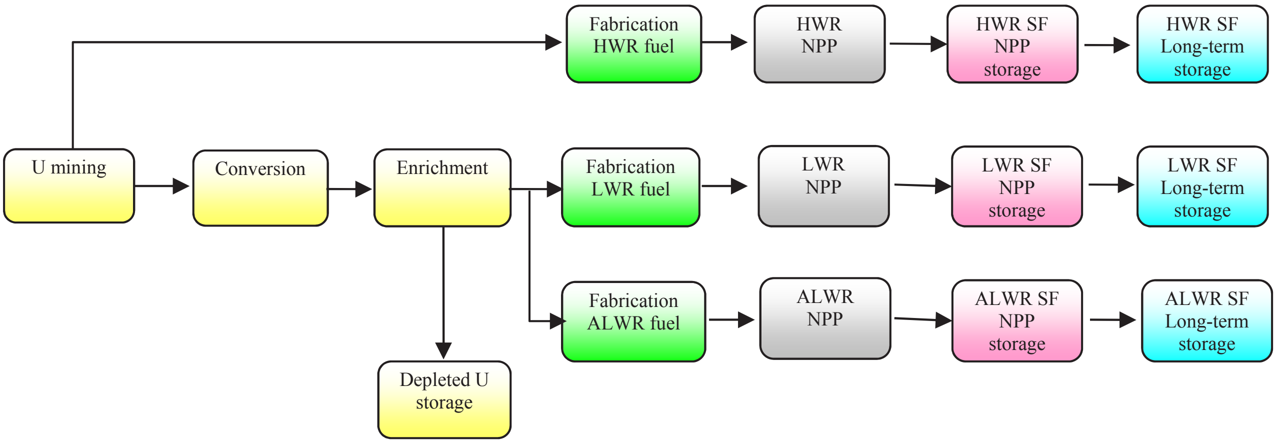


FIG. 6.2. Basic diagram of a once-through cycle system (BAU+ scenario).

The fuel cycle conditions for the BAU+ scenario are as follows:

- — Reactor types and their fuel burnup: Three kinds of reactors, LWR (L1), ALWR (L2) and HWR (H1), as previously mentioned are operated.
- — Proportions of each reactor type and their change with time: The share of two reactors of HWR (H1) and LWR (L1) is modelled based on historical data from 1970 to 2008, and from 2008 the share of ALWR + LWR (L1 + L2) to HWR (H1) is assumed to be 94:6. ALWR (L2) is introduced and replaces LWR (L2) from 2015 as LWR (L1) reactors reach the end of their plant lifetimes.
- — Tails assay of uranium enrichment: In the BAU+ scenario, the tails assay is first set as 0.3% and is changed to 0.2% from 2015 accompanied by ALWR (L2) introduction.
- — Cooling time for SF in nuclear power plant storage: In the BAU+ scenario, the cooling time in nuclear power plant storage is 6 years.

6.3.2. Combined system of a once-through cycle and a fast reactor closed cycle

Owing to the limitation of natural uranium resources in the world, the commercialization of FRs is a promising option for future nuclear development, as these reactors use natural uranium over 60 times more effectively than a once-through cycle system with thermal reactors. Several countries, such as China, France, India, Japan and the Russian Federation, are seriously engaged in FR development. Although in an actual transition phase from a once-through fuel cycle system with thermal reactors to a closed cycle system, the plutonium reprocessed from SF of thermal reactors can partly be put back to be used in the same reactors, in GAINS transition scenarios, all reprocessed plutonium from a once-through fuel cycle system is assumed to be used for the first loading of FRs.

Figure 6.3 shows the basic diagram of a combined once-through cycle and FR closed cycle system. As shown in the figure, the combined system has a reprocessing facility, plutonium (and MA), uranium recycle and a radioactive waste management facility for the FR closed cycle. The reprocessed uranium from an LWR or ALWR can be used as the feed for re-enrichment or the matrix uranium for FR driver fuel and HWR fuel.

6.3.2.1. BAU with a fast reactor (BR: ~1.0) scenario (framework base case) and its fuel cycle conditions

In the BAU with an FR (BR: ~1.0) scenario (Fig. 6.4), the major parameters which feature in the fuel cycle system are as follows:

- — Reactor types and their fuel burnup;
- — Plant lifetime and load factor of each reactor type;
- — Proportions of each reactor type and their change with time;
- — Process time of each step in up-stream (often treated as lead time);
- — Tails assay of uranium enrichment;

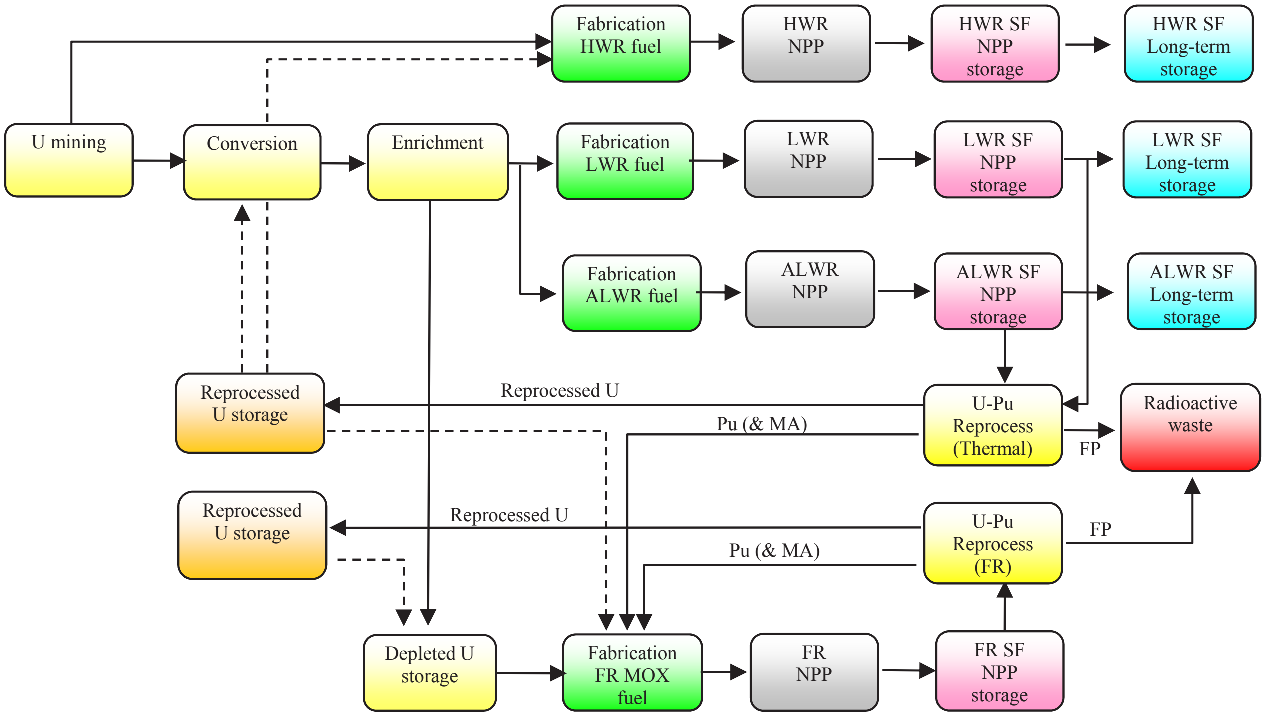


FIG. 6.3. Example of a combined system of a once-through cycle and an FR closed cycle system.

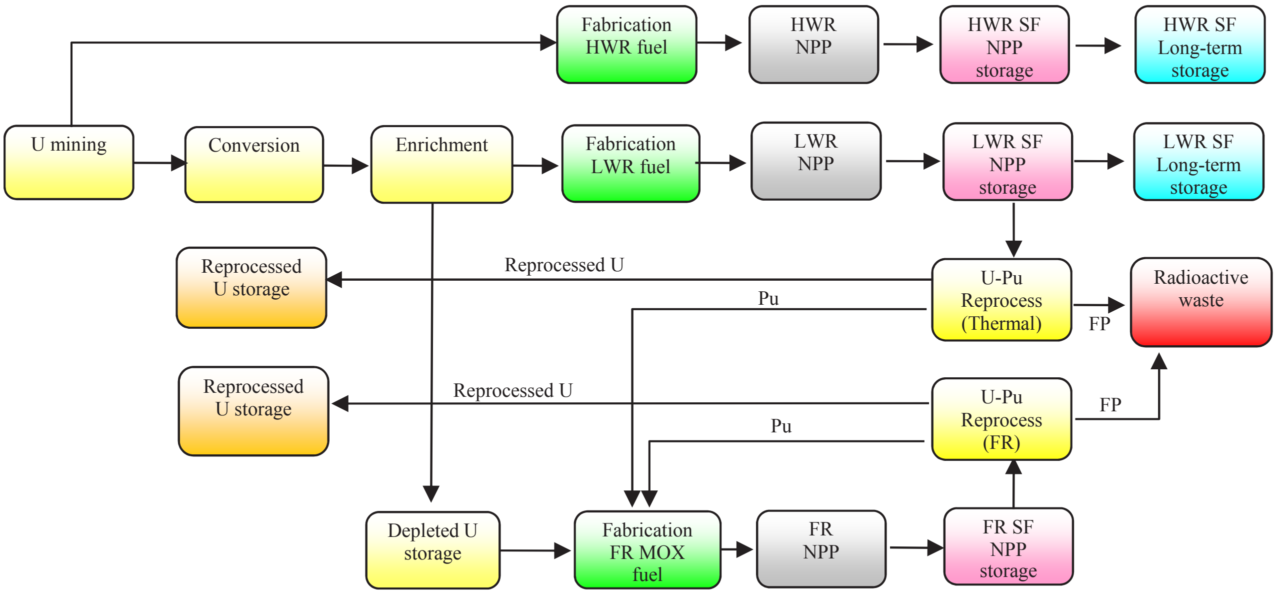


FIG. 6.4. Fuel cycle system of BAU with an FR (BR: ~1.0) scenario (framework base case).

— Cooling time for SF in nuclear power plant storage;
— Process time of reprocessing;
— Process time of fuel fabrication with recovered Pu;
— Capacity of reprocessing facilities;
— HM loss in reprocessing.

For plutonium recycling, the last five items are particularly important. In the FR closed cycle, the total cooling time for SF in nuclear power plant storage, the process time of reprocessing and the process time of fuel fabrication with recovered plutonium is usually defined as 'out of reactor time'.

The fuel cycle conditions for BAU with an FR (BR: ~1.0) scenario are as follows:

- Reactor types and their fuel burnup: Three kinds of reactor, LWR, HWR and FR (BR: ~1) as mentioned in the previous section as break-even FRs are to be operated.
- Plant lifetime and load factors: The plant lifetime and load factor for both LWRs and HWRs are 60 years and 85%, respectively. Plant lifetime and load factor of FRs are 60 years and 85%, respectively, the same as for LWRs and HWRs.
- Proportions of each reactor type and their change with time: The share of LWR and HWR reactors from 1970 to 2008 is modelled based on historical data; from 2008, the share of HWRs is assumed to be 6% of global total power production. The introduction rate (power demand) of FRs is settled by the project participants as follows:
  - 2021–2030: 1 GW(e) demand growth per year (total demand: 10 GW(e) in 2030).
  - 2031–2050:
    (i) High case: 19.5 GW(e) demand growth per year (total demand: 400 GW(e) in 2050).
    (ii) Moderate case: 9.5 GW(e) demand growth per year (total demand: 200 GW(e) in 2050).
  - From 2051: Adjust to maximum FR introduction rate consistent with plutonium availability.
- Process times: The time interval from uranium mining to fresh fuel loading is assumed to be 2 years, the same as for the BAU scenario.
- Tails assay of uranium enrichment: The tails assay is 0.3% for the whole period.
- Cooling time for SF in nuclear power plant storage: The cooling time of SF from thermal reactors (LWRs and HWRs) in nuclear power plant storage is 5 years.
- Out of reactor time: Concerning thermal LWRs, the out of reactor time is 6 years, which consists of 5 year cooling in nuclear power plant storage and a process time of 1 year for reprocessing and FR fuel fabrication. The out of reactor time of FRs is 3 years, which consists of 2 years cooling in nuclear power plant storage and a process time of 1 year for reprocessing and fuel fabrication.
- Capacity of reprocessing facilities: No limitation of the capacity of reprocessing facilities for LWRs and FRs is assumed in the current phase of the GAINS project, so that FR introduction speed can be limited only with plutonium availability.
- HM loss in reprocessing: The plutonium and uranium loss during reprocessing are assumed to be 1%.

6.3.3. Combined system of a once-through cycle and a fast reactor, and/or an ADS/MSR closed cycle

Other than the combined system of thermal reactors, a once-through cycle and an FR closed cycle, ADSs and MSRs have been proposed, focusing on the effective incineration of MAs. Figure 6.5 shows the diagram of the combined system which introduces both types of advanced reactors. The closed cycle for ADSs is almost the same as for FRs. On the other hand, the MSR has its own closed fuel cycle circuit in the reactor system, and there is basically no HM discharge from the system.

6.3.4. Combined system of a once-through cycle and a fast reactor (U–Pu)/Th closed cycle

From the viewpoints of the limitation of natural uranium resources, enhancement of nuclear proliferation resistance and the reduction of long term SF decay heat, the introduction of a thorium cycle to the global NFC system has been proposed.

One feature of the NFC system with Th–U (^{233}U) is that the Th–U system has to coexist independently with the U–Pu system, and the initial state of Th–U system introduction has to be supported by a preceding U–Pu system through an MOX fuel feed. As a result, the NFC system which introduced the Th–U system tends to become complicated compared to the preceding U–Pu system. Figure 6.6 shows a diagram of Th–U system introduction as one example.

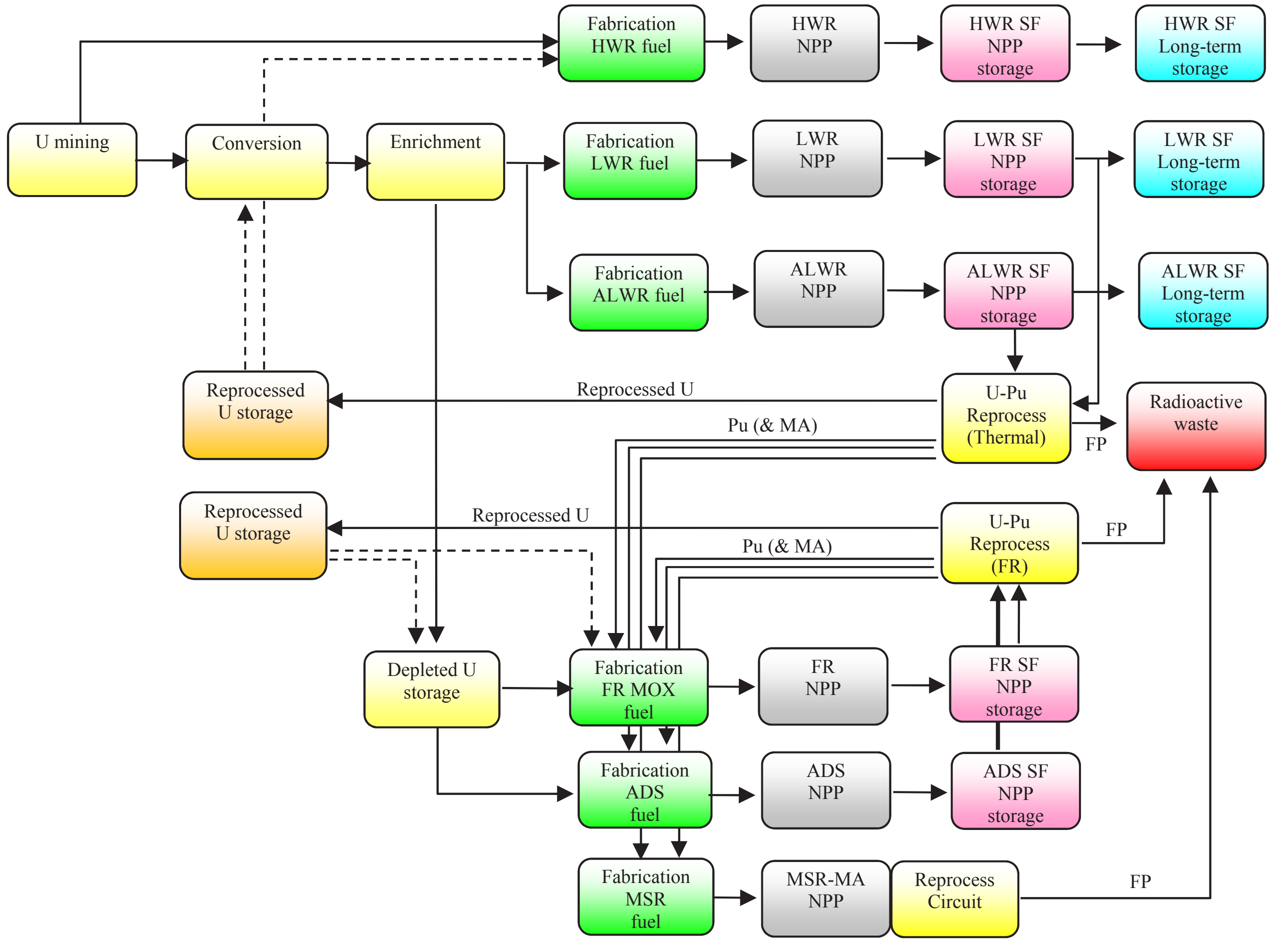


FIG. 6.5. Example of a combined system of a once-through cycle and an FR and/or an ADS and MSR closed cycle.

6.4. SUMMARY

The data of different reactor systems and the conditions of fuel cycle systems which are necessary to evaluate the mass flow in future nuclear power systems were compiled for the scenario studies in GAINS.

The reactor data spread from existing reactors, LWRs and HWRs, to advanced reactors, such as FRs, ADSs, Th fuelled HWRs and MA fuelled MSRs. Major specifications and the compositions of fresh and discharged fuel were provided for the reactors based on design work by members.

Concerning NFC systems, some typical fuel cycle concepts were chosen for GAINS scenario studies. Then, major cycle conditions were settled on the framework base cases of the BAU, and BAU with FR scenarios.

These reactor database and fuel cycle concepts and the conditions can be used not only in the current GAINS framework but also in future programmes on a global nuclear vision.

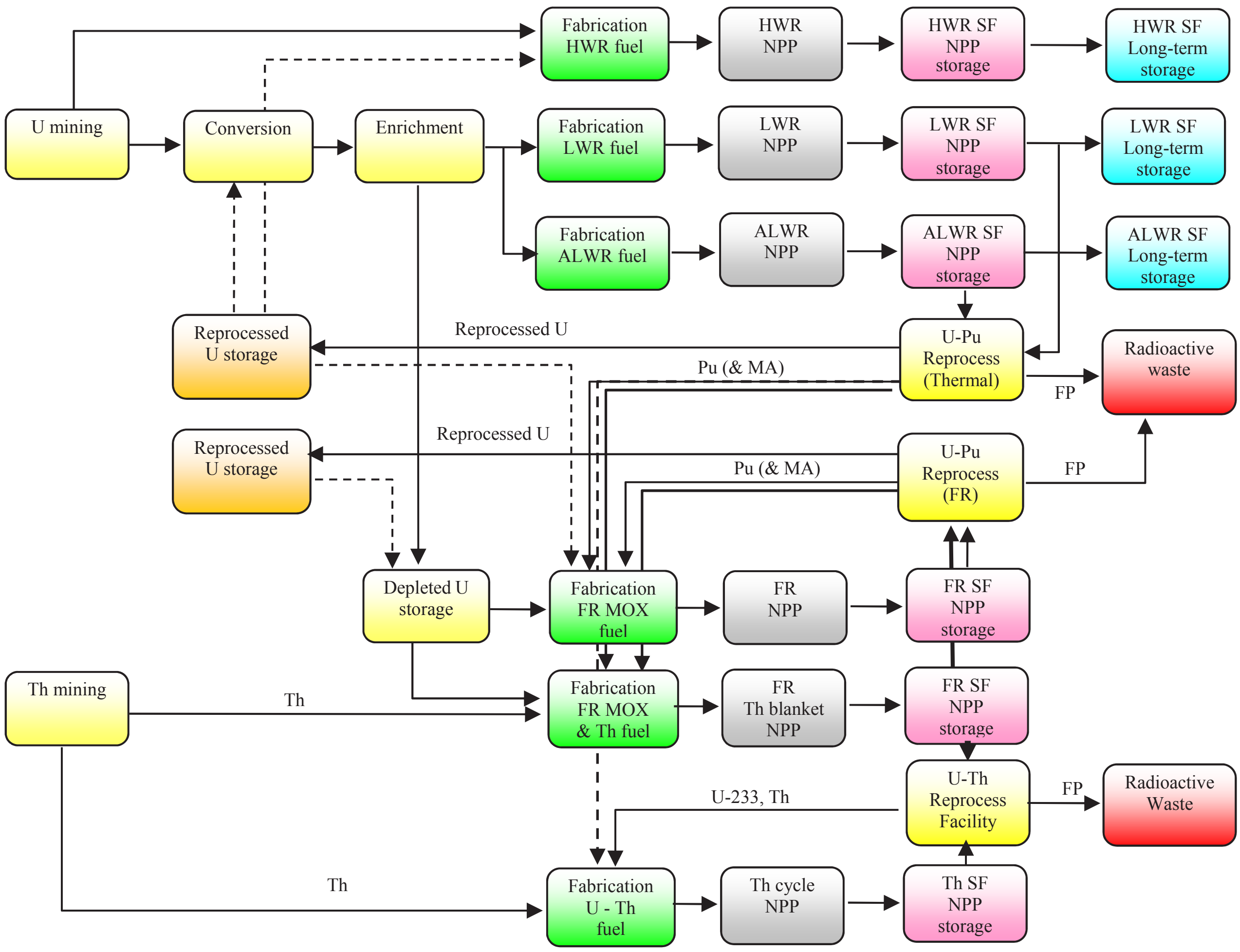


FIG. 6.6. Example of a combined system of a once-through cycle and a closed FR (U–Pu) cycle and a closed Th cycle.

REFERENCE TO SECTION 6

[6.1] ARGONNE NATIONAL LABORATORY, Preliminary Core Design Studies for the Advanced Burner Reactor over a Wide Range of Conversion Ratios, ANL-AFCI-177, Argonne Natl Lab., IL (2006).

7. FRAMEWORK BASE CASES

7.1. PURPOSE

As stated in Section 1, the overall objective of GAINS is to develop a standard framework for assessing future NESs taking into account sustainable development, and to validate the results through sample analyses. In Section 3, homogeneous and heterogeneous story lines of future global NESs were introduced as a basis for the framework. Section 4 discussed indicators of sustainability for NESs. Together, these sections developed the basis for the GAINS framework and the method to evaluate the relative sustainability of a fuel cycle strategy. Section 5 then introduced the basis for the energy demand curves to be used in the framework. Section 6 provided descriptions and parameters for a number of reactors and fuel and scenario specifications for several NFC options.

A subset of the NFCs in Section 6.3 were identified as framework base cases. The framework base cases are intended to provide simple reference examples of the homogeneous and heterogeneous story lines. Users of the GAINS framework can start from these simple examples to develop more elaborate NESs and more detailed scenarios. The non-framework cases in Section 6.3 are examples of these more elaborate systems, as they include modifications of scenario parameters (e.g. higher burnup LWRs, different tails enrichment), additional reactor/transmuter types (e.g. MSRs, ADSs) and additional fuel materials (thorium, MAs).

The purpose of this section is to provide a full description of the framework base cases and document their sustainability performance, so that they may serve as a reference by framework users. This includes a full description of the input parameters and expected outputs for each base case, as well as sustainability results for the framework base cases using the sustainability indicators.

The eight framework cases are:

- Homogeneous world:
  - BAU scenario:
    - (i) Moderate growth case;
    - (ii) High growth case.
  - BAU with FR (BAU–FR) scenario:
    - (i) Moderate growth case;
    - (ii) High growth case.
- Heterogeneous world for the BAU–FR scenario:
  - Non-synergistic:
    - (i) Moderate growth case;
    - (ii) High growth case.
  - Synergistic:
    - (i) Moderate growth case;
    - (ii) High growth case.

Each of these cases is presented in this section. To assist framework users, the section also includes insights into considerations that went into the development of the base cases and observations on the sources of variations in results for these cases based on code capabilities and modelling options employed by the analyst.

The framework base cases should not be considered as benchmarks but rather as reference cases. Users of the framework should not expect to generate results that exactly match those shown here. Owing to variations in methods used in fuel cycle codes, as well as interpretations on how to implement assumptions, it is very difficult to get exact matches with independently developed analyses. Some of the reasons for these differences are discussed in Section 7.2.3.

Users of the framework should first run a framework base case on their own code and compare results to ensure that the scenario is the same as in the framework base case and that the behaviour is very similar. Typically, primary values such as electricity generation by reactor type and mass flows should be within a few per cent and changes in trend lines should occur within 1–2 years of the results documented here. The user's version of the framework base case should then be used for comparison to alternative cases analysed with the same fuel cycle code.

7.2. GENERAL CHARACTERISTICS OF FRAMEWORK CASES

The purpose of the framework cases is to illustrate application of the GAINS framework for two scenarios that are relatively easy to model, to give users of the framework insights into what it can provide, and to provide analysts adapting the framework to different scenarios a basis for comparison. These framework cases integrate the information described in earlier sections of this report by applying the homogeneous and heterogeneous global models (Section 3) to analyse the BAU and BAU–FR scenarios (Section 6) for high and moderate nuclear power demand profiles (Section 5), and calculate KIs and EPs for NES sustainability (Section 4).

Historically, a large number of different reactor designs have been employed. Some were shut down after only a few years of operation while others have operated for nearly 40 years and are licensed to at least 60 years. The average fuel discharge burnup from these reactors has evolved, with much higher burnup today than when most of the reactors were first started up. The average electrical generating capacity of a typical LWR reactor has also increased from the initial reactors of well below 500 MW(e) to a current average nearing 1000 MW(e), and new construction averages even larger capacity. Projecting into the future, these differences will likely continue, with reactors of different sizes using different fuel burnups. While detailed modelling of every type of reactor or even every specific reactor is possible, average values are much easier to model and provide sufficiently accurate results for most types of analyses.

To make the framework cases easy to model by a range of fuel cycle codes, only two reactor types are used to model the history of nuclear energy to the present. This includes one LWR (‘L1’ from Section 6.1) to represent all of the BWRs and PWRs, and one HWR (‘H1’) to represent all of the heavy water moderated reactors. For simplicity, other reactor types such as gas cooled reactors have been replaced with the LWR. The GAINS representatives decided to model the HWRs separately due to their large installed base, the very different burnup compared with other thermal reactors, and the likelihood that a decision to recycle would probably not include HWR SF due to the low plutonium content.

Also for simplicity, L1 and H1 both have the same average load factor and average operating life, and use fuel burnup levels that are higher than the historical average but lower than projected levels. Thus, the framework cases do not exactly model either the past or the present. However, they do provide a simple representation of both the past and present that is easy to model.

The framework cases with FR introduction use ‘F1’, a break-even FR. While many users of the framework will be interested in modelling breeder FRs (FRs that produce more plutonium than they consume), others will model burners (FRs that consume more plutonium and transuranics than they produce). The break-even reactor provides a neutral point of reference, while also avoiding the issue of what BR to select. It also simplifies analysis of material recycled from FRs, as it does not generate excess material like a breeder or require make-up material like a burner. Again, the same average load factor and operating life is used as for the HWR and LWR. However, the thermal efficiency used is higher, reflecting the significantly higher outlet temperatures expected from liquid metal cooled reactors.

For all three reactor types, a load factor of 0.85 is used on all framework base cases. The reactor information in Section 6 indicates a lower load factor for the LWR and HWR based on available data. In discussions, the GAINS representatives felt that there was insufficient justification for projecting a higher load factor for new FRs than for the well established HWR and LWR technologies. HWRs have always enjoyed high load factors, while the load factors for existing LWRs have been increasing steadily and are now above the 80% level. Figure 7.1 shows the load factor trends for the commercial reactor fleet in the USA [7.1], which represents ~30% of total global energy generation.

The framework base cases also use the same plant life for all three reactor types, which is set at 60 years. This value was adopted to reflect the current status of LWRs. Nuclear power is still a relatively new industry and plant operating licences primarily reflect when governments require performance re-evaluation rather that any known technical driver for plant life. Where licence renewals have been requested by the reactor owners, extensions to 60 years have been granted following regulatory review.

Finally, the framework base cases all use the same tails enrichment for depleted uranium of 0.3% throughout the analysis. A sensitivity analysis was performed to determine the impact of lower tails enrichment on uranium consumption and the difference was found to be significant (see Fig. 7.2). Dropping from 0.3 to 0.2% could reduce natural uranium consumption by 20%. The ideal tails enrichment for a specific enrichment facility can be determined using an economic formula that takes into account the cost of yellowcake, the cost of conversion to

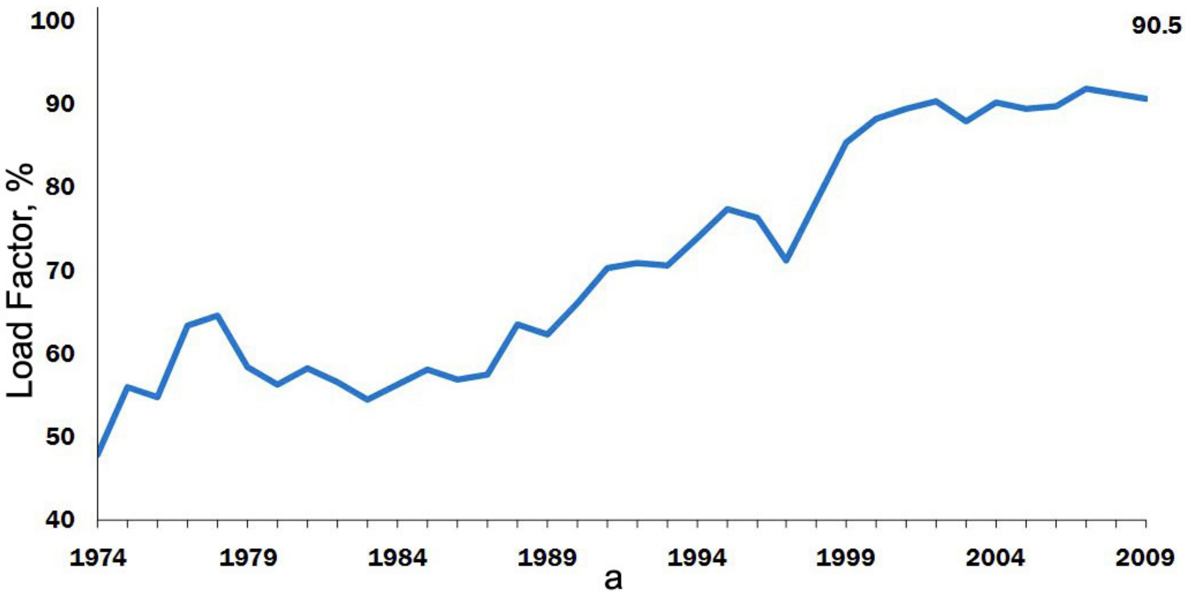


FIG. 7.1. US nuclear industry average load factors, 1974–2009.

UF_6, the cost of SWUs, and the desired isotopic enrichment of the UO_2 fuel. Some of these parameters can vary daily, so projecting a value 100 years into the future involves considerable uncertainty. For simplicity, the value was assumed to be constant for the framework base cases. This provides a point of departure for future analyses that may examine changes in tails enrichment.

(It should be noted that some of the sensitivity analyses presented in this report were developed while the framework base cases were still evolving and the results may not exactly match the final base cases.)

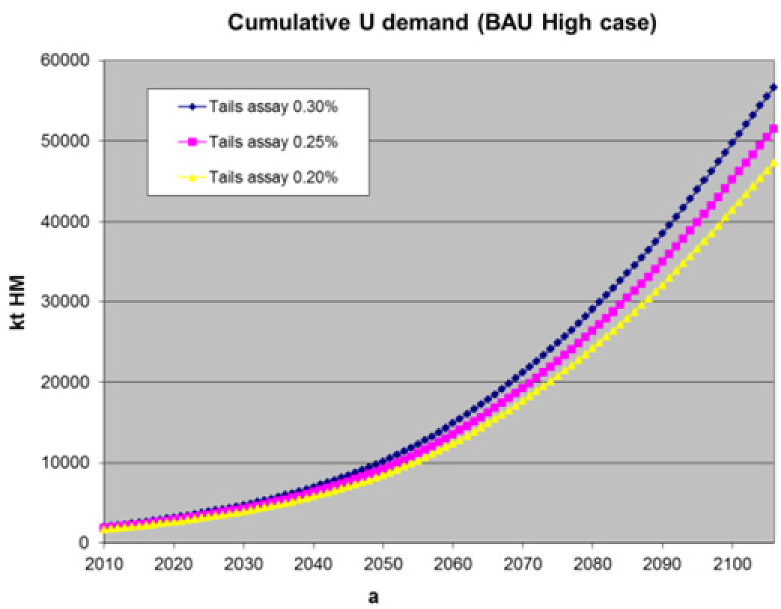


FIG. 7.2. Tails assay and cumulative uranium consumption (BAU high case).

7.2.1. Initialization data

A reporting template has been developed for the GAINS framework that indicates standard output data to be provided by fuel cycle simulation codes. The template reports results for the 100 year period from 2010 to 2110. Prior to 2010, an initial fleet of reactors is established and existing inventories of SF are developed. The GAINS history begins in 1970 and is based on historical capacity information through 2008. After 2008, the growth curves discussed in Section 5 are applied. The annual data used for the growth curves is contained in Annex IV.

Table IV–1 in Annex IV documents the GAINS framework history from 1970 to 2010 for both the moderate and high cases. This history is used for all homogeneous framework base cases. For the heterogeneous framework base cases, the same history is used but the reactor capacities are split between groups NG1 and NG2 as discussed later.

While information in the GAINS template is not presented prior to 2010, it is preferred that the codes simulate the full history for several reasons. First, this allows for a retirement history to be established. In the GAINS framework base cases, the first reactors are assumed to have been built beginning in 1970, with a 60 year life, so the first reactors will retire in 2030. Second, some codes have the capability to model isotopic decay in stored material, including SF. By modelling the history beginning in 1970, this decay can be included. Finally, most codes require a short period at the start of the simulation to initialize the code. For this reason, the analyst may want to start the simulation a few years prior to 1970.

Analysts should note that the scenarios are driven by demand for electricity, shown here in GW·a (8766 GW·h), which is the electric energy generated by 1 GW(e) of electrical generating capacity operating for a full year at a 100% load factor. This value was adopted instead of the installed power capacity to avoid potential issues when using reactors with different load factors. While the framework base cases use the same load factor for all reactors, other analyses may not. By using demand as the scenario basis, the total electricity generation will be the same for all scenarios running the same growth curve, independent of the load factors used for different reactor types. As the historical load factors were lower than 0.85, this simplified history understates the historical capacity. It also understates the historical SF generation due to both the difference in load factor and the use of a fixed burnup for the LWRs. Historically, LWR burnup was lower than the 45 000 MW·d/t used for the base cases.

7.2.2. Growth curves

The GAINS growth curves from Section 5 (Fig. 5.5) start with current global nuclear electricity generation as of 2008, with growth to 2100. After 2100, the generation level is held at a fixed level (2500 GW·a for the moderate case and 5000 GW·a for the high case). This fixed level was added to allow analyses to come into balance and determine whether the system resulting in 2100 was sustainable.

Table IV–2 in Annex IV provides the growth profiles for the moderate and high GAINS cases for the BAU framework base case. Specific features of these cases include a fixed electricity generation ratio of LWRs to HWRs of 94:6 during four separate growth periods. Each growth period is modelled as linear growth (not exponential) to reach a specific level of generation by the end of the period:

(a) 2009–2030: reaching 600 GW·a for the moderate case and 700 GW·a for the high case.
(b) 2031–2050: reaching 1000 GW·a for the moderate case and 1500 GW·a for the high case.
(c) 2051–2100: reaching 2500 GW·a for the moderate case and 5000 GW·a for the high case.
(d) 2101 and beyond: no additional growth, only replacement of retiring reactors. (Depending on the scenario, retiring reactors may be replaced by the same type or a different type of reactor.)

The GAINS template graphics reports results through 2110 to show the impact of the transition from steady growth to a fixed level for the total system. However, the data columns continue to 2130 to allow for additional analysis time beyond the reported period. When a scenario ends, the look-ahead algorithms in some fuel cycle simulation codes go through a coast-down period when, for example, replacement reactors may no longer be placed in a construction queue. This can result in perturbations in other parts of the simulation, such as fuel ordering and enrichment. The extra 20 years of simulation beyond the graphed period allow for the analyst to examine these perturbations without affecting the results in the displayed period (2010–2110).

For the BAU–FR framework cases, the same growth curves are used. However, FRs can displace LWRs after the FRs are introduced. The HWRs continue at 6% of the total generation independent of the FR introductions. The same growth curves are also used for the heterogeneous cases, with the allocation by NFC group explained later in this section.

7.2.3. Sources of variance in results for different codes

This section discusses some of the simulation code capability options that can be a source of differences in results versus those shown in this section when running the framework base cases. Many of these differences were identified by the GAINS participants or as part of other benchmarking efforts [7.2, 7.3].

An exact match year-by-year to the values in the Annex IV tables should not be expected when using a simulation code that does not allow fractional reactors. Some codes allow fractional construction (e.g. ‘3.68 reactors’) and can, therefore, model a specific growth curve very accurately. Other codes require whole reactors (e.g. ‘4 reactors’) and will, therefore, vary slightly from the indicated values, especially during the early part of the simulation when the total number of reactors is small. Even for a code requiring whole reactors, the per cent difference should be small (maximum: 1–2%) during the reporting period of 2010–2110.

Depending on when a code reports results, values may be offset by one year. For example, some codes report the status as of the start of the year, while others report as of the end of the year. Some codes model discrete steps that occur on a shorter time scale, resulting in any spikes in process values potentially falling a year earlier or a year later versus other codes. For example, when acquiring the fuel for the initial core of a new reactor, a spike may move through conversion, enrichment and fabrication steps, with some of these activities reported the year before the new reactor starts operation and others occurring in the same year as operations. The time to transit fuel acquisition was set to 2 years to accommodate codes that model these steps discretely.

Some codes have the ability to model isotopic decay of stored material, such as SF. These codes will have lower plutonium inventory values due to ^{241}Pu decay, which can impact plutonium enrichment levels in recycled fuel, proliferation indicators and other factors. In the scenarios involving transition to FRs, these codes will also build slightly fewer FRs. Some codes have no decay capabilities. For these codes, the GAINS template supports some adjustment of inventory values due to decay via a specification of the decayed content mix (U, Pu, MAs, FPs) at the time of separation.

Some codes have the ability to model startup cores differently from standard refuelling, to account for the lower burnup of the fuel discharged for the first few refuellings. Depending on the code, the implementation may simply be a lower enrichment of the initial core, or it may also include the difference in isotopic content of the resulting SF. Some codes will also model the final core differently for similar reasons.

At the burnup of the LWR used in the framework base cases (45 GW·d/t), the first core average enrichment is only ~60% of the equilibrium core enrichment. A sensitivity study was performed to determine the impact on natural uranium usage of not modelling the first core separately. The difference was found to be only a few per cent over the life of the reactor. Reactor life was found to be the largest driver for this difference, with reactors with a 40 year life having up to a 5% difference (see Fig. 7.3) while reactors with a 60 year life having a smaller difference. It was also found that the higher the burnup of the equilibrium core, the smaller the difference in initial enrichment, but this factor had less than a 1% impact for the range of burnups modelled.

As the first core impacts were small, for the framework base cases, the simplest approach was used — that the startup core has the same composition for both fresh fuel and SF as the equilibrium core. This allows the framework base cases to be run by fuel cycle codes that do not have the capability to model first cores differently and also eliminates differences arising from how other codes implement first and final cores.

7.3. HOMOGENEOUS WORLD CASES

Homogeneous modelling allows estimating the evolution of nuclear demand on a global level. Since similar estimations have already been done within other projects (see Section 2), it will be useful to check the results obtained within the CP GAINS with previous studies and evaluate the existing analytical codes for modelling NESs to establish a specification to improve such tools.

The homogeneous framework cases include a BAU case involving only LWRs and HWRs, and no recycling of used nuclear fuel, as well as a case for transitioning from the BAU case to one that includes recycling and introduction of FRs (BAU–FR) fuelled by recycled material.

By design, both sets of cases have been kept simple with most parameters held constant, so that they may serve as a starting point for analysts interested in examining the impact of changing those parameters. The base case results provide a performance reference for comparison to other analyses.

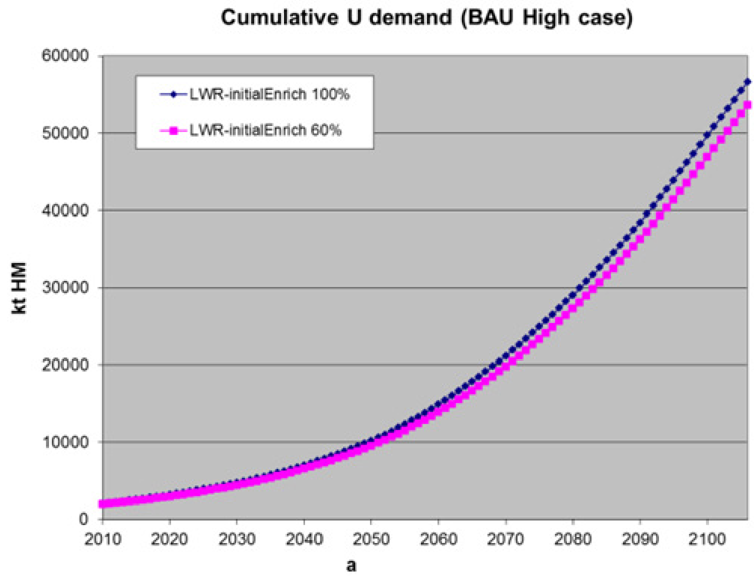


FIG. 7.3. Effect of initial core enrichment adjustment (BAU high case).

The base cases are in no way optimized to try to solve any particular sustainability issue. The BAU case provides a reference for continuing a once-through fuel cycle. The BAU–FR case provides a simple reference for transitioning to recycling, which can serve as a starting point for analyses investigating different BRs, different transmutation systems (e.g. burner FRs, ADSs, fission–fusion hybrids), or different transition timing.

7.3.1. BAU

The basic features of the BAU case are sustained growth of thermal reactors (LWRs and HWRs) with accumulation of SF and no recycling. Figure 7.4 shows the growth profile that is characteristic of all of the framework base cases. For both the high growth case and the moderate growth case, this includes the four separate growth periods up to 2030, 2030–2050, 2050–2100 and beyond 2100. The figure also shows the 94% LWR, 6% HWR generation shares. This figure is sustainability KI KI-1, as described in Section 4. The high case tops out at 5000 GW·a, while the moderate case has the same basic shape, but tops out at 2500 GW·a. As they have the same history up to 2008, the moderate case appears in the graph to start higher due to using a different scale on the year axis.

Figure 7.5 shows the commissioning rate of new reactors (sustainability EP EP-1.1). The period between 2010 and 2030 is almost constant. The slight ripple in the LWR line during this period is due to the code used (VISION) modelling whole reactors and needing to add an additional reactor every few years to stay on the growth

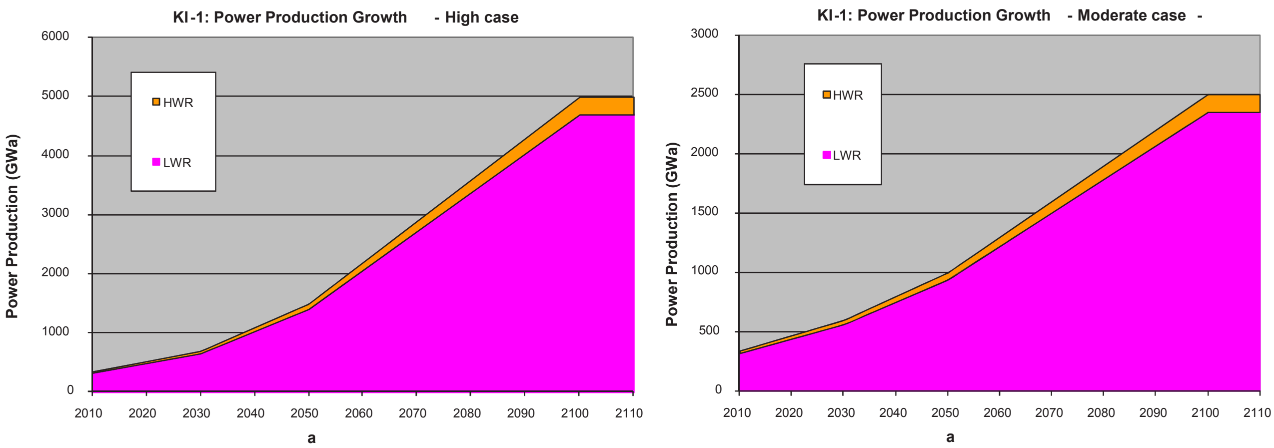


FIG. 7.4. Power production (BAU high case — left, moderate case — right).

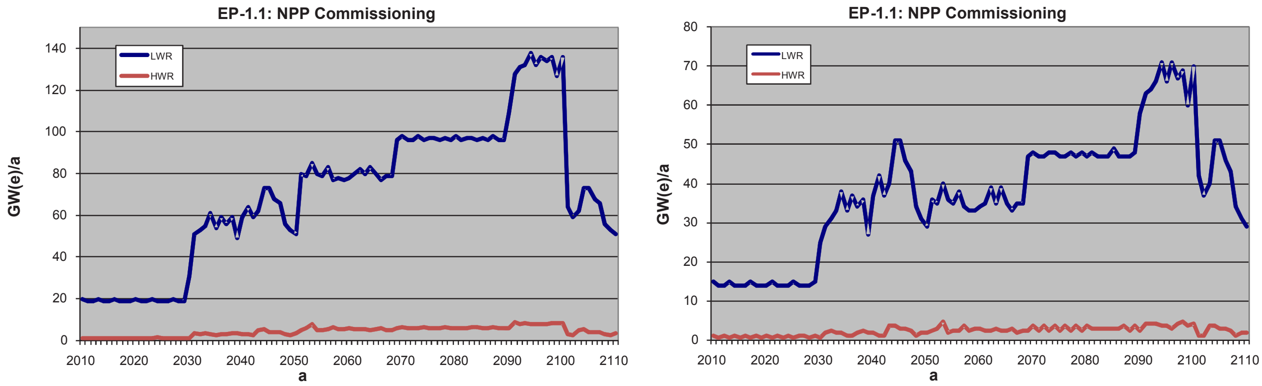


FIG. 7.5. New power plant commissioning rate (BAU high case — left, moderate case — right).

curve. After 2030, the growth rate increases and retirement of the reactors initially built in the 1970s begins. Both contribute to a higher commissioning rate of new reactors. The larger variation in this period is due to replacement of retiring reactors, with the magnitude greater in the moderate case because the underlying growth rate is lower. In 2050, a new higher growth rate is introduced, while the rate of retirements slows.

Around 2070, the commissioning rate again increases, due to retirement of the reactors built in the initial growth phase around 2010. In 2090, the replacements increase to make up for retirement of reactors built in 2030 — note the similar ripple pattern, with the values in the 2090s echoing those from the 2030s. The ripples are not an exact match due to the whole reactor effect, where retirement of a reactor may or may not trigger construction of a replacement depending on how close the remaining capacity is to the growth curve. The total value in the 2090s is higher than in the 2030s due to the high underlying growth rate. The underlying growth ends abruptly in 2100, resulting in a significant decrease in new reactor commissioning — after 2100, all new reactors are replacements of retiring reactors, and the peak around 2045 is echoed at the same magnitude around 2105.

The second KI for use in the GAINS framework area is the average net energy production per unit of natural uranium/thorium used (KI-2 in Table 4.1). The left graph of Fig. 7.6 shows the result of this indicator applied to the BAU high case. The figure exhibits four features:

— First, the overall trend is flat, which is to be expected given that there is no change in reactor type, fuel burnup or fuel cycle strategy during the simulation.
— Second, the HWRs outperform the LWRs due to two factors. First, the LWR more efficiently uses the fissile ^{235}U in its fuel, but because in the LWR case a portion of ^{235}U is left in the tails from the enrichment process, the HWR actually uses more of the natural ^{235}U. Second, the in-core breeding and consumption of fissile ^{239}Pu from the fertile ^{238}U in the fuel contributes to energy production in both reactor types. However, due to neutron spectrum differences, the HWR produces and uses ^{239}Pu more efficiently and, therefore, generates more of its energy from Pu. The system total closely tracks the LWR numbers due to the LWRs providing 94% of the total energy generation.

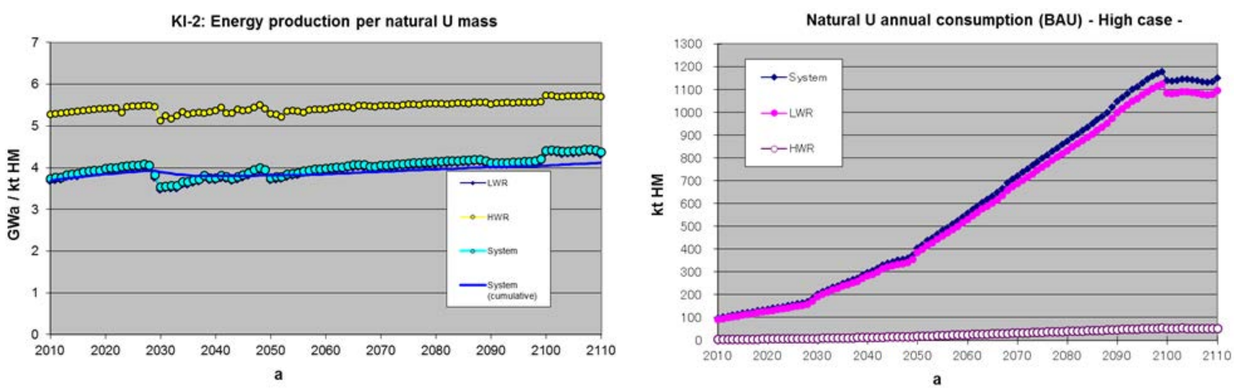


FIG. 7.6. Energy production normalized to uranium use (left) and annual uranium use without normalization (right) (BAU high case).

— The third feature is a very slight improvement over the course of the simulation, due primarily to the uranium required for the initial cores of new reactors. Uranium use improves the longer a reactor operates, as the first core is averaged over more total generation from that reactor.
— The final feature is the dips around 2030 and 2050, and the slight jump in 2100. This is also due to first cores when the growth curve changes, but the impact is amplified by the delay between when the uranium is mined versus when it starts generating energy — when there is an increase in the growth rate of energy generation (2030 and 2050), the uranium mining reduces the indicator value, but when the growth rate slows (2100), the opposite effect is seen.

The right graph in the figure shows the annual natural uranium usage (kilotonnes of HM for each year) without normalization. The usage closely follows the growth curve in Table IV–2 of Annex IV with minor fluctuations caused by variations in the rate of reactor retirements and replacements due to the uranium needed for initial cores of the replacement reactors. The impact of initial cores is most easily observed just before 2100, when the growth rate goes to zero and the rate of new reactor startups drops. The result is a small step drop in annual uranium usage two years before 2100. The two years is due to the lead time between when uranium is mined and when it is loaded into the reactor, which is part of the framework specification.

The top of Fig. 7.7 shows the cumulative uranium usage since 2008, which is the sustainability EP EP-2.1. The 2008 value was selected as a starting point rather than cumulative usage since the start of commercial nuclear power for two reasons — it facilitates comparison to the most recent estimation of resources from the Red Book [7.4], and it does not include non-commercial usage of uranium.

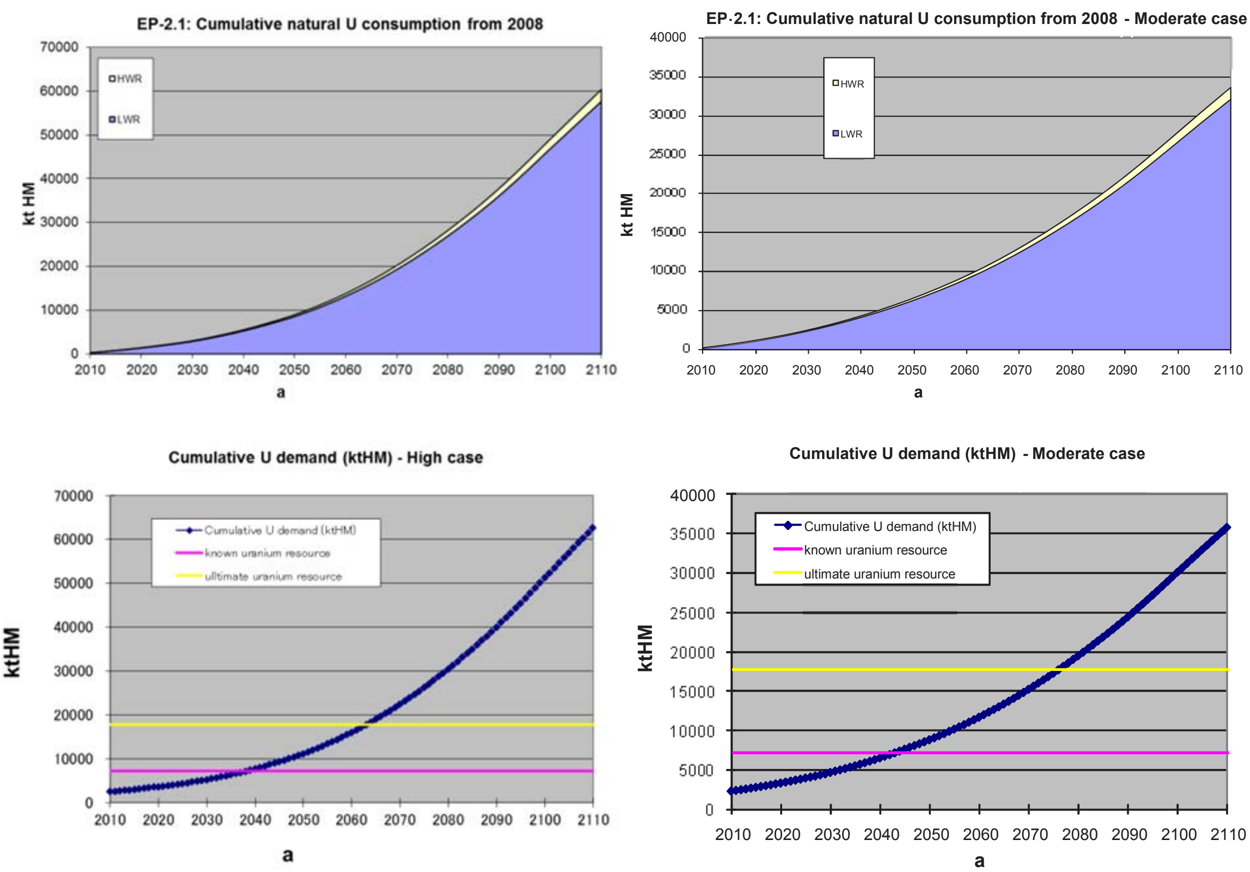
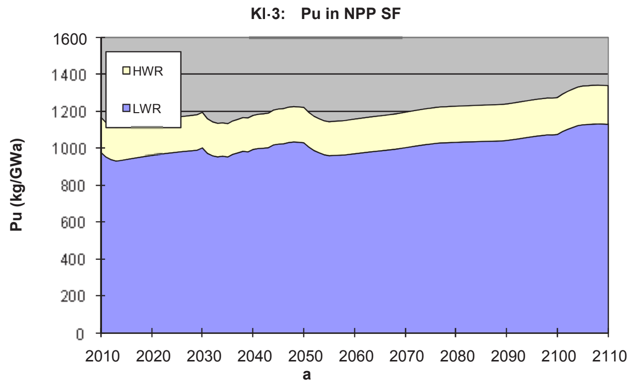


FIG. 7.7. Total cumulative uranium usage and comparison to projected resources (BAU high case — left, moderate case — right).

The bottom of Fig. 7.7 shows cumulative demand versus Red Book projected conventional resources ('known resources' of 7.24 Mt and 'ultimate resources' of 17.8 Mt). It should be noted that the cumulative uranium demand significantly exceeds current projected resources, with only a few decades of resources left. This suggests a future shortage unless a different fuel cycle is adopted, a position supported by some of the GAINS participants. However,



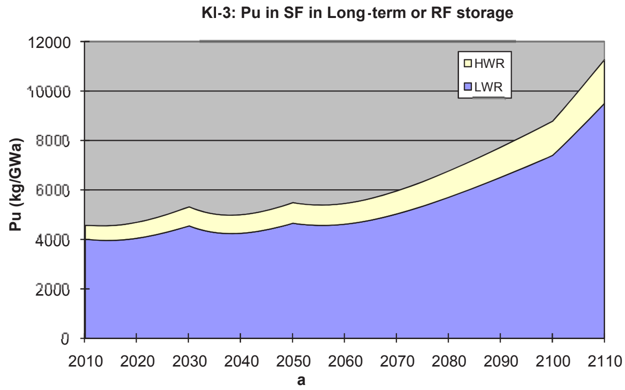


FIG. 7.8. Normalized plutonium in spent fuel in storage at reactors (left) and in long term storage (right) (BAU high case).

other participants support the view that more resources will be identified in the future. Since the Red Book's first publication in 1965, the projected resource base has grown even as the nuclear power industry has drawn more uranium from the ground. The Red Book Retrospective [7.5] documents this trend. The same trend is seen in many other natural resources, including oil, natural gas and a number of minerals. How long this trend will continue is unknown. With future uranium resources uncertain, the GAINS framework supports assessment assuming current projected resources or other values selected by the analyst.

KI-3 addresses an area of proliferation risk by measuring the amount of plutonium normalized per annual unit of energy produced in the global system. For the BAU case, the areas of interest are SF storage at the nuclear power plant, while the fuel is cooling and storage of cooled fuel. Figure 7.8 shows the result for the BAU case. When normalized to annual energy production, the same trend is observed as in uranium use, where changes in the growth rate result in changes in the normalized level that even out over time. In this case, higher growth means that the existing inventory is divided by a higher annual output, resulting in a lower Pu/GW·a value shortly after the growth rate transition. For the BAU case, SF stays in the system in long term storage. As it accumulates, the Pu/GW·a grows, especially after the growth rate goes to zero in 2100. The HWR share of total plutonium is ~13% because there is slightly over twice as much plutonium per unit of energy produced in HWR SF versus LWR SF.

Figure 7.9 shows the result of KI-4 — SF generation normalized to energy production. Owing to much lower burnup (7 GW·d/t versus 45 GW·d/t) and slightly lower thermal efficiency (30% versus 33%), the HWR produces much more SF than the LWR. The system average is close to the LWR value because LWRs represent 94% of the system. (Note: for the rest of this subsection, only the high case results are shown.)

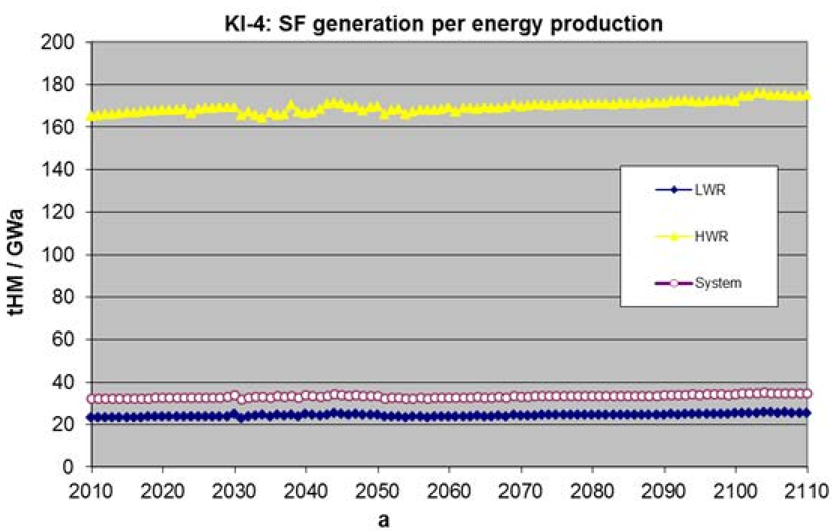


FIG. 7.9. Spent fuel generation normalized to energy produced (BAU high case).

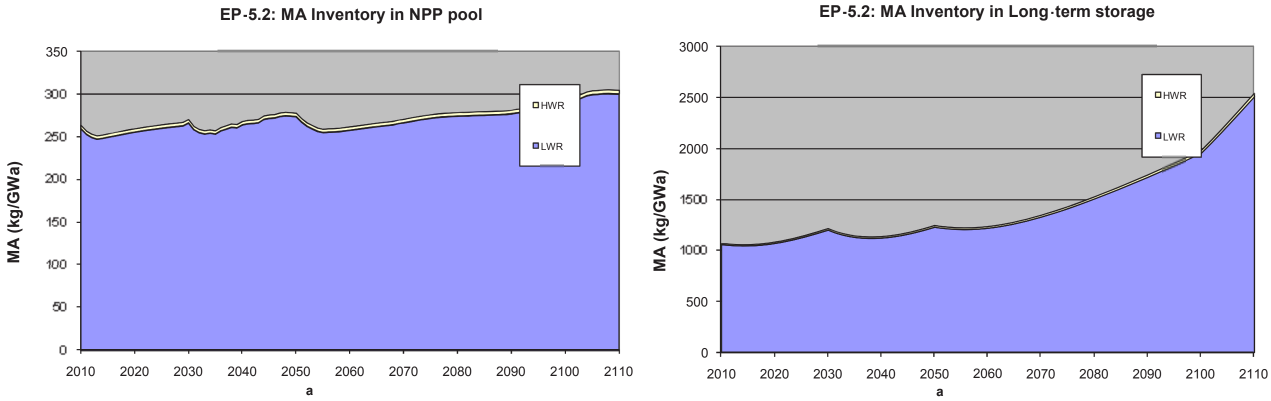


FIG. 7.10. Normalized minor actinide inventory in spent fuel in storage at reactors (left) and in long term storage (right) (BAU high case).

KI-5 addresses waste inventories. For the framework, a method for calculation of waste volume was not established, so KI-5 is not reported. EPs within KI-5 include radiotoxicity and MA content of wastes. MA content is shown in Fig. 7.10. Since all SF eventually goes to waste in the BAU scenario, the MAs are shown for both the fuel in cooling storage at the reactors and the cooled fuel in long term storage. Note the difference between the graphs in Figs 7.10 and 7.8. The MA content in HWR SF is much lower on a per cent basis than in LWR SF due to the lower burnup which does not allow as much fluence for buildup of the heavier isotopes.

The final indicator of interest for the BAU case is KI-6, SWUs per energy production, as shown in Fig. 7.11. The left graph shows the same behaviour as the uranium utilization graph, where the lead time for enrichment prior to fuel loading results in a small increase in the normalized value when the growth rate increases (just prior to 2030 and 2050) and a small decrease when the growth rate decreases (just prior to 2100). The magnitude and timing of this difference depends on the time difference between when enrichment occurs and when fuel is loaded (especially the first core). This difference varies with different fuel cycle codes. The other feature of the figure is the lower system average value versus the LWR value, which is due to averaging in the energy produced by the HWRs. Since the HWRs are running on natural uranium, there is no enrichment and, therefore, no SWUs for the HWR fuel. The right graph shows the SWUs without normalization, which is dominated by growth in the total number of LWRs.

In addition to the KIs and EPs presented above, there are several other parameters of interest for fully describing the framework case. Several of these have already been presented when values without normalization were provided. Four additional parameters are presented in Figs 7.12 and 7.13.

Figure 7.12 shows the annual fuel fabrication and fuel discharge amounts, which are almost identical except for the time delay for the period when the fuel is in the reactor. The fabrication rate shows the impact of initial cores, including the drop just before 2100 when the rate of new reactors drops as the growth rate goes to zero. The discharge rate does not show this drop because the rate of fuel discharge is constant once the reactors are operating. The length of the delay between fabrication and discharge will vary depending on the approach used by the fuel cycle code. Some codes explicitly model each fuel batch, and will not show discharge from the LWRs until ~1.5 years after initial startup. Other codes model fleet average behaviour, and will show a continuous discharge and reload rate as soon as the reactor is operating.

Figure 7.13 shows the SF inventories in the system. In the GAINS base case scenarios, SF is assumed to be stored at the nuclear power plant only until the minimum cooling time is met, after which it is moved to long term storage or recycled. Even though the HWRs only represent 6% of the total system electricity generation, the relatively low burnup of the HWR fuel results in a significant share of the total SF quantities coming from the HWRs.

7.3.2. BAU with introduction of fast reactors (BAU–FR)

The next set of framework cases includes introduction of FRs. The initial transition to FRs is controlled (specified) in the framework for the first 30 years. The FRs are initially introduced starting in 2021 at a low rate of 1 GW·a of energy generation per year for the first ten years. The rate of increase is significant beginning in 2031,

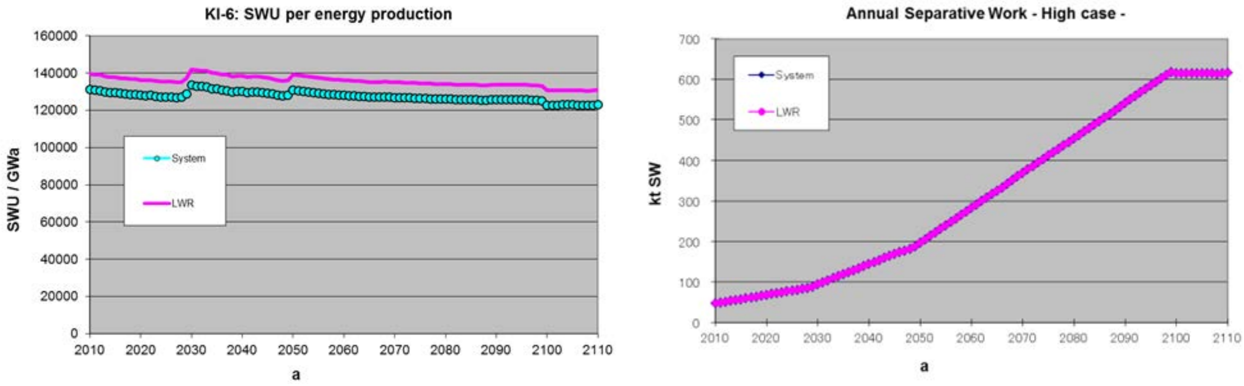


FIG. 7.11. SWU load normalized per unit of energy (left) and without normalization (right) (BAU high case).

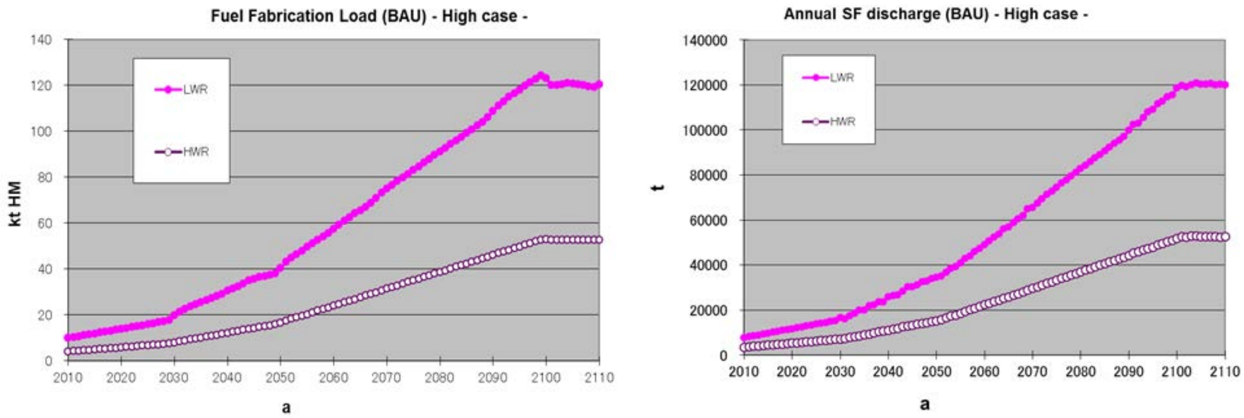


FIG. 7.12. Fuel fabrication rates (left) and discharge rates (right) (BAU high case).

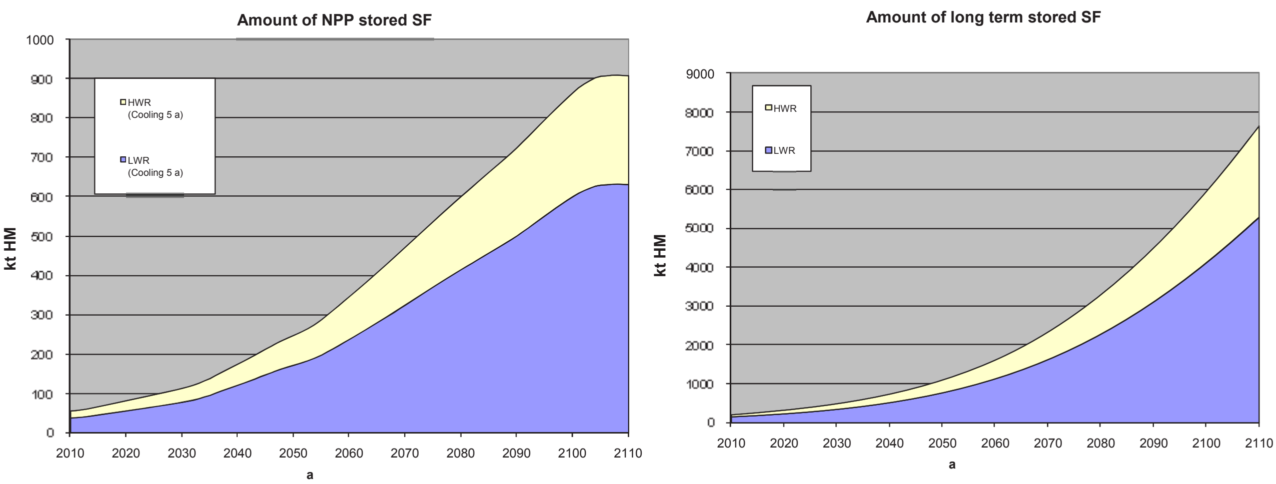


FIG. 7.13. Spent fuel inventories in cooling storage (left) and long term storage (right) (BAU high case).

but is still limited through 2050 — to 9.5 GW·a of generation per year for the moderate case and 19.5 GW·a for the high case. After 2050, the FR build rate is only limited by the amount of plutonium available and the overall growth rate.

In the framework case, HWRs represent a separate community of reactor owners from LWRs. FRs are supplied plutonium for initial cores from the recycling of LWR fuel, while HWR fuel is not recycled. For this reason, in the framework base case, FRs are coupled to, and replace, LWRs but not HWRs. Therefore, the overall growth curves (Table IV–1 in Annex IV) retain a 6% share for HWRs with the other 94% split between the LWRs and FRs.

The annual FR build rate during the 30 year introduction period from 2021 to 2050, is shown in Table 7.1. The rate is provided in whole reactors (870 MW(e)) for fuel cycle codes that use whole reactors and in partial reactors for codes that model fractional reactors. The fractional reactor rate stays closer to the growth curve. Analysts will need to recalculate these rates when modelling alternate cases using a different size of FR or a different load factor. The objective is to have a total generation rate of 10 GW·a from FRs in 2030 for both growth cases and a total in 2050 of 400 GW·a for the high case and 200 GW·a for the moderate case.

TABLE 7.1. FAST REACTOR SPECIFIED INTRODUCTION RATES FOR THE BAU–FR FRAMEWORK BASE CASES

New reactors per year							
	Moderate and High cases			Moderate case		High case	
Year	Whole reactors	Partial reactors	Year	Whole reactors	Partial reactors	Whole reactors	Partial reactors
2021	2	1.3523	2031	13	12.8465	26	26.3692
2022	1	1.3523	2032	13	12.8465	27	26.3692
2023	1	1.3523	2033	13	12.8465	26	26.3692
2024	2	1.3523	2034	12	12.8465	26	26.3692
2025	1	1.3523	2035	13	12.8465	27	26.3692
2026	1	1.3523	2036	13	12.8465	26	26.3692
2027	2	1.3523	2037	13	12.8465	26	26.3692
2028	1	1.3523	2038	13	12.8465	27	26.3692
2029	1	1.3523	2039	13	12.8465	26	26.3692
2030	2	1.3523	2040	13	12.8465	26	26.3692
Total	14	13.523	2041	12	12.8465	27	26.3692
			2042	13	12.8465	26	26.3692
			2043	13	12.8465	26	26.3692
			2044	13	12.8465	27	26.3692
			2045	13	12.8465	26	26.3692
			2046	13	12.8465	26	26.3692
			2047	12	12.8465	27	26.3692
			2048	13	12.8465	26	26.3692
			2049	13	12.8465	26	26.3692
			2050	13	12.8465	27	26.3692
			Total	257	256.930	527	527.384

The ‘F1’ FR specification is based on a uranium/plutonium fuel. Sufficient plutonium is needed to support the FR introduction rates. Plutonium supply is dependent on the LWR fuel reprocessing rate. The framework specification is for ‘unlimited separations’ starting as early as 2010, which provides excess plutonium quantities. However, using unlimited separations also results in high separation rates for a very short period while the stored inventory built up since 1970 is reprocessed, followed by a much lower level based only on the current rate of discharge and cooling. It is unrealistic to build expensive reprocessing facilities and only use them for a few years. The sudden increase of recovered plutonium also creates unrealistic problems with storage inventories.

To develop a more practical and potentially more realistic reprocessing rate, the introduction of new LWR reprocessing capacity was limited to units achieving reprocessing of 800 t/a of SF (counting only the HMs and FPs). At this unit size, the Rokkasho reprocessing plant in Japan is approximately one unit in size while the La Hague reprocessing plant in France is approximately two units in size. (Note: load factors for the reprocessing facilities were not specified, only the actual annual throughput.) Also, once added, capacity is not removed until the unit has operated for a full life (40 years or more).

Capacity for reprocessing of FR fuel was not limited, as at this point there is no basis for establishing a unit plant size. Instead, modellers should always provide sufficient capacity to process used FR fuel as soon as it has cooled.

The specific introduction rate for LWR SF reprocessing capacity should be determined by the analyst based on the specifics of their fuel cycle code. One consideration is how long after reprocessing before fuel is available for charging or recharging an FR. The framework specification for recycling is nominally half a year for reprocessing and half a year for new fuel fabrication or a total of 1 year after cooling before fuel is available for use. If a fuel cycle code takes longer than 1 year in total for recycling, extra time should be taken from the SF cooling time prior to recycling, such that the total out of reactor time is maintained, since it is important to preserve the total duration for fuel recycle (from reactor discharge to insertion back into a reactor). Variations in this duration can significantly affect results, as will be shown via a sensitivity analysis provided in the next section.

The objective should be to provide just the amount of reprocessing capacity sufficient to support the specified FRs during the specified introduction period. Each of the break-even ‘F1’ FRs requires ~9 t of plutonium from LWR SF for startup before they become self-sufficient through the reprocessing of their own SF. This includes just under 3 t of plutonium for the initial core plus ~2 t/a for reloads for the 3 years it takes for the initial discharged fuel to cool (2 years), be reprocessed (half a year) and fabricated into new fuel (half a year). Each tonne of ‘L1’ LWR SF reprocessed yields ~10 kg of Pu.

A second consideration in determining the LWR SF reprocessing capacity to use is to avoid having any significant inventories of excess separated material at the end of the FR introduction period. If there is a significant excess, then a large number of FRs can be built immediately after 2050 but the rate cannot be sustained and a spike occurs. If there is little excess, then the FR rate after 2050 is driven by the rate of LWR SF reprocessing, which should be relatively constant, and the FR build rate will be fairly steady.

The third consideration is to avoid a sharp decrease in LWR SF reprocessing throughput rates when the excess inventory of used LWR fuel is depleted. Ideally, the current rate of discharge should grow to almost the same magnitude as the reprocessing capacity around the time the excess inventory is fully worked off, resulting in a ‘soft landing’ and avoiding a sudden drop in FR build rates. To achieve this soft landing, it may be necessary to retire some reprocessing capacity — but only after it has operated for a full facility life. In the moderate growth case, this approach is used with a facility life of 40 years. In the high growth case, capacity retirement was not necessary due to continued growth.

Finally, if the analyst is using a fuel cycle code which models the construction period for reprocessing and fabrication facilities, then sufficient lead time must be included when ordering capacity.

Figure 7.14 shows the actual reprocessing loads used for the results shown in this section (both are shown on the same vertical scale). The flat areas for the LWR SF lines are when the reprocessing capacity is limiting the load. A minimal capacity of a single 800 t/a unit is introduced to supply plutonium for the first part of the FR introduction phase when only 1 GW·a of generation is added per year. The capacity is increased significantly for the second part of the introduction phase, to 21 600 t/a (26 additional units, 800 t/a each) for the high case to support 19.5 GW·a/a of new FRs, and to 10 400 t/a (12 additional units) for the moderate case to support 9.5 GW·a/a of new FRs.

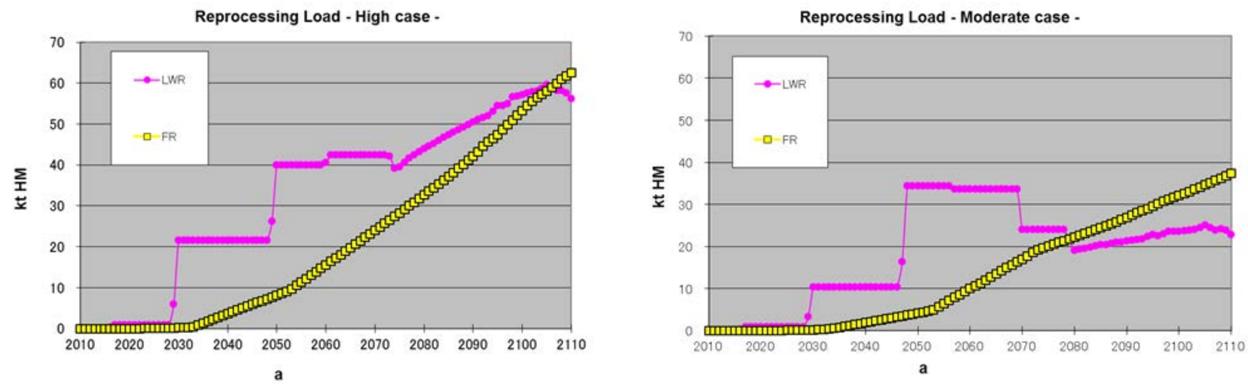
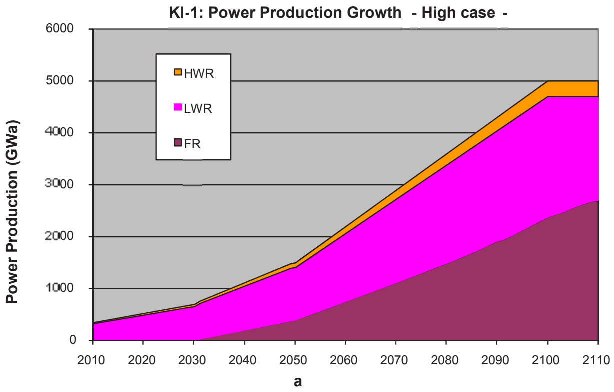


FIG. 7.14. Spent fuel reprocessing annual throughput (BAU–FR high case — left, moderate case — right).



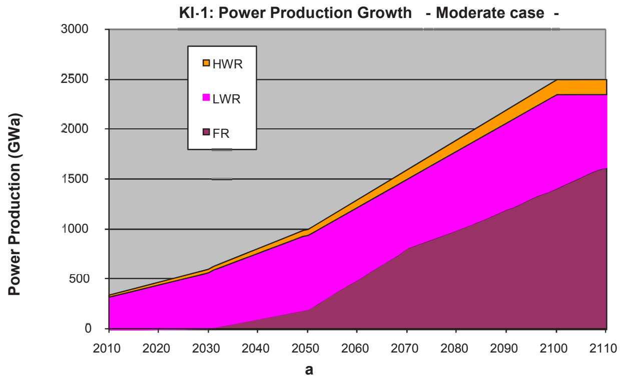


FIG. 7.15. Power production (BAU–FR high case — left, moderate case — right).

Near the end of the FR introduction period, the reprocessing capacity is increased again to work off the remaining excess inventory of cooled fuel. In the high case, the capacity is increased to 40 000 t/a (23 additional units) and then to 42 400 (3 additional units) until the excess SF in storage is all reprocessed shortly after 2070. At that point, there is a slight dip when the reprocessing capacity is not fully utilized (the 'soft landing'), but the total capacity (53 units) is maintained through replacement of retiring facilities. Once the excess inventory is worked off, the capacity is set to an unlimited value and the rate is determined by the rate of discharge from the LWRs (including retired cores).

The FR reprocessing capacity is always set as unlimited. For this reason, there should not be any significant inventory of FR fuel in long term storage. Instead, as soon as the fuel is cooled it should move through reprocessing.

To achieve the same result of working off the excess used LWR fuel in the moderate case, the capacity was increased to 34 400 t/a (30 additional units). However, the longer term fuel discharge rate is lower in the moderate case and this level of separations cannot be sustained. Thus, the capacities added to support the FR introduction phase were allowed to retire after 40 years. The slight dip prior to 2060 is the retirement of the initial single unit and the larger drop around 2070 retires an additional 12 units. The remaining capacity of 24 000 t/a (30 units) continues to operate for the remainder of the simulation. The excess SF is depleted around 2080, after which time the capacity is set to an unlimited value, as was done with the high growth case.

Figure 7.15 shows the generation rates for the BAU–FR framework base cases. Comparing these to Fig. 7.4, the total generation and HWR generation is the same, but a portion of the LWRs are offset by FRs. The results of the two growth rates appear similar up to 2050. This is because the y axis scale for the high growth case is twice that for the moderate growth case and the FRs specified to be built during the introduction phase are doubled in the high case. However, the overall growth through 2050 in the moderate case is closer to two thirds that of the high case. The higher relative growth in LWRs for the moderate case prior to 2050 results in relatively more plutonium for startup of FRs after 2050, and the growth curve for FRs is steeper for the 20 year period from 2050 to 2070. This coincides with the hump in the reprocessing rate during this time period (Fig. 7.14). As the reprocessing rate slows after 2070, so does the FR growth rate. Owing to the higher relative growth, the FR share is higher in the moderate growth case after 2050. While the energy generation ratio in 2100 is ~50:50 between LWRs and FRs in the high case, in the moderate case the ratio is ~40:60.

Figure 7.16 shows the new reactor commissioning rates. The two phases of the FR introduction period are the relatively flat areas on the FR lines (yellow lines) during the 2020s and from 2030 to 2050. The commissioning of FRs after 2050 is generally limited by plutonium availability because the FR used is a break-even reactor rather than a breeder. Each new reactor requires enough plutonium for the initial core plus at least three years' worth of recharge while the first fuel discharged from the reactor is cooled and recycled. The FR ('F1') has a relatively low burnup, resulting in the need for ~2 t of plutonium per year for recharge.

In the high case, the build rate after 2050 is still fairly flat while the flow of material from reprocessing of LWR fuel is constant. When the backlog of LWR fuel is worked off, the flow increases due to increasing numbers of LWRs and increasing retirements of LWRs. FRs are only able to support part of the total system growth and additional LWRs continue to be built until 2100 when growth stops.

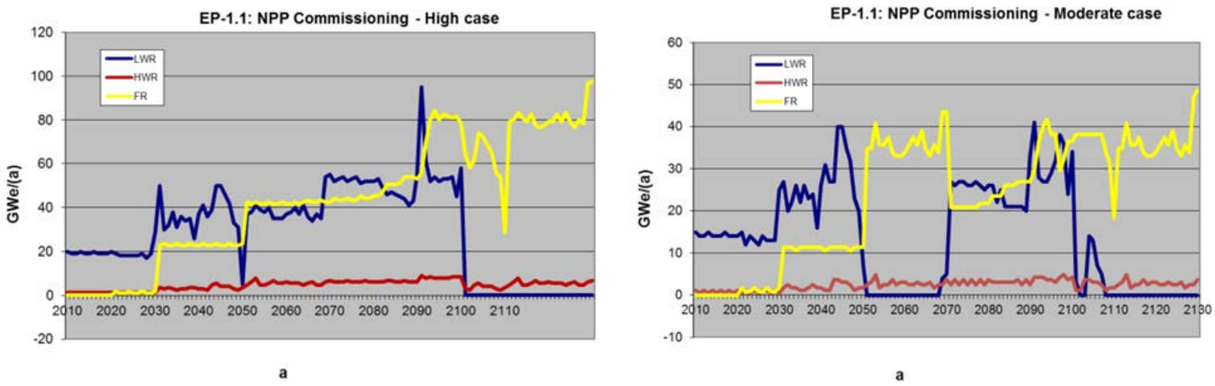


FIG. 7.16. New nuclear power plant commissioning rate (BAU–FR high case — left, moderate case — right).

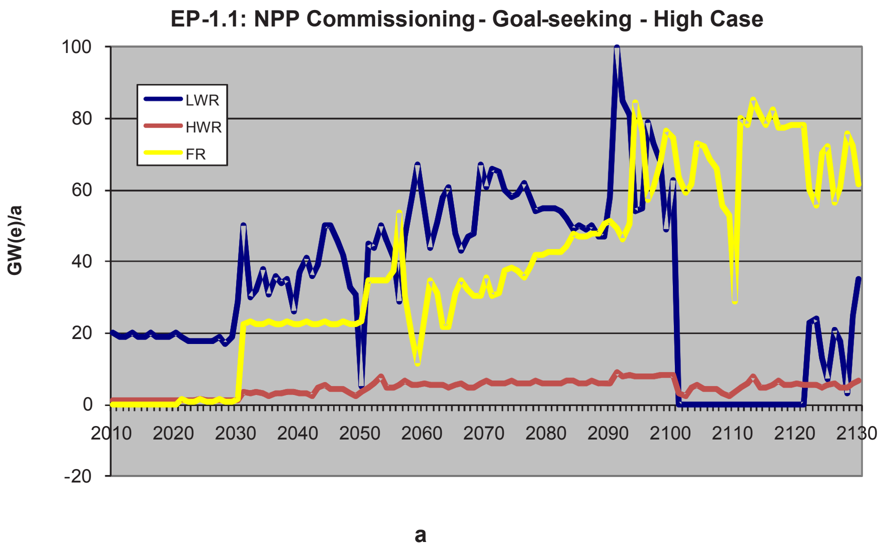


FIG. 7.17. Nuclear power plant commissioning rate using a goal-seeking algorithm — BAU–FR high case.

In the moderate case, there is enough plutonium available that FRs can support all of the growth and also replace any retiring LWRs. This continues for ~20 years until the reprocessing rate drops, after which the behaviour is similar to the high case.

The graphs in Fig. 7.16 were extended to 2130 to better show the behaviour after growth stops. Once growth stops, there is generally sufficient plutonium to replace all retiring LWRs with FRs, other than short periods with the retirement rate spikes (moderate growth case around 2115). This results in a rapid decrease in the number of LWRs. However, this is self-limiting because as the total number of LWRs continues to decrease, the plutonium supply from LWRs also decreases and the rate of LWR replacement by FRs will eventually slow (not shown on the graph).

The framework base case uses a break-even FR, where the used FR fuel provides just the amount of plutonium needed for fresh fuel. The behaviour shown in Fig. 7.16 will change if in an alternative analysis a breeder or burner FR is used. In the case of a breeder FR, once established, the new FRs will begin to generate more plutonium than they use and will become a second source of plutonium for new FRs. The result will be more FRs and fewer LWRs. In the case of a burner FR, the used FR fuel will only provide part of the plutonium needed and the existing fleet of FRs will require 'make-up' plutonium from the LWRs. This will reduce the plutonium available for new FRs, resulting in fewer FRs and more LWRs.

Depending on the fuel cycle code and the FR ordering strategy, the FR commissioning rate may initially oscillate after 2050 when there is no longer a specified rate to follow. Some fuel cycle codes require the analyst to specify the number of reactors to build at all times, while others also have goal-seeking algorithms built in that attempt to follow a strategy specified by the analyst. The results shown in Fig. 7.16 used the analyst-specified number of reactors method. Figure 7.17 shows the high case using the same code but employing a goal-seeking algorithm.

When using the goal-seeking method, some oscillations occur just after the FR introduction period ends when the FR orders are no longer specified (in this case around 2055). This is due to the transition in ordering strategy as the goal-seeking algorithm takes over and initially needs to find the equilibrium ordering rate. The oscillations quickly dampen out as this occurs. The oscillation may be retriggered if there is an abrupt change in the flow of LWR fuel through separations, such as a step change in separations capacity or when a large number of LWRs retire, as occurs in the framework case around 2090. The algorithm then needs to determine whether the change was a spike versus a new long term trend, after which the oscillations again dampen out. Some goal-seeking algorithms can be adjusted to be more conservative, resulting in much less oscillation but also in fewer total FRs being commissioned.

Figure 7.18 shows the annual SF discharges. All of the discharges grow annually in the high case until 2100, when replacement of LWRs by FRs results in the LWR discharges dropping while the FR discharges continue to rise. The HWR discharges level out after 2100 due to the growth rate dropping to zero. In the moderate case, the same behaviour is observed except for that during the period from ~2050 to 2070 the LWR rate slightly decreases because enough FRs are being built to support all of the required growth and also to replace the retiring LWRs.

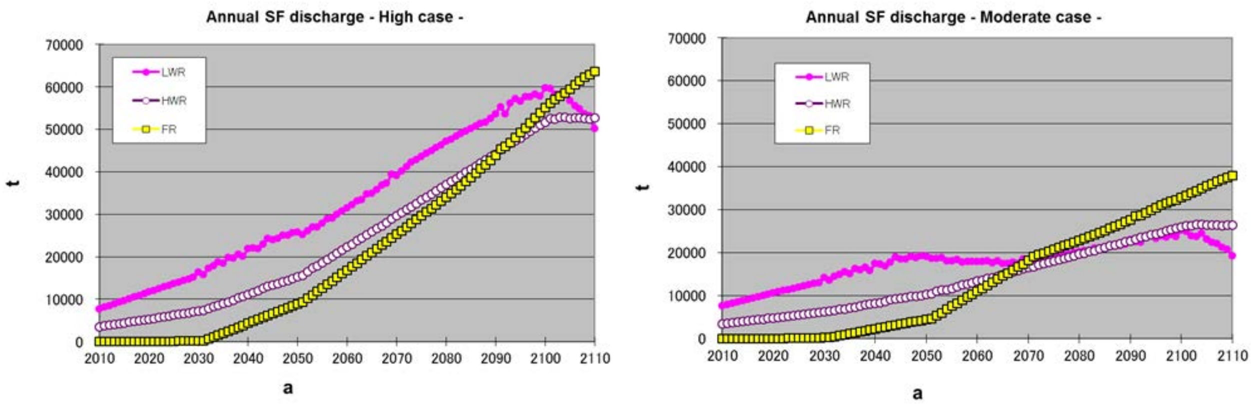


FIG. 7.18. Spent fuel discharge (BAU–FR high case — left, moderate case — right).

Figure 7.19 is a group of graphs showing the SF inventories, with the high case on the left and the moderate case on the right. The top row of graphs is total inventory of SF, while the middle and bottom rows are the SF cooling at the nuclear power plants and the cooled fuel in long term storage ready for either reprocessing or disposal.

All of the discharged fuel from Fig. 7.18 initially goes to storage. The amount at the nuclear power plant grows with the total fleet size until after 2100, when the number of LWRs begins to decrease rapidly. While the number of FRs is growing after 2100, the combination of higher burnup fuel and shorter cooling time results in a much smaller amount of FR fuel versus LWR fuel for the same level of energy generation.

Reprocessing does not directly impact the nuclear power plant storage inventory, but does impact the long term storage. When reprocessing begins, the LWR reprocessing rate (Fig. 7.14) is below the discharge rate and inventories continue to grow. However, as the reprocessing rate increases, the inventory begins to decrease, and the impacts of the large changes in separation capacity can be seen on the LWR inventory. Around 2070 (high case) to 2080 (moderate case), the remaining inventory of cooled LWR fuel is depleted and the reprocessing rate is determined only by the fuel cooling at the nuclear power plants. In the moderate case only, there is a dip in both total inventories and long term inventories after 2050, which reflects the period when FRs are replacing LWRs and the LWR discharge rate is dropping. During this same period, the LWR SF in long term storage is dropping quickly due to the relatively high reprocessing rate.

The HWR fuel is not recycled, so the inventory of stored HWR fuel continues to grow. Fast reactor fuel is reprocessed as soon as it has cooled, so there is never any significant inventory in long term storage.

Figure 7.20 shows the result for KI-2 energy production per natural uranium use. It should be noted that while the moderate case growth is only half that of the high case, the normalized results are very similar. By using normalized values for the KIs, cases with different overall growth rates can be compared directly.

The normalized performance of the LWRs and HWRs has not changed versus the BAU case (see Fig. 7.6). However, the figure shows an overall improvement in system performance on this indicator once FRs are introduced.

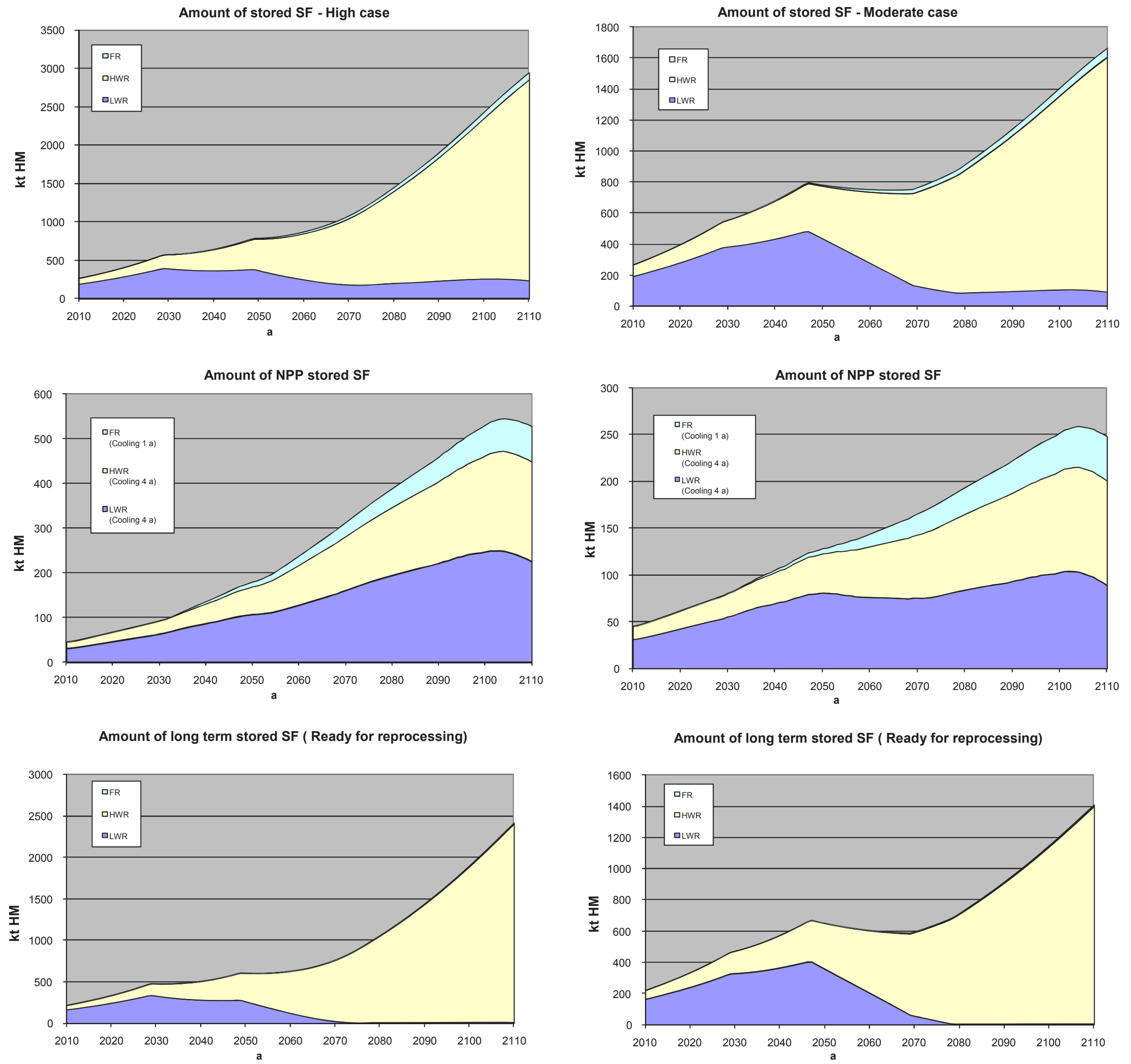


FIG. 7.19. Spent fuel storage, including total in storage (top) and the portions stored (cooling) at the nuclear power plant (middle) and in long term storage (bottom) (BAU–FR high case — left, moderate case — right).

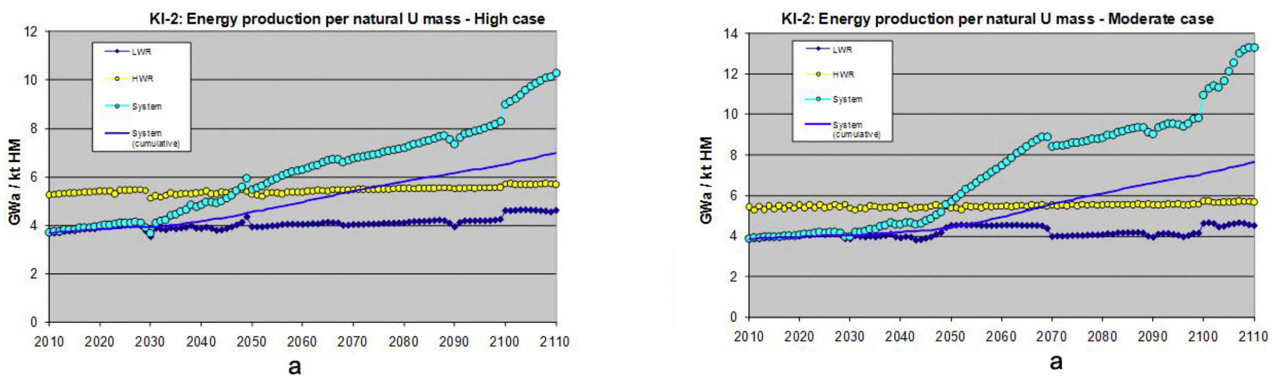


FIG. 7.20. KI-2 energy production normalized to uranium use (BAU–FR high case — left, moderate case — right).

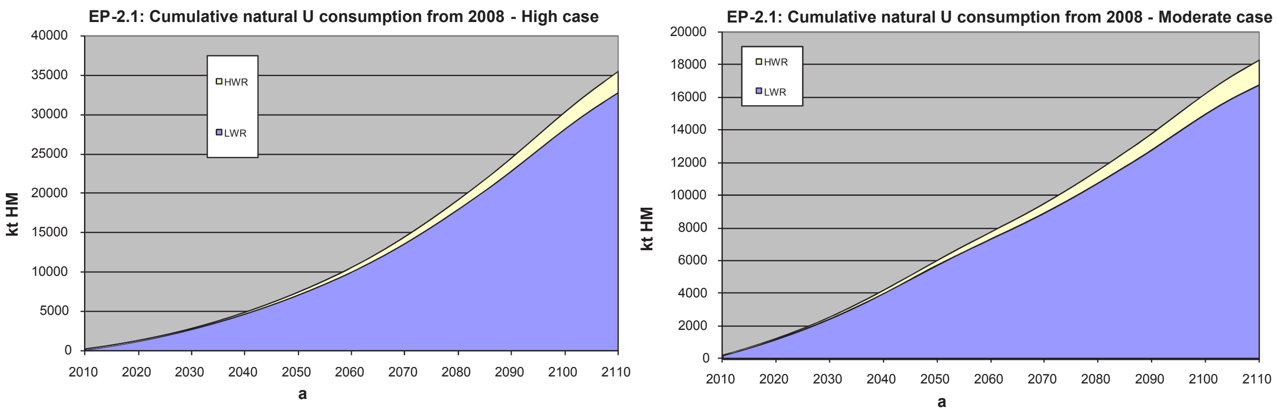


FIG. 7.21. EP-2.1 cumulative natural uranium use (BAU–FR high case — left, moderate case — right).

This is because the FRs are primarily fuelled by depleted uranium left over from enrichment for LWR fuel, which is otherwise not utilized. Since the FRs are generating additional electricity without mining additional natural uranium, total system energy production per natural uranium use increases. The faster share growth of FRs in the moderate case is reflected in better overall system performance.

The reduction in depleted uranium stocks parallels the total change in natural uranium usage, shown by reactor type in Fig. 7.21 using the framework EP EP-2.1. The impact of the HWR fuel is small due to the small fraction of total energy produced by the HWRs, but also because no enrichment is required for the HWR fuel.

Figure 7.22 shows the total natural uranium usage versus the identified and projected resources in the current Red Book. These results can be compared to the BAU case (shown previously in Fig. 7.7).

With the introduction of recycling and a break-even FR, the scenario shows total uranium usage exceeding projected resources for the high case, while roughly equalling the currently identified total projected uranium resource through the end of the century in the moderate growth case, and even in that scenario the resource value is eclipsed early in the next century.

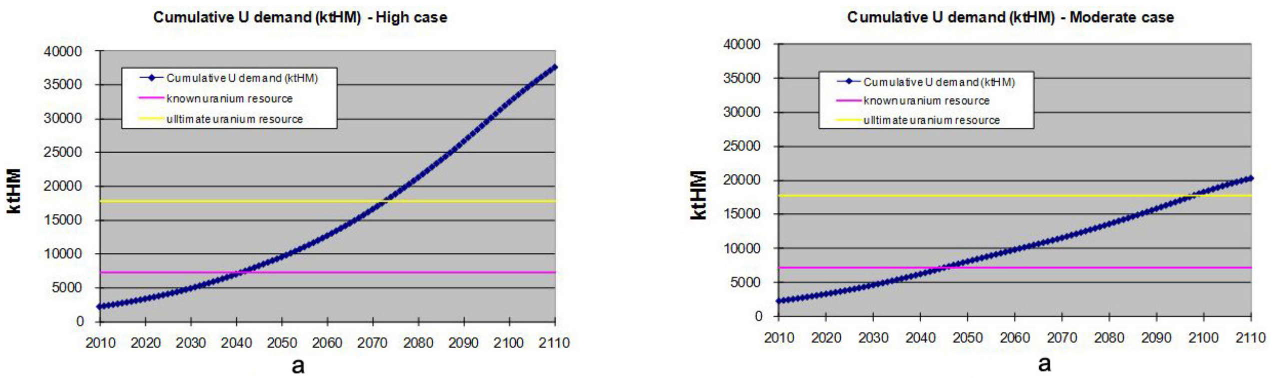


FIG. 7.22. Cumulative natural uranium use versus current uranium resources (BAU–FR high case — left, moderate case — right).

The framework provides a starting point for analysing more sophisticated fuel cycle options and a reference for analysts investigating the impact of recycling and FRs on uranium utilization. The introduction of breeder FRs and increases in projected resources are two primary areas for analysis.

KI-3 is the normalized quantity of plutonium, measured at a number of points in the fuel cycle. Figure 7.23 shows at four of these points for the BAU–FR high and moderate cases — in the reactor cores, in storage at the nuclear power plant, in long term storage and in reprocessing. In each case, the vertical scale for both the high and moderate cases are the same to facilitate comparison.

Plutonium in-core is dominated by the FR fuel, which has ~10 times as much plutonium as the LWR fuel at the end of irradiation. The moderate case does slightly worse on this indicator due to the higher share of FRs.

The same result is seen in the at nuclear power plant storage, except that the HWR and LWR shares are relatively higher due to the longer cooling time. (Note: For the cases shown, to achieve six years recycle time for

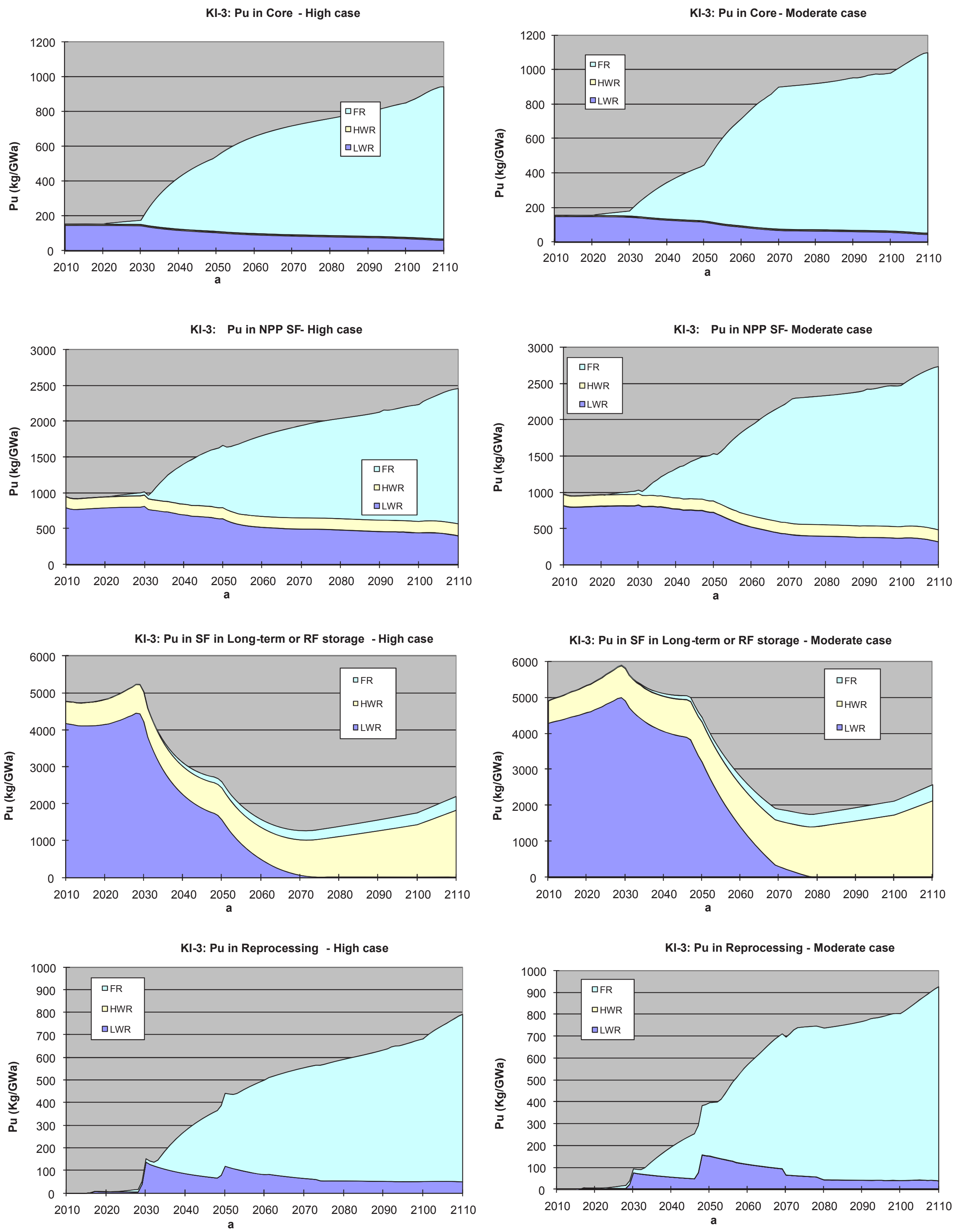


FIG. 7.23. KI-3 plutonium inventories in reactor cores (top), at reactor storage (top middle), in long term storage (bottom middle) and in reprocessing (bottom) (BAU–FR high case — left, moderate case — right).

LWR fuel and three years for FR fuel the at reactor storage times are only four years for LWR and HWR fuel, and one year for FR fuel.)

The high case continues to perform slightly better than the moderate case for long term storage and for material in reprocessing, again due to the higher share of FRs in the moderate case and the higher concentration of plutonium in FR fuel. In reprocessing, the impact of the reprocessing rate changes the result in step changes for the energy-normalized LWR values that become smaller over time as the amount of energy generation increases.

The results for nuclear power plant storage and long term storage can be compared to Fig. 7.8 for the BAU case (different scales). The BAU case is much better than the BAU–FR case for plutonium in fuel stored at the nuclear power plants, due to the higher concentration of plutonium in FR fuel. However, the BAU–FR case is much better than the BAU case for fuel in long term storage because both the LWR and FR fuel is recycled.

KI-4, the normalized SF generation rate is shown in Fig. 7.24. The same scale is used for both graphs as was used for the BAU case in Fig. 7.9. The LWR and HWR rates are essentially unchanged versus the BAU case, with the FR rate slightly lower than the LWR rate. While the FR burnup is lower than the LWR (37.7 versus 45 GW·d/t), the higher thermal efficiency (41.4 versus 33%) allows the FR to generate ~5% more electricity per tonne of discharged fuel (37.7 × 0.414 versus 45 × 0.33).

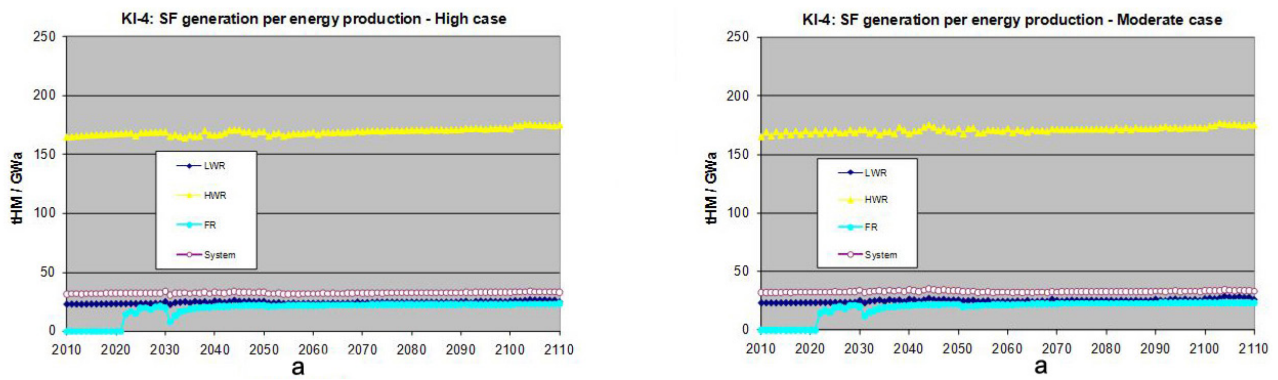


FIG. 7.24. KI-4 spent fuel generation normalized per annual energy generation (BAU–FR high case — left, moderate case — right).

As discussed previously, the method for calculating KI-5 has not been established, so only one of the EPs is shown here — EP-5.2, the MA inventory at different points in the system. The results shown in Fig. 7.25 are similar in form to those in Fig. 7.23, showing the MA content in the reactors, in discharged fuel stored at the reactors, in SF in long term storage, and that that has been separated and is in waste.

The in-core inventories of MA decrease as FRs replace LWRs. For the fuel and reactors used in the framework cases (L1 and F1), there are no MAs in fresh fuel and the per cent of MAs at discharge are very similar (0.1% for the LWR and 0.11% for the FR). However, the much larger core fuel loading in the case of the LWR (77 versus 28 tHM/GW(e)) results in a much higher amount of MAs in the LWR cores.

The moderate case shows slightly lower parameter values in-core due to the higher share of FRs. However, the values in long term storage remain high longer due to relatively higher LWR SF quantities. As relatively more LWR fuel is reprocessed to support fuelling of the higher share of FRs, the MA in waste value is slightly higher.

Compared to the BAU case (Fig. 7.10), the normalized nuclear power plant inventory parameter values are lower in 2010 due entirely to the shorter cooling time used prior to reprocessing. However, the values stay fairly constant in the BAU case throughout the scenario, while they drop as FRs are introduced in the BAU–FR cases. The parameter values in long term storage are higher and rise steadily in the BAU case, but drop in the BAU–FR cases as LWR fuel is reprocessed and the MAs move to the waste.

Two additional proliferation indicators are contained in KI-6. The first is SWUs per energy production (shown in Fig. 7.26). The initial values in 2010 are the same as the BAU case (Fig. 7.11). However, as FRs are introduced, the values improve significantly, with the moderate case improving more due to the higher final FR share.

The other KI-6 indicator is separation of direct use material (Pu) per energy production (shown in Fig. 7.27). There is no equivalent for this indicator in the BAU case. Both BAU–FR cases show similar values for this indicator, with the growth of FRs resulting in increasing amounts of material separated per unit of energy produced. The moderate case values are somewhat higher due to the higher FR share resulting in a relatively higher amount

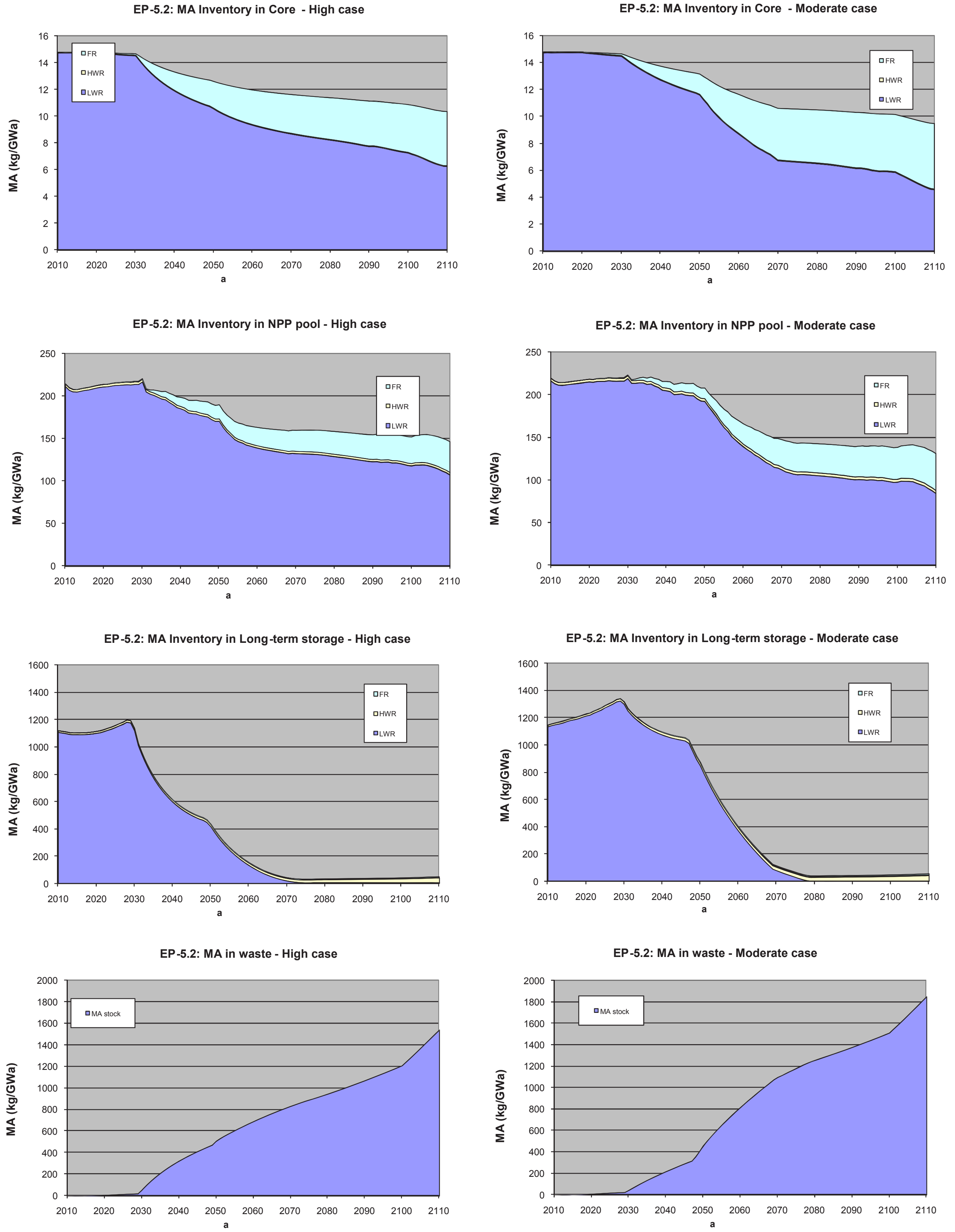


FIG. 7.25. EP 5.2 MA inventories in reactor cores (top), at reactor storage (top middle), in long term storage (bottom middle) and in waste (bottom) (BAU–FR high case — left, moderate case — right).

of separations. As discussed previously, the combination of lower burnup but higher thermal efficiency results in ~5% less SF per unit of energy produced by the FRs. However, the FR SF contains 12 times as much plutonium per tonne (12.0 versus 1.04%).

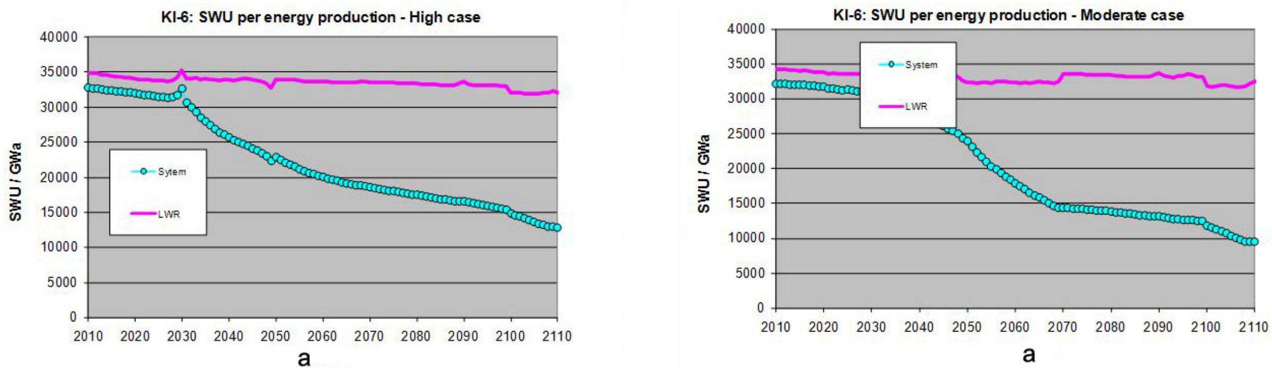


FIG. 7.26. KI-6 SWUs per energy production (BAU–FR high case — left, moderate case — right).

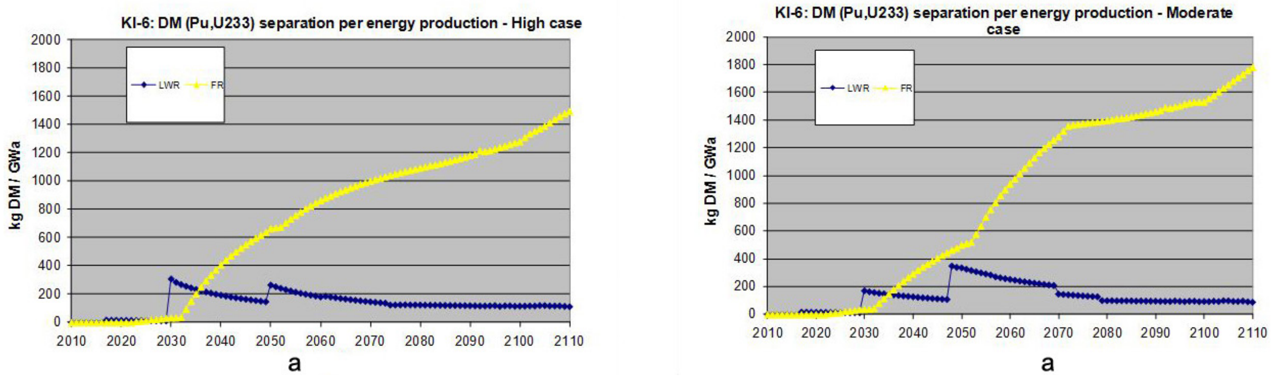


FIG. 7.27. KI-6 separated direct use material per energy production (BAU–FR high case — left, moderate case — right).

7.4. HETEROGENEOUS WORLD CASES

Special attention is focused on heterogeneous modelling of nuclear energy deployment. The heterogeneous description of the world is more realistic and reveals regional peculiarities of nuclear energy deployment. The CP GAINS approach to the heterogeneous world growth modelling is based on the grouping of countries or reactor owners using similar fuel cycle strategies. A focus of the GAINS project is to consider synergetic development of global nuclear energy with interactions between these groups in order to understand how synergy impacts on the main indicators such as uranium cumulative demand, SWU, SF amount and plutonium availability.

The heterogeneous models extend the BAU–FR homogeneous case by dividing the world into three non-geographic groups where the countries within a group all adopt the same fuel cycle strategy. The first nuclear power group (NG1) adopts recycling and a transition to FRs, as in the homogeneous BAU–FR case. NG2 continues with the BAU strategy of a once-through fuel cycle without recycling. NG3 starts with no NESs and introduces LWRs beginning in 2008. The 6% global HWRs are all modelled as part of NG2.

The non-synergistic and synergistic heterogeneous story lines were described in Section 3. Figure 3.2 showed fuel cycles for the three groups for the non-synergistic scenario. Figure 3.3 showed the addition of possible material movements between groups in the synergistic case. For the synergistic framework base cases, only the solid lines of material movement between groups are used (fresh fuel and SF), and NG3 simplifies to only LWRs (see Fig. 3.5).

7.4.1. Initialization data and growth curves for heterogeneous cases

The heterogeneous cases use the same overall nuclear electricity demand curves for high growth and moderate growth as the homogeneous cases.

Until 2008, the heterogeneous framework base cases assume 50% of world nuclear power generation is in the recycling fuel cycle group (NG1) and 50% in the once-through fuel cycle group (NG2), so total global generation is the same as shown in Table IV–1 of Annex IV, but split 50:50 between NG1 and NG2. However, HWR fuel is assumed not to be recycled and, therefore, 100% of the HWRs are in NG2. This results in more LWRs in NG1 than in NG2.

For the non-synergistic case, no movement of fuel (fresh or used) occurs between NG1 and NG2. This limits the amount of LWR SF available in NG1 for starting FRs.

After 2008, LWRs are introduced in NG3. The growth rate results in the equivalent of an additional 5% generation in 2030, 10% in 2050 and 20% in 2100 in NG3 versus NG1 and NG2 combined. To provide the specified energy levels at the key points of 2030, 2050 and 2100 listed in Section 5 using the same total energy generation as used in the homogeneous case, the total energy generation in the heterogeneous case is split into 21 shares in 2030, with one share to NG3 (the additional 5%) and 10 each to NG1 and NG2. In 2050, the split is 11 shares, again with one share (the additional 10%) to NG3 and the remainder split between NG1 and NG2. In 2100, the split is 5 shares, to arrive at the final energy mix of 40% NG1, 40% NG2 and 20% NG3. The area between the starting point of 2008 and each key point is modelled as linear growth in each NG. Table IV–3 in Annex IV provides the GW·a levels for each reactor type for each NG for both the moderate and high growth cases. To get the required reactor capacity in GW(e), these numbers are divided by the load factor of 0.85.

The introduction rates for FRs in NG1 in the heterogeneous cases are the same as for the homogeneous BAU–FR case, as shown in Table 7.1. However, as will be shown later, for the high growth case there is not sufficient LWR SF available to NG1 to build the full number of specified FRs during the introductory phase. The location and, therefore, the availability for reprocessing of used LWR fuel is the primary difference between the homogeneous BAU–FR and the heterogeneous cases.

7.4.2. Non-synergistic cases

In the heterogeneous non-synergistic cases, there is no movement of material between NGs. Each group has its own fuel cycle facilities to mine, convert, enrich and fabricate fresh fuel and to store and/or dispose of SF. The only fuel available to reprocess for FRs is the fuel in NG1.

The LWR SF reprocessing rates used are shown in Fig. 7.28.

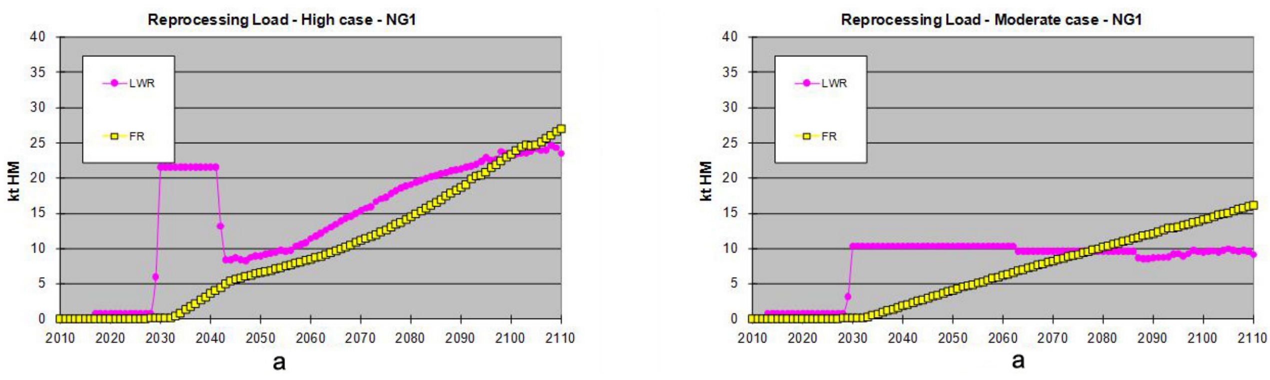


FIG. 7.28. Spent fuel reprocessing annual throughput (heterogeneous non-synergistic high case — left, moderate case — right).

In the high growth case, the same initial reprocessing rate is used as in the homogeneous BAU–FR case. This is because the same numbers of FRs are to be built and so they require the same amount of SF processed. However, a shortage of SF occurs after 2040, causing the actual reprocessing rate to fall well below what is needed. While 400 GW·a of electricity is desired to be generated by FRs by 2050, only ~300 GW·a is achieved.

Figure 7.29 shows the inventory of cooled LWR fuel available for processing in NG1. Because so many reactors are in NG2, there is not as much of an inventory of SF in NG1 as there was in the homogeneous BAU–FR case when reprocessing begins, and the ongoing LWR discharge rate is insufficient to keep up with the needed reprocessing rate. The result is the inventory approaching zero shortly after 2040. (The small inventory after 2040 is due to the code used requiring fuel to spend a one time step in dry storage before it is sent to reprocessing. This was not observable in previous cases because the vertical scale was much larger.)

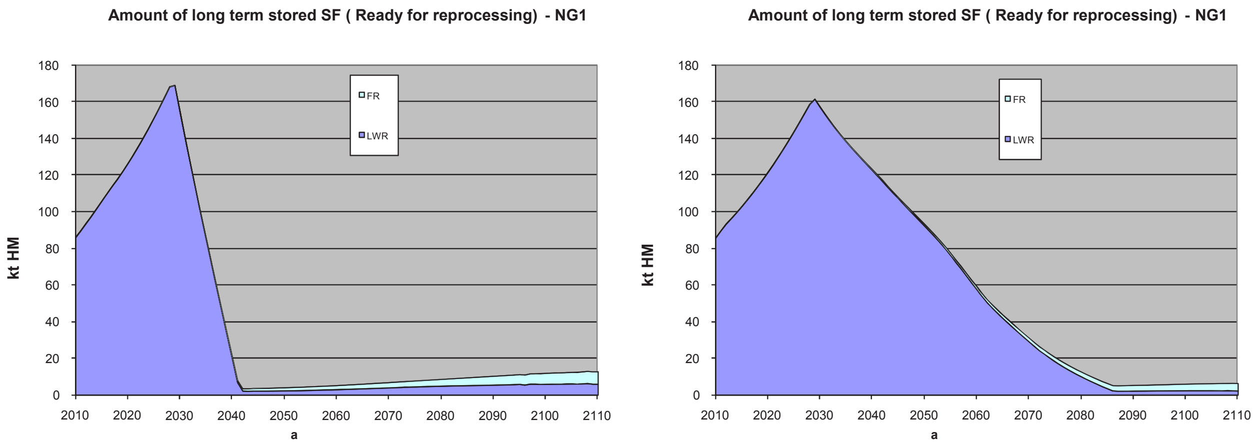


FIG. 7.29. Spent fuel in long term storage (heterogeneous non-synergistic high case — left, moderate case — right).

In the moderate growth case, the FR introduction rate is half as large (200 GW·a by 2050), so less used LWR fuel reprocessing is needed. While the growth rate after 2008 is somewhat lower, there is still sufficient discharge to maintain the lower reprocessing rate until ~2085. In this case, the initial reprocessing unit is allowed to retire after 50 years, but the additional capacity is maintained.

Figure 7.30 shows KI KI-1 for each group and globally (all three groups combined). NG1 and NG2 have twice the generation of NG3 (note difference in vertical scale). NG1 shows the transition to FRs, which can be compared to the BAU–FR homogeneous world case in Fig. 7.15. The share of FRs in NG1 is higher than the global share in the homogeneous case, but the total global share (bottom row of Fig. 7.30) is much lower. This has an implication for infrastructure in NG1, where countries that adopt recycling will have a larger transition than the world on average. The non-synergistic scenario supports investigation of this issue. NG2 is similar to the BAU case (Fig. 7.4), except for the larger share of HWRs. NG3 shows the growth curve for the group of countries that begin to add nuclear power to their energy mix. The GAINS framework can be used to investigate different growth rates for NG3 by varying the fraction of world growth assigned to NG3.

The reduced availability of plutonium for commissioning of FRs in the NG1 high case can be seen where the rate of the FR generation growth is initially high and then quickly moderates. The net impact is a slightly smaller final share of FRs than in the moderate growth case which does not run into this limitation. In reality, it is likely that excess reprocessing capacity would not be constructed in NG1 and that the FR commissioning rate would, therefore, be more gradual without the initial steep growth.

Figure 7.31 shows EP-1.1, the commissioning rate of reactors in each group. NG1 shows the introduction rate for FRs as they offset LWRs. In the high growth case, there is initially a high introduction rate matching the specification, but the rate drops off after 2040 and then slowly builds as more LWRs provide additional Pu. The FR growth follows the separations rate (Fig. 7.28) closely until after 2100, when the FRs commissioned in the 2030s retire and are replaced. In the moderate growth case, the FR commissioning rate is much steadier, again following the separations rate.

In NG2, the LWR commissioning rate line is almost identical in shape to that in the BAU case (Fig. 7.5), as is the HWR line. Only the magnitudes are different. NG3 follows the growth curve much more closely because there are no retirements until 2070. The three growth rates (from 2008–2030, from 2031–2050 and after 2050) can clearly be seen, especially on the moderate growth graph. After 2070, retirements of plants begin and the

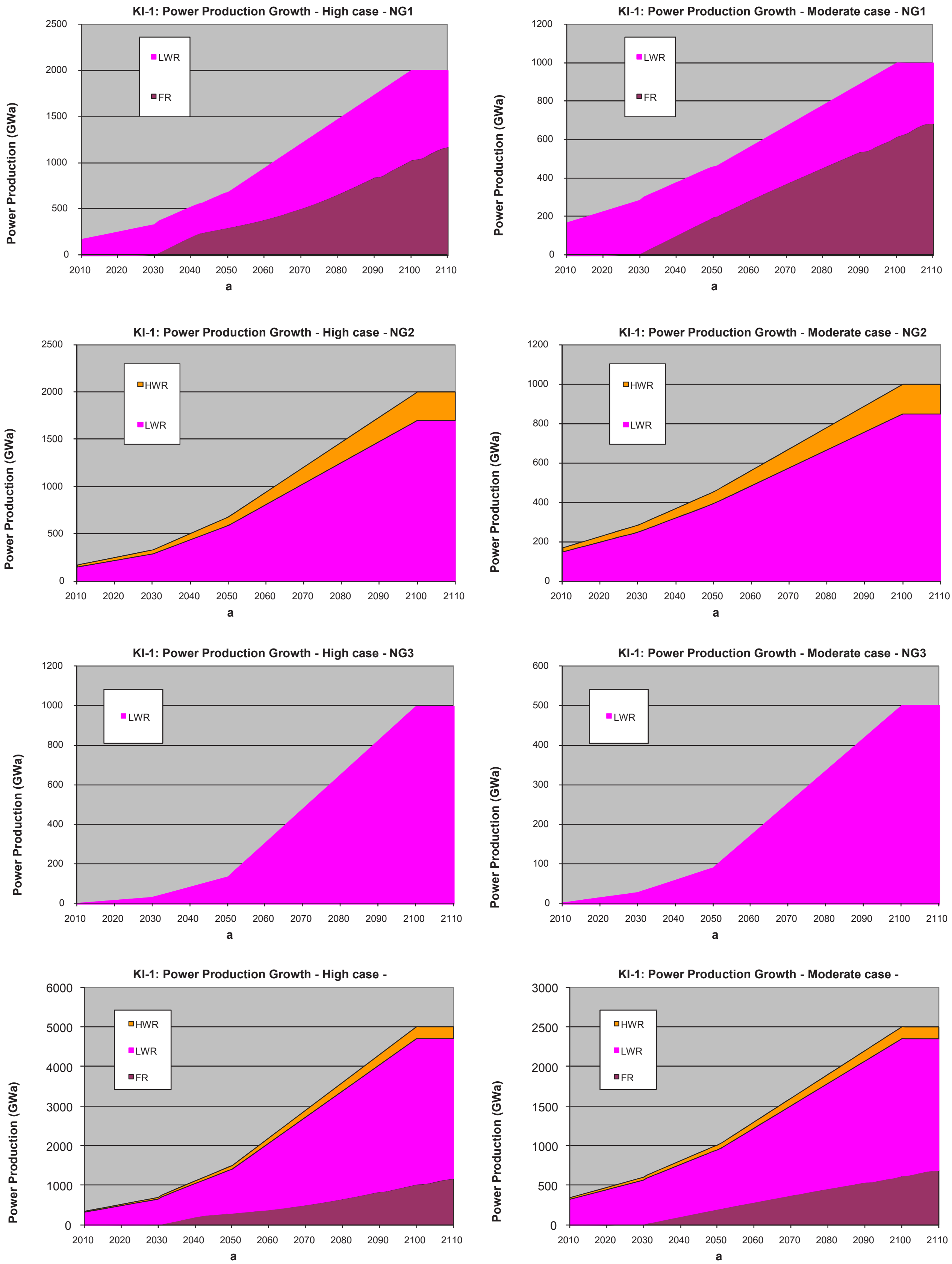


FIG. 7.30. KI-1 power production, heterogeneous non-synergistic high case (left) and moderate case (right) showing production by group and total (bottom).

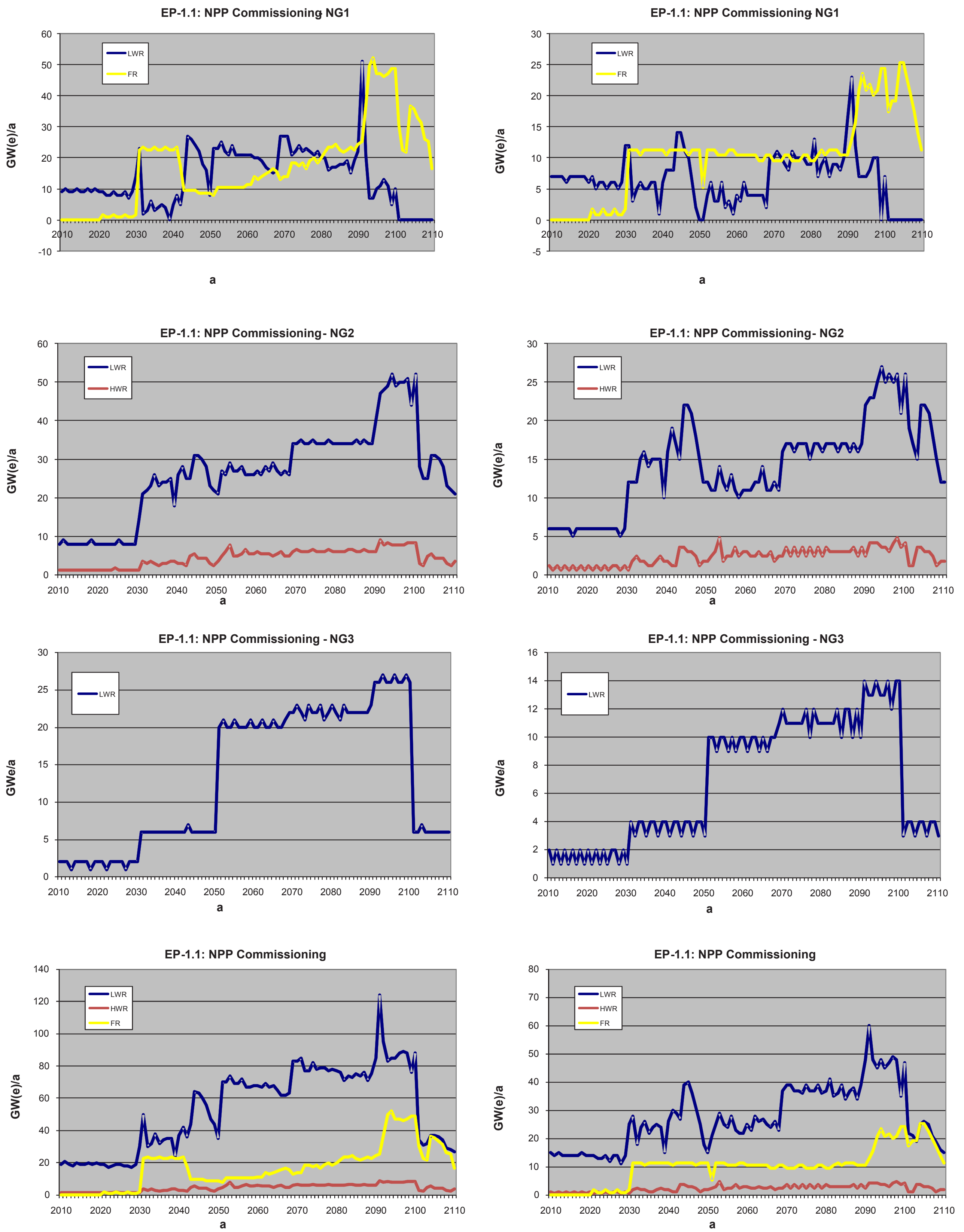


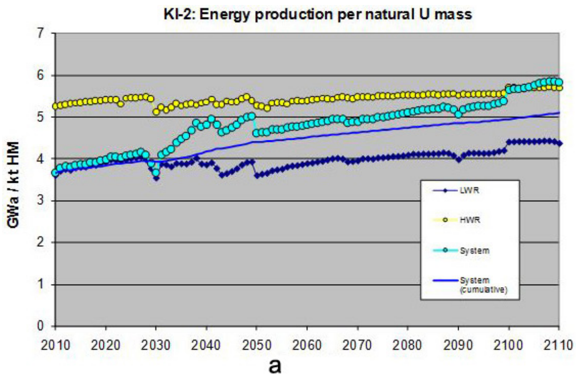
FIG. 7.31. EP-1.1 new power plant commissioning rate, heterogeneous non-synergistic high case (left) and moderate case (right) showing production by group and total (bottom).

commissioning rate is the combination of growth and replacement. After 2100, only replacement occurs and the commissioning rate falls sharply.

The global composite (bottom graphs) can be compared to the BAU–FR case (Fig. 7.16), with significantly lower FR growth overall due to less than half of the world's LWR SF being available to reprocess. The FR share of total electricity generation in 2100 is ~20% in the high growth case and ~25% in the moderate growth case.

For the next sets of indicators, only the global composite values are shown. The values for NG2 are similar to the BAU case and the NG3 values are similar to the LWR values of the BAU case. The composite values show the impact of the lower overall FR share versus the homogeneous BAU–FR case.

Figure 7.32 shows energy production per unit of natural uranium. The same general trend is seen as in the homogeneous BAU–FR case (Fig. 7.20), except significantly muted by the lower FR share. The moderate case does slightly better due to the higher share of FRs.



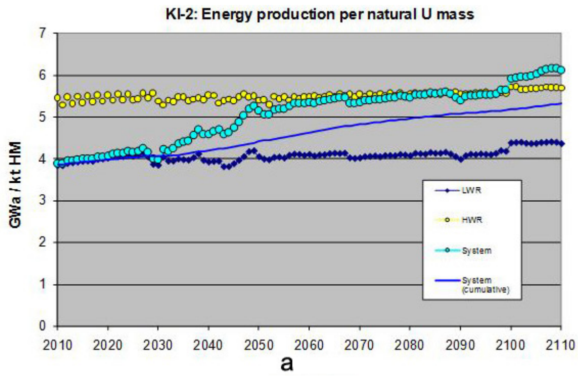
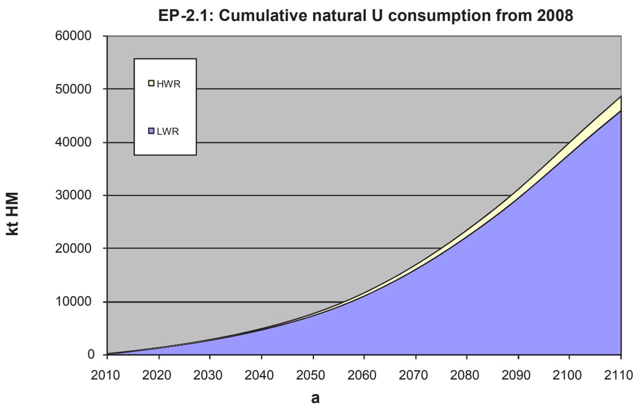


FIG. 7.32. KI-2 normalized uranium utilization, heterogeneous non-synergistic high case (left) and moderate case (right).

Figure 7.33 documents the cumulative natural uranium use. The values are between those of the BAU cases (Fig. 7.7) and the homogeneous BAU–FR cases (Fig. 7.21). Again, the improved uranium utilization of the homogeneous BAU–FR cases is muted by only a portion of the world participating in recycling.



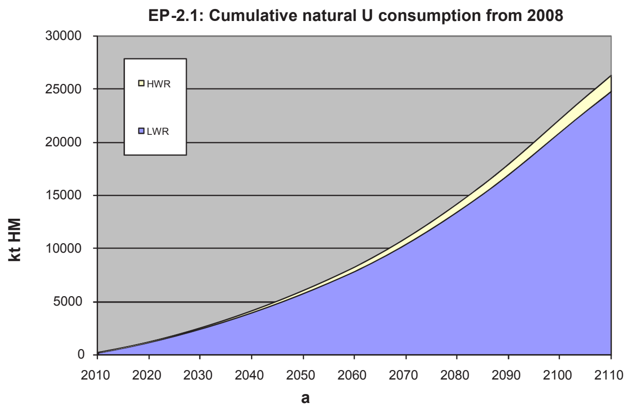


FIG. 7.33. EP-2.1 cumulative uranium utilization, heterogeneous non-synergistic high case (left) and moderate case (right).

Figure 7.34 shows the results for KI-3, 'Pu in the system'. Initially, the in-core values are similar to those in the homogeneous BAU–FR cases (Fig. 7.23), but the FR value levels off when new FR commissioning is reduced. The amount of plutonium in on-site reactor storage is in between the BAU cases (Fig. 7.8) and the homogeneous BAU–FR cases, with more LWR SF and less FR SF. The amount of plutonium in long term storage again shows values between those of the BAU and homogeneous BAU–FR cases, but the general shape is that of the BAU case. This is because there is no significant FR fuel in long term storage. The plutonium in fuel in reprocessing is dominated by the higher concentration of plutonium in FR fuel, the same as in the homogeneous BAU–FR cases,

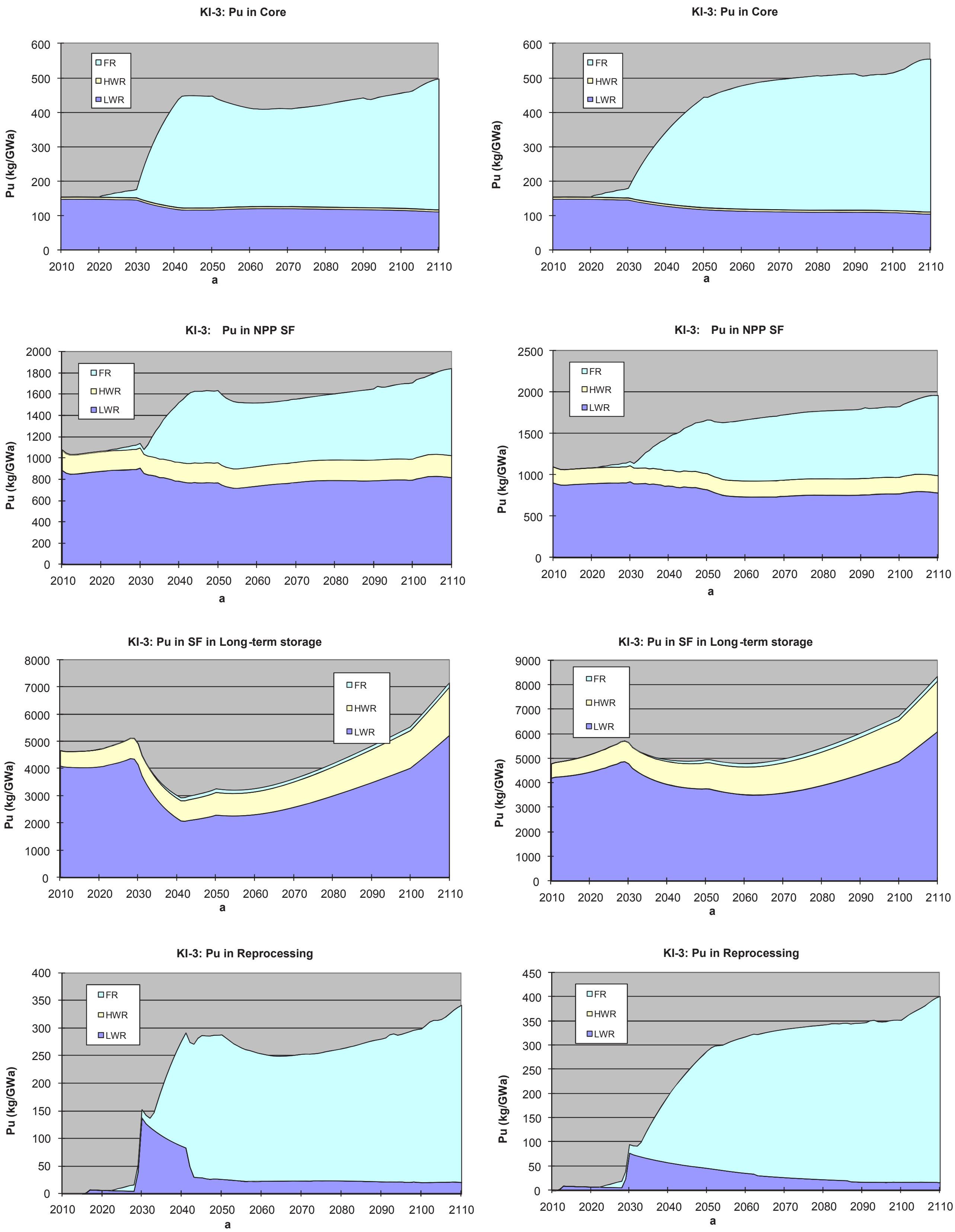


FIG. 7.34. KI-3 plutonium inventories in: reactor cores (top), on-site reactor storage (top middle), long term storage (bottom middle) and reprocessing (bottom) (heterogeneous non-synergistic high case — left, moderate case — right).

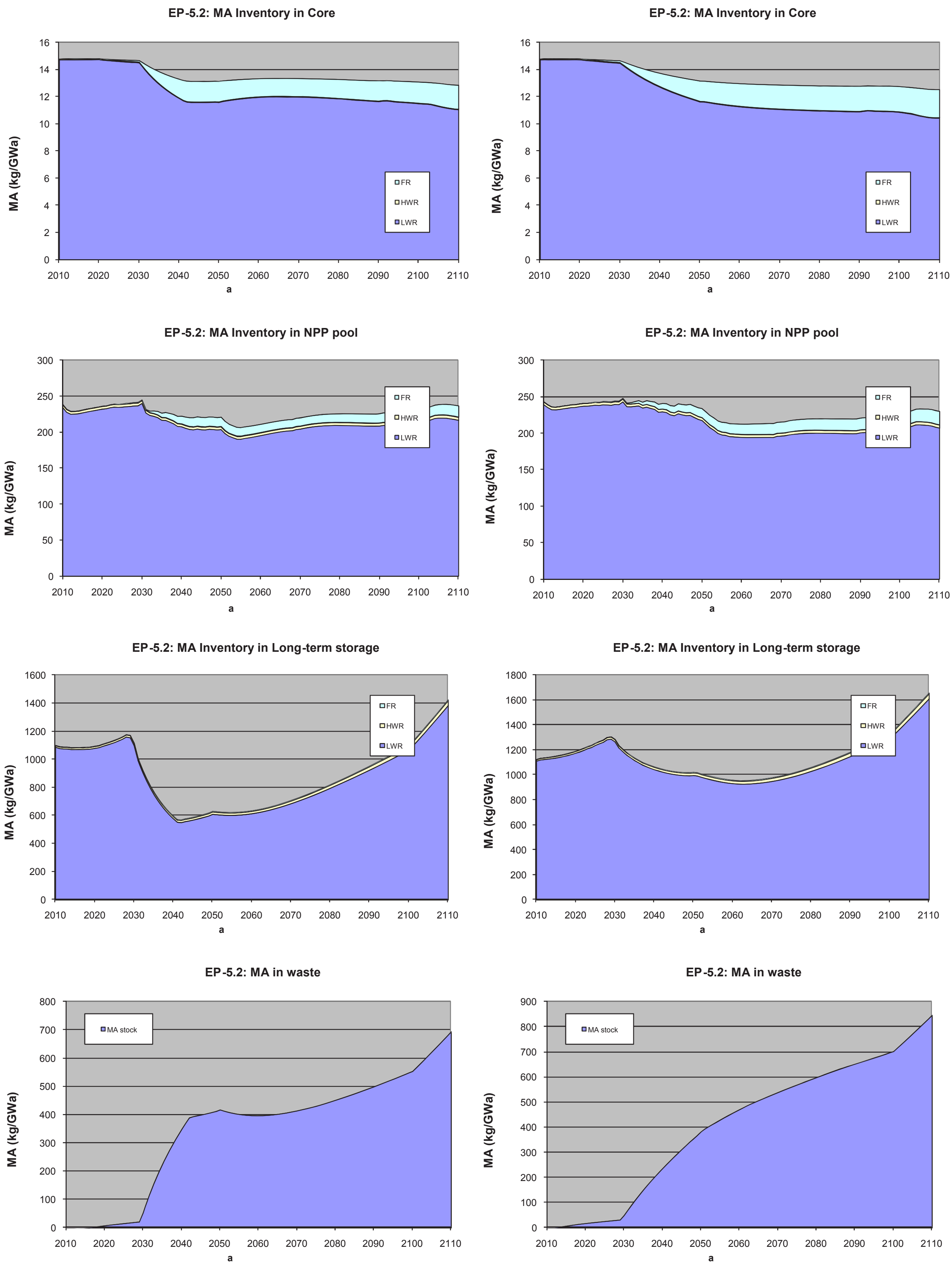


FIG. 7.35. EP-5.2 MA inventories in reactor cores (top), at reactor storage (top middle), in long term storage (bottom middle) and in waste (bottom) (heterogeneous non-synergistic high case — left, moderate case — right).

but the values are lower due to fewer FRs. The initial values are impacted by LWR SF reprocessing rates before there are enough FRs for FR SF to dominate the indicator.

KI-4, cumulative SF inventories per unit energy generated, shows values similar to the homogeneous BAU–FR cases, and so are not shown.

The values for normalized quantities of MAs, EP-5.2, are shown in Fig. 7.35. The equivalent indicator values for the BAU and homogeneous BAU–FR cases are in Figs 7.10 and 7.25, respectively.

The MA in-core values and in storage at the nuclear power plant values show the same trend as was seen in the homogeneous BAU–FR case, where introduction of FRs improves the indicator value due to the larger core in the LWR containing more MAs. In both cases, the impact is diluted by the relatively small FR share of total energy generation. The long term storage values are a combination of the BAU and homogeneous BAU–FR cases, with a dip in the value when reprocessing is working off the excess inventory of cooled, used LWR fuel, followed by rising values as seen in the BAU case as the total inventory in SF in NG2 and NG3 grows. The MAs in waste values are similar to the homogeneous BAU–FR case in shape but lower in magnitude, except that the high case shows faster growth initially as noted for other indicators.

The result of the KI-6 indicator for normalized SWUs is shown in Fig. 7.36. The global result shows steady improvement as FRs are introduced, but the improvement is not as significant as for the homogeneous BAU–FR case in Fig. 7.26. The values for NG1 are much closer to the homogeneous results, showing the dilution impact of the other two groups.

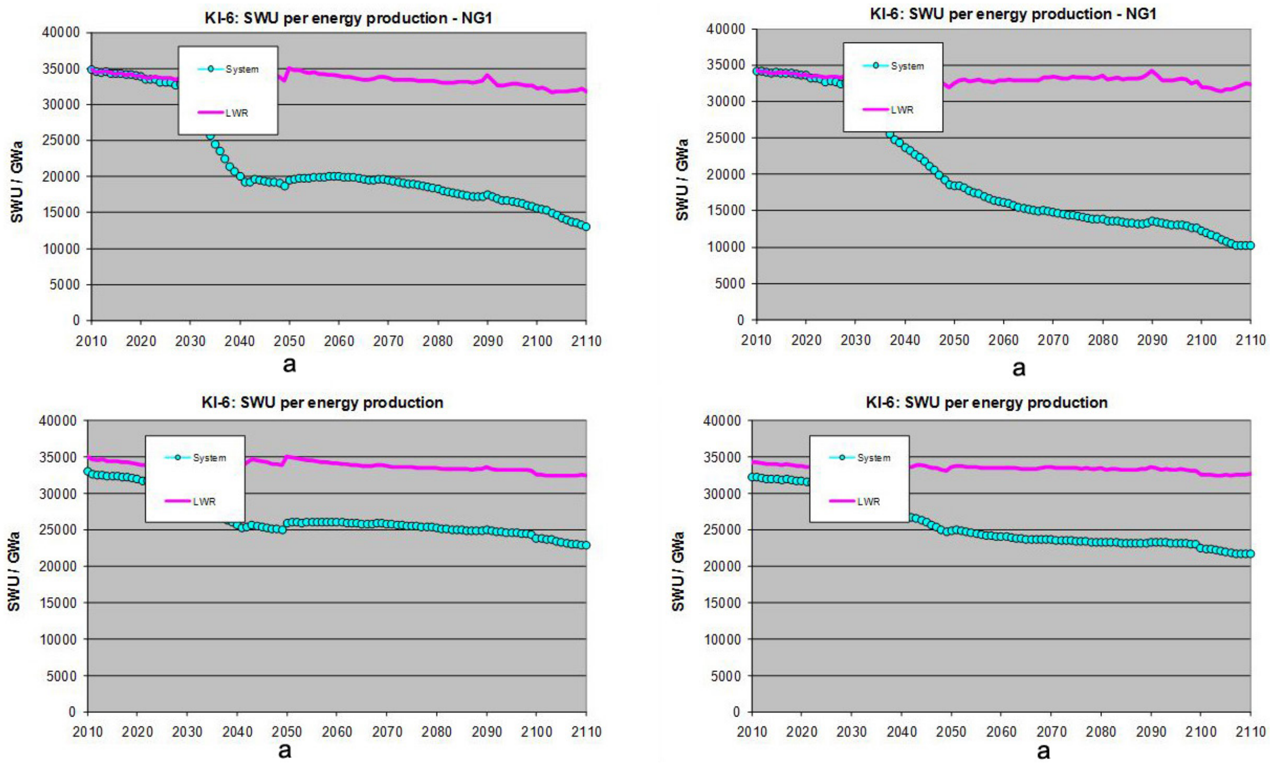


FIG. 7.36. KI-6 SWUs per energy production for NG1 (top) and total global (bottom) (heterogeneous non-synergistic high case — left, moderate case — right).

Figure 7.37 shows the results for the other KI-6 indicator, the normalized amount of direct use material separated. The graphs are all plotted on the same vertical scale to aid comparison to the homogeneous BAU–FR cases in Fig. 7.27.

In the NG1 results, the FR impact is slightly smaller than the homogeneous BAU–FR cases, especially for the moderate growth scenario. This is due to a higher share of FRs in NG1, as the FR SF has higher plutonium content. The LWR results for NG1 show the impact of reprocessing rates, including a step drop as the inventory of long term stored SF is depleted shortly after 2040.

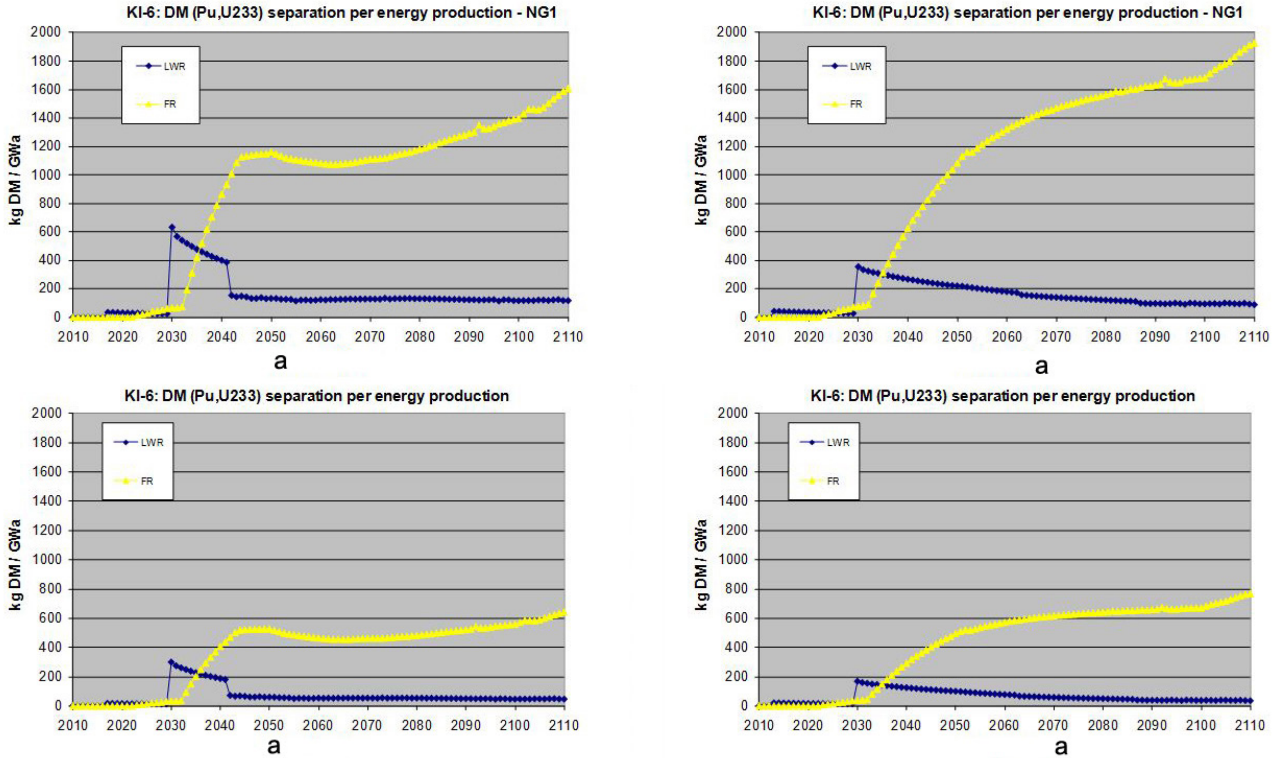


FIG. 7.37. KI-6 separated direct use material per energy production for NG1 (top) and total global (bottom) (heterogeneous non-synergistic high case — left, moderate case — right).

The global results are significantly lower that the homogeneous BAU–FR case due to two of the three groups not participating in reprocessing.

7.4.3. Synergistic cases

The heterogeneous synergistic framework cases build on the non-synergistic cases of the previous subsection. All of the primary input parameters are the same. The key difference is that movement of material is allowed between the NGs (synergism). This movement of material may result in improving the ability of each group to follow their selected fuel cycle strategies.

For the framework base cases, the NG3 group is assumed to follow a strategy to limit infrastructure investments by only building reactors and obtaining fuel cycle services from NG1 and NG2. In this scenario, this includes front end services of mining, converting and enriching uranium and fabricating fresh LWR fuel, and back end services of taking back used LWR fuel after it has cooled. The only movement modelled is the shipment of fresh and cooled SF. NG3 benefits by not having to develop, site and construct NFC facilities, including those related to the disposition of highly radioactive SF.

NG1 and NG2 must augment their fuel cycle infrastructure to support this strategy. In return, NG1 gains a source of additional used LWR fuel to support its strategy of transitioning to FRs. Benefits to NG2 are also common to other groups, including supporting the global growth of nuclear power for economic development and reduced greenhouse gas emissions while seeking to reduce proliferation risks. For simplicity in the framework base cases, NG1 and NG2 equally share the fuel cycle services for NG3, with each providing 50% of the fresh fuel and taking back 50% of the SF. In this scenario, any waste generated by reprocessing of NG3 SF for use in reactors in NG1 is kept in NG1.

Figure 7.38 shows the modified reprocessing rates for NG1 to account for the additional SF provided by NG3. Since NG3 first introduces reactors after 2008, this flow of fuel is initially small but grows throughout the scenario. The reprocessing rates compare to the non-synergistic rates shown in Fig. 7.28. In the high growth case, the LWR reprocessing rates increase later in the scenario, but there is still a shortage of SF to support the

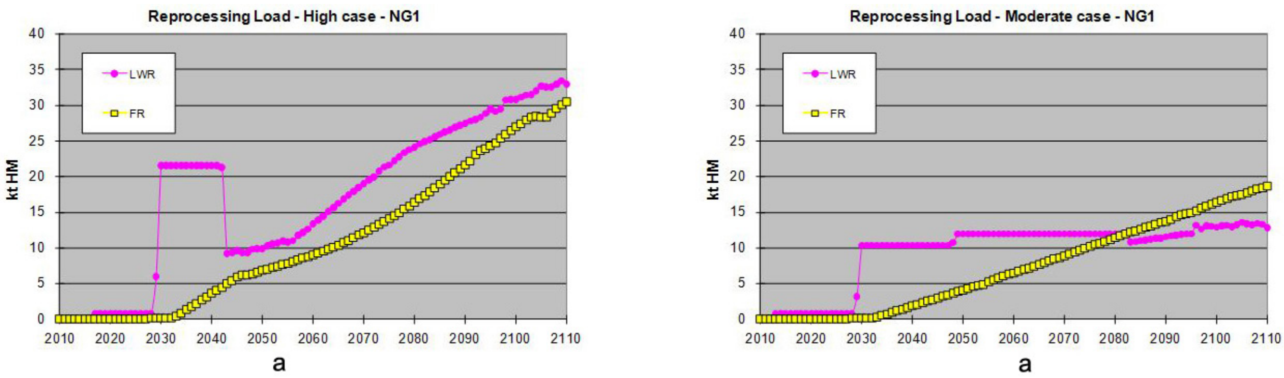


FIG. 7.38. Spent fuel reprocessing annual throughput (heterogeneous synergistic high case — left, moderate case — right).

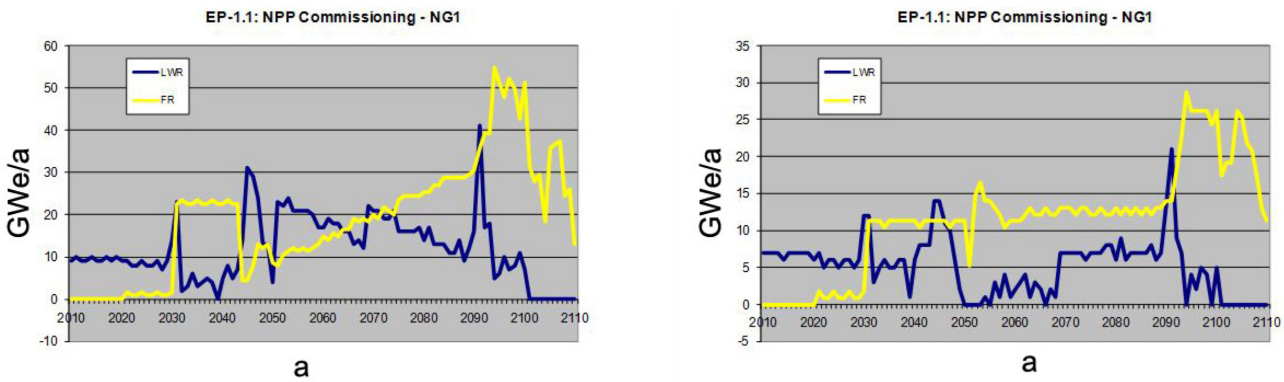


FIG. 7.39. EP-1.1 new power plant commissioning rate for NG1 (heterogeneous synergistic high case — left, moderate case right).

initial FR introduction. In the moderate growth case, two additional separations plants are commissioned after 2050 and the initial reprocessing plant is replaced to maintain the higher processing rate. With the modifications to the reprocessing rates, there is not a significant change to the amount of cooled fuel stored in NG1 (Fig. 7.29).

The impact of the higher LWR SF reprocessing rates is to increase the FR commissioning rate. Figure 7.39 shows the NG1 new reactor commissioning rate. When compared to the non-synergistic cases (Fig. 7.31 top), the high growth case shows a steeper slope for FRs after 2050. There is also a slight improvement in the number of FRs constructed during the introduction phase, resulting in a 365 GW·a FR generation rate in 2050 versus the goal of 400 GW·a. In the moderate growth case, the sustained commissioning rate in the latter half of the century is higher than the rate during the second phase of the introduction period. During the 2050–2070 time frame, very few LWRs are built. In the non-synergistic case, the FR commissioning rate over the latter half of the century was slightly lower than during the 2030–2050 period.

Figure 7.40 shows the NG1 reactor mix. The FR levels are somewhat higher versus the non-synergistic case. At the global level, the FR share of electricity generation in 2100 has increased 4% to ~24% for the high growth case and to ~29% in the moderate growth case.

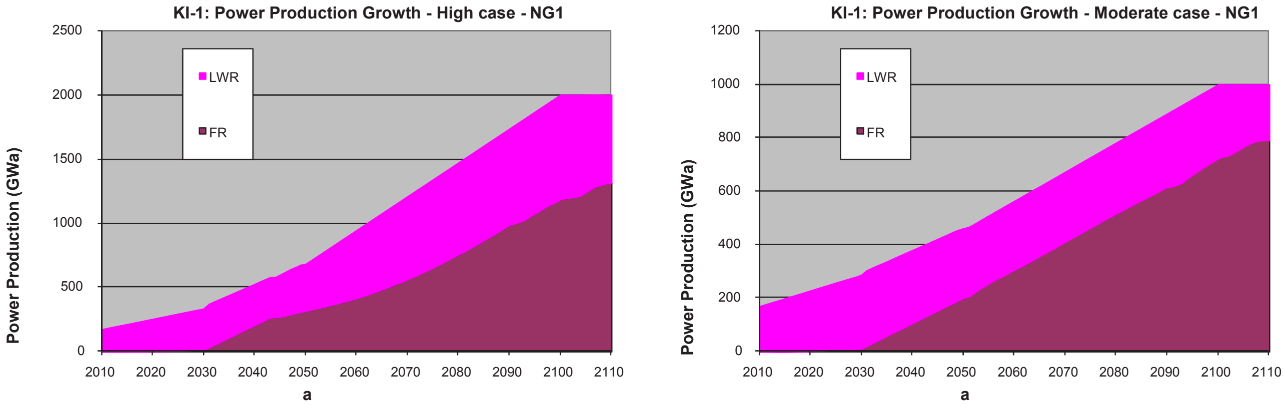


FIG. 7.40. KI-1 power production in NG1 (heterogeneous synergistic high case — left, moderate case — right).

The global impacts on most of the KIs and performance parameters are very small, so they are not presented. As an example, global values for cumulative uranium utilization, EP-2.1, decrease by 2.5% for the high growth case and by 2.7% for the moderate growth case versus the non-synergistic scenario.

The synergistic scenario introduces one new sustainability indicator, KI-7, the amount of fuel and waste material transported between groups as shown in Fig. 7.41. The amount of fuel shipped exceeds the amount returned while growth occurs. This is due to two factors, the fuel needed for new cores and the time delay between shipping fresh fuel and returning cooled SF. There is a step increase in fresh fuel shipments when the growth rate increases due to new cores being shipped at a higher rate. This can best be seen on the high growth case graph around 2050.

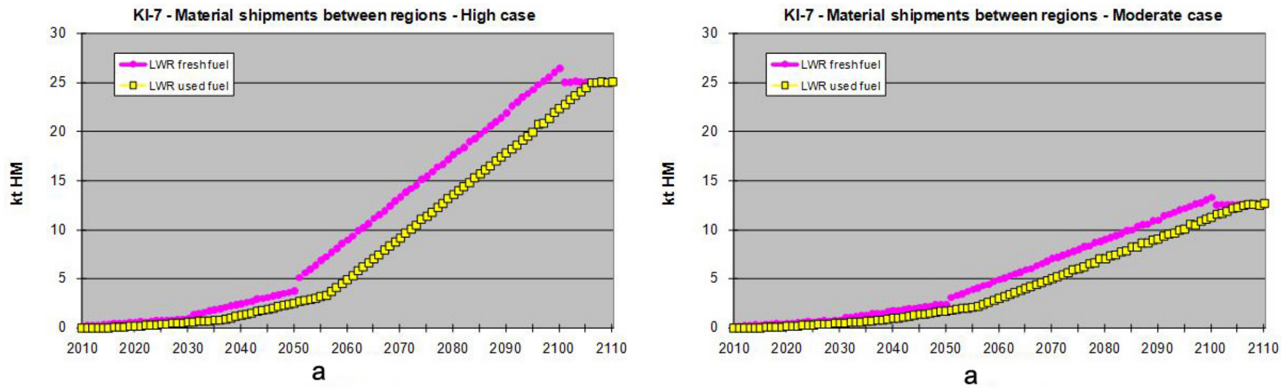


FIG. 7.41. KI-7 material shipped between groups (heterogeneous synergistic high case — left, moderate case — right).

There is a step decrease for the same reason when the growth rate flattens out around 2100. After the step decrease around 2100, the fresh fuel shipments continue to exceed SF shipments for a few more years, which is due to the impact of the time delay. Shortly after 2100, fresh fuel shipments are at a level to support a sustained rate of energy generation, but the SF shipments are still based on the generation level a few years before growth ended. After 2105, the SF shipments rise to match the fresh fuel shipments. At this point, the system is in equilibrium, with a constant number of reactors. The fuel loading rate equals the fuel discharge rate and discharged cores from retiring reactors equal new cores for replacement reactors.

7.4.4. Comparison of heterogeneous cases

As mentioned in the previous subsection, the changes in the global values of indicators between the non-synergistic and synergistic cases in the variant of interaction between groups under consideration are small. However, there are several significant changes within the NGs. This subsection compares the heterogeneous cases by showing some of these differences using comparison graphs in the GAINS results template. To aid comparisons,

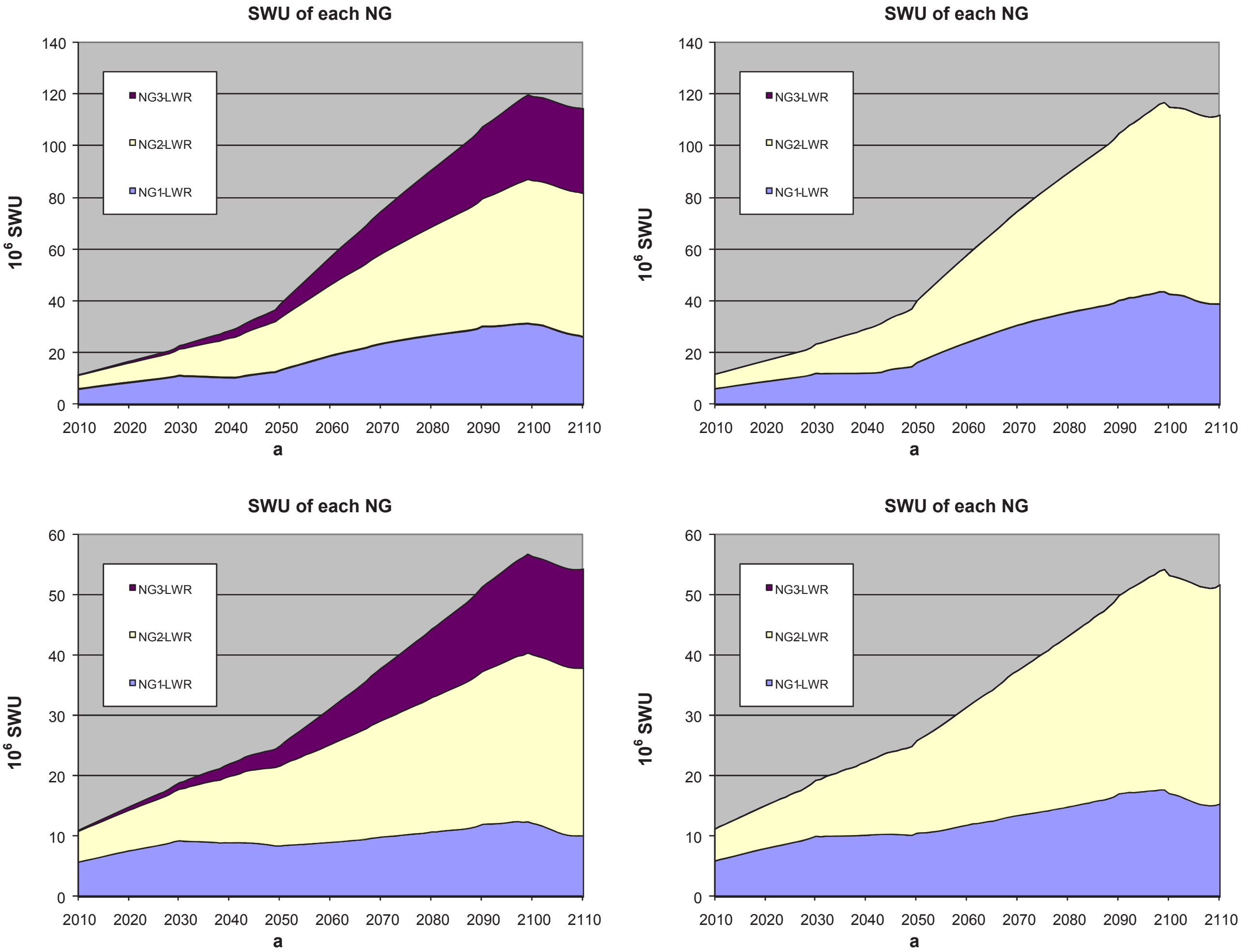


FIG. 7.42. SWUs by NG, heterogeneous cases — non-synergistic (left) and synergistic (right), high growth (top) and moderate growth (bottom).

in this section, each figure shows the non-synergistic cases on the left and the synergistic cases on the right. When both growth cases are shown, the high growth cases are on the top and the moderate growth cases on the bottom. Within a growth case, the same scale is used for the y axis.

Figure 7.42 shows the SWUs required in each NG. SWUs are only needed for the LWRs. These graphs clearly show the impact of enrichment services being provided by NG1 and NG2 in the synergistic cases, with the total SWUs nearly the same but the NG3 SWUs split between NG1 and NG2. The total SWUs are slightly lower for the synergistic case due to more FRs being built in NG1.

Figure 7.43 shows the fuel discharge amounts in each NG for the high growth case. The moderate case is similar other than the vertical scale. The synergistic and non-synergistic cases are very similar, with the only differences being the amount of LWR and FR fuel in NG1 due to the slightly higher number of FRs in the synergistic case.

Figure 7.44 shows the reprocessing loads in NG1. The annual loads are higher in the synergistic cases, reflecting the additional LWR SF coming from NG3. There is also a slightly higher amount of FR fuel reprocessing, reflecting the larger number of FRs constructed in the synergistic cases.

Figure 7.45 shows total stored SF, both at reactors and in cooled storage. The stored amounts in NG1 are small for both the non-synergistic and synergistic cases because this fuel is reprocessed. There is some NG1 LWR SF in the first part of the scenario until the excess stored fuel is reprocessed. Fuel stored in NG2 dominates total storage, with the quantity of low burnup HWR fuel nearly matching the quantity of higher burnup LWR fuel due to all of the HWRs being in NG2. In the synergistic case, the only fuel stored in NG3 is the small amount cooling at the reactors prior to shipment to NG1 and NG2.

The next set of graphs (Fig. 7.46) shows the plutonium in stored SF, which is part of KI-3. The high and moderate growth cases are similar other than the magnitude of amounts, so only the high case is shown. The

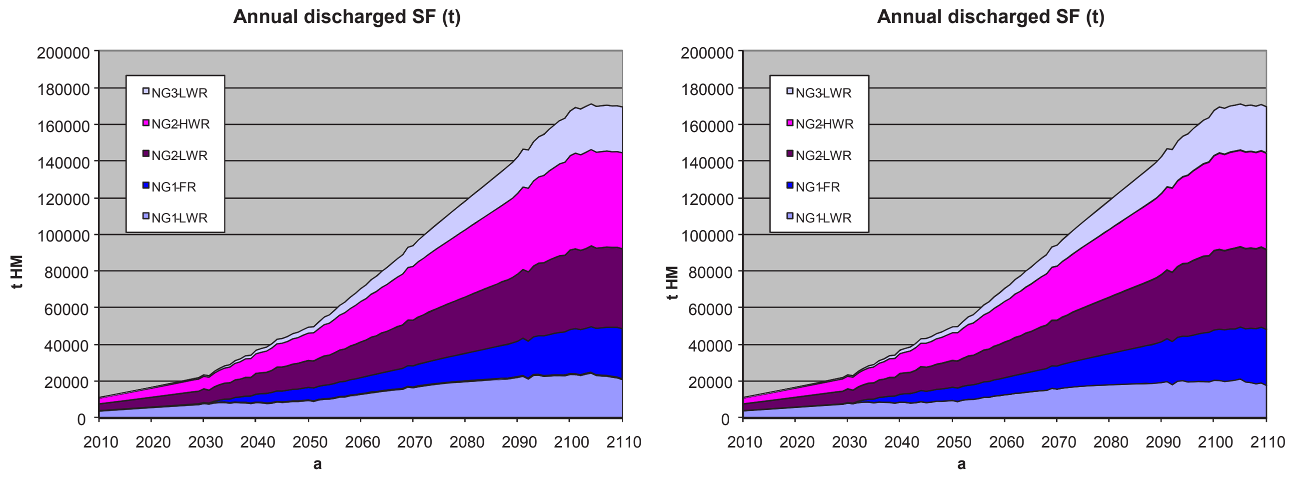


FIG. 7.43. Fuel discharge rates by NG, heterogeneous high cases — non-synergistic (left) and synergistic (right).

synergistic case total values are lower because half of the LWR fuel from NG3 is reprocessed. However, the NG2 values are higher in the synergistic case, reflecting the other half of the NG3 fuel that is stored. The plutonium in NG3 in the synergistic case is due to the fuel at reactors cooling prior to being shipped.

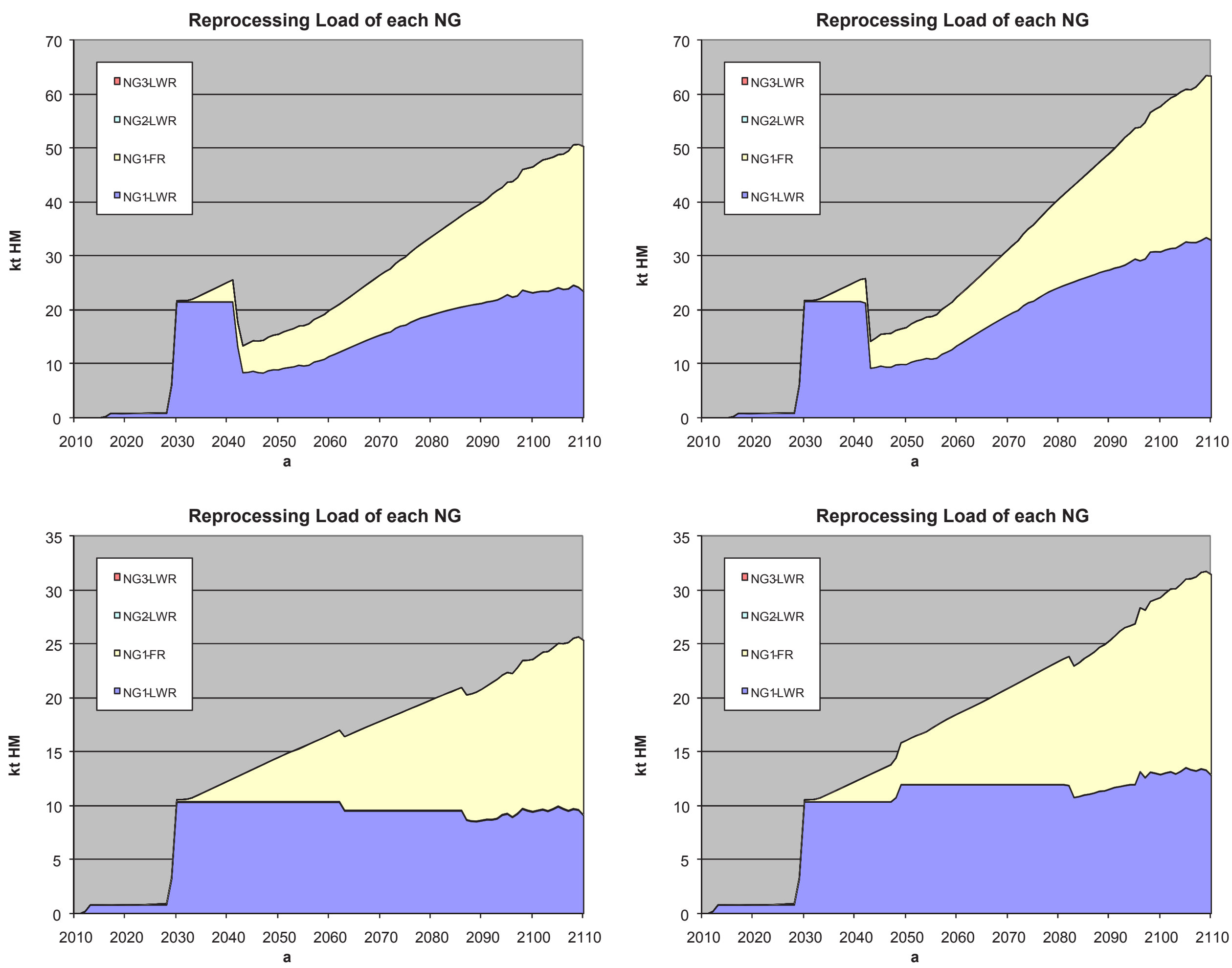


FIG. 7.44. Reprocessing loads, heterogeneous cases — non-synergistic (left) and synergistic (right), high growth (top) and moderate growth (bottom).

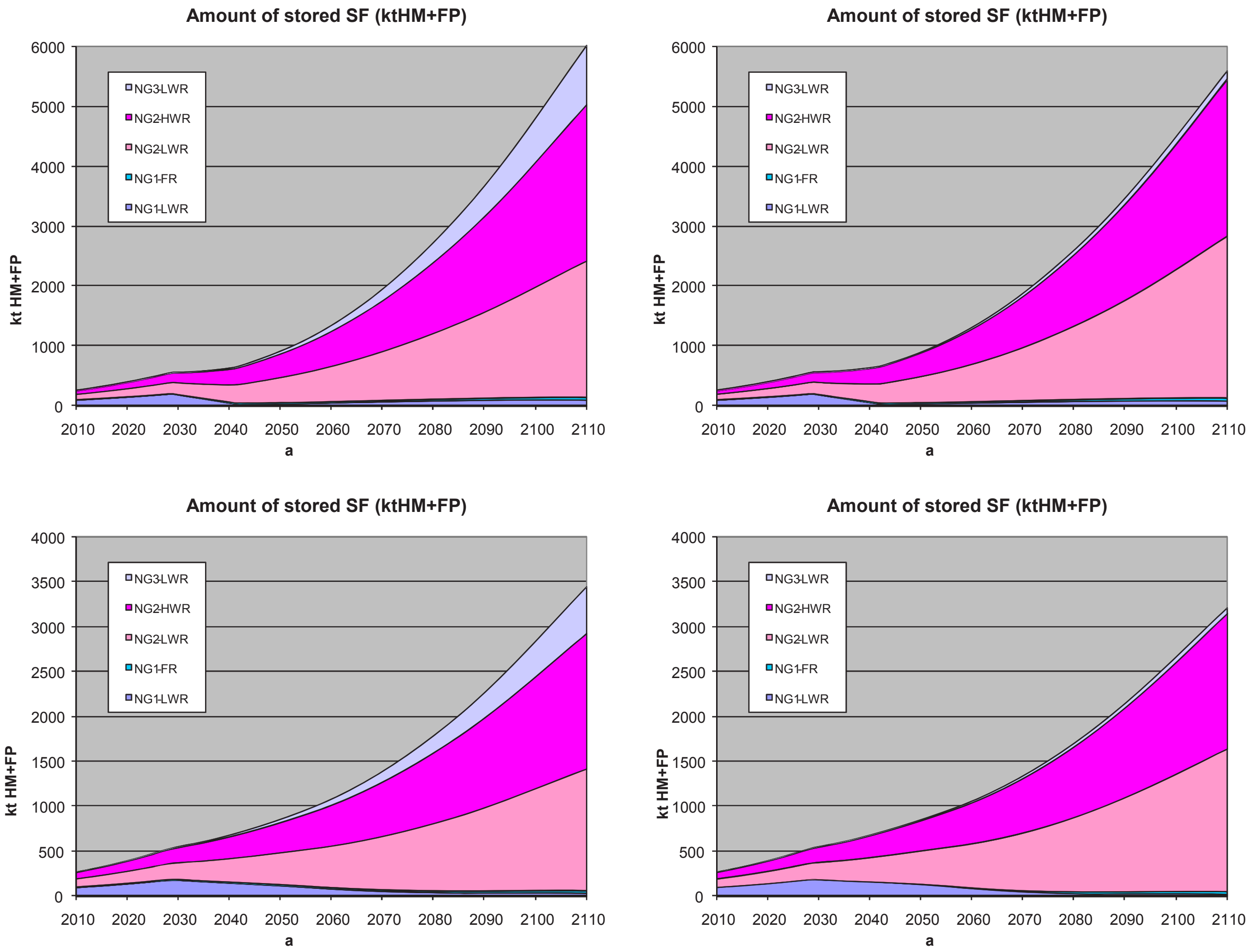


FIG. 7.45. Spent fuel stored at reactors and in long term storage, heterogeneous cases — non-synergistic (left) and synergistic (right), high growth (top) and moderate growth (bottom).

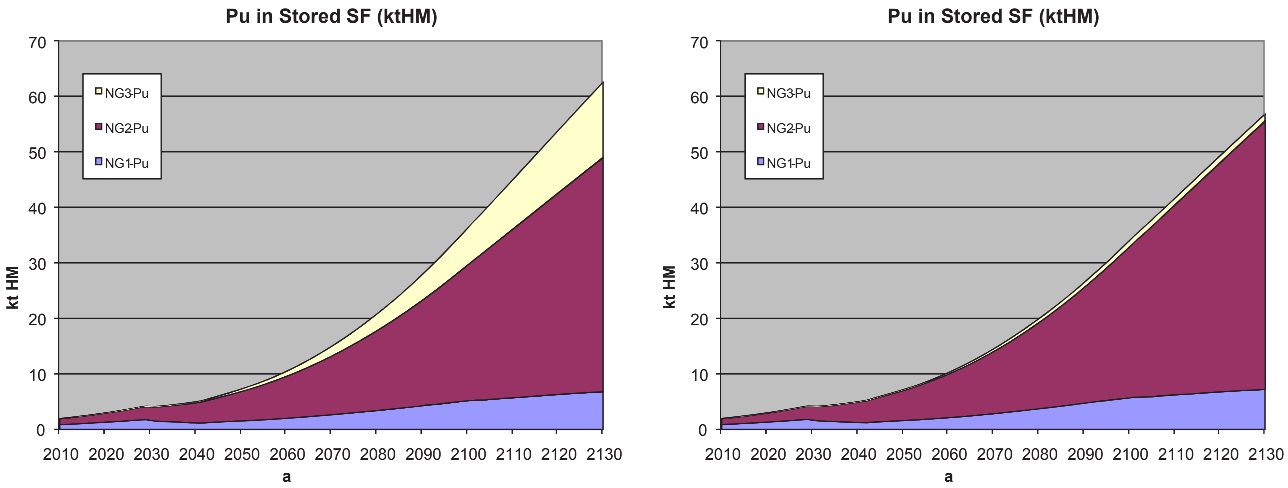


FIG. 7.46. Plutonium in stored spent fuel by NG, heterogeneous high cases — non-synergistic (left) and synergistic (right).

Figure 7.47 shows the total plutonium in the system normalized versus annual energy production (KI-3). Only the high case is shown. NG2 begins higher than NG1 due to the plutonium content in used HWR fuel, while NG3 starts at zero (no reactors). In both cases, NG2 shows much higher accumulation than NG1 throughout the scenario, due primarily to NG1 avoiding some plutonium production because some Pu-producing LWRs are

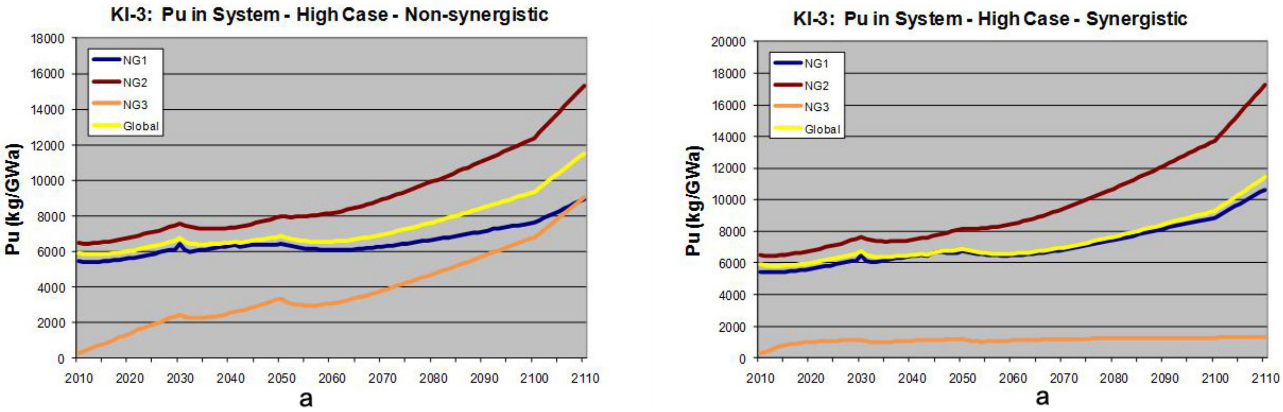


FIG. 7.47. Plutonium in system (normalized), heterogeneous high cases — non-synergistic (left) and synergistic (right).

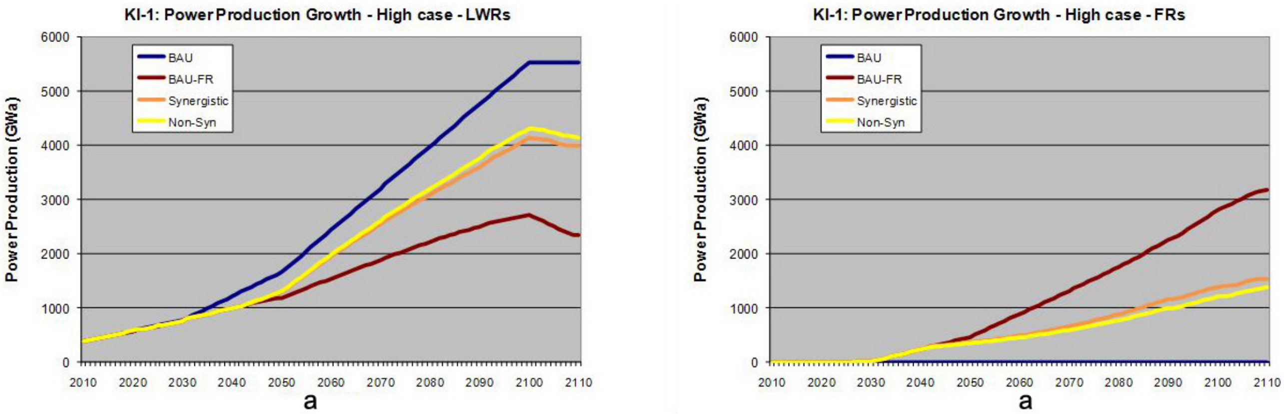


FIG. 7.48. Power production for homogeneous BAU and BAU–FR, and heterogeneous synergistic and non-synergistic cases (LWR — left, FR — right).

replaced with break-even converter FRs. In the non-synergistic case, plutonium accumulates quickly in NG3, while in the synergistic case, normalized NG3 quantities level out because used fuel is exported after five years cooling. The total plutonium in NG3 is growing, as shown in Fig. 7.47, but so is energy production. The fuel exported from NG3 does not impact the NG1 and NG2 lines as much because it is normalized against higher energy production in those two NGs.

7.5. BASE CASE COMPARISONS

This subsection compares fuel cycle performance for the homogeneous and heterogeneous world cases. All comparisons are based on the high growth case.

Figure 7.48 compares the power production (KI-1) for LWRs and FRs, indicating their relative fleet sizes for all four high growth cases. Since the HWRs are constant in all cases, this provides two views of the level of replacement of LWRs with FRs. In the BAU case, no FRs are built and the LWRs level out in 2100 when overall growth levels out. The other three cases depart from the BAU case starting in 2020 (hardly noticeable) and more obviously after 2030 when the FR introduction rate increases. All three follow the same initial curve as FRs are introduced at the same rate until the stocks of stored used LWR fuel are depleted, when they diverge.

After this point, the rate of FR growth is dependent on the proportion of the world providing LWR used fuel for new FRs. For the BAU–FR case, this is 100%, while for the heterogeneous cases, this starts at 50%. The value stays at 50% for the synergistic case, but drops slowly to 40% by 2100 for the non-synergistic case. The result is that the heterogeneous cases are roughly half way between the BAU and BAU–FR cases, with the synergistic case showing slightly higher FR growth and, therefore, slightly lower LWR growth. Note the negative slope for

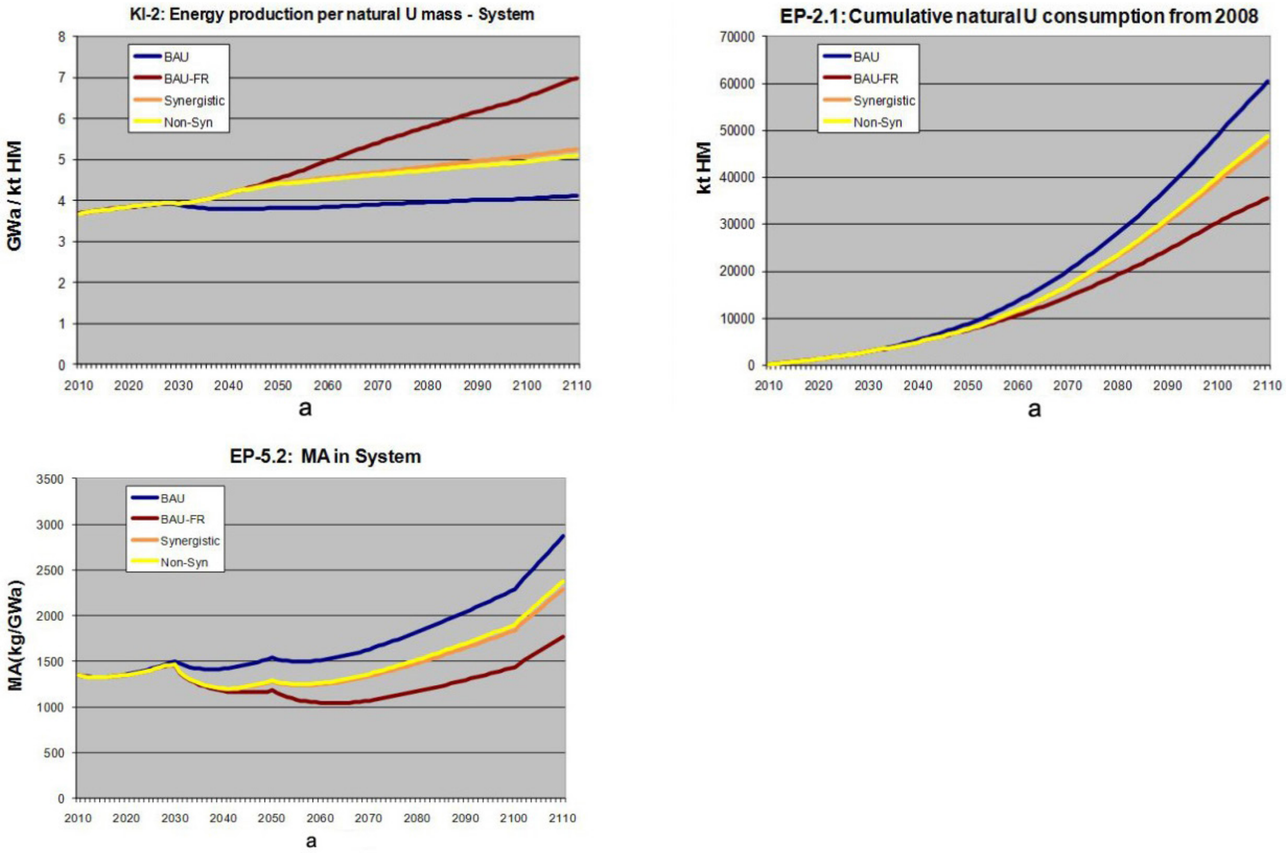


FIG. 7.49. Resource utilization and direct use material for homogeneous BAU and BAU–FR, and heterogeneous synergistic and non-synergistic cases (energy per uranium — left, total uranium — right, total MAs — bottom).

LWRs after 2100 for the BAU–FR and heterogeneous cases. While overall growth has levelled out at this point, the replacement of retiring LWRs with new FRs continues.

Figure 7.49 shows a similar behaviour for normalized uranium utilization (KI-2), cumulative uranium used (EP-2.1) and MAs (EP-5.2). The greater the power generated from FRs, the higher the normalized energy generation per unit of uranium, the lower the total uranium usage, and the lower the total MA in the system. Again, the BAU curve diverges around 2030 and the BAU–FR curve after 2040, with nearly identical results from the two heterogeneous cases. There is no period of negative slope after 2100, indicating that the differences in behaviour are driven primarily by the FRs.

These comparison graphs all show the same basic behaviour of the heterogeneous case, its performance being between that of the two homogeneous cases. This illustrates the value of the heterogeneous world model incorporated in the GAINS framework for exploring the area between the two bounding cases of a global open cycle (BAU) and a global closed cycle (BAU–FR). Not all countries will follow the same strategy or timing, so it is important to be able to assess and understand the impacts of simultaneously following different strategies. According to the GAINS ToR, a standard framework has been developed for the assessment of a wide range of transition scenarios, including those involving combinations of fuel cycle strategies.

The two heterogeneous cases modelled result in similar performance due to several factors. First, the size of NG3 is small, including only 20% of the world in 2100. A larger share in NG3 would result in larger impacts. Second, in the synergistic case, half of NG3 used fuel goes to NG2, where it does not participate in reprocessing. The difference between the non-synergistic and synergistic cases would be greater if more of the NG3 used fuel went to NG1. Third, the story lines and figures in Section 3 included the possibility of NG2 used fuel also being exported to NG1 for reprocessing.

This would significantly increase the difference between the non-synergistic and synergistic cases. If all NG2 and NG3 fuel were sent to NG1 and reprocessed, the global result would be the same as the homogeneous BAU–FR case (which is why this was not included as a base case). Many other options and combinations of strategies are possible, including using other types of reactors/transmuters such as FRs with breeder ratios other than 1.0 or using MSRs or ADSs. Other permutations include a different timing for technology/strategy transitions and other splits of growth rates between the NGs.

7.6. SUMMARY

This section has documented the eight GAINS framework base cases, including fully describing the cases and presenting the results for each case against the KIs and EPs in Section 4. The eight cases include four using the moderate growth and four using the high growth projections described in Section 5. The cases are divided into two sets to cover the homogeneous and heterogeneous story lines discussed in Section 3.

First, the BAU cases model a homogeneous world scenario with only LWRs and HWRs and no reprocessing. Next, the BAU–FR cases extend the BAU cases to include the introduction of FRs starting in the first half of the century and slowly replacing LWRs in the second half of the century. The rate of introduction of FRs is specified to 2050, after which they are commissioned based only on the availability of plutonium for their startup.

The heterogeneous world scenario considers three separate nuclear groups based on their fuel cycle strategies. Two of the groups, NG1 and NG2, are modelled to represent the existing global nuclear infrastructure, split between countries pursuing a closed fuel cycle with recycling and FRs (NG1) and countries continuing to use a once-through fuel cycle without reprocessing (NG2). The third group (NG3) represents new nuclear growth and is modelled on the non-synergistic scenario as a stand-alone group that develops its own fuel cycle facilities using a strategy that stores SF. In the synergistic scenario, NG3 works together with NG1 and NG2, obtaining fuel cycle services to support reactor deployment and operations. A principal feature of the synergistic scenario is that some of the fuel discharged from the NG3 reactors is disposed of in NG2, while some of the discharged fuel is recycled for use in NG1 FRs.

The results in this section show the impact of these different fuel cycle strategies while providing reference base cases for future users of the GAINS framework. A large number of scenario alterations are possible and can be used to assess different strategies, different technologies and different assumptions about the possible futures of nuclear power. The results can be compared to those documented here to assess where these alternate cases perform differently versus the sustainability indicators.

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8. SENSITIVITIES OF BASE CASE PARAMETERS AND EFFECTS OF BREEDING AND BURNUP PERFORMANCES OF FAST REACTORS

8.1. BACKGROUND

In Section 7, the framework base case, BAU and FR introduction scenarios with a break-even FR (‘F1’) were studied with the reactor systems of LWR (‘L1’), HWR (‘H1’) and break-even FR (‘F1’), and analysis conditions of 0.3% of uranium enrichment tails assay, one uranium enrichment model of an LWR, a five year SF cooling time for LWRs and HWRs, and two year cooling and one year reprocessing including fabrication for an FR. The plant load factor and the lifetime of the reactors were set to 85% and 60 years, respectively for all reactor types.

This section describes the results of sensitivity analyses, in which the effect of the analysis condition changes, and the influences from the reactor performance advancement in the future are studied with regard to the indices: natural uranium consumption, uranium enrichment separative work, amount of SF discharge, reprocessing load, plutonium balance and so on.

8.2. TAILS ASSAY

The assumption of the tails assay in uranium enrichment facilities is expected to affect the uranium consumption rate. A sensitivity calculation was made under the same conditions as the BAU scenario mentioned in Section 6.3 (framework base case), in which the tails assay is constant from the beginning to the end and initial core enrichment was treated the same as that for the equilibrium core.

8.2.1. Natural uranium consumption

As shown in Figs 8.1 and 8.2, the cumulative uranium consumption will decrease by 17% from the reference case when the tails assay decreases from 0.3 to 0.2% in the BAU scenario (for both high and moderate demand cases). The cumulative value for 2100 decreases from 49.8 Mt when the tails assay is 0.3% to 41.5 Mt when the tails assay 0.2% in the high case, for example.

8.2.2. Separative work

Contrary to the decrease in natural uranium consumption, the separative work increases when the tails assay is lowered. As shown in Figs 8.3 (high case) and 8.4 (moderate case), the annual necessary separative work rises by around 24%. For example, the annual separative work rises from 672×10^6 to 834×10^6 SWU for 2100 in the high case.

However, taking account of recent technology developments for uranium enrichment, the above separative work increase seems to be easily cancelled out by the development of new economic enrichment facilities, such as centrifuges.

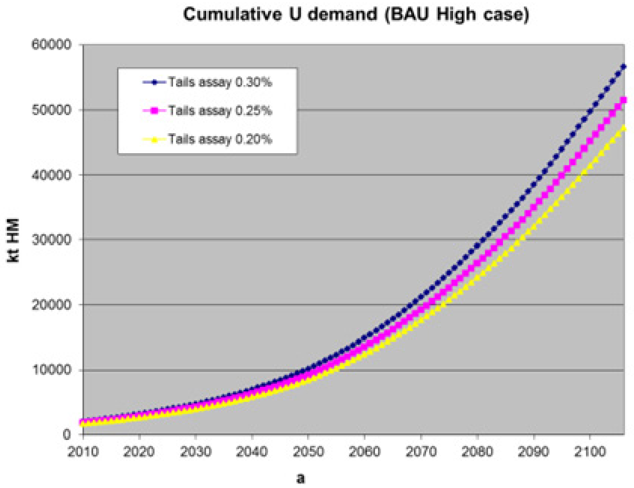


FIG. 8.1. Tails assay and cumulative uranium demand (high case).

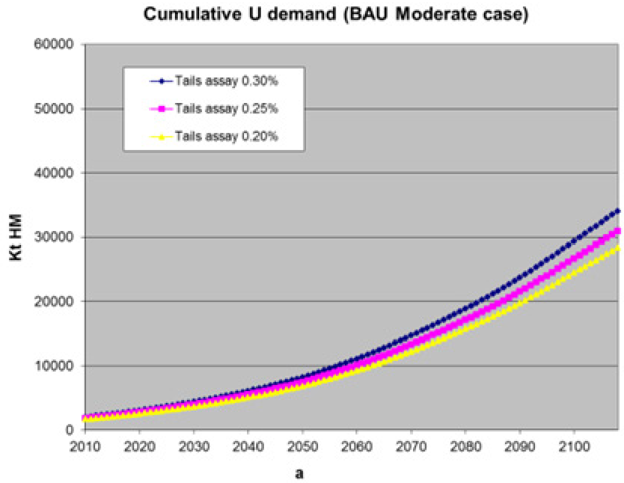


FIG. 8.2. Tails assay and cumulative uranium demand (moderate case).

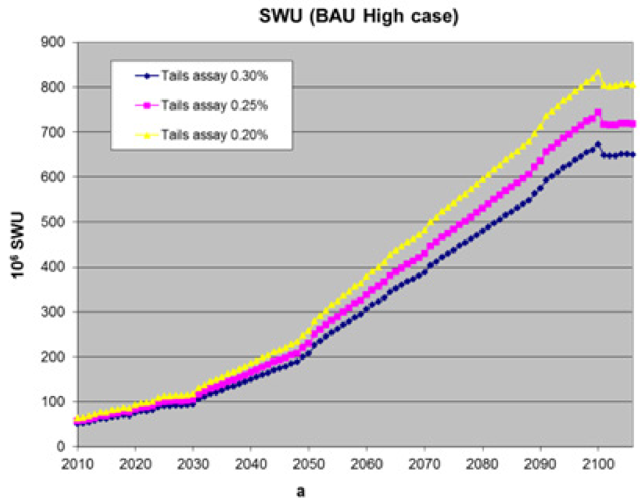


FIG. 8.3. Tails assay and annual separative work (high case).

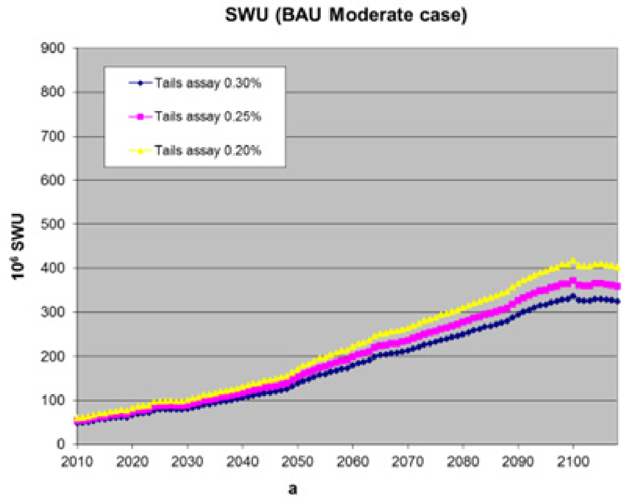


FIG. 8.4. Tails assay and annual separative work (moderate case).

8.3. MODEL OF INITIAL CORE URANIUM ENRICHMENT

In the previously mentioned BAU scenario (framework base case), the initial core uranium enrichment was modelled the same way as the equilibrium core, presuming that the effect would be small in long term fuel cycle analyses. In reality, according to the current operation experience of PWRs, for a case of a four batch refuelling scheme, four levels of uranium enrichment occur in the initial core loadings, depending on the residence time of each fuel subassembly. The average uranium enrichment of the initial core loading is around 60% of that of equilibrium cores. The effect of an initial core with 60% equilibrium enrichment was, therefore, examined.

8.3.1. Natural uranium consumption

Figures 8.5 and 8.6 show the effect of the initial core enrichment model on the cumulative uranium demand in the BAU case. The uranium consumption rate decreases around 5.7% in the high case and 5.3% in the moderate case in 2100 by modelling the reduction of the initial core uranium enrichment.

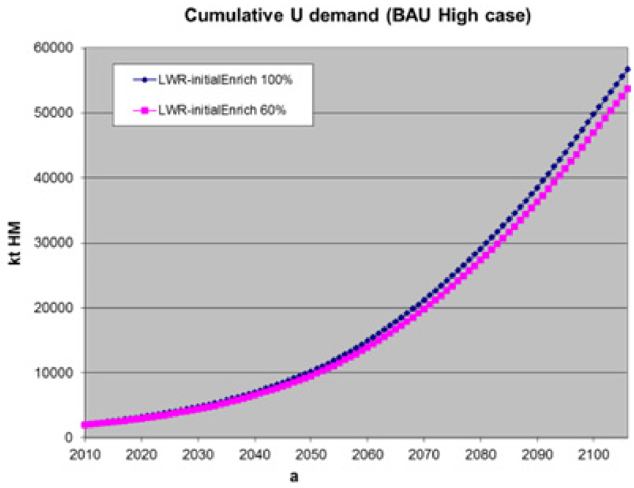


FIG. 8.5. *Effect on uranium consumption by initial core enrichment model (high case).*

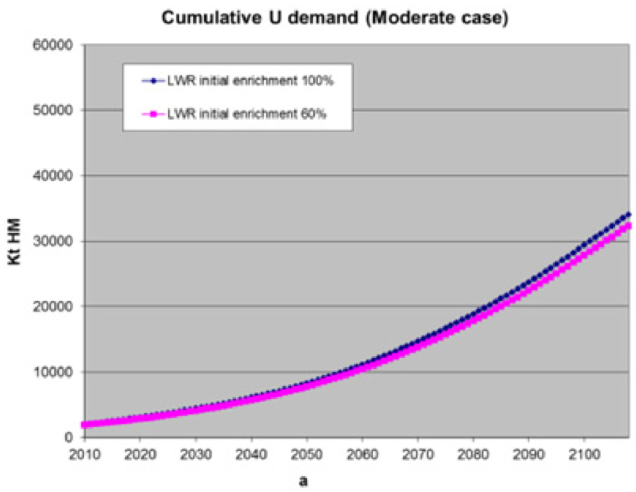


FIG. 8.6. *Effect on uranium consumption by initial core enrichment model (moderate case).*

The effect is rather small because the enrichment reduction affects only the initial core loading during the plant lifetime of 40 years. The size of the effect can be expected to increase during periods of rapid nuclear capacity growth, when many reactors are newly installed over a short time period.

8.3.2. Separative work

As well as the above uranium consumption decrease, the annual separative work also decreases by around 7% in 2100 in both the high case and the moderate case (Figs 8.7 and 8.8).

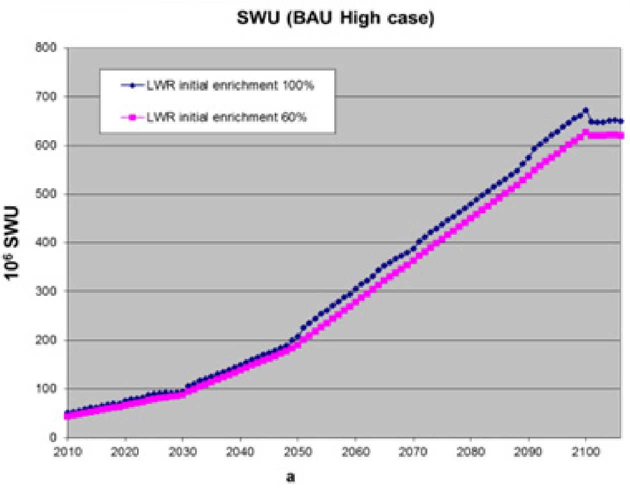


FIG. 8.7. *Effect on separative work by initial core enrichment model (high case).*

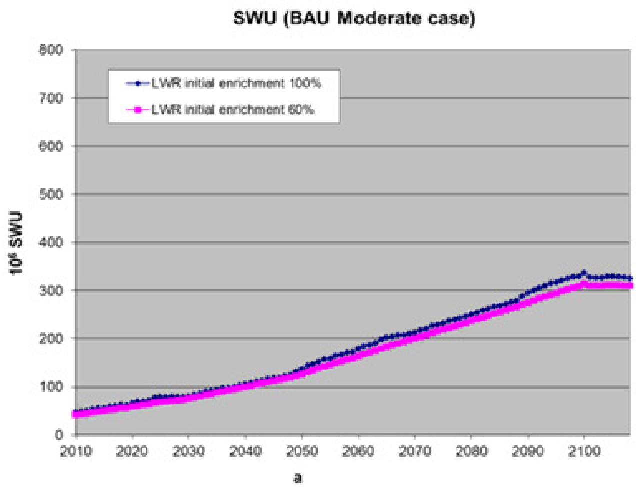


FIG. 8.8. *Effect on separative work by initial core enrichment model (moderate case).*

8.4. LIGHT WATER REACTOR BURNUP

The fuel burnup in LWRs has a significant impact on a number of sustainability indicators. At double the burnup, the SF generation is cut in half for the same energy generated by the same reactor. As burnup increases, it also has impacts on fresh and spent fuel compositions. For higher burnup, the enrichment level in fresh fuel is higher, impacting uranium and SWU requirements. In SF, the amount of FPs increases linearly with increasing burnup while the amount of transuranics also increases, but in a non-linear manner.

The burnup of 45 GW·d/t used in the framework BAU base cases is near the current global average. However, average LWR burnup has been steadily increasing as fuel fabrication techniques improve. This sensitivity study assesses the impact of increasing the LWR burnup versus the framework BAU high growth base case. The burnup values used are 45, 60, 75, 90 and 100 GW·d/t. No other input parameters are changed.

8.4.1. Natural uranium consumption

The change in uranium consumption is small, with the best result (lowest consumption) at 60 GW·d/t burnup. The total variation from the highest consumption (100 GW·d/t) to the lowest consumption (60 GW·d/t) is ~2.5%. Figure 8.9 shows the annual and cumulative uranium consumption for the five cases.

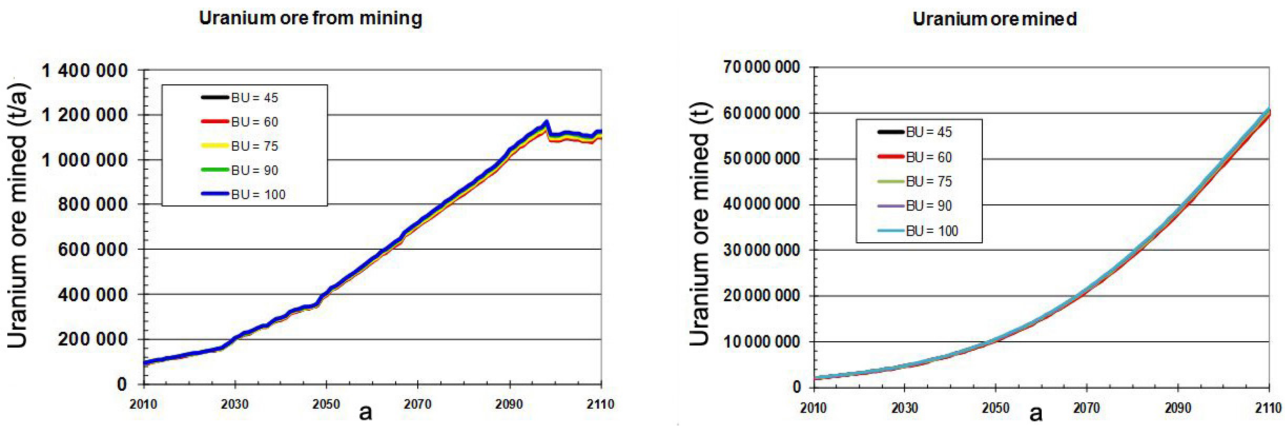


FIG. 8.9. Annual uranium usage (left) and EP-2.1 cumulative uranium usage (right) for the BAU case with LWR burnup varied from 45 to 100 GW·d/t.

The primary drivers for these results are the amount of uranium that ends up in fuel versus that as depleted uranium and the rate of conversion of ^{238}U to plutonium in the core. As burnup increases, the required enrichment for fresh fuel also increases. The relationship between uranium ore requirements with burnup is shown in Fig. 8.10. The conversion of ^{238}U to plutonium is discussed in Section 8.4.4.

Some other drivers of less importance are the amount of plutonium transmuted from ^{238}U (increases with burnup), which reduces uranium requirements, and the amount of neutron absorption occurring by FPs (also increases with burnup), which increases uranium requirements. Overall, the uranium utilization improves slightly for moderate burnup versus either lower or higher burnups.

8.4.2. Enrichment

As shown in Fig. 8.10, LWR fresh fuel enrichment levels increase with increasing burnup. Figure 8.11 shows the impact on annual SWU requirements. There is a 26% increase in SWUs for the 100 GW·d/t burnup case versus the 45 GW·d/t burnup case.

Figure 8.12 shows the impact on enriched uranium flows, which can be compared to the natural uranium flows from Fig. 8.9. As enrichment increases, the amount of low enriched uranium decreases. There is a 55% decrease in low enriched uranium for the 100 GW·d/t burnup case versus the 45 GW·d/t burnup case.

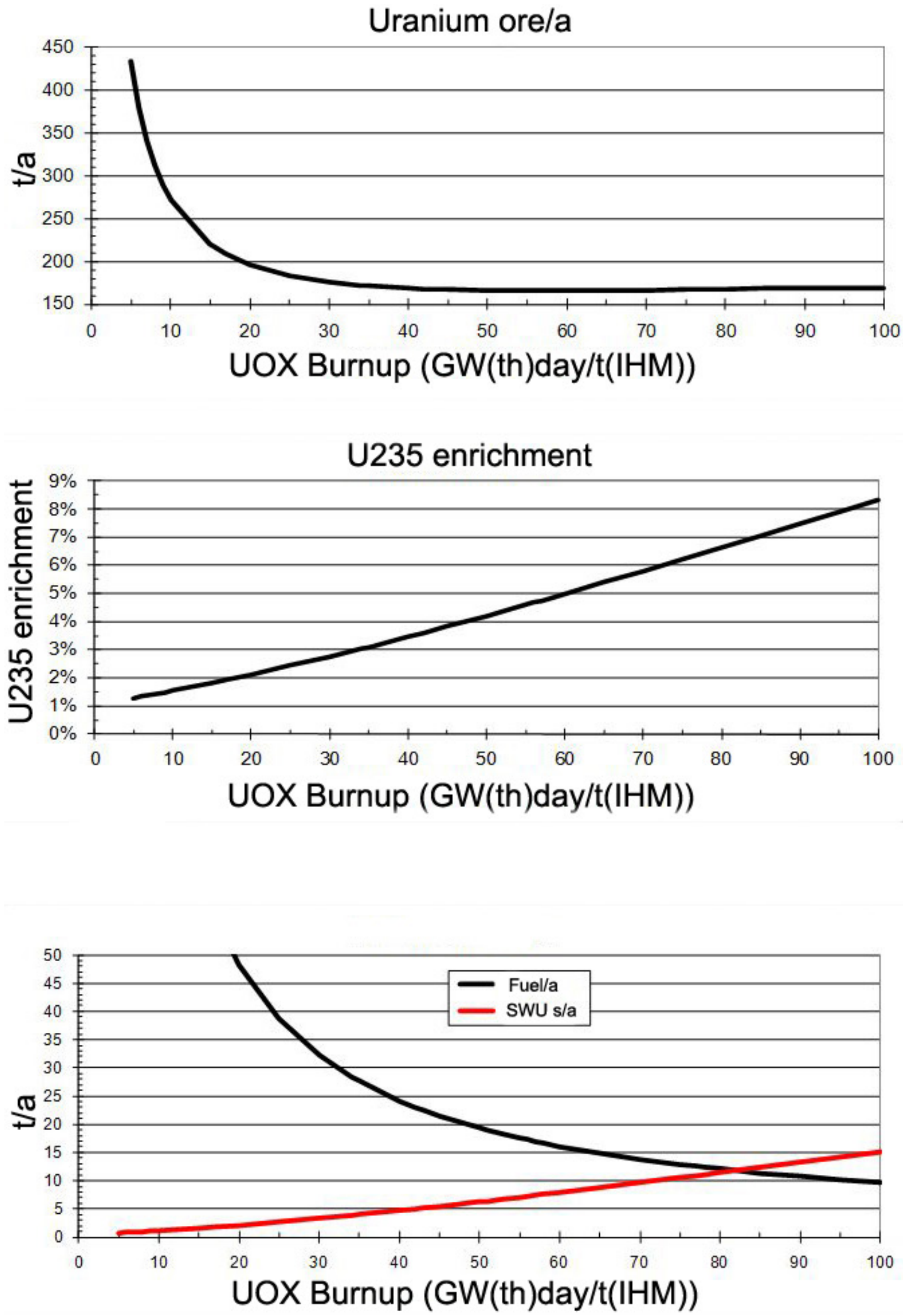


FIG. 8.10. Relationship between uranium requirements, ^{235}U enrichment levels, tonnes of fuel per year and SWU/a as a function of LWR UOX burnup.

8.4.3. Fuel discharges

Higher enrichment equates to less mass of fresh fuel required per energy produced and, therefore, less SF discharged. Figure 8.13 shows the cumulative amount of SF (LWR and HWR) in long term storage. The overall reduction is 40% for the 100 GW·d/t burnup case versus the 45 GW·d/t burnup case.

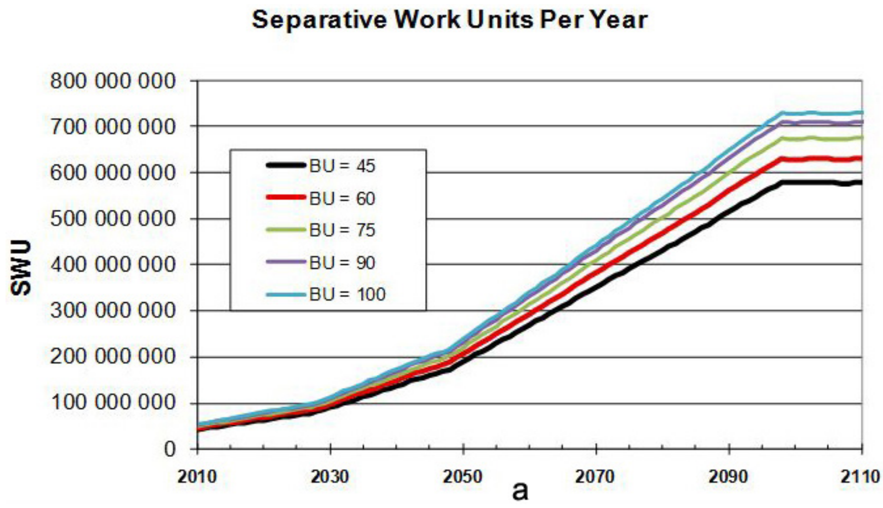


FIG. 8.11. Annual SWU requirements for the BAU case with LWR burnup varied from 45 to 100 GW·d/t.

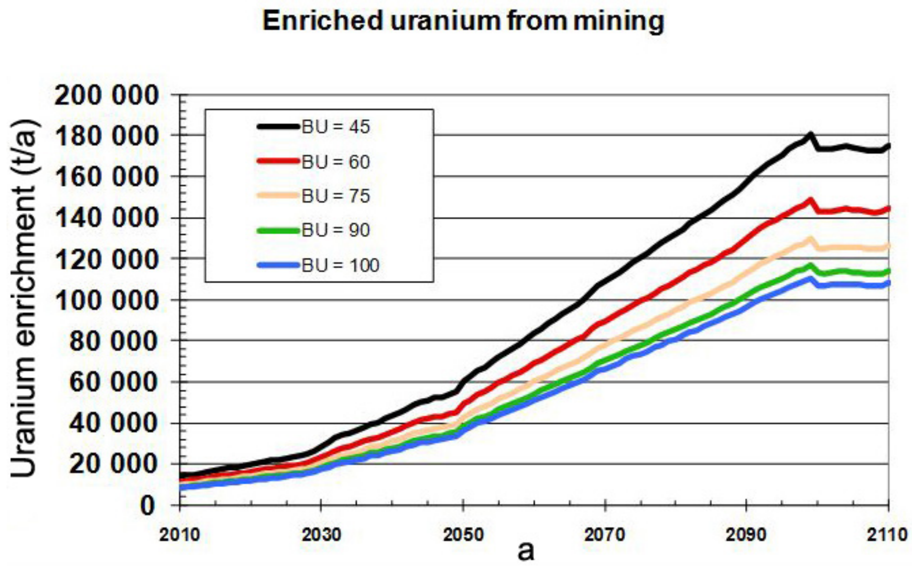


FIG. 8.12. Enriched uranium flows for the BAU case with LWR burnup varied from 45 to 100 GW·d/t.

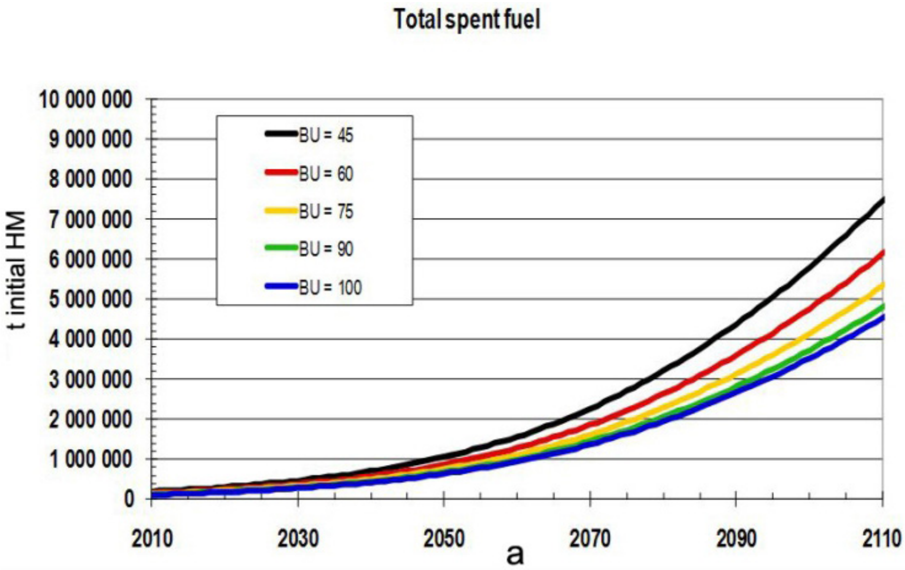


FIG. 8.13. Total spent fuel (LWR + HWR) in long term storage for the BAU case with LWR burnup varied from 45 to 100 GW·d/t.

8.4.4. Plutonium

As burnup increases, transmutation of ^{238}U into plutonium also increases. As the plutonium content of the fuel rises, some of the plutonium fissions and contributes to the overall energy generation. The result is that as burnup increases the net plutonium content of the fuel increases too but at a slower rate. The impact on the system is to reduce the total plutonium content for the same level of energy generation. The plutonium inventory in the system is shown in Fig. 8.14. There is a 30% reduction in total plutonium for the 100 GW·d/t burnup case versus the 45 GW·d/t burnup case.

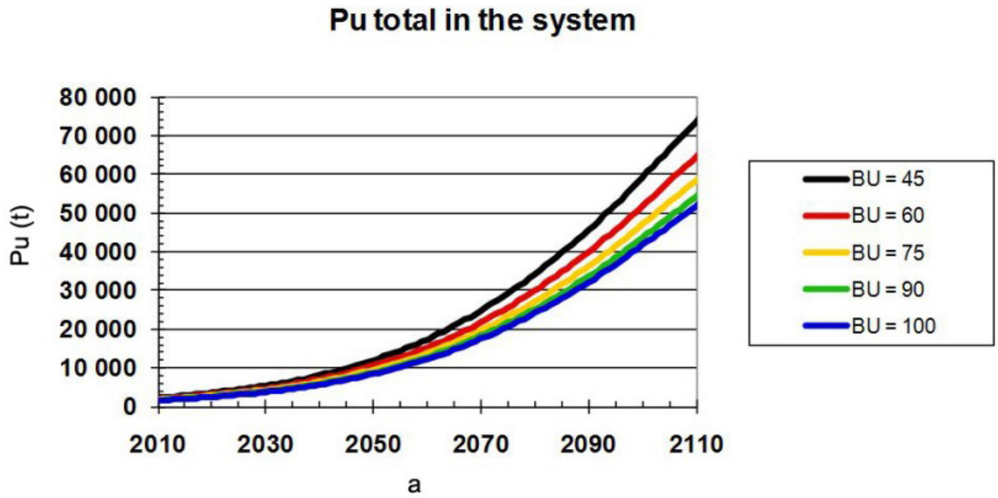


FIG. 8.14. Plutonium inventories for the BAU case with LWR burnup varied from 45 to 100 GW·d/t.

In summary, as burnup increases the amount of SF drops but the total uranium needs are nearly constant due to higher enrichment needs. Higher enrichment results in more SWUs. The SF contains more plutonium per tonne, but less plutonium overall.

8.5. COOLING TIME

Cooling time for SF prior to reprocessing has a significant impact on the flow of plutonium for construction of new FRs. This sensitivity analyses assesses the impacts of longer cooling times for both LWR and FR SF. The analysis is based on the homogeneous BAU–FR high growth framework base case. Minimum cooling time for SF is increased by 5 and 10 years for both LWR and FR SF.

8.5.1. Fast reactor share of energy production

Figure 8.15 shows the impact on power production growth by cooling LWR and FR fuel an extra 5 years versus the BAU–FR framework base case result. The number of FRs commissioned by 2100 drops by over 50% when the cooling time is increased by 5 years and by over 62% when the cooling time is increased by 10 years.

The longer cooling time for LWR SF results in more of the fuel being tied up in wet storage as it cools longer. This delay in availability for reprocessing slows the rate of plutonium becoming available for new FRs. The longer cooling time for the FR SF results in each new FR needing much more plutonium for startup — twice as much with 5 years more cooling and three times as much with 10 years more cooling. The increase in plutonium needs is also due to each FR requiring ~2 t of plutonium per year for refuelling. With a longer cooling time, the time before the first SF is reprocessed, fabricated into new fuel, and loaded back into the reactor, increases. The break-even FR is not self-sufficient until recycling of its fuel is established.

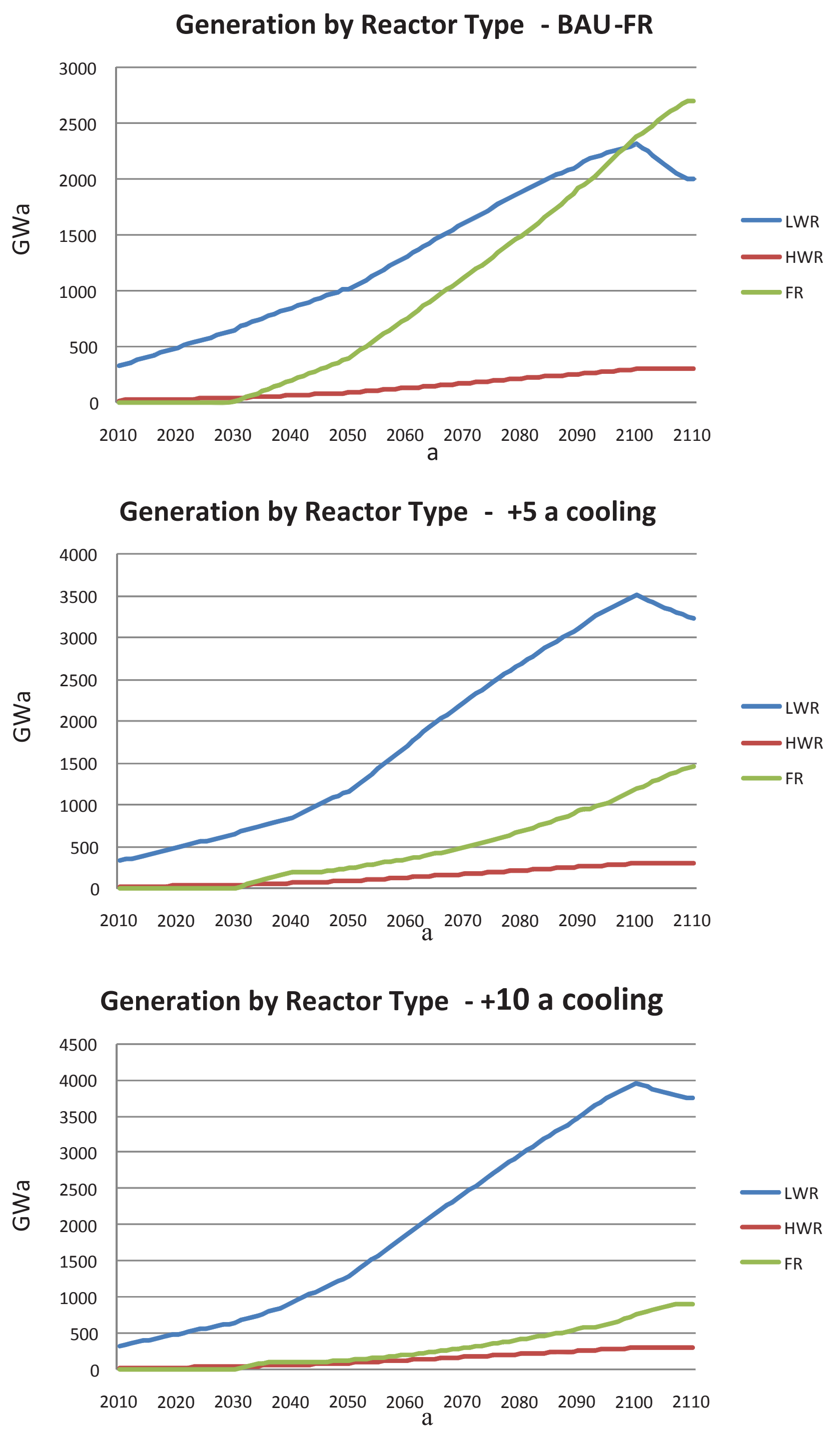


FIG. 8.15. Electricity generation by reactor type — comparison of BAU–FR, BAU–FR + 5 years extra cooling time and BAU–FR + 10 years extra cooling time.

8.5.2. Uranium usage

Figure 8.16 shows the impact on total uranium usage. Owing to fewer FRs being built, more LWRs are required to meet the total energy demand.

Uranium usage by 2100 increases by 29% when LWR and FR SF are both cooled an additional 5 years over the cooling time of the framework base case. The increase is 40% with 10 years of additional cooling.

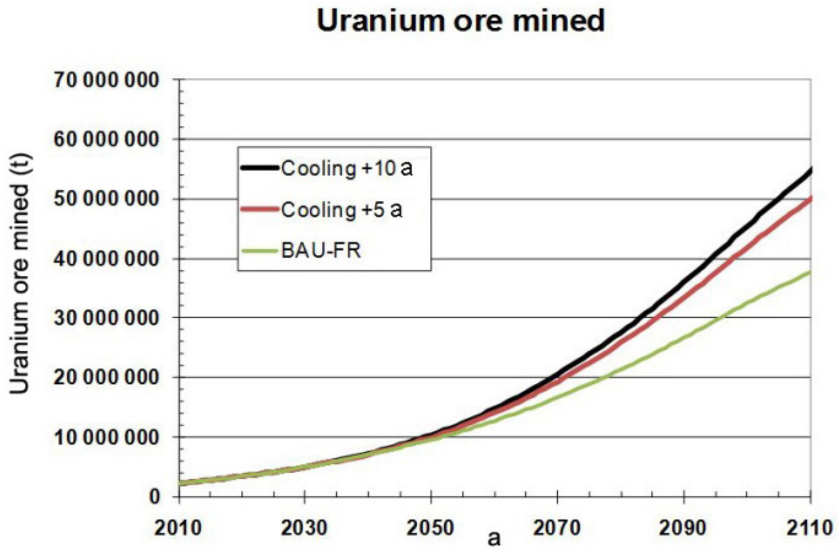


FIG. 8.16. Uranium usage for BAU–FR, BAU–FR + 5 years cooling and BAU–FR + 10 years cooling.

8.5.3. Reprocessing loads

Figure 8.17 shows the impact of longer cooling times on reprocessing loads. With longer cooling times, the initial reprocessing capacity must be raised to support the FR introduction rate. However, there is insufficient used LWR fuel in storage to support these higher reprocessing capacities and actual throughput rates drop off prior to 2045 in the 5 year additional cooling case and just after 2035 in the 10 year additional cooling case. This results in less plutonium available to support new FRs. Since fewer FRs are built, the LWRs grow more quickly and the sustained reprocessing throughput rate rises until it is higher than the framework base case late in the simulation. However, the additional plutonium is not sufficient to overcome the impact of the longer cooling time for the FR SF, so sustained commissioning rates remain below that of the base case.

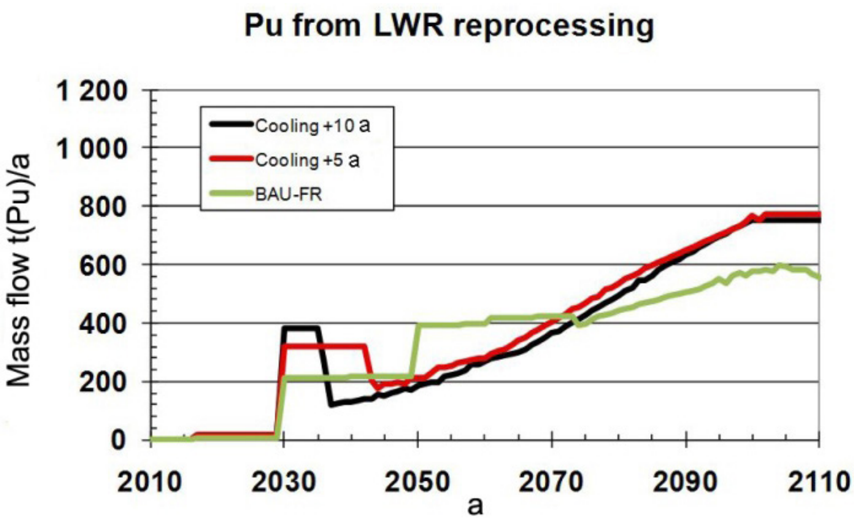


FIG. 8.17. Plutonium separated from LWR spent fuel for BAU–FR, BAU–FR + 5 years cooling and BAU–FR + 10 years cooling.

8.5.4. Plutonium inventories

Figure 8.18 shows the impact on total plutonium in the system. As cooling times increase, fewer break-even FRs are built and more LWRs are built. Since LWRs produce more plutonium than they consume, this results in an increase in the total plutonium in the system.

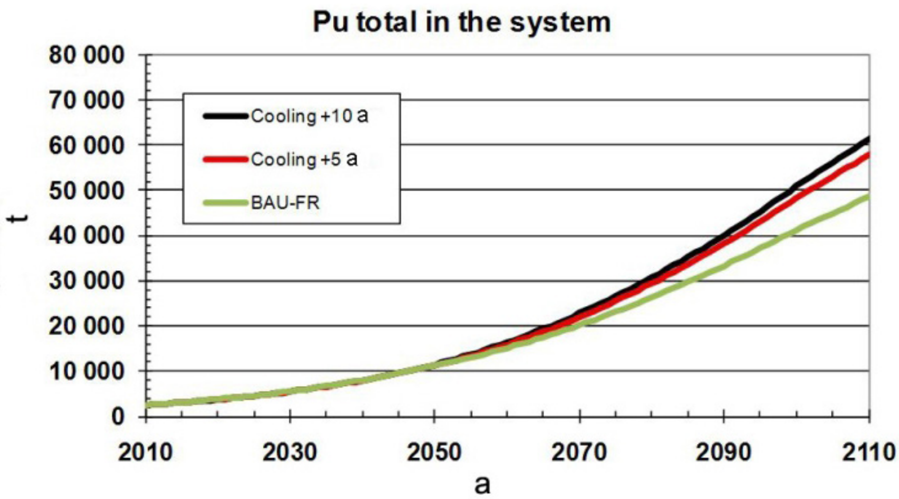


FIG. 8.18. Total plutonium in the fuel cycle for BAU–FR, BAU–FR + 5 years cooling and BAU–FR + 10 years cooling.

For the case of an additional five years cooling, the increase in 2100 is 17%, while for the case of an additional ten years cooling, the increase is over 23%.

In summary, longer cooling times for SF result in longer recycle times. As plutonium is spending more time out of reactors, there is less available to construct FRs and the FR share of total electricity generation drops. With more LWRs, there is increased uranium usage and increased total plutonium in the system.

8.6. COMPARISON OF BURNER FAST REACTORS

In Section 7, the once-through fuel cycle of the BAU scenario and the scenario of 'break-even' FR (BR: ~1.0) introduction to the BAU scenario were studied as 'framework base cases'. The break-even reactor produces as much plutonium as it consumes and, therefore, neither requires nor provides additional plutonium once recycling of its own fuel is established after the first few years of operation.

In this section, the impact of using a 'burner' FR is assessed. Burner FRs are designed to consume more plutonium than they produce, as a means of reducing overall plutonium inventories (BR < 1.0). Burner FRs are often also used to consume MAs to reduce the long term decay heat and radiotoxicity of waste. The FRs used for this sensitivity study burn both plutonium and MAs, and a TRU CR is used instead of the typical fissile CR or BR in order to take into account the rates of production and destruction of the TRU isotopes.

The FRs used for this analysis are based on a core conceptual design developed by the Argonne National Laboratory (ANL) in the USA [8.1, 8.2]. The CR is varied from 0.0 to 1.0, with a CR of 1.0 being approximately equal to a BR of 1.0. Figure 8.19 shows the core arrangements at each CR level used in this analysis.

The framework base case used is the BAU–FR high growth case documented in Section 7.3.2. Besides the difference in FR used, the reprocessing is changed to recover both plutonium and MAs. Reprocessing capacity for LWR SF is unlimited after the initial FR introduction period. A goal-seeking method is used with the fuel cycle code to determine the number of FRs to commission. Owing to the above differences in CR versus BR, recycle of MAs, different reprocessing rates and use of a goal-seeking algorithm, the results of the CR = 1.0 case exhibit the same behaviour as the BAU–FR framework base case but do not exactly match the base case values.

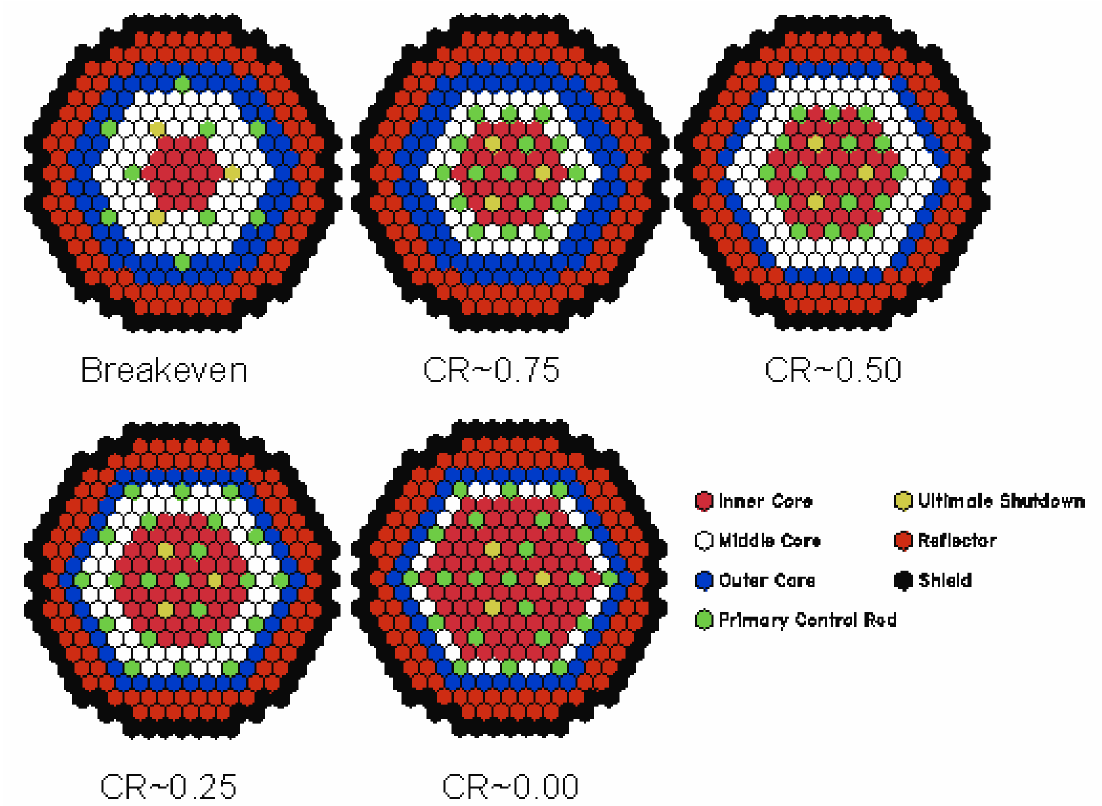


FIG. 8.19. Core configurations for burner fast reactors used in the sensitivity study.

The principal feature of burner FRs is the continuing need for make-up transuranics from reprocessing of LWR SF. For comparison in the sensitivity analyses only the CR is varied. The values used are CR = 1.0, 0.75, 0.5, 0.25 and 0.0. (In practice, it may not be possible to fabricate the fuel needed to achieve CRs below 0.5.) As the reprocessing capacity is unlimited, the actual reprocessing loads vary with CR. At lower CR, the FRs require more make-up transuranics, leaving less for commissioning of new FRs. This results in fewer FRs being built overall.

8.6.1. Fast reactor share of energy production

Figure 8.20 shows the energy generation by reactor type (KI-1) by CR. The same behaviour is observed as for the BAU–FR framework base case, with the share of FRs slowly growing while both LWRs and HWRs also grow until 2100. After 2100, FRs replace retiring LWRs and the LWR share decreases. At lower CRs, more transuranics are needed to fuel existing FRs. This results in fewer new FRs.

8.6.2. Uranium usage

Since fewer FRs are built at lower CR, more uranium is needed to support the additional LWRs. Figure 8.21 shows the uranium ore mined for the five cases along with the SWU requirements. The values for CR = 1.0 are very close to those of the BAU–FR base case.

8.6.3. Reprocessing loads

The additional LWRs result in more LWR SF to be reprocessed and more TRU recovered. Figure 8.22 documents the additional separations of LWR (thermal reactor) SF by showing the increased amounts of transuranics recovered as the CR drops.

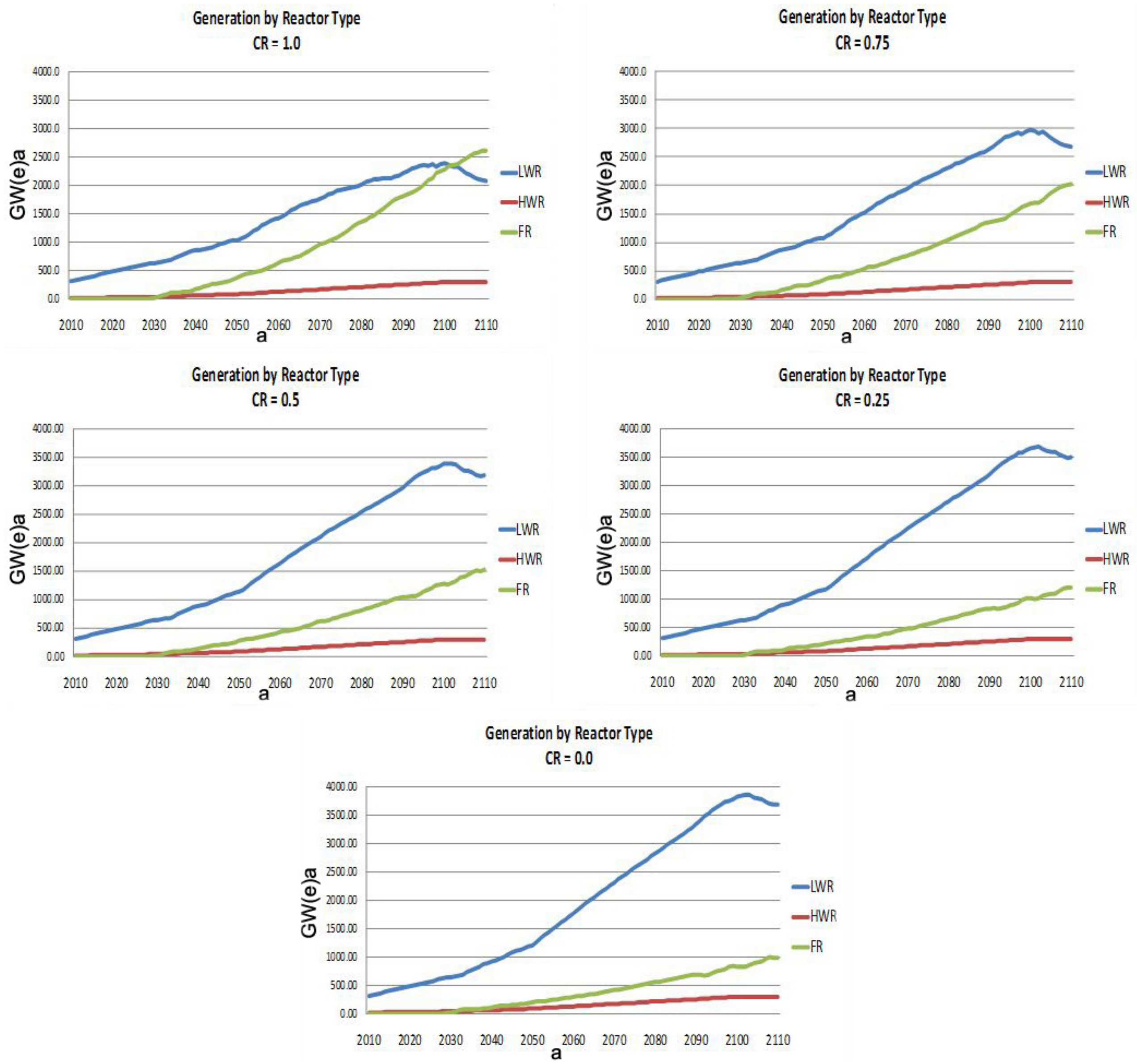


FIG. 8.20. Electricity generation by reactor type for CR = 1.0, 0.75, 0.5, 0.25 and 0.0.

8.6.4. Plutonium inventories

One objective of using burner FRs in a fuel cycle is to reduce the total plutonium in the fuel cycle. Figure 8.23 shows the change in total plutonium in the complete fuel cycle system as a function of CR.

In summary, burner FRs do reduce the amount of total plutonium and total transuranics by net consumption in the process of producing energy. As the CR drops, more of the transuranics are consumed by the FRs, resulting in the need for processing additional transuranics produced by the LWRs. To the extent there is limited LWR SF, this results in a lower FR deployment rate with lower CR.

8.7. COMPARISON OF BREEDER FAST REACTORS

In Section 7, a once-through fuel cycle of the BAU scenario and the scenario of break-even FR (BR: ~1.0) introduction to the BAU (BAU–FR scenario) were studied as framework base cases.

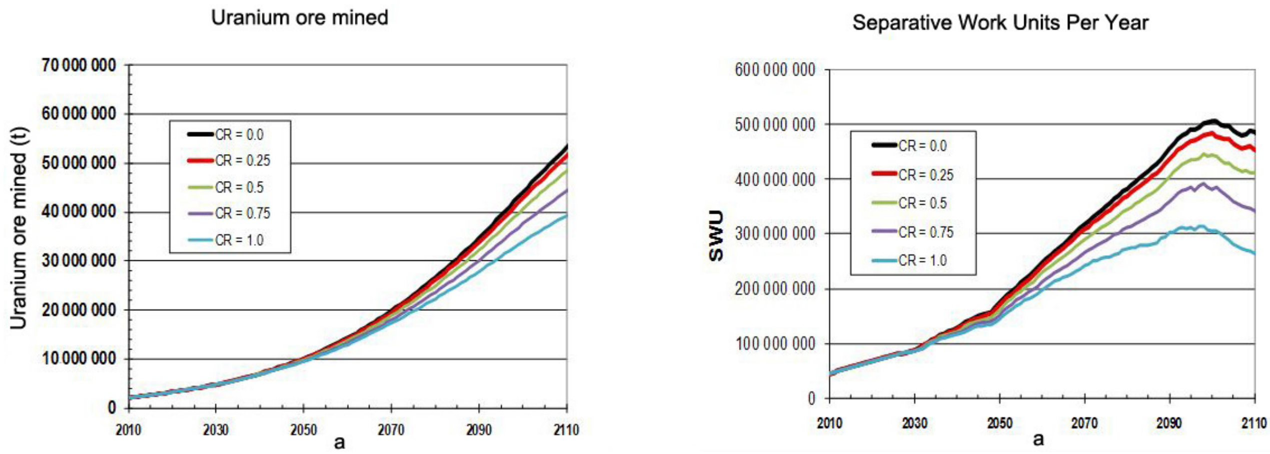


FIG. 8.21. Uranium ore mined (five cases) with SWU requirements.

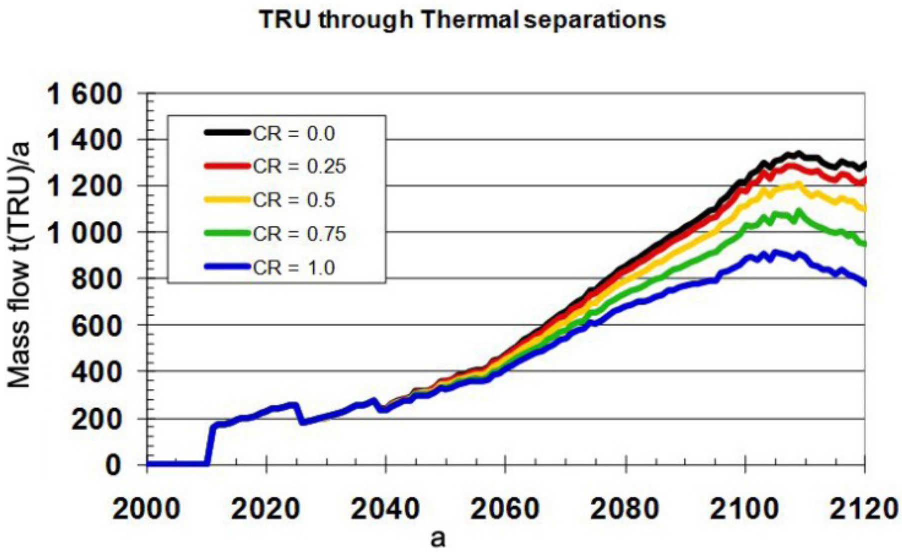


FIG. 8.22. Transuranics separated from LWR SF for CR = 1.0, 0.75, 0.5, 0.25 and 0.0.

In this section, a study is made of the impact on mass flow analysis results due to variations in FR performance, such as BR and average discharged FR burnup, and the way these differences affect the transition scenario from LWR to FR in the middle of the twenty-first century. In order to compare the performances of FRs, the BAU+ scenario was taken as the base line, because various efforts would be applied to lessen the electricity generation cost of LWRs and to save natural uranium resources in the situation where FR introduction starts on a worldwide scale.

8.7.1. Comparison of breeding performances of three fast reactors

Table 8.1 compares the breeding performances of three FRs. As shown in the table, the BRs of the two breeders are almost at the same level. On the other hand, the operation cycle lengths and the initial fissile inventories of two reactors are considerably different. Generally, if a breeder design is aimed at high FR introduction speed, the operation cycle length becomes short and the core inventory small. On the other hand, from the viewpoint of economics, high burnup and long operation cycle length are generally preferable.

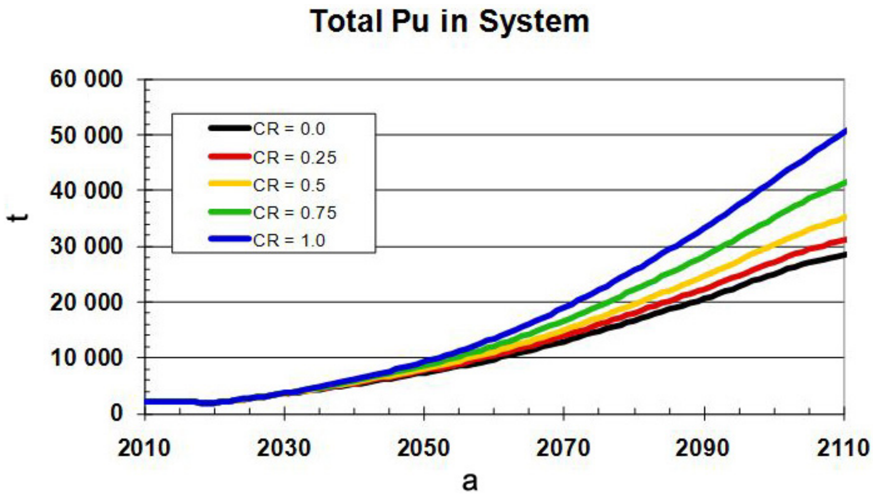


FIG. 8.23. Total plutonium in the fuel cycle for CR = 1.0, 0.75, 0.5, 0.25 and 0.0.

TABLE 8.1. BREEDING PERFORMANCES OF THREE FAST REACTORS

FR name	Av. burnup (MWd/t) whole reactor	Operation cycle length (day)	Pu-fissile initial inventory (ton/GWe)	Reactor doubling time* (y)	BR
Break-even FR (F1) (Demonstration reactor)	37677	140	2.25	-	~ 1.0
Breeder FR (F2) (Prototype reactor)	31061	180	2.94	18.5	~ 1.2
High burnup breeder FR (F3) (Commercial reactor)	53526	540	4.36	27.4	~ 1.2

Reactor doubling time (y) = Initial Pu-fissile inventory (t) / Pu-fissile gain (t/y)
* assuming load factor of 100%

8.7.2. Homogeneous model analysis

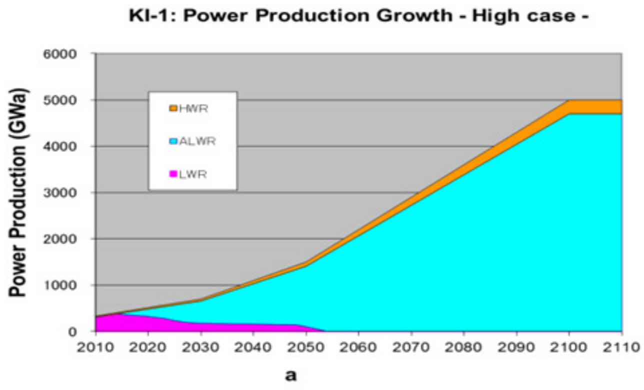
8.7.2.1. BAU+ scenario for the comparison of fast reactors

As previously mentioned, in order to compare three FRs, the BAU+ scenario was chosen as the base line. The mass flow analysis results of the BAU+ scenario are, therefore, described first. The analysis conditions of the scenario are already mentioned in Section 6.2.

8.7.2.1.1. Reactor power share

Figures 8.24 and 8.25 show the reactor power share of BAU+ high case and moderate case. As shown in the figures, LWR is replaced with ALWR from 2015 and disappears by 2055 completely.

The share of HWR is settled as 6% of total nuclear power capacity.



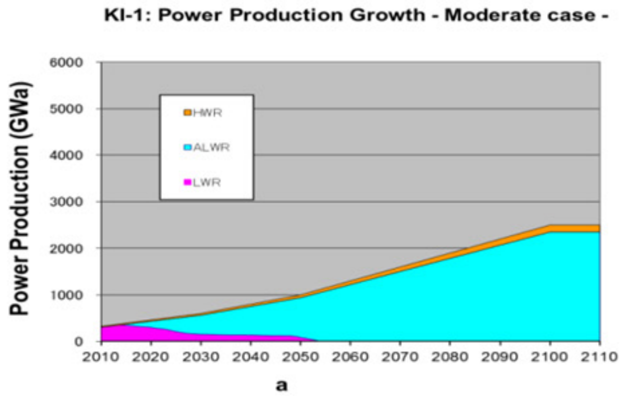
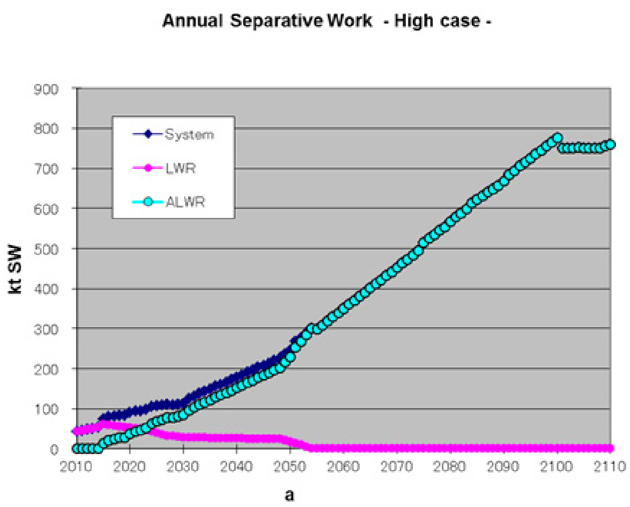


FIG. 8.24. Reactor power share of the BAU+ scenario (high case).

FIG. 8.25. Reactor power share of the BAU+ scenario (moderate case).

8.7.2.1.2. Uranium separative work

Figures 8.26 and 8.27 show the annual separative work for uranium enrichment. As shown in the figures, the separative work reaches around 800 × 10^6 SWU in 2100 in the high case, and 400 × 10^6 SWU in the moderate case, respectively.



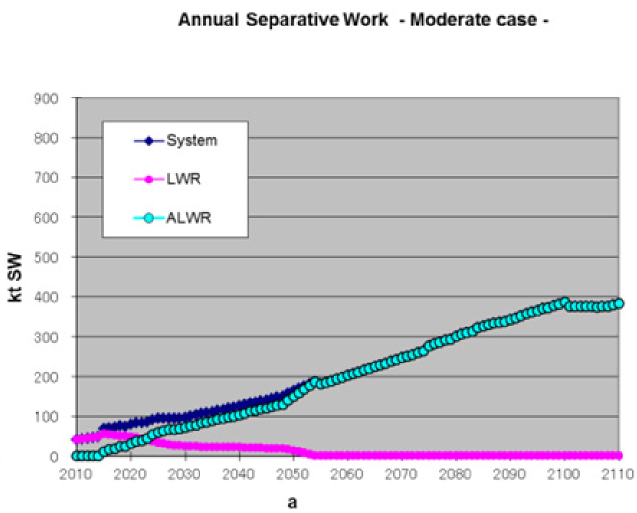


FIG. 8.26. Annual uranium separative work for the BAU+ scenario (high case).

FIG. 8.27. Annual uranium separative work for the BAU+ scenario (moderate case).

8.7.2.1.3. Fuel fabrication load

Figures 8.28 and 8.29 show the annual fuel fabrication load in the high case and moderate case, respectively. The fabrication load monotonically increases along with the nuclear power demand. In the high case, the fabrication load for ALWRs becomes around 90 kt/a HM in 2100, and that for HWRs becomes 50 kt/a HM, much higher than its power share. In the moderate case, the fabrication load is half of that in the high case.

8.7.2.1.4. Natural uranium consumption

Figures 8.30 and 8.31 show the cumulative natural uranium demand. The cumulative natural uranium demand reaches 37.8 Mt in 2100 in the high case and 22.6 Mt in the moderate case, respectively.

The 2009 Red Book [8.3] reports ‘identified resources’ and ‘undiscovered resources’ for natural uranium resources. According to the report, in the category of identified resources, cheap natural uranium resources at a cost

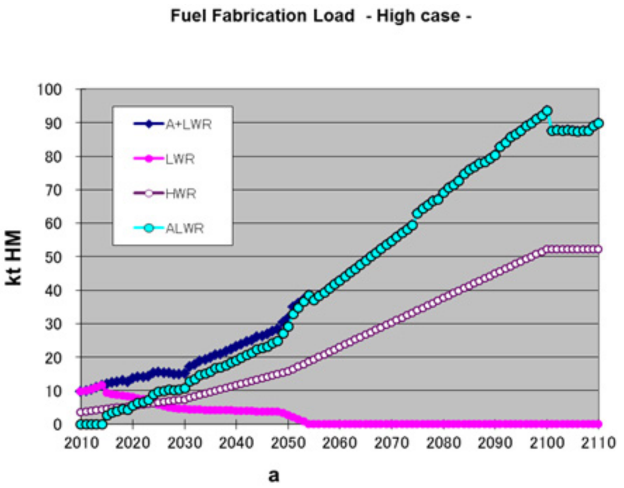


FIG. 8.28. Fuel fabrication load for each reactor in the BAU+ scenario (high case).

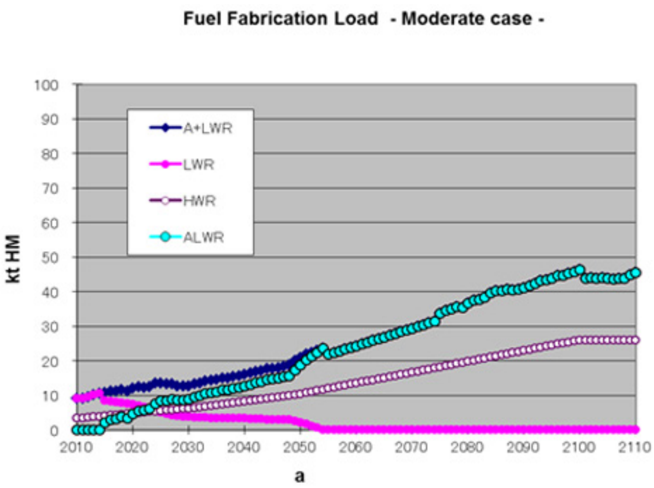


FIG. 8.29. Fuel fabrication load for each reactor in the BAU+ scenario (moderate case).

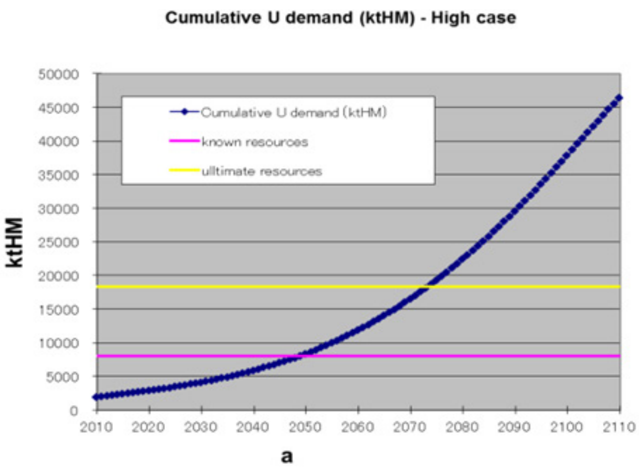


FIG. 8.30. Cumulative natural uranium demand in the BAU+ scenario (high case).

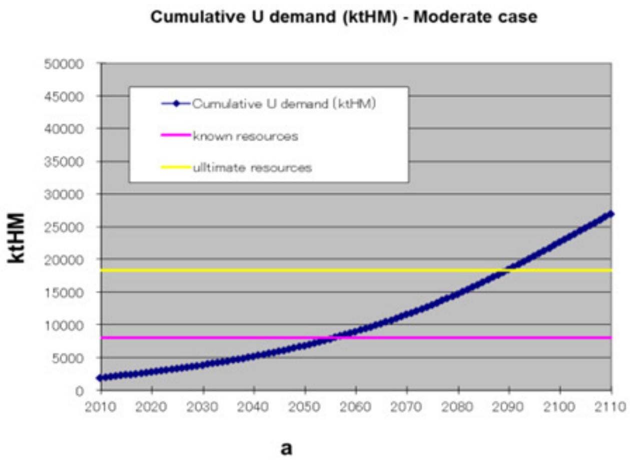


FIG. 8.31. Cumulative natural uranium demand in the BAU+ scenario (moderate case).

of below \$130/kg U are estimated as 5.40 Mt, and 6.30 Mt for those at a cost below \$260/kg U. Hereinafter, we call these resources 'known (conventional) resources'. The report states that undiscovered resources consist of 6.55 Mt of cheap uranium resources (below \$130/kg U) and 3.76 Mt of more expensive resources (below \$260/kg U or more). So, an additional 10.3 Mt of natural uranium resources are estimated as conventional resources. The total amount of conventional resources of 16.6 Mt is assumed to include identified resources and undiscovered resources. This amount is also referred to as 'ultimate (conventional) resources'.

Based on the report, horizontal lines of an additional 6.3 Mt and an additional 16.6 Mt are indicated in the graphs of the cumulative uranium demand. The conventional natural uranium resources will be exhausted around 2070 in the high case and around 2090 in the moderate case, assuming only a once-through fuel cycle strategy.

8.7.2.2. Analysis conditions of fast reactor introduction scenario

We first studied an FR introduction scenario in the homogeneous global model, where all countries approve an FR introduction strategy and free trade of fuel material is guaranteed, and there are no restrictions on the transport of fuel or SF. The following analysis conditions were assumed to analyse the global mass flow, aiming at the maximum FR introduction case.

8.7.2.2.1. Fast reactor introduction speed

The FR introduction speeds are provided the same way as the one in the BAU–FR scenario of the framework base case in Section 7. The speeds after 2051 are to be maximized according to plutonium availability related to the breeding performances of FRs. In the fuel cycle analyses using the NFCSS, the maximum FR introduction capacity is obtained with the strategy below.

8.7.2.2.2. Spent fuel reprocessing strategy

The maximum FR introduction speed (after 2050) was estimated using the following set of assumptions;

(a) The SF from ALWRs is reprocessed first. The out-of-reactor time of ALWRs is 6 years.
(b) The SF from LWRs is reprocessed next. As much of the LWR SF will have cooled for more than 6 years before being required for FRs, the isotopic composition used for LWR SF was that of 30 year cooling (although assumed to be available after 6 years).
(c) Reprocessing of SF from LWRs and ALWRs is to be undertaken to get a zero annual plutonium balance without assuming any limitation on reprocessing capacity.

8.7.2.3. Break-even fast reactor (‘F1’)

The mass flow difference between the BAU+ and the ‘break-even FR’ maximum introduction case is considered first.

8.7.2.3.1. Maximum fast reactor power share

Figures 8.32 and 8.33 show the power production growth curves of each reactor type. The maximum FR introduction is restrained by the zero breeding performance of the break-even FR. The power share of the break-even FR reaches around 44% of total nuclear power in 2100 in the high case, and 53% in the moderate case. The reason why the moderate case gives a higher FR power share is that the relative amount of LWR SF to the total nuclear power capacity becomes larger in the moderate case than in the high case (the historical reactor capacity does not change in the two cases).

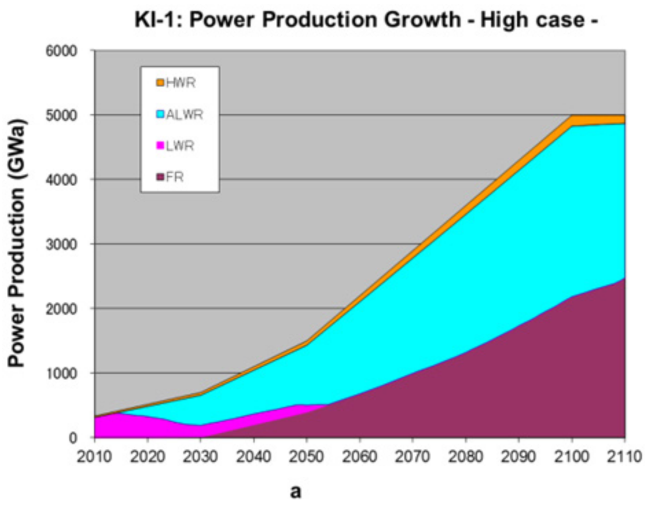


FIG. 8.32. Power share in the FR introduction scenario (F1, high case).

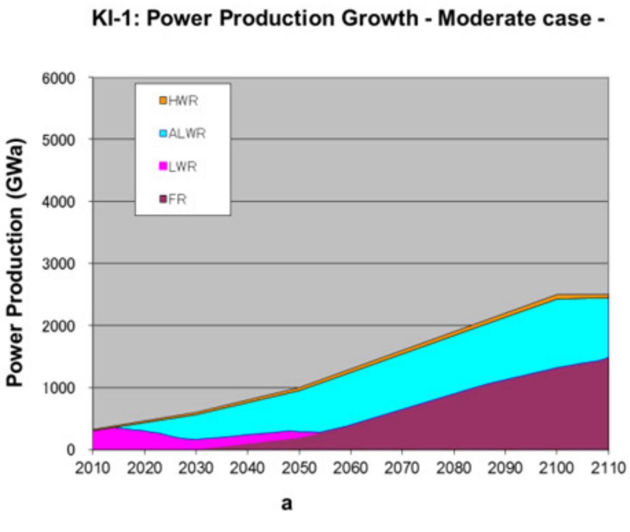


FIG. 8.33. Power share in the FR introduction scenario (F1, moderate case).

8.7.2.3.2. Uranium separative work

Figures 8.34 and 8.35 show annual uranium separative work in the high case and moderate case, respectively. Depending on the FR introduction rate, the values go down from the BAU+ scenario case shown in Figs 8.25 and 8.26. The values decrease from 750 × 10^6 to 450 × 10^6 SWU in the high case, and 400 × 10^6 to 180 × 10^6 SWU in the moderate case in 2100.

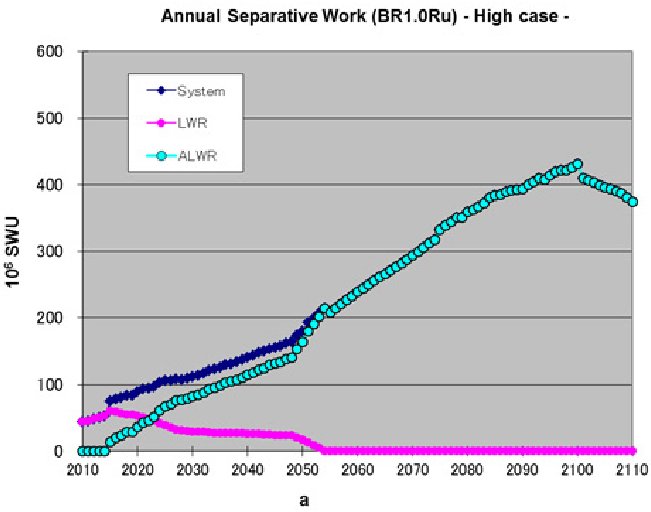


FIG. 8.34. Uranium separative work in the FR introduction scenario (F1, high case).

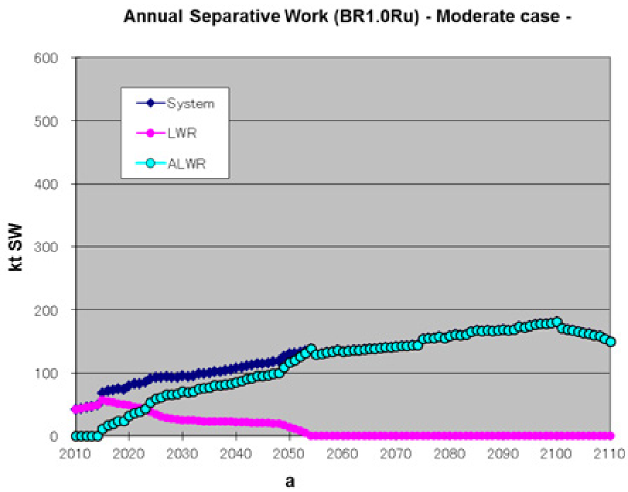


FIG. 8.35. Uranium separative work in the FR introduction scenario (F1, moderate case).

8.7.2.3.3. Reprocessing load and plutonium balance in the system

Figures 8.36 and 8.37 show the annual reprocessing load for SF from LWRs, ALWRs and FRs. The requirement for the reprocessing of SF from LWRs and ALWRs rises stepwise depending on the FR introduction speed change, and the peak value is around 45 kt/a HM in the high case and 30 kt/a HM in the moderate case. The requirement for FR SF reprocessing increases monotonically along with the FR capacity increase and reaches almost the same level as for the ALWR at the end of the century.

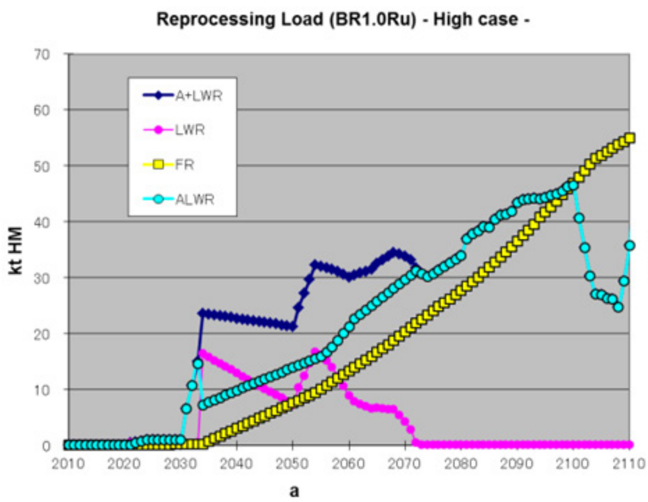


FIG. 8.36. Reprocessing load in the FR introduction scenario (F1, high case).

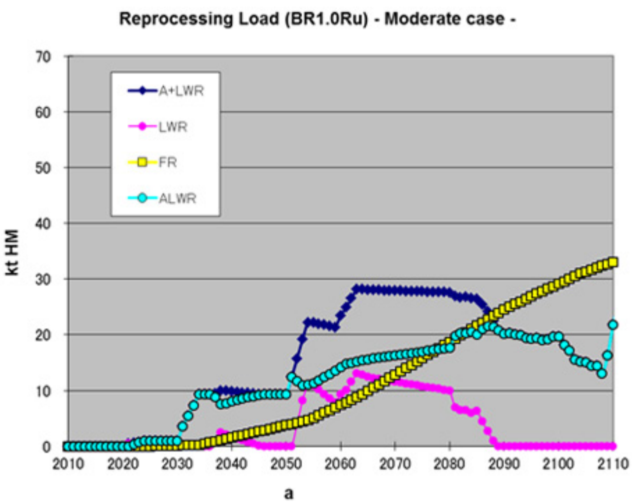


FIG. 8.37. Reprocessing load in the FR introduction scenario (F1, moderate case).

Figures 8.38 and 8.39 show the annual plutonium balance between LWRs, ALWRs and FRs. As shown in the figures, the plutonium supply from LWRs and ALWRs reaches around 600 t/a in the high case, and 300 t/a in the moderate case in 2100. On the other hand, as shown in Figs 8.40 and 8.41, the amount of plutonium being recycled globally each year in the closed FR cycle is very large, around 8000 t/a in the high case and 4000 t/a in the moderate case in 2100.

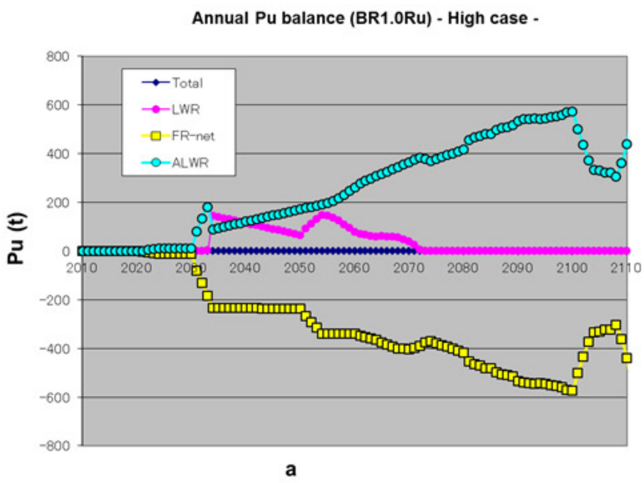


FIG. 8.38. Plutonium balance between reactors in the FR introduction scenario (F1, high case).

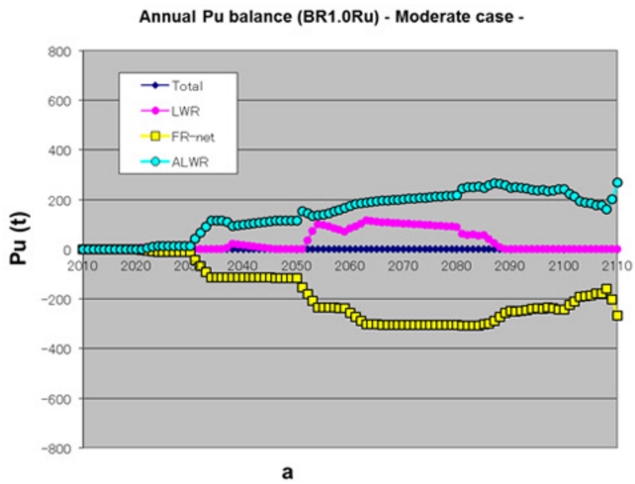


FIG. 8.39. Plutonium balance between reactors in the FR introduction scenario (F1, moderate case).

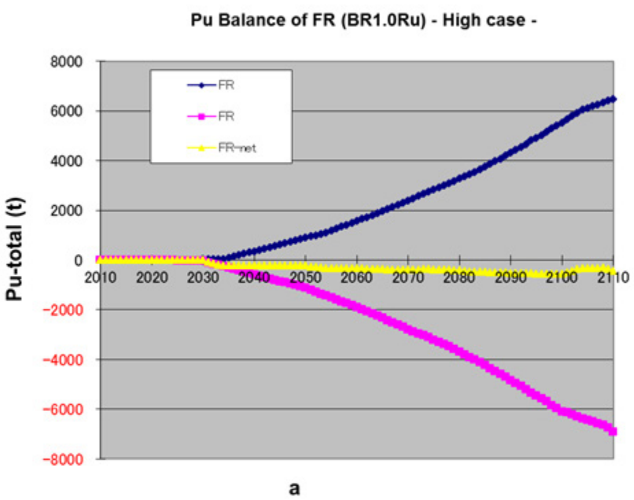


FIG. 8.40. Amount of plutonium managed for the FR scenario (F1, high case).

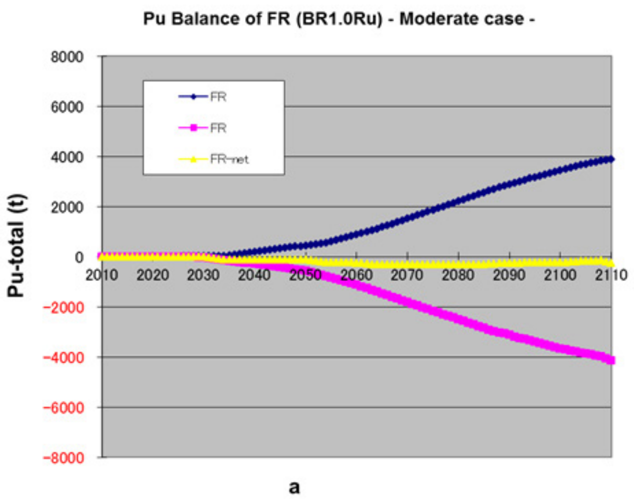


FIG. 8.41. Amount of plutonium managed for the FR scenario (F1, moderate case).

8.7.2.3.4. Fuel fabrication load

Figures 8.42 and 8.43 show the annual fuel fabrication load for each reactor type in the high and moderate cases, respectively. In the high case, the fuel fabrication load for ALWRs decreases to 50 ktHM in 2100 from 90 ktHM in the BAU+ scenario (Fig. 8.28). On the other hand, the fuel fabrication load for FR increases to almost the same level as for ALWRs in 2100. The total fuel fabrication load for all reactors remains the same at 130 ktHM as for the BAU+ scenario.

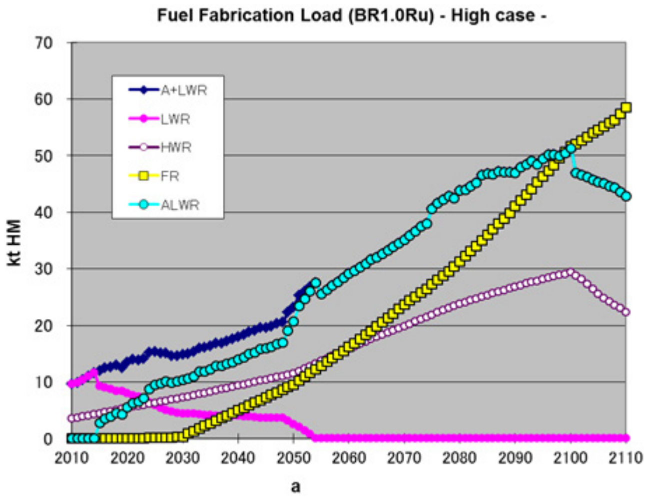


FIG. 8.42. *Annual fuel fabrication load for each reactor (F1, high case).*

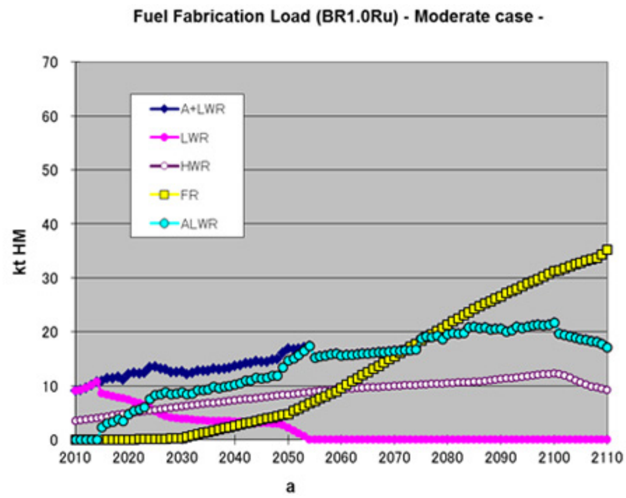


FIG. 8.43. *Annual fuel fabrication load for each reactor (F1, moderate case).*

8.7.2.3.5. Natural uranium saving

Figures 8.44 and 8.45 show the cumulative natural uranium demand. In order to make natural uranium saving effects clearer, the natural uranium cumulative demand is shown in comparison with the BAU+ scenario in Figs 8.30 and 8.31. In the high case, the uranium saving effect reaches around 12 Mt in 2100, and in the moderate case around 8 Mt.

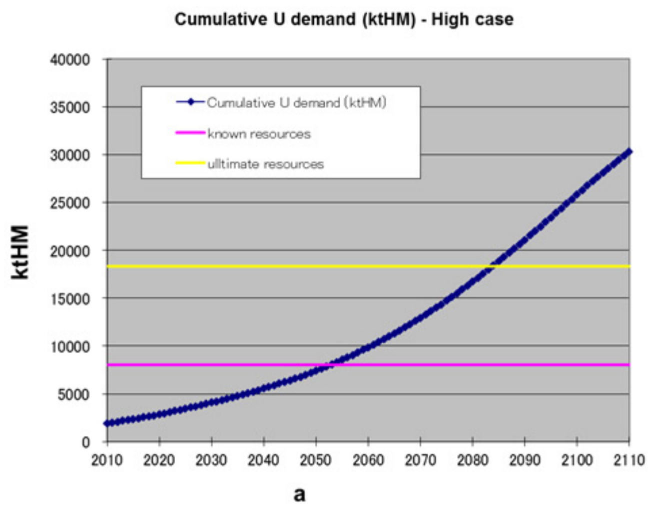


FIG. 8.44. *Cumulative natural uranium demand in the FR introduction versus the BAU+ scenario (F1, high case).*

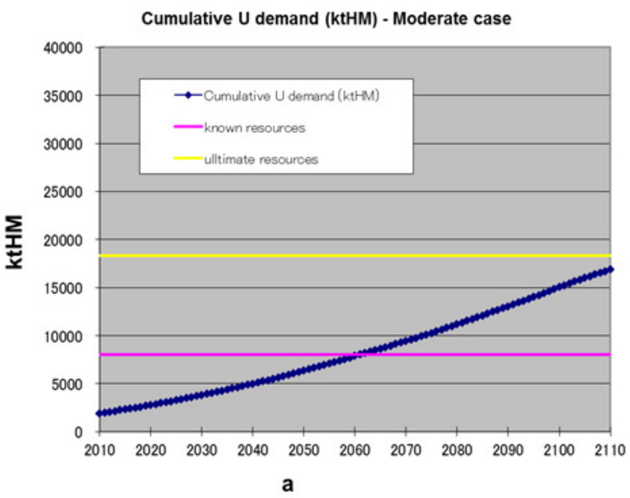


FIG. 8.45. *Cumulative natural uranium demand in the FR introduction versus the BAU+ scenario (F1, moderate case).*

8.7.2.4. Comparison with breeder fast reactor ('F2') and high burnup breeder fast reactor ('F3')

In Sections 8.7.2.3.3–8.7.2.3.5, the differences of the mass flow between the BAU+ scenario and the break-even FR (F1) maximum introduction scenario were described. Natural uranium saving and uranium separative work reduction are the main benefits of FR introduction. On the other hand, it was understood that the large SF reprocessing load and the management of a large amount of plutonium in the closed FR fuel cycle will appear as new issues to be solved. These issues are expected to vary, depending on the breeding performance and average discharge burnup of the FRs. The effects from breeding performance and burnup characteristics are studied by comparison between the break-even FR (F1) and two breeder FRs (F2, F3). For the sake of convenience, only the high nuclear demand growth case is analysed here.

8.7.2.4.1. Maximum fast reactor power share

Figures 8.46 and 8.47 show the power production curves of each reactor type in the two breeder FR (F2, F3) maximum introduction cases. Owing to the breeding performance of F2 and F3, the power shares of FRs can be increased relative to the case having F1. The power shares of F2 and F3 reach 59 and 54% in 2100, respectively compared to 44% for the F1 case. The difference between F2 and F3 comes from the doubling time difference as shown in Table 8.1, in which the BRs are almost at the same level but the doubling time of F2 is shorter than that of F3 because of the small inventory and the short operation cycle length.

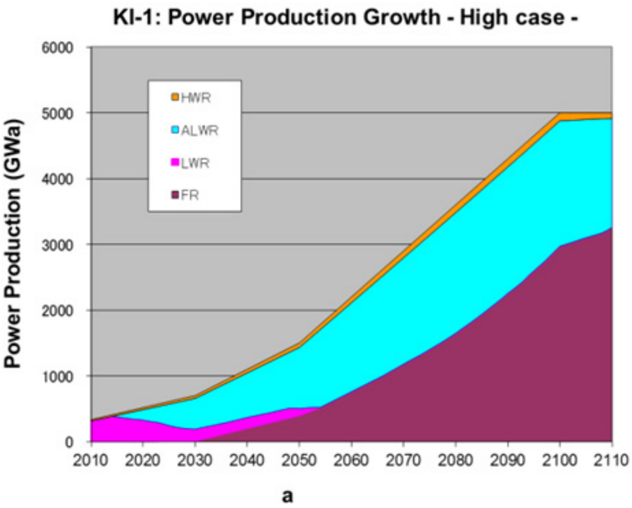


FIG. 8.46. Power share in the FR introduction scenario (F2, high case).

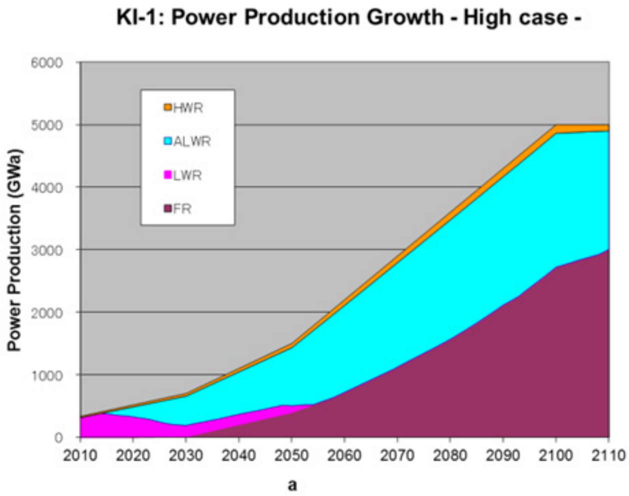


FIG. 8.47. Power share in the FR introduction scenario (F3, high case).

8.7.2.4.2. Uranium separative work

Owing to the larger power share of FRs than in the F1 case, the annual separative work (Figs 8.48 and 8.49) for uranium enrichment for LWRs and ALWRs decreases in the F2 and F3 cases. The peak value changes to 320×10^6 SWU in the F2 case and to 350×10^6 SWU in the F3 case, compared with 450×10^6 SWU in the F1 case. The differences are due to the larger displacement of ALWR reactors by FRs.

8.7.2.4.3. Reprocessing load and plutonium balance in the system

The reprocessing load of LWR and ALWR SF is linked directly to the breeding performance of the FR adopted in the NES. As shown in Section 8.7.2.3.3 of the F1 case (Fig. 8.36 is linked directly to the breeding performance), the reprocessing load of SF from LWRs and ALWRs continues to increase to 45 ktHM in 2100, when nuclear power growth ends, because there is no plutonium breeding in F1, so the ALWR share must increase to support an expanding FR population. In the breeder F2 and F3 cases, the reprocessing load of LWR and ALWR SF peaks at around 40 ktHM because of the breeding of plutonium in the FRs (Figs 8.50 and 8.51). The peak value of the reprocessing load of LWR and ALWR SF is almost the same level for the F2 and F3 cases, because F2 and F3 have the same BR.

On the other hand, the reprocessing load for FR SF varies considerably depending on the average discharge burnup of the FRs. In the case of F1, the load in 2100 becomes 50 kt/a HM, corresponding to an FR power share of 44%. The F2 case has a load of 80 kt/a HM, corresponding to an FR share of 59% in 2100, while the F3 case requires 40 kt/a HM, corresponding to a share of 54% in 2100. In the breeder F2 and F3 cases, the plutonium supply from LWR or ALWR SF becomes unnecessary after 2100, when the nuclear power demand becomes constant.

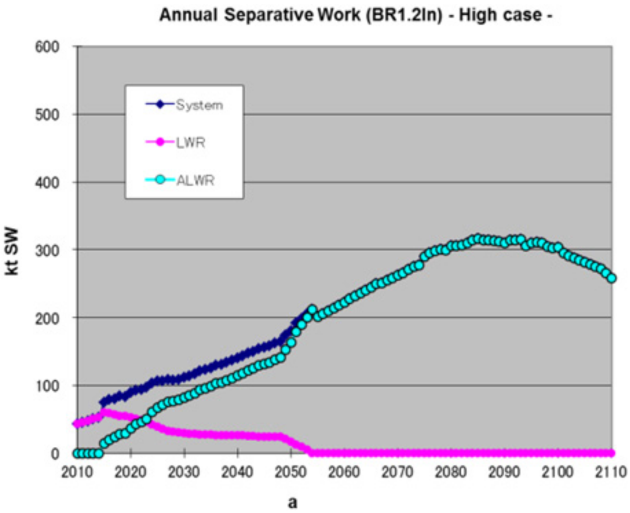


FIG. 8.48. Uranium separative work in the FR introduction scenario (F2, high case).

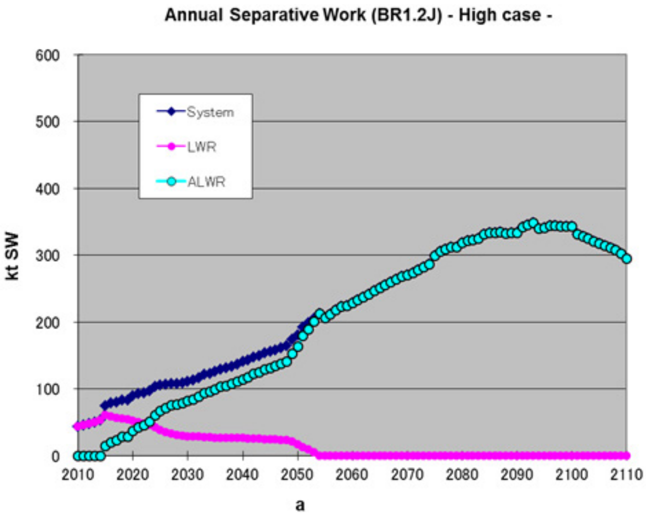


FIG. 8.49. Uranium separative work in the FR introduction scenario (F3, high case).

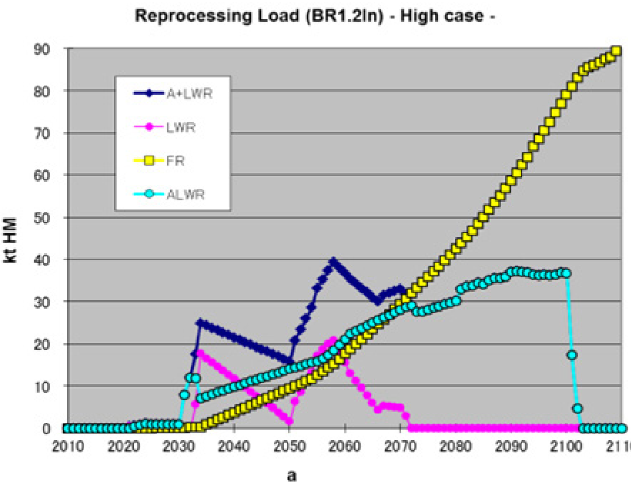


FIG. 8.50. Reprocessing load in the FR introduction scenario (F2, high case).

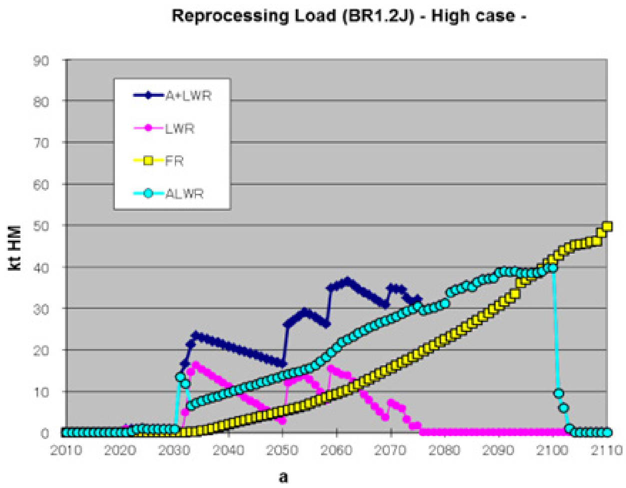


FIG. 8.51. Reprocessing load in the FR introduction scenario (F3, high case).

The FR breeding performance adopted in the NFC system affects the amount of plutonium from the LWRs and ALWRs managed in the system. As mentioned above, in the case of the break-even FR, F1, around 500 t of plutonium has to be managed in 2100, while in the case of breeder FRs, F2 and F3, the value declines to about 400 t in spite of the larger power share of F2 and F3 relative to F1 (Figs 8.52 and 8.53).

The average discharge burnup of FR SF affects the amount of plutonium managed in the closed FR fuel cycle (Figs 8.54 and 8.55). The annual amount of plutonium managed for the high burnup breeder FR, F3, is around 4000 t in 2100 in the high case. On the other hand, the amount of plutonium managed for the medium burnup breeder FR, F2, reaches 8000 t in 2100 in the high case.

8.7.2.4.4. Fuel fabrication load

Figures 8.56 and 8.57 show the annual fuel fabrication loads of each of the F2 and F3 introduction cases, respectively. As shown in Fig. 8.56, the fuel fabrication load of F2 increases steeply and reaches 90 kt/a HM in 2100, while the fabrication load of F3 reaches 50 kt/a HM at this time, around half that of F2.

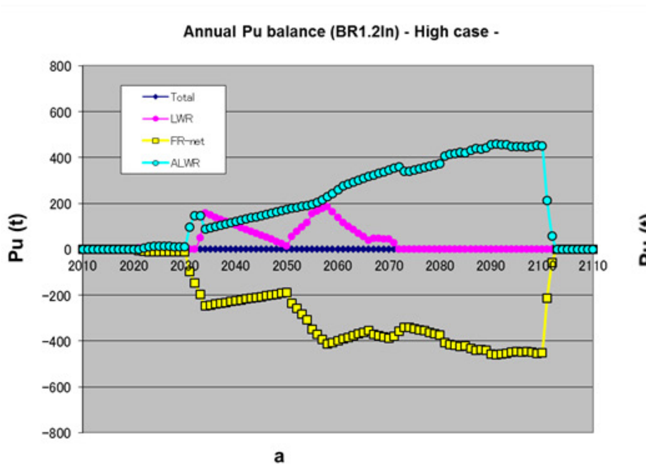


FIG. 8.52. Plutonium balance between reactors in the FR introduction scenario (F2, high case).

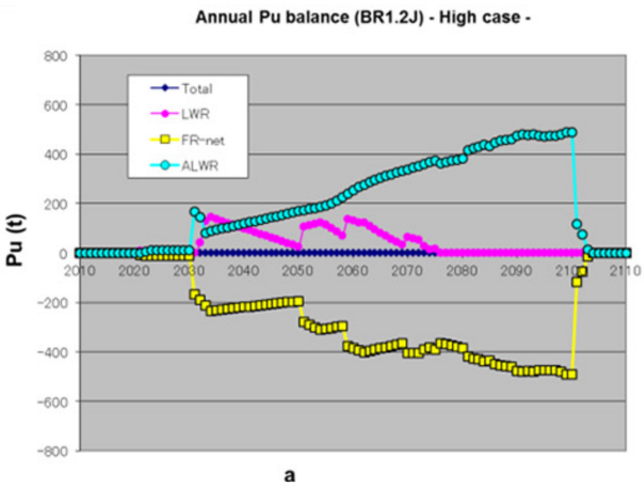


FIG. 8.53. Plutonium balance between reactors in the FR introduction scenario (F3, high case).

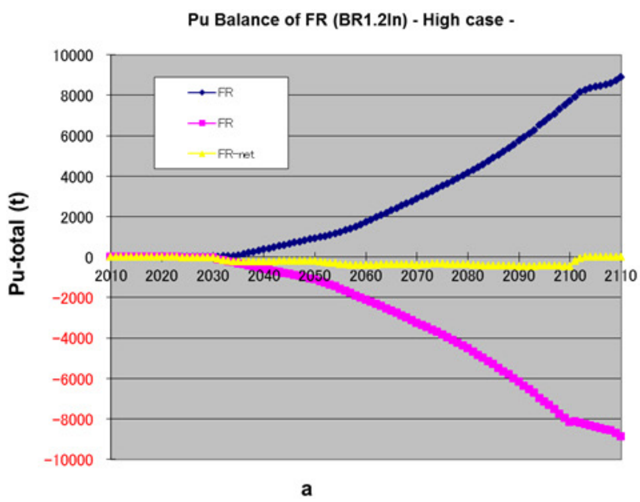


FIG. 8.54. Amount of plutonium managed for the FR scenario (F2, high case).

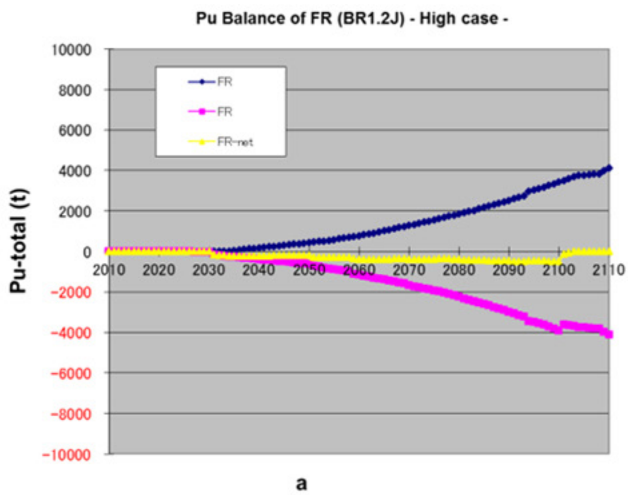


FIG. 8.55. Amount of plutonium managed for the FR scenario (F3, high case).

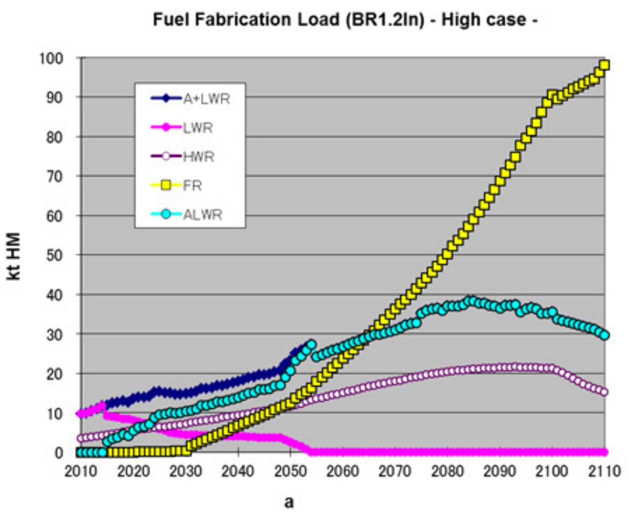


FIG. 8.56. Annual fuel fabrication load for each reactor (F2, high case).

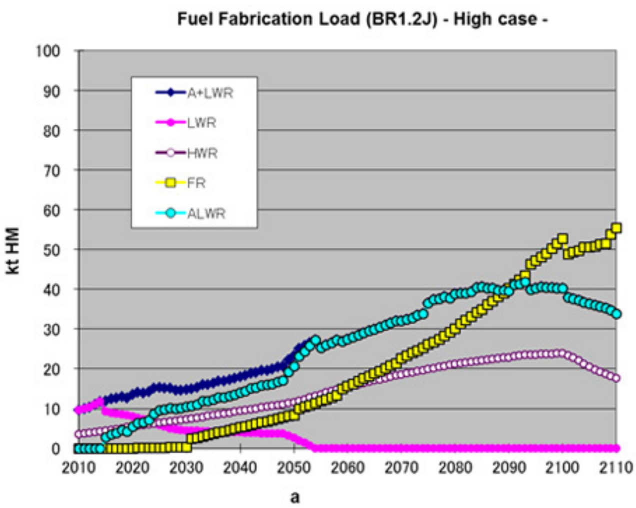


FIG. 8.57. Annual fuel fabrication load for each reactor (F3, high case).

8.7.2.4.5. Natural uranium saving

Natural uranium cumulative demands of the F2 and F3 cases are shown in Figs 8.58 and 8.59 compared to the BAU+ case, respectively. In comparison with the F1 case of 12 Mt of natural uranium saving (Fig. 8.44), the cumulative natural uranium savings reach around 15 Mt in 2100 in the F2 case, and around 14 Mt in the F3 case, depending on the FR power shares.

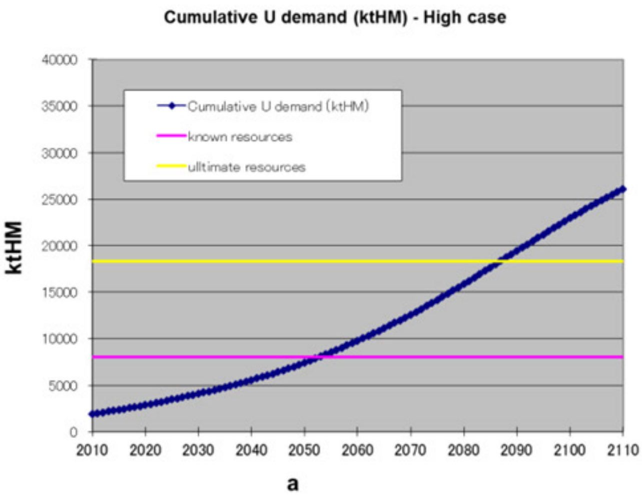


FIG. 8.58. Cumulative natural uranium demand in the FR introduction versus the BAU+ scenario (F2, high case).

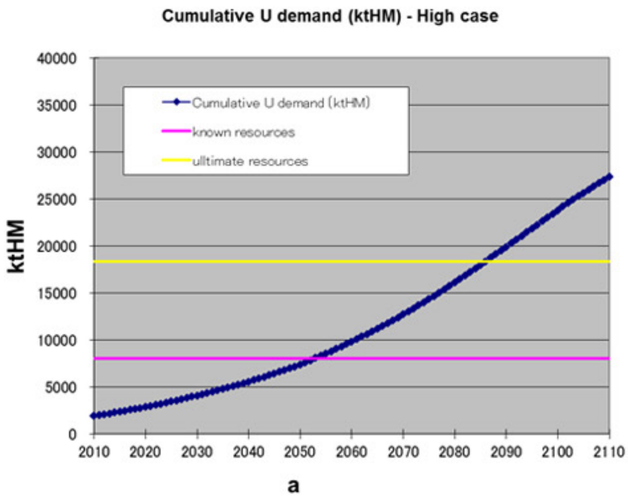


FIG. 8.59. Cumulative natural uranium demand in the FR introduction versus the BAU+ scenario (F3, high case).

8.7.3. Heterogeneous model analysis

As mentioned in Section 3, one of the goals of the CP GAINS is to define a process for the study of the actual heterogeneous world. An NG model was first proposed in this regard. In the model, each country included in the same NG has a similar fuel cycle approach, especially for back end (reprocessing) strategies.

8.7.3.1. Analysis conditions for mass flow calculations

8.7.3.1.1. Nuclear power growth of each NG

As the nominal case of power demand growth of each NG, it was first assumed that the power demand share of each NG in 2100 is NG1:NG2:NG3 = 40%:40%:20%.

Concerning NG3, its power share increases linearly from 0% in 2008 to 20% in 2100. Figure 8.60 shows the nuclear power demand growth of each NG for the high and moderate cases.

8.7.3.1.2. Fast reactor introduction speed in NG1

FRs are to be introduced only into NG1 and the introduction speed is fixed from 2021 to 2050. After 2051, the speed is maximized according to plutonium availability. To avoid the conflict with the plant lifetime condition, the fixed introduction speed from 2021 to 2050 is the same in the high and moderate cases. The speeds are:

— 2021–2030: 1 GW(e) FR demand growth a year (total demand 10 GW(e) at 2030).
— 2031–2050: 9.5 GW(e) FR demand growth a year (total demand 200 GW(e) at 2050).
— After 2051: adjust to maximum FR introduction to follow plutonium availability.

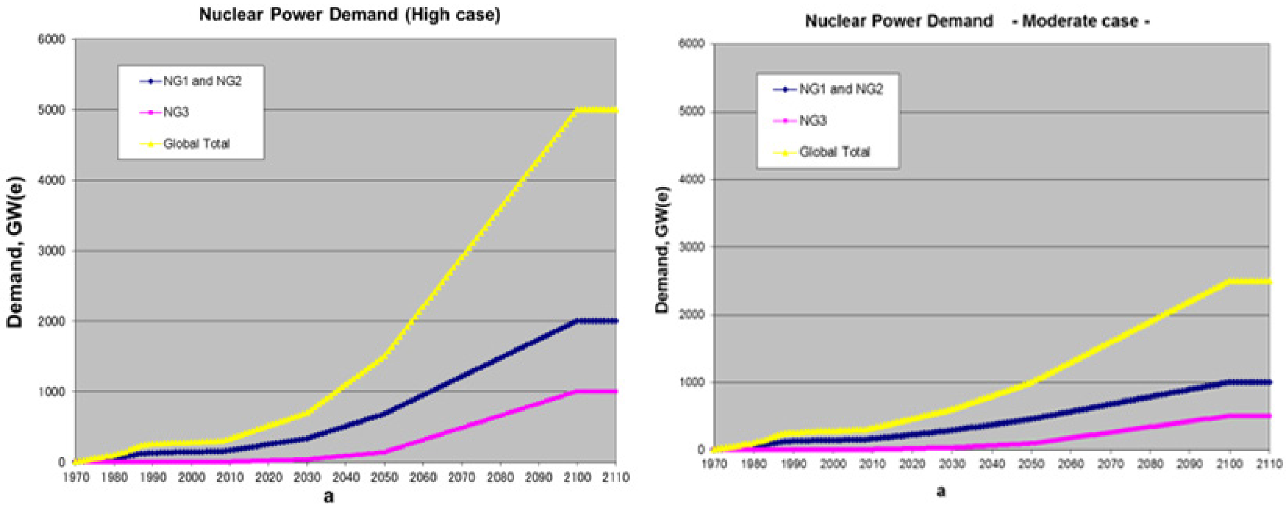


FIG. 8.60. Nuclear power demand growth of each NG for the high and moderate cases.

8.7.3.1.3. Spent fuel reprocessing strategy

The strategy of SF reprocessing is, in principle, the same as in the previously mentioned homogeneous model (Section 8.7.2). However, the availability of LWR and ALWR SF is different between the 'separate case' and the 'synergistic case'.

— Separate case: Reprocessing of SF is limited only to that originated in NG1.
— Synergistic case: All SF from LWRs and ALWRs in the world is available for reprocessing and can be used in NG1.

8.7.3.2. Separate model case

In the separate model, it is assumed that there is no mass flow between NGs. Therefore, global mass flow analysis becomes easy to solve just by adding the three independent calculation results for each NG, and the mass flows in NG2 and NG3 of the once-through fuel cycle do not change.

8.7.3.2.1. Maximum fast reactor power share

Following the total nuclear power demand of each NG in the nominal case, the maximum FR introduction in NG1 and BAU+ in NG2 and NG3 are calculated. The FR power share in NG1 varies depending on the FR breeding performance. As the maximum FR power share in NG1 can be estimated based on the results of the homogeneous model, the power shares are near those of the homogeneous global model. In Figs 8.61–8.63, the reactor power shares are shown for three FR cases — globally and in groups (NG1, NG2, NG3).

The power share of FRs in NG1 is larger than that of the homogeneous global calculation, because the share of the NG1 group is 40% of the global demand while the historical LWR capacity is 50%. The FR power shares in NG1 in 2100 are 46% in the high case for the break-even FR (F1), 64% for the breeder FR (F2) and 58% for the high burnup breeder FR (F3). Naturally, the global FR power share is easy to estimate by multiplying the power share of NG1 by 40%; that is, the maximum global FR power shares in 2100 stay only at 18% in the high case for F1, 26% for F2 and 23% for F3.

8.7.3.2.2. Reprocessing load and plutonium balance in the system

The previously mentioned maximum FR power shares of each FR in NG1 are a result of the breeding performances of each FR, and it was assumed that there is no capacity limitation for the reprocessing of LWR, ALWR and FR SF. However, it will actually be the key factor in the feasibility of FR introduction in NESs.

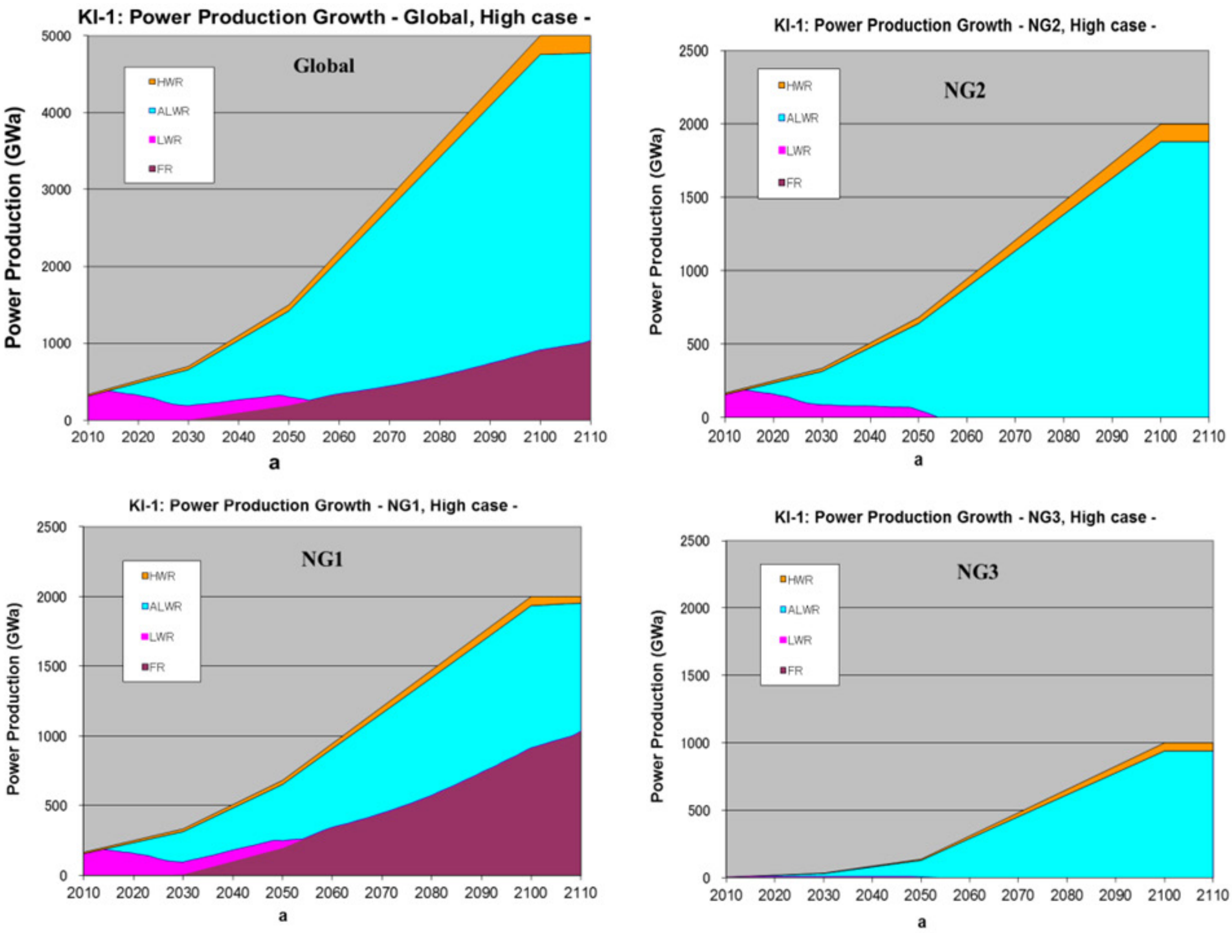


FIG. 8.61. Reactor power share of each NG at maximum FR introduction into NG1 (F1, separate model, high case).

Figures 8.64–8.66 show the reprocessing load and plutonium balance in NG1 of the three FRs of the F1, F2 and F3 introduction cases, respectively. The reprocessing load for LWR and ALWR SF is almost the same level as in the three FR introduction cases, while the FR power shares in 2100 vary from 46 to 64% in F1 and F2. This means that F1 needs more support in plutonium supply from LWRs and ALWRs for the FR introduction because of the lack of breeding. Concerning the reprocessing load of SF from FRs, the F2 case needs around 35 ktHM in 2100, while F1 and F3 need around 20 ktHM in 2100. As the FR capacities of the F2 and F3 cases are at the same level, the effect is due to the lower average fuel burnup of F2. Regarding the amount of plutonium treated for the FR, the F2 case has to manage around 34 ktHM of Pu, while for F3 it is around 18 ktHM. The large difference is due to differences in fuel requirements and Pu content in the fuel.

8.7.3.2.3. Natural uranium saving

Figures 8.64–8.66 also show the cumulative natural uranium demand from 2008. As shown in the figures, the F1 case results in 30.7 Mt, the F2 case 29.5 Mt and the F3 case 29.9 Mt. As the BAU+ case gives 36.1 Mt (Fig. 8.30 includes the contribution from 1970 to 2008), the natural uranium saving in 2100 comes to 5.4 Mt for the F1 case, 6.6 Mt for the F2 case and 6.2 Mt for the F3 case. Thus, the natural uranium saving does not reach as high a level as in the separate model case because of the small FR power share globally.

8.7.3.3. Synergistic model case

As shown in the previous section, as long as no mass flow is assumed between NGs, FR introduction in NG1 is restrained by the limitation of available plutonium even in the case of breeders which have a BR of 1.2. This

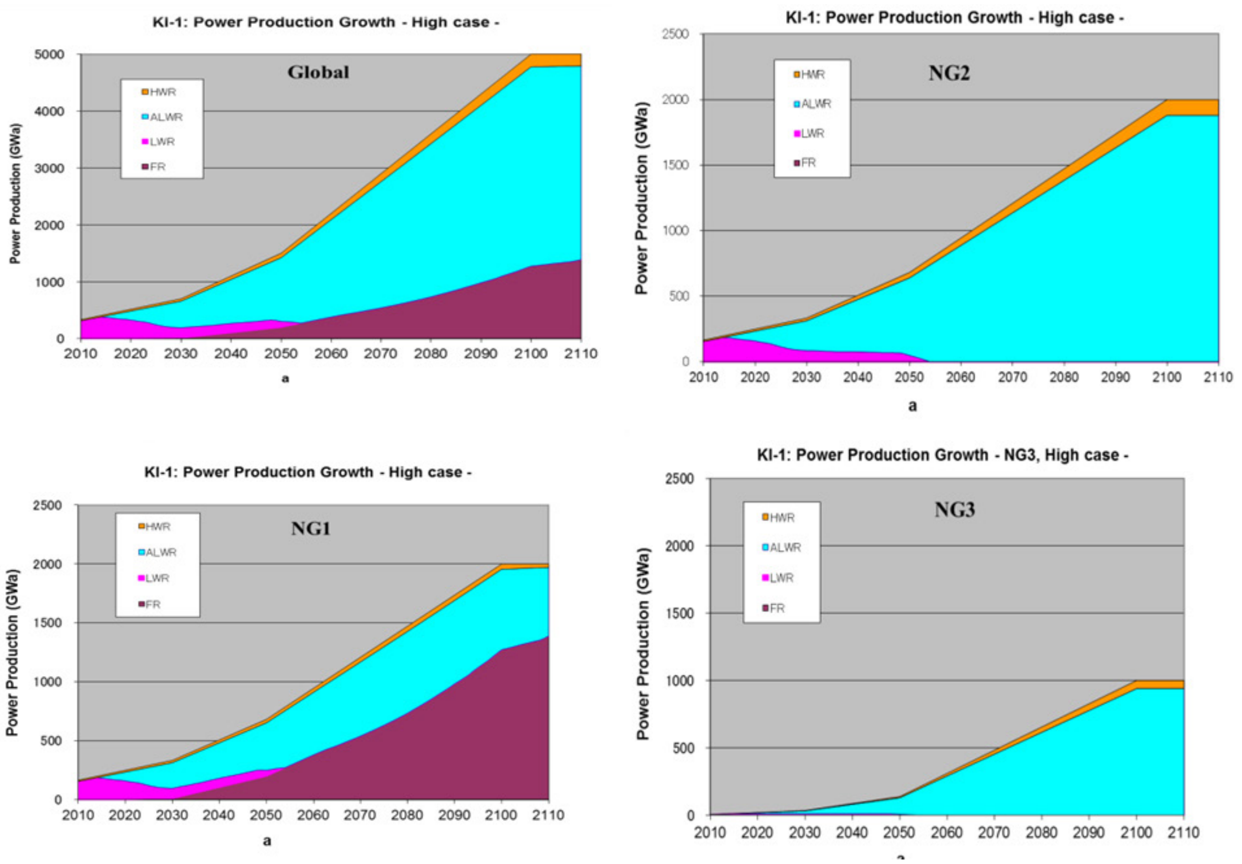


FIG. 8.62. Reactor power share of each NG at maximum FR introduction into NG1 (F2, separate model, high case).

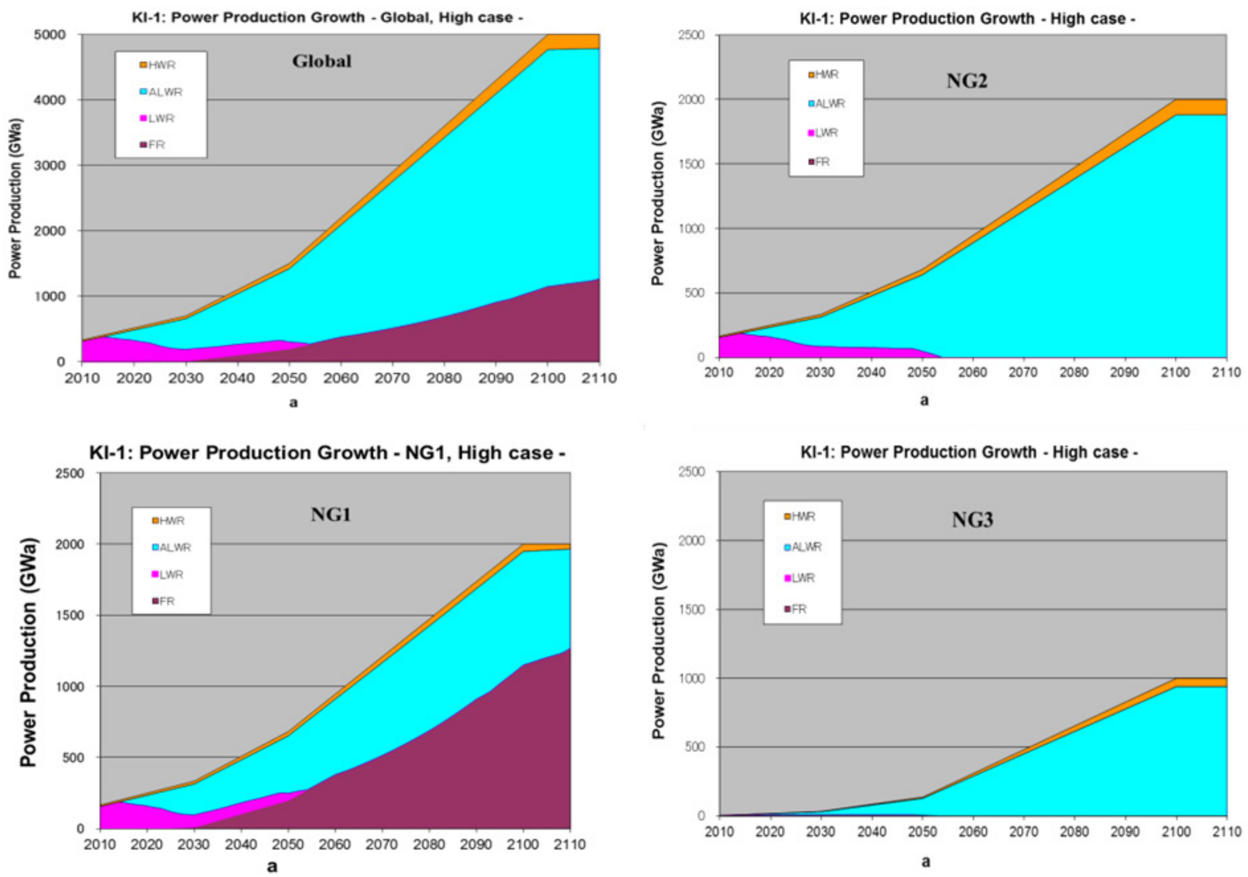


FIG. 8.63. Reactor power share of each NG at maximum FR introduction into NG1 (F3, separate model, high case).

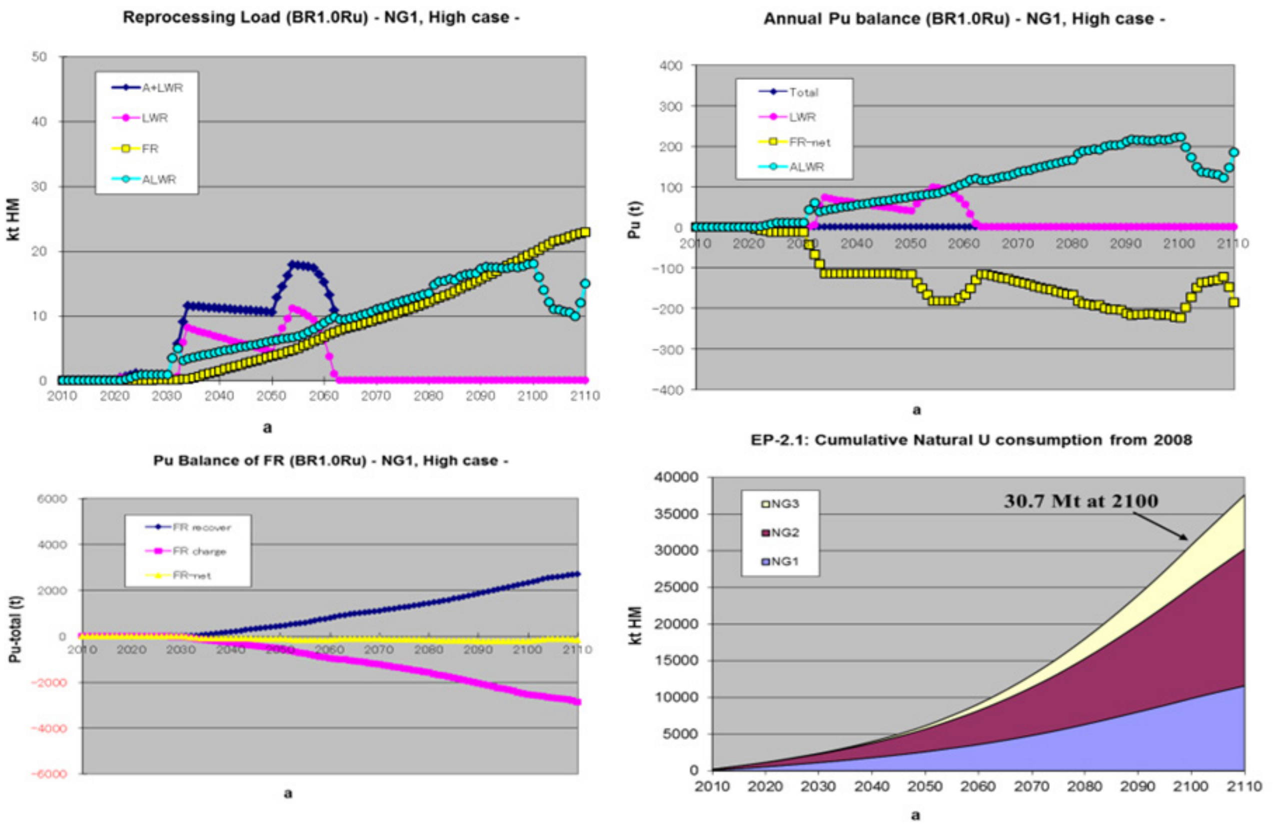


FIG. 8.64. *Reprocessing load and plutonium balance in NG1, global cumulative natural uranium demand in global (F1, separate model, high case).*

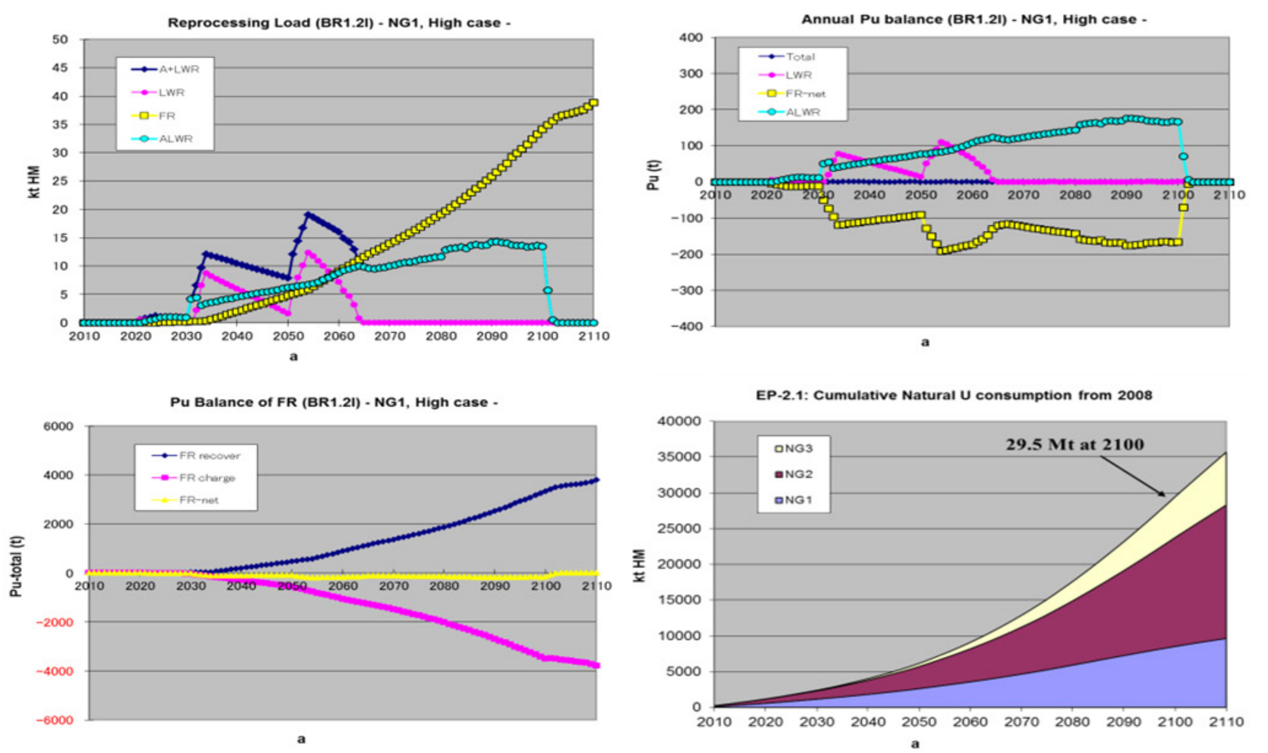


FIG. 8.65. *Reprocessing load and plutonium balance in NG1, global cumulative natural uranium demand (F2, separate model, high case).*

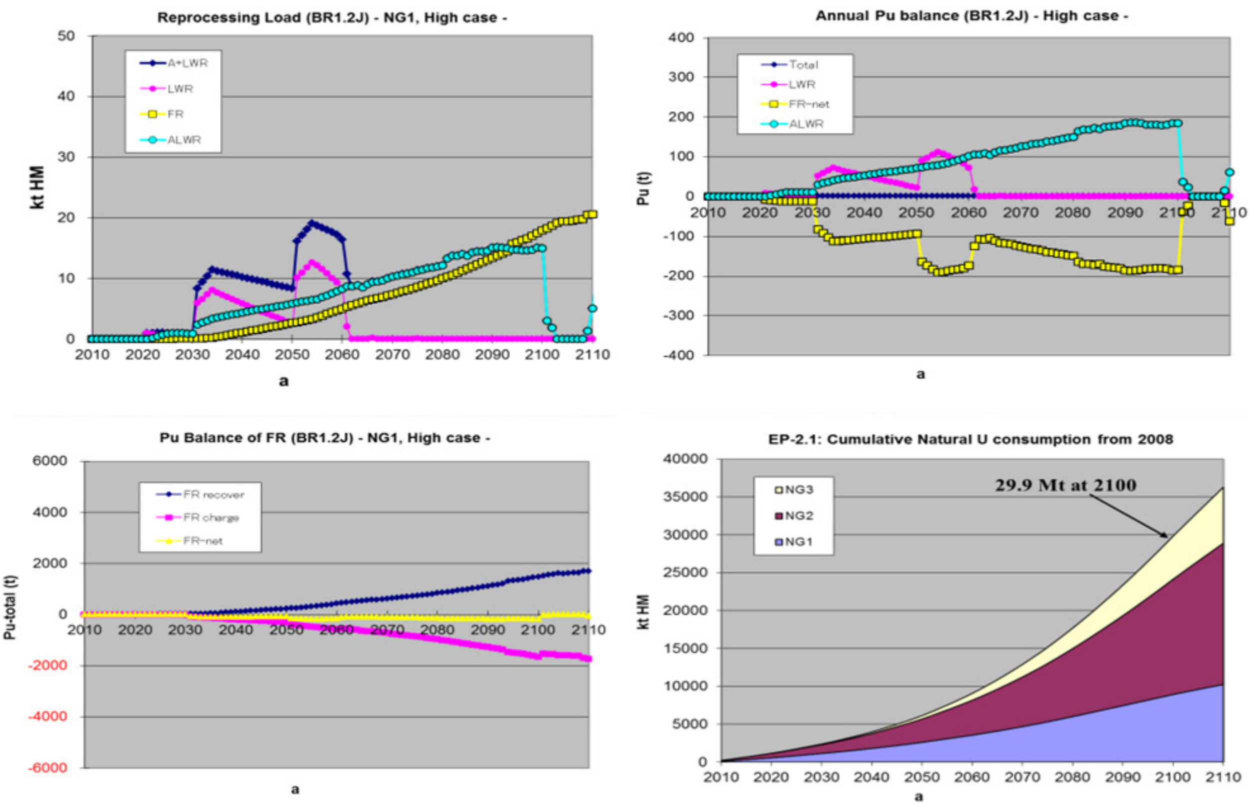


FIG. 8.66. Reprocessing load and plutonium balance in NG1, global cumulative natural uranium demand (F3, separate model, high case).

situation comes from the fact that much plutonium for initial core loading is necessary for the large capacity of new FR nuclear power plants under the assumption of a high nuclear power growth rate, and the plutonium debt for initial core loading will not be compensated within a period of over 30 years corresponding to the doubling time of FRs.

However, once the free transport of SF or reprocessed plutonium into NG1 from NG2 or NG3 is assumed, the problem of a lack of plutonium will be solved and many more FRs can be installed in NG1. Therefore, such synergistic situations are analysed in this section.

8.7.3.3.1. Maximum fast reactor power share

Figure 8.67 shows the maximum FR power share of the high case in the synergistic NG model case. The figure is common for F1, F2 and F3, because almost 100% of FR power share in NG1 in 2100 will be able to be achieved with no limitation of ALWR SF transport from NG2 or NG3. The reason for the FR power share to be a little less than 100% comes from the ALWR plant lifetime assumption of 60 years. (In the analysis, ALWRs cannot be removed before the end of their lifetime, even if there is enough plutonium for new FR installation.) Thus, almost 40% of FR power share can be achieved globally not depending on the breeding or burnup performances of FRs, although mass flows would be different depending on breeding and burnup.

8.7.3.3.2. Reprocessing load and plutonium balance in the system

As previously mentioned, the maximum FR power shares in NG1 for the three FR cases are the same and almost 100% at the end of the century. However, in order to achieve such a high FR introduction rate in NG1, ALWR SF has to be transported from NG2 or NG3 into NG1 and be reprocessed and the fuel for FRs fabricated. Figures 8.68–8.70 show the reprocessing load and plutonium balance in NG1 of the three F1, F2 and F3 introduction cases, respectively. As shown in the figures, the reprocessing load and the plutonium balance become feasible assuming huge amounts of SF are transported from NG2 and NG3, and reprocessing for NG1.

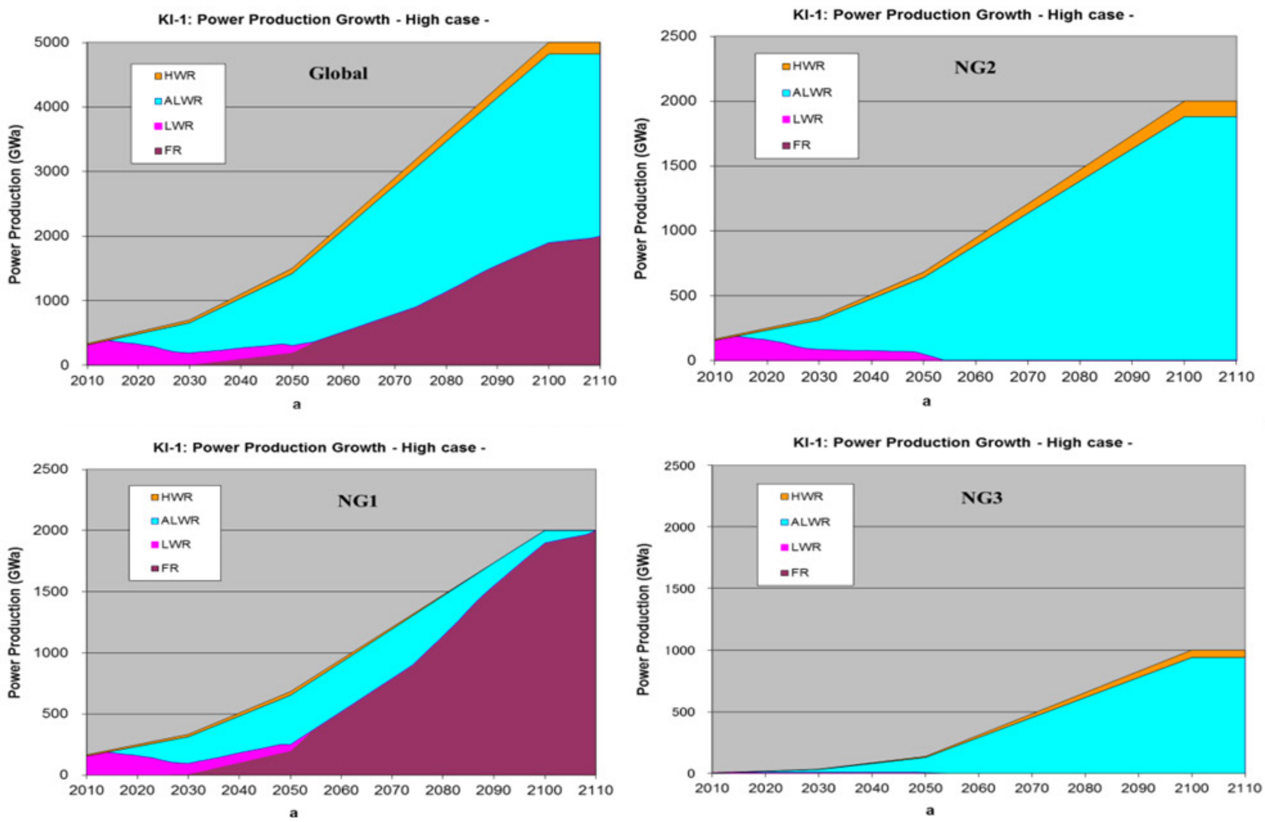


FIG. 8.67. Reactor power share of each NG at maximum FR introduction into NG1 (F1, F2 and F3, synergistic model, high case).

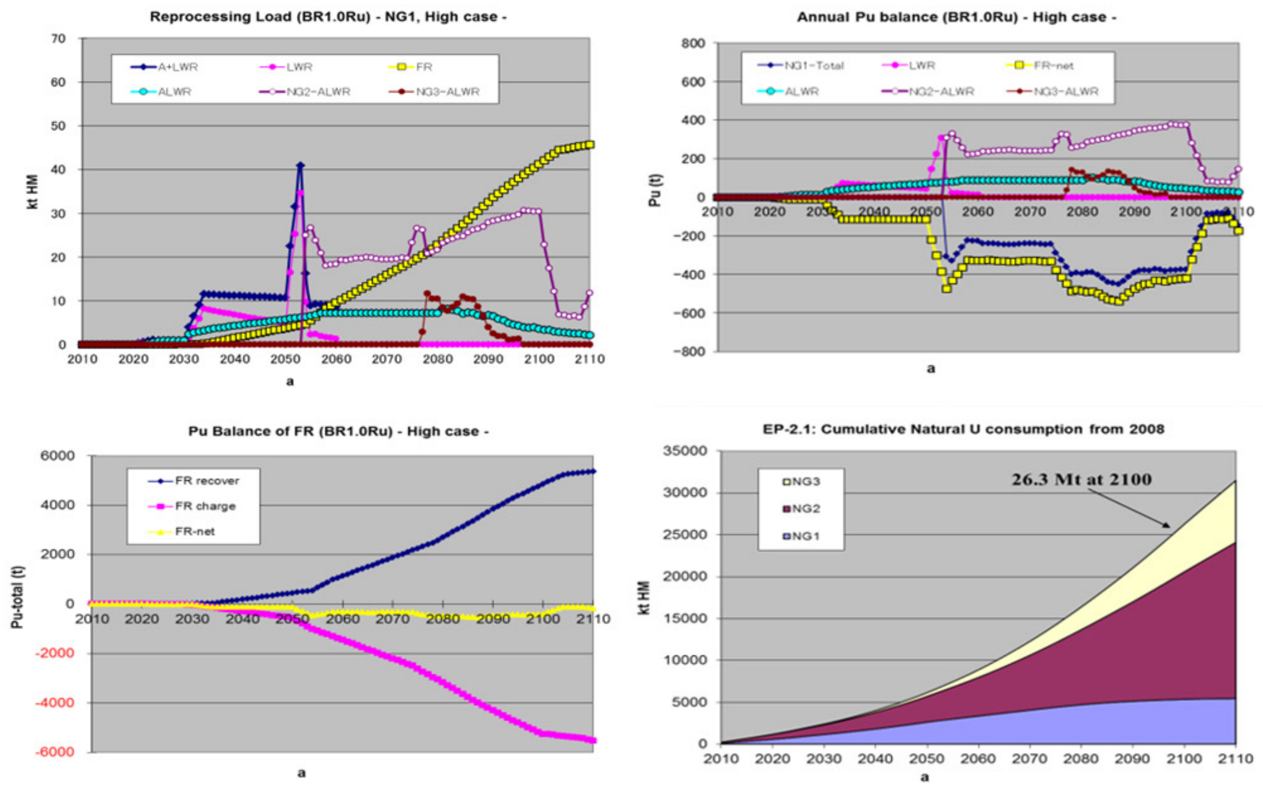


FIG. 8.68. Reprocessing load and plutonium balance in NG1, global cumulative natural uranium demand (F1, synergistic model, high case).

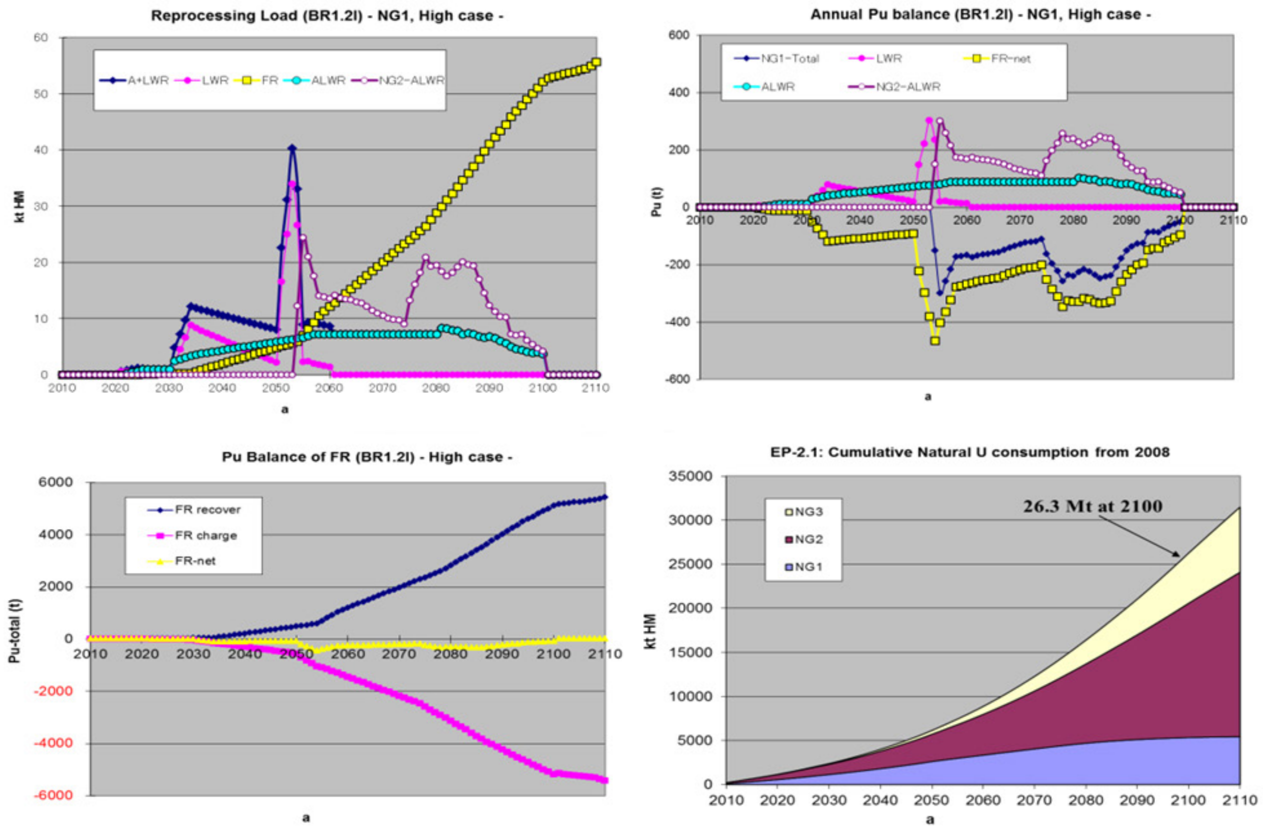


FIG. 8.69. Reprocessing load and plutonium balance in NG1, global cumulative natural uranium demand (F2, synergistic model, high case).

The amount of SF transport and the resulting SF reprocessing load vary depending on the performance of the FRs.

The reprocessing load of LWR and ALWR SF originated in NG1 is at almost the same level in the three FR cases. However, as shown in Fig. 8.68 in the F1 case, a much larger amount of ALWR SF from NG2 and NG3 has to be transported and reprocessed. The peak value of the amount is around 40 ktHM annually from 2075 to 2090 because of there being no breeding of F1.

The situation is mitigated in the cases of F2 and F3 as shown in Figs 8.69 and 8.70, respectively. The ALWR SF transport from NG3 is not necessary in the two FR cases because of the BR of 1.2 of the two FRs and the amount of SF transport from NG2, and the reprocessing load becomes around 20 ktHM annually from 2075 to 2090 as the peak value.

The amount of transport and reprocessing load for SF from NG2 or NG3 are linked to the breeding performance of the FR deployed in NG1. Thus, the two FRs of F2 and F3 give the same result because of the same level of the BR. On the other hand, the reprocessing load of FR SF varies depending on the average discharged burnup of FRs. As shown in Figs 8.64–8.66, F1 and F2 need around 45 and 55 ktHM of reprocessing load, respectively, for FR SF and to manage around 5000 t of plutonium annually in 2100, while F3 needs around 30 ktHM of reprocessing load of FR SF and to manage around 2000 t of plutonium annually in 2100.

8.7.3.3.3. Natural uranium saving

As the natural uranium saving entirely depends on the FR share in global nuclear power, the saving is common in the three FR cases and amounts to around 10 Mt in 2100, corresponding to an FR power share of 40% in 2100.

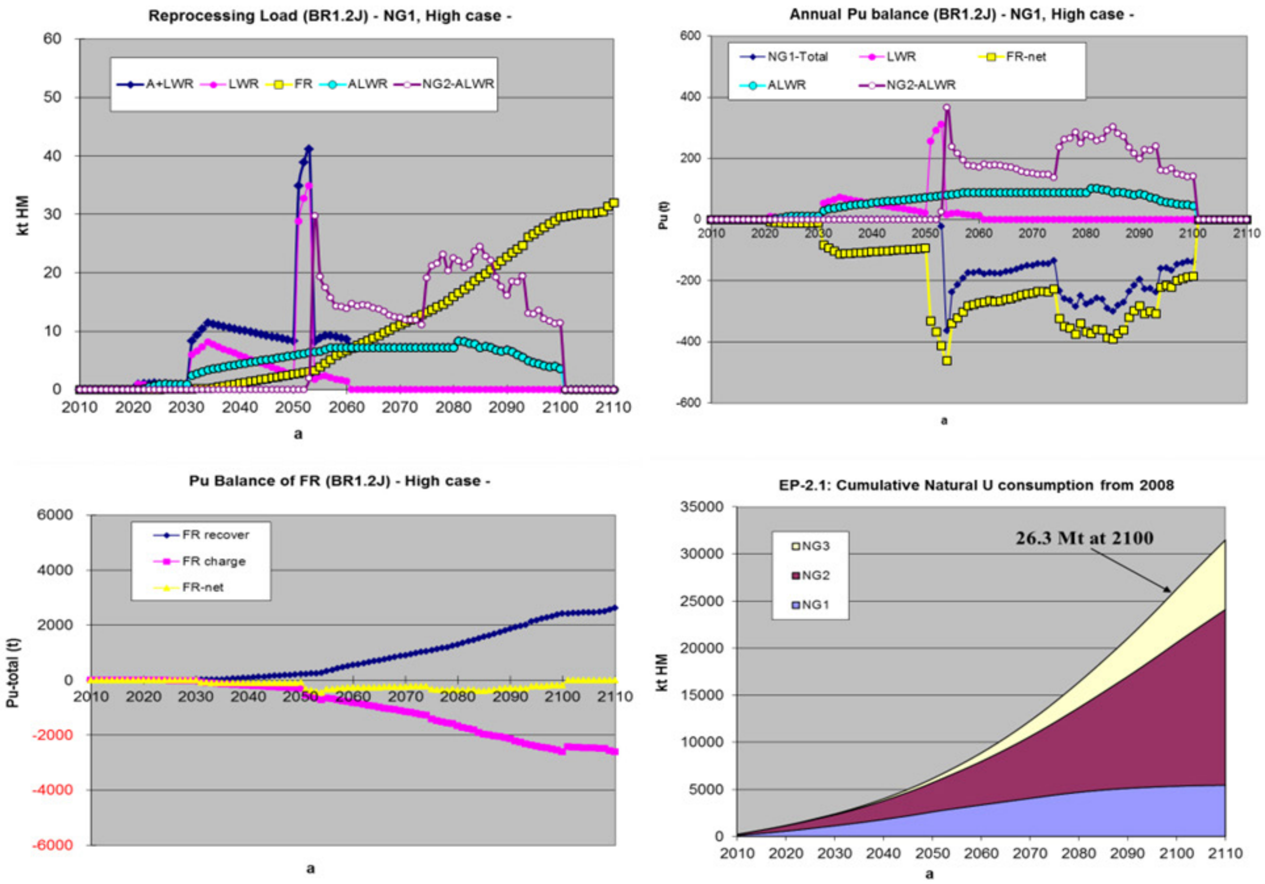


FIG. 8.70. Reprocessing load and plutonium balance in NG1, global cumulative natural uranium demand (F3, synergistic model, high case).

8.7.4. Remarks

With the BAU+ scenario, the three FRs are compared on the basis of the resulting mass flows for maximum FR introduction cases derived from their breeding or burnup performances. A summary comparison is shown in Table 8.2.

The following issues were clarified through the comparison:

— The homogeneous global model gives the ideal maximum FR introduction rates, where the FR power share in 2100 reaches 44, 59 and 54% in the high case, and 53, 73 and 67% in the moderate case for F1, F2 and F3, respectively, depending on the breeding performances. As a result of FR introduction, 12–15 Mt of natural uranium saving in the high case and 7–9 Mt in the moderate case could be achieved under such ideal conditions.

— As the heterogeneous world model may provide a more realistic representation, the model was applied for comparison purposes. In the separate (non-synergistic) case of the NG model, the FR power share is a small percentage due to limited plutonium inside the recycle group (NG1), and the natural uranium saving is only 6 Mt in the high case and 4 Mt in the moderate case.

— Assuming the synergistic situation between NG groups, almost 100% of the FR power share in 2100 in the recycle group (NG1) is possible, and 10 Mt in the high case and 6 Mt in the moderate case of natural uranium can be saved. However, in order to make the synergistic situation realistic, lessening the reprocessing load for both LWRs and FRs will become the key issue. Thus, both high breeding and high burnup FRs will be strongly desired for future commercial FRs from a global perspective for natural uranium saving.

TABLE 8.2. COMPARISON OF GLOBAL MASS FLOWS OF THREE FRs AND GLOBAL MODELS

		Homogeneous model				NG model Separate case				NG model Synergistic case			
		Global FR share at 2100 (%)	Natural U demand 2008 ~ 2100 (Mt)	Reprocessing Peak Load 2008 ~ 2100 (kt HM/a)		Global FR share at 2100 (%)	Natural U demand 2008 ~ 2100 (Mt)	Reprocessing Peak Load 2008 ~ 2100 (kt HM/a)		Global FR share at 2100 (%)	Natural U demand 2008 ~ 2100 (Mt)	Reprocessing Peak Load 2008 ~ 2100 (kt HM/a)	
				FR	A+LWR			FR	A+LWR (NG1)			FR	A+LWR (NG1) (NG2+3)
BAU+	High	-	**36.1**	-	-	-	**36.1**	-	-	-	**36.1**	-	-
	Moderate	-	**20.9**	-	-	-	**20.9**	-	-	-	**20.9**	-	-
F1 Breakeven FR	High	**44**	24.1	47	**47**	**18**	30.8	20	18	**38**	26.3	41	41 33
	Moderate	53	13.4	29	28	21	17.2	12	12	**40**	14.9	22	21 22
F2 Breeder FR	High	59	21.2	**79**	37	26	29.4	34	19	**38**	26.3	**52**	41 24
	Moderate	**73**	11.3	50	35	**30**	16.2	21	16	**40**	14.9	28	22 15
F3 High burnup beeder FR	High	54	22.1	**42**	40	23	29.9	18	19	**38**	26.3	**30**	41 30
	Moderate	67	11.8	26	33	27	16.5	12	16	**40**	14.9	16	21 16

8.8. SEPARATION PROCESSING RATE

8.8.1. Background

The scenarios studied in GAINS have been calculated by assuming that there is no limitation in the fuel cycle infrastructure. Several types of fuel cycle infrastructure should be involved in the fuel cycle studies, such as:

— Mining;
— Conversion plant;
— Enrichment plant;
— Fabrication plant;
— Interim storage for SF;
— Interim storage for nuclear material (for instance plutonium, high and intermediate level waste);
— Reprocessing plant;
— Transport;
— Geological disposal.

All of these infrastructures have some limited capacity. For instance, in the case of an enrichment plant, the annual enrichment capacity is defined and limited by the size of the plant and the size of the equipment inside the plant. These limitations can have an impact on the possibility to deploy nuclear energy production. For instance, in the case of the FR front end fuel cycle shown in Fig. 8.71, the fresh fuel arriving in the reactor is the result of a chain of fabrication involving several plant storages and transport, and the capacity of the global chain is limited by the element of the chain having the minimum capacity.

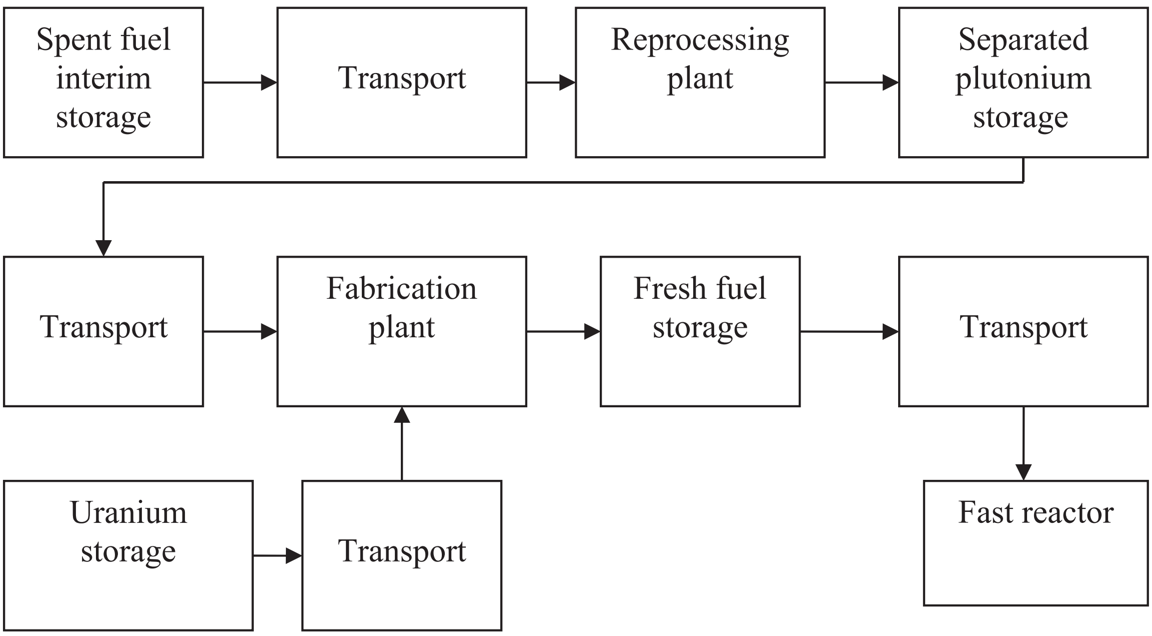


FIG. 8.71. Fast reactor front end fuel cycle.

Among all the possible limitations, the reprocessing capacity seems the most interesting to study because reprocessing plants can be considered as the most complex and expensive facilities, depending on the unit size and particular attributes.

Thus, in the case of break-even FR (F1) introduction, high case, and the homogeneous model previously mentioned in Section 8.5.2.3, we have tried to estimate at which level the reprocessing capacity could be a limitation for FR introduction.

8.8.2. Fuel cycle code

The scenario code used for this estimation is the COSI 6 code [8.4]. COSI 6 has the possibility to simulate different types of reprocessing assumptions.

8.8.2.1. Reprocessing on demand

In this case, there is no limitation in the reprocessing capacity. Thus, annual reprocessing depends on plutonium demand without limitation except for SF availability.

8.8.2.2. Industrial reprocessing

In this case, the reprocessing capacity is fixed by the user, adding a second limitation to the previous case.

If a lack of plutonium for fabrication appears, COSI can take the plutonium from a fictive infinite stock of plutonium. The cumulative amount of plutonium taken from this stock represents the total lack of plutonium for FR deployment.

In order to compare the possible reprocessing demand in the FR introduction scenario and constrained industrial reprocessing plant capacity, the COSI code used the same FR introduction curve as the maximum FR introduction scenario analysed by the NFCSS code mentioned in Section 10.2.3 (break-even FR, high case). Then, some reprocessing limitations were introduced in the calculations to study the effect of these limitations on FR deployment.

8.8.3. Reprocessing on demand and industrial reprocessing

8.8.3.1. Scenario assumptions

The reprocessing plants have a fixed capacity defined by the user. Thus, the annual plutonium production depends both on SF availability and reprocessing plant capacity.

One additional reprocessing plant is installed each year from 2020 with a capacity of 850 tHM/a, which corresponds to the capacity of the UP2 or UP3 plants at La Hague.

Each plant is replaced after 40 years lifetime. Thus, after 2060, two plants are commissioned each year.

Three cases are considered with three different final reprocessing capacities:

(a) 18 times La Hague or 30 600 tHM of SF per year;
(b) 24 times La Hague or 40 800 tHM of SF per year;
(c) 45.5 times La Hague or 77 350 tHM of SF per year.

8.8.3.2. Results

Figure 8.72 presents the plutonium lack of the three ‘industrial reprocessing’ cases in comparison with the ‘reprocessing on demand’ case. The light blue line shows the reprocessing on demand case. As shown in the figure, a lack of plutonium occurs even in the reprocessing on demand case. Although this plutonium lack comes from the difference in plutonium balance calculation between COSI and NFCSS, the difference is below 5% of the total plutonium handled in the fuel cycle system.

As shown in Fig. 8.73, in the cases of industrial reprocessing, the plutonium lack is much higher than in the case of reprocessing on demand. When the reprocessing capacity has a limited value, it becomes the limiting factor for FR deployment. Moreover, the lack occurs very early in the simulation, indicating that important reprocessing capacities are necessary as soon as the FR deployment starts.

Figure 8.74 presents the SF inventory for LWRs + ALWRs and FRs. It represents the sum of the SF in nuclear power plant storage, reprocessing plants storage and transport. The lack of reprocessing capacity leads to an increase of non-reprocessed SF, compared to the scenario ‘reprocessing on demand’.

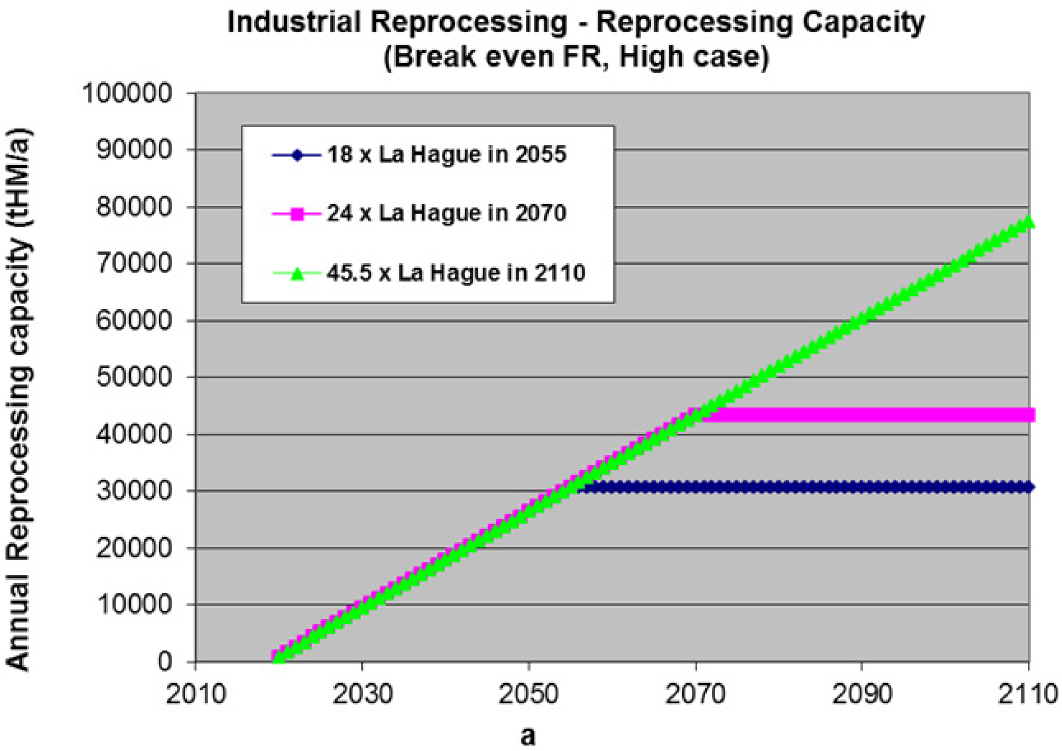


FIG. 8.72. Industrial reprocessing — reprocessing capacity.

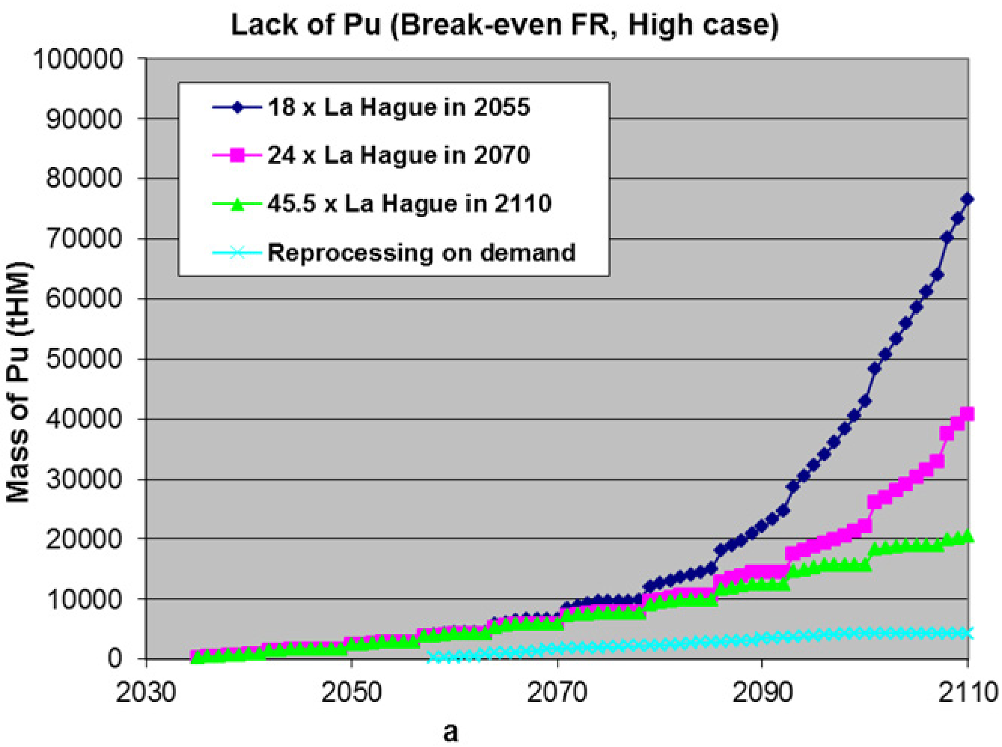


FIG. 8.73. Plutonium lack in the 'industrial reprocessing case' (break-even FR, high case).

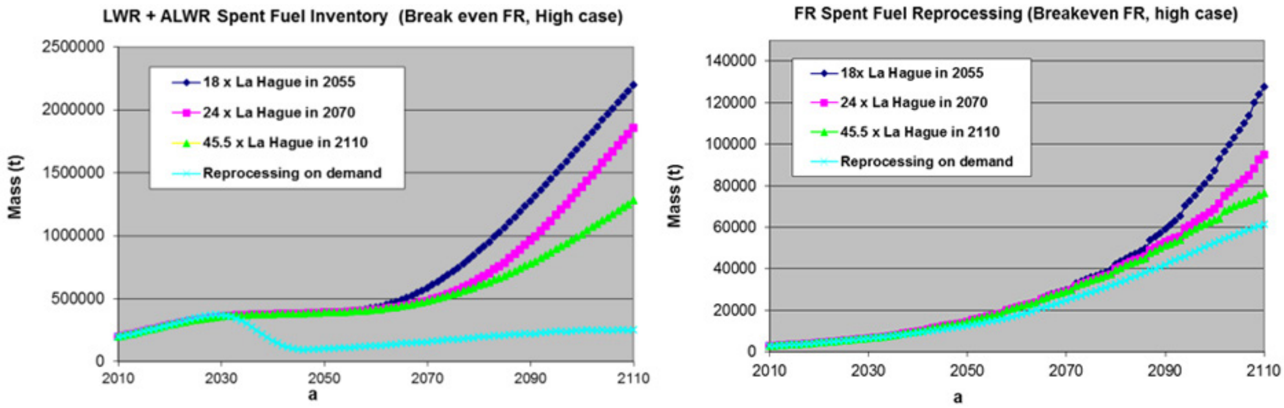


FIG. 8.74. Spent fuel inventories in the 'industrial reprocessing case' (break-even FR, high case).

8.9. SUMMARY

The sensitivities of the tails assay of the uranium enrichment plant, the uranium enrichment model of LWR initial cores, the average discharge LWR burnup and the cooling time of SF on mass flow analysis results were investigated in the early part of this section. The effects of the breeding and burnup performances of FRs on the mass flow analysis results during the transition period from LWRs to FRs were studied in the latter part of the section.

The results are summarized as follows:

— The natural uranium consumption rate varies by 20–30% according to the uranium enrichment facility tails assay (0.2–0.3%) and the treatment of LWR initial core enrichment. At the same time, the required separative work (SWU) varies inversely with the uranium consumption rate.
— The effects of a change in LWR discharge burnup from 45 to 100 GW·d/t on the natural uranium consumption rate and the required separative work are small. The advantages of the higher LWR discharge burnup are the reduction in the fuel fabrication load and the amount of discharged SF. The optimum point of the discharge burnup seems to be at 60 GW·d/t.
— The effect of cooling time of LWR and FR SF is very large. It operates the same way as large breeding performance changes.
— The study on the effect from a change in CR of the break-even FR (F1) and burner FRs shows that the FR power share in 2100 decreases from 46 to 16%, and that the cumulative natural uranium consumption rises from 34 to 45 Mt, according to the change of TRU CR from 1.0 to 0.0.
— The study on the break-even FR (F1) and medium breeding FRs (F2, F3) shows that the average discharge burnup of FR SF is more important than the BR of the FRs, because the FR introduction speed is actually limited by the reprocessing capacity for FR SF.
— To maximize FR power share and to save natural uranium resources in the world, the synergy between the 'recycle group' and the 'once-through group' becomes important through the smooth transport of LWR SF or reprocessed plutonium according to the plutonium availability of the recycle group.
— As the reprocessing capacity will limit the FR introduction speed, a realistic assumption on the reprocessing facility planning is important to make a feasible long term global transition scenario from LWRs to FRs.

REFERENCES TO SECTION 8

[8.1] HOFFMAN, E.A., et al., Preliminary Core Design Studies for the Advanced Burner Reactor over a Wide Range of Conversion Ratios, Rep. ANL-AFCI-177, Argonne Natl Lab., IL (2006).

[8.2] HOFFMAN, E.A., Updated Design Studies for the Advanced Burner Reactor over a Wide Range of Conversion Ratios, Rep. ANL-AFCI-189, Argonne Natl Lab., IL (2007).

[8.3] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, NUCLEAR ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Uranium 2009: Resources, Production and Demand, OECD, Paris (2010).

[8.4] MEYER, R., et al., "New developments on COSI 6, the simulation software for fuel cycle analysis", SAS Global Forum (Proc. Washington, DC, 2009) (2009).

9. ADDITIONAL ILLUSTRATIVE CASES

9.1. PURPOSE

When simulating the development of a global NES within the time frame of a century, components with different levels of maturity should be included in the model. Meanwhile, the degree of credibility of the results of the scenario modelling and assessment of an NES against criteria of sustainability depends on the validity of the input data; those data are higher for existing and near term systems than for more advanced components. This section provides examples of the use of the GAINS framework to analyse the introduction of several different advanced technologies which could be implemented in the coming century. These technologies are in the stages of conceptual development or feasibility demonstration. Thus, the level of uncertainty of the input data necessary for assessing their compatibility with the objective of sustainable energy is higher than that in the base cases and sensitivity studies described in Sections 7 and 8. Sections 9.2–9.6 deal with the analysis of the material flows in the systems dedicated to transmutation of MAs and in thorium breeders. Sections 9.7 and 9.8 address some economic issues connected with the deployment of a global NES.

9.2. TYPES OF SCENARIO ANALYSED

9.2.1. Dedicated minor actinide burners

Concerns about environmental preservation have increased the demand for more efficient management, and environmentally sound and sustainable development of nuclear energy. The appropriate management of radioactive waste arising at the back end of the fuel cycle is considered to be a crucial issue of long term environmental concern in relation to the NFC. A typical 1 GW(e) LWR generates about 20–30 t of SF per year. The basis of environmental concern lies in whether the harmful components of this waste can be isolated from the biosphere for at least tens of thousands of years, or even longer.

Most of the SF mass (generally more than 98.5%) from commercial LWRs is composed of uranium, and short lived FPs, which do not pose a long term radiological hazard. Of the remainder, about 0.4 wt% of the SNF is in the form of long lived FPs, such as isotopes of caesium, strontium, technetium and iodine, whose disposition path is a matter of some debate at the moment. Approximately 1% of the SNF is in the form of plutonium and MAs (together known as transuranics), many of which are long lived and create issues of radiotoxicity, heat load and proliferation. As of the year 2006, it is estimated that about 110 t of MAs are contained in SF storage worldwide, and an additional 40 t are contained in high level waste products from reprocessing. In the absence of a plan for their transmutation, the amount of MAs will double by 2020.

Currently, in some jurisdictions, both plutonium and MAs are destined for underground storage or permanent disposal in repositories. However, the idea of re-using them in advanced fuel cycles with advanced reactor technologies is attractive because it could minimize the volume of radioactive waste destined for repositories. As a bonus, advanced technologies have the possibility of increasing the operating safety of NESs, enhancing the proliferation resistance of the NFC, increasing the efficiency of natural resource utilization, and potentially increasing the economic competitiveness of nuclear power plants.

The estimated MA accumulation (assuming no specific plan for MA transmutation) for the BAU+ with FR case when the FR is the fast reactor break-even (BR: ~1.0) type is shown in Fig. 9.1 (moderate demand) and Fig. 9.2 (high demand). The estimated MA accumulation for the FR breeder (BR: ~1.2) and for the high burnup FR breeder is shown in Figs 9.3–9.6. The fuel for the high burnup FR breeder contains MAs and, hence, this reactor contributes to MA burning.

Section 9.3 discusses the implementation of an ADS, which comprises a subcritical fissionable assembly driven by a spallation neutron source. Section 9.4 instead considers the introduction of MSRs. These are high temperature reactors cooled by a molten salt and in which the fuel can either be conventional rods, or be in suspension in the coolant. Both types of facility have a fast spectrum (in the fuel) which can achieve very high burnup of MAs.

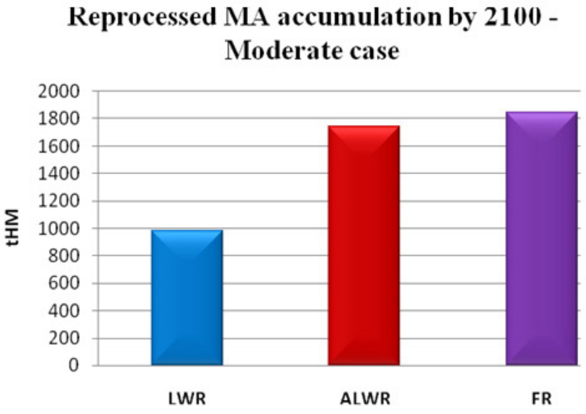


FIG. 9.1. Reprocessed MAs for the BAU+ with 'break-even FR' scenario, moderate case.

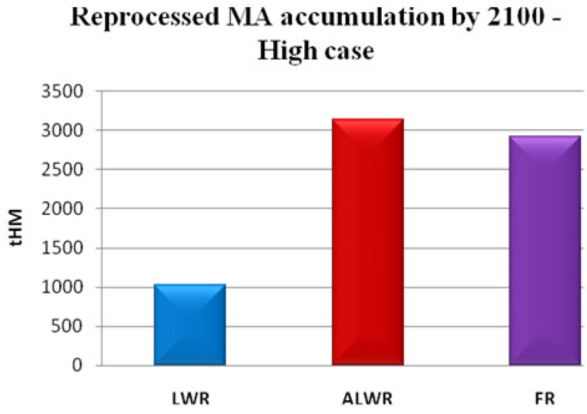


FIG. 9.2. Reprocessed MAs for the BAU+ with 'break-even FR' scenario, high case.

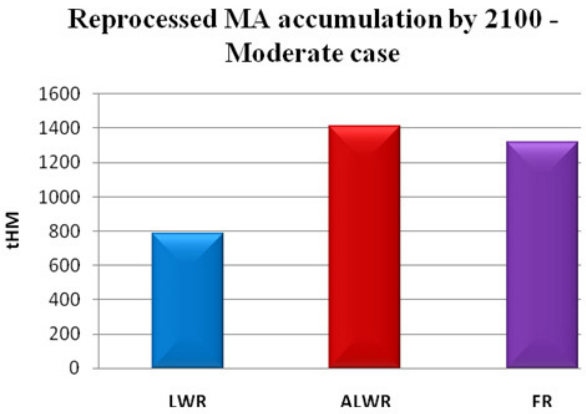


FIG. 9.3. Reprocessed MAs for the BAU+ with 'breeder FR' scenario, moderate case.

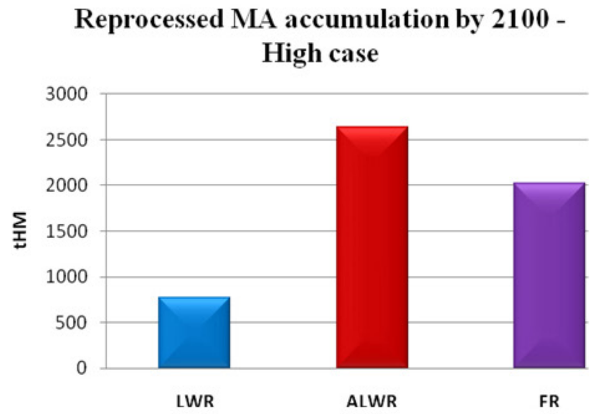


FIG. 9.4. Reprocessed MAs for the BAU+ with 'breeder FR' scenario, high case.

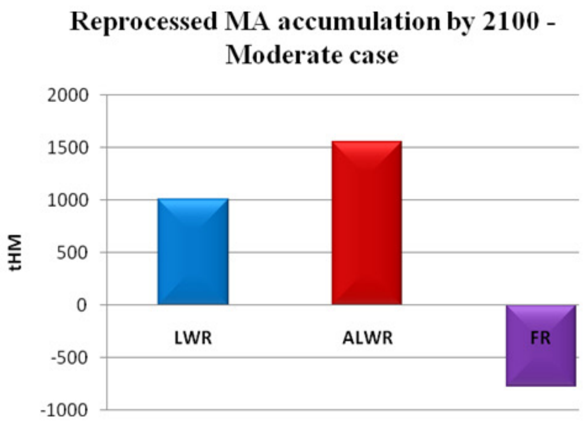


FIG. 9.5. Reprocessed MAs for the BAU+ with 'high burnup breeder FR' scenario, moderate case.

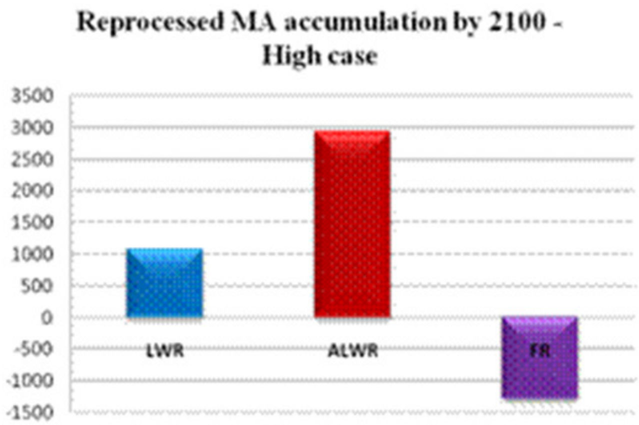


FIG. 9.6. Reprocessed MAs for the BAU+ with 'high burnup breeder FR' scenario, high case.

9.2.2. Thorium breeders with near break-even breeding ratio (BR: ~1)

Another interesting technology is the implementation of the thorium fuel cycle in either a thermal or fast spectrum heavy water cooled reactor. Current known reserves of easily extractible natural uranium are ~5.5 × 10^6 t.[6] Undiscovered resources are estimated to be ~10.5 × 10^6 t, while the somewhat more expensive, unconventional resources (e.g. phosphates) may be ~22 × 10^6 t.[7] All resources, known, undiscovered and unconventional, can be seen to sum to only 38 × 10^6 t. The simple BAU scenario (high case), with once-through uranium fuel cycles, requires ~46.4 × 10^6 t of natural uranium by 2130 (see Fig. 9.7, produced using DESAE 2.2) and, therefore, exceeds current estimates of available uranium in the world (excluding that available in sea water). One solution to this resource issue is the adoption of FBRs, and a calculation for break-even FR (BR: ~1.0) introduction is shown in Fig. 9.7, indicating that savings of around one third of natural uranium resources are available in this scenario.

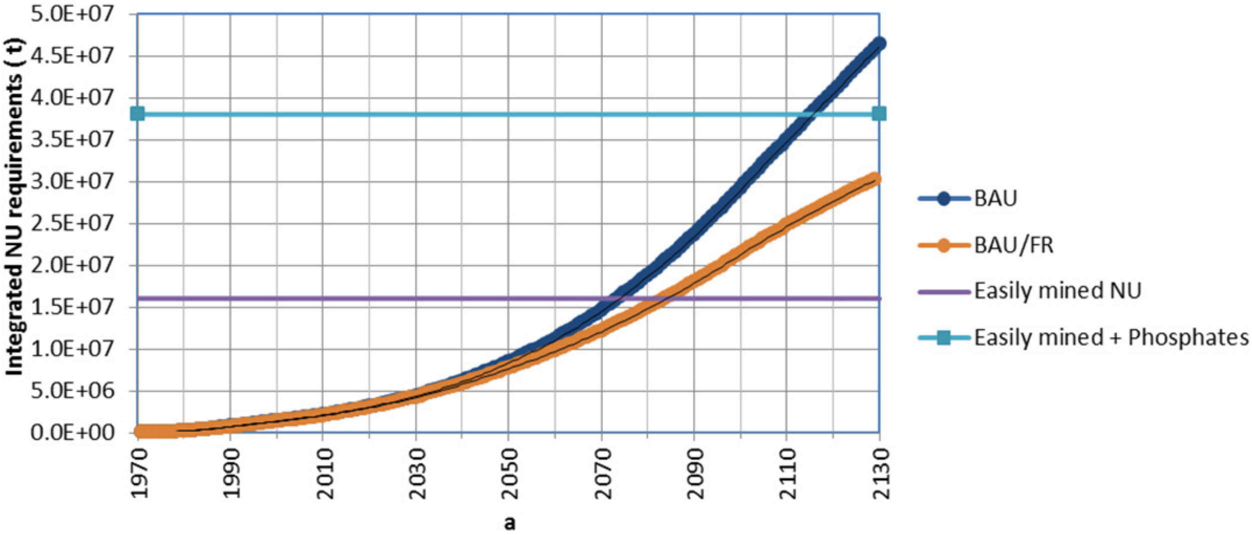


FIG. 9.7. LWR and HWR contributions to BAU total uranium requirements.

The fissile material, for thorium breeders, as envisioned here, is a combination of plutonium from spent LWR and ^{233}U from thorium breeding. One of the advantages of all of the recycling scenarios is the reduction of some of the actinides which may cause significant difficulties for repositories by virtue of their contribution to the heat load. Recycling scenarios generally include all plutonium isotopes, which are not separated, and possibly some of the MA elements — americium in the case of the high burnup breeder FR and all of them in the ADS and MSR scenarios. If it is assumed that unburnt MAs are placed in geological repositories, then, since heat migrates slowly through rock, the most important parameter is probably the total decay energy of the fuel over a span of one or two thousand years. This parameter was approximated by making the metric for heat load the SNF decay power after 1000 years of internment.

In the BAU scenario (high case), uranium can be used more efficiently by reducing the tails enrichment, or by switching LWRs to HWRs. In the first case, the feed-to-product ratio

$$\frac{F}{P}=\frac{x_p-x_t}{x_f-x_t}$$

for x_p (LWR fuel enrichment) = 4%, x_f (natural uranium weight fraction of ^{235}U) = 0.711% can be improved from 9.00 (for a tails enrichment, x_t = 0.3 wt%) to 7.44 (for x_t = 0.2 wt%), a saving of 15.7% in natural uranium

[6] From the OECD–IAEA Red Book [9.1].

[7] This estimate is from the 2005 Red Book.

requirements[8] (7.3×10^6 t). In the second case, producing the entire electrical demand with (CANDU) HWRs would require 34.8×10^6 t of natural uranium, a saving of 25% (11.6 t). These savings would mitigate the problem.

An alternative to the adoption of break-even FRs (BR: ~1.0) would be to switch to thorium breeder reactors (BR: ~1.0), which do not require natural uranium. In a thorium breeder reactor using ^{233}U, fuel utilization is improved by having surplus neutrons[9] which would otherwise have been captured parasitically in the fuel or coolant, available to breed ^{233}U from ^{232}Th. The ^{233}U can then be subsequently fissioned, either in the same fuel cycle (extending the fuel residence time in the reactor), or subsequently, after reprocessing and being added to fresh fuel. The increased supply available by switching to a material estimated to be four times more common in the Earth's crust is another advantage of these reactors.

Sections 9.5 and 9.6 examine two possibilities for the implementation of the thorium cycle in a thermal-spectrum HWR (such as a CANDU): a once-through fuel cycle and a fuel cycle with ^{233}U recycling. Both of these concepts assume that plutonium (with the isotopic composition of LWR SF) is added to the initial fuel. Finally, in Section 9.7, the effects of a fast-spectrum HWR breeder are examined.

9.3. SCENARIO: ACCELERATOR DRIVEN SYSTEMS FOR MINOR ACTINIDE TRANSMUTATION

9.3.1. Overview of accelerator driven systems

Within the EURATOM Sixth Framework Programme, the EUROTRANS project has developed an ADS called EFIT, with the aim of demonstrating the technical feasibility of MA transmutation at the industrial level (Fig. 9.8) [9.2]. The maturity level of EFIT, according to the terminology of INPRO, is conceptual feasibility.

EFIT has a subcritical core of 400 MW thermal power with a k_{eff} of 0.97. The sub-criticality level is chosen to make certain that the core always remains subcritical under all plant conditions. The fission reactions are driven by an accelerator which delivers a proton beam of 800 MeV and 20 mA (16 MW) to a lead target where spallation reactions occur that release neutrons. Outside the target region, the core is divided into three zones, in which the fuel is composed of MAs and plutonium in a metal–ceramic fuel (enriched Mo-matrix; see Fig. 9.9). The plutonium and MA vectors are derived from reprocessed LWR SF (and, for the GAINS scenarios, also from the spent FR fuel). Plutonium comprises about 46.5% of the fuel, with the remainder being MAs. In these proportions, the BR of plutonium is about one and there is a net reactivity swing of only a few hundred per cent per million per cycle. Thus, fresh plutonium is required for the initial core only, and for subsequent cycles merely fresh MAs are added. As a consequence, the low plutonium consumption permits the available plutonium to be used in FRs instead.

The requirement of nearly zero burnup reactivity swing is dictated by the wish not to rely too heavily on the proton beam accelerator for the generation of extra spallation neutrons for reactivity swing compensation. This gives the designer the freedom to tailor the accelerator for a narrower range of proton beam current and to avoid oversizing. Hereby, safety is also improved since only very limited beam power excursions can occur due to the narrow operating range of the accelerator.

The core is designed with a high degree of radial power flattening, resulting in a high average power density and, as a consequence, high average burnup. This result can be obtained while respecting present technological limits such as the maximum allowable temperature for pellets (about 1380°C) and cladding (550°C) integrity. The major specifications of the reactor are shown in Table 9.1.

The objective for EFIT is transmutation of MAs, and this is how it is employed in the studies of GAINS. The MA burning in dedicated ADSs eliminates the need for transmutation of MAs in a critical FR, which means that the latter can be optimized for other objectives. However, ADSs can, in principle, have a broader mission, such as for instance transmutation of additional isotopes (e.g. selected FPs) in order to reduce the waste burden further, or production of fissile material for later use in critical systems by irradiation of fertile elements [9.3].

[8] A saving of 17.4% applied only to LWRs and only after 2015.

[9] Most thorium breeder designs are based on HWRs, which have more surplus neutrons than other reactor designs.

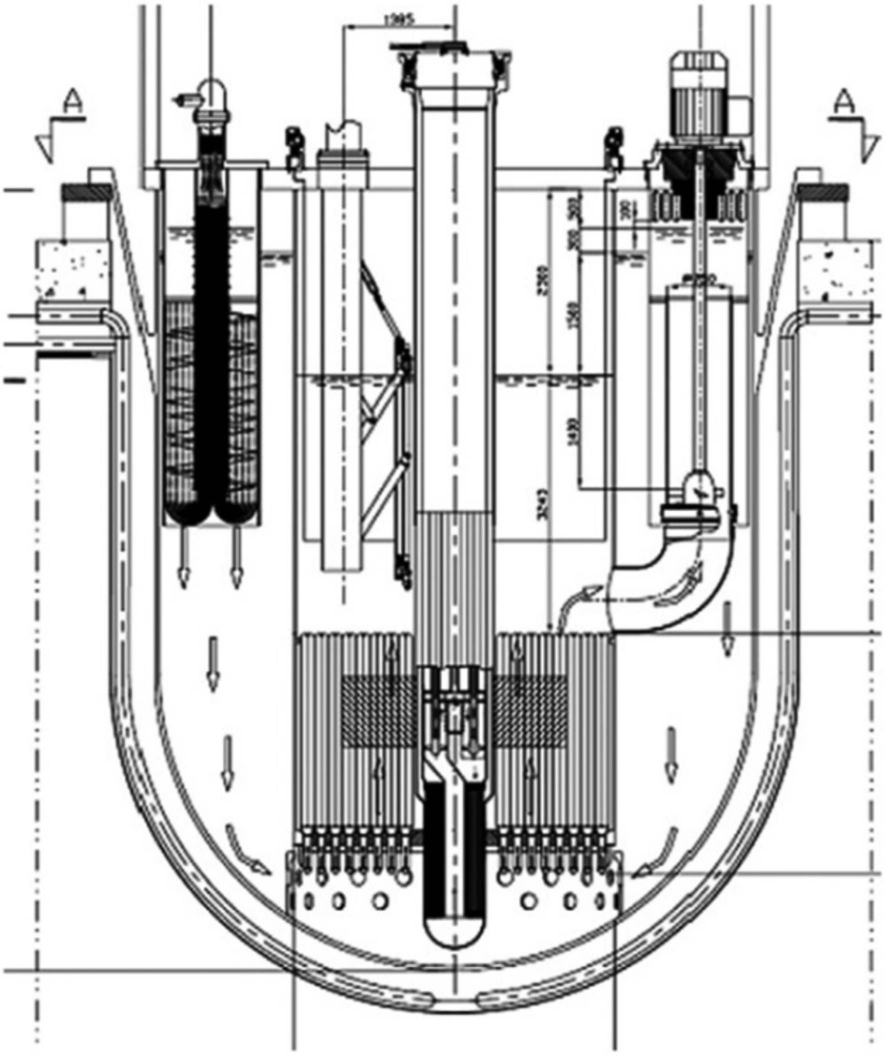


FIG. 9.8. EFIT primary system layout.

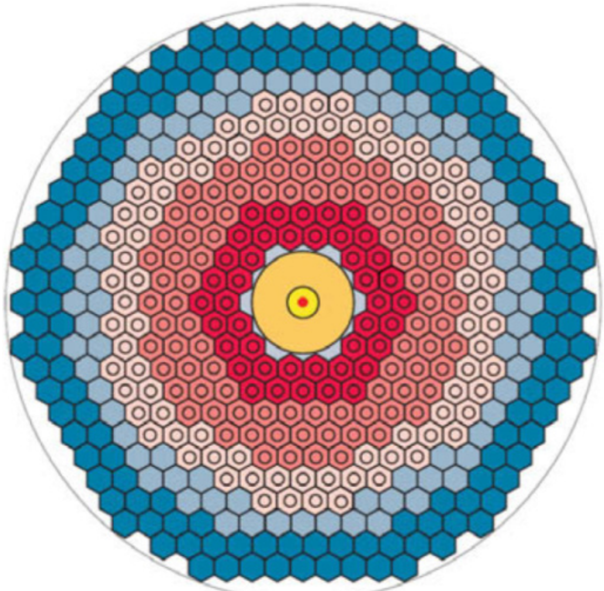


FIG. 9.9. Cross plane of three-zone core with metal–ceramic fuel [9.2]. In the centre is the proton beam target. Then, zones 1 and 2 are two assembly rings each, and zone 3 is five rings.

TABLE 9.1. MAJOR SPECIFICATIONS OF THE EUROPEAN FACILITY FOR INDUSTRIAL TRANSMUTATION

Reactor net electric output	MW	160		
Reactor thermal output	MW	400		
Average load factor	%	95		
Plant lifetime	a	60		
Operation cycle length	EFPD	1800		
		Core	Axial blanket	Radial blanket
Power share of each region	%	100	0	0
Fuel residence time	EFPD	1800		
Specific power density	MW/t	56.69		
Average discharged burnup	MW·d/t	107 447		

9.3.2. Fuel cycle

The fuel cycle system of the scenarios is illustrated in Fig. 9.10. In this figure, the red lines indicate pure MA flows, and the line connecting the ‘MA’ box to the ‘fabrication FR MIX fuel’ box is there only when the FR is a high burnup breeder FR.

9.3.3. Scenario assumptions

Both GAINS medium and high demand cases (see Fig. 5.5) were analysed. The scenario conditions under which the effects of the ADS were estimated are described below, and were slightly different than the final GAINS framework in some respects (as noted):

(a) Nuclear power plant load factor and lifetime:
   (i) LWR and HWR: plant load factor: 80%, plant lifetime: 40 years (GAINS: 85% load factor and 60 year lifetime).
   (ii) ALWR: plant load factor: 80%, plant lifetime: 60 years (GAINS: 85% load factor).
   (iii) FR: plant load factor: 85%, plant lifetime: 60 years.
   (iv) ADS: plant load factor: 95%, plant lifetime: 60 years (GAINS: 85% load factor).
   (v) U enrichment tails assay = 0.3 wt% first and 0.2 wt% after 2015 (GAINS: 0.3 wt% tails throughout).
(b) Reprocessing:
   (i) LWR: cooling + reprocessing time[10]: 6 years, composition of SF: 30 years decay after discharge from reactor[11].
   (ii) ALWR: cooling + reprocessing time: 6 years.
   (iii) FR: cooling + reprocessing time: 3 years.
   (iv) ADS: cooling + reprocessing time: 3 years.
   (v) Reprocessing losses were not considered (GAINS: 1% losses of all material during reprocessing).

[10] Cooling + reprocessing time is the time between reactor exit and the availability of the isotopes for fabrication of new fuel.

[11] NFCSS calculates decays for fuel in on-site storage (i.e. in cooling pools), but not after reprocessing. Since reprocessed LWR fuel may not be required for a significant time, its decay after reprocessing is approximated considering its composition 30 years after discharge at the time of its removal from the reactor.

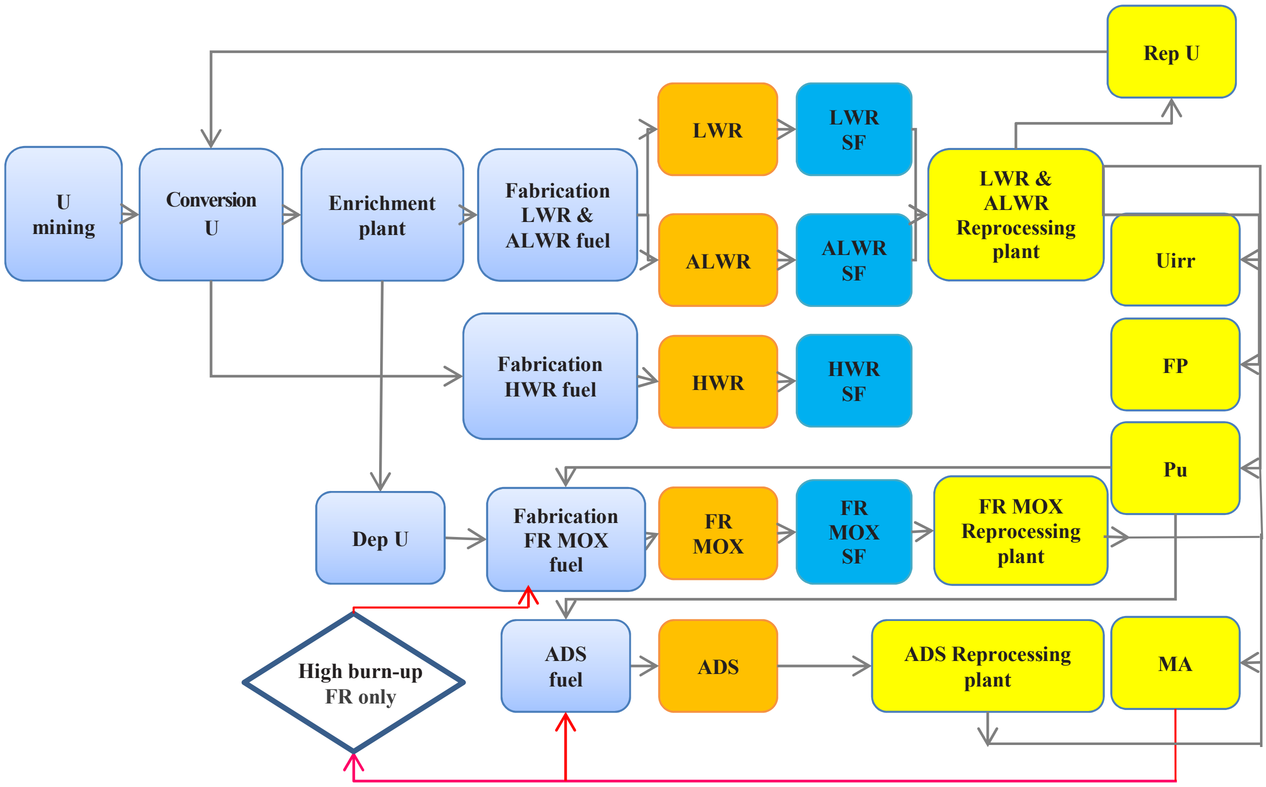


FIG. 9.10. Fuel cycle system of the BAU+ scenario and FRs and ADSs.

(c) Lead time:
   (i) Lead times, such as mining, conversion and fabrication process time, were not taken into account in the analysis (GAINS: 2 year lead time from mining to fuel fabrication).

(d) Reactor introduction speed:
   (i) LWR introduction: decrease from 2015 and disappears completely in 2055 because of a plant lifetime of 40 years.
   (ii) ALWR introduction: replaces the LWR.
   (iii) HWR introduction: 6% share of total nuclear power of ALWR + LWR (GAINS: HWR is 6% of all nuclear capacity, including FR).
   (iv) FR introduction:
   - 2021–2030: 1 GW(e) FR demand growth a year (total demand: 10 GW(e) in 2030).
   - 2031–2050:
     - High case: 19.5 GW(e) FR demand growth a year (total demand: 400 GW(e) in 2050).
     - Moderate case: 9.5 GW(e) FR demand growth a year (total demand: 200 GW(e) in 2050).
   - After 2051: maximum FR introduction consistent with plutonium availability.
   (v) ADS introduction from 2075 until 2100: consistent with plutonium availability and in line with MA balance needed for ADS operation during the plant lifetime.

9.3.4. Codes used for scenario calculations

Calculation studies of ADS introduction were carried out using the MESSAGE (Model for Energy Supply Strategy Alternatives and their General Environmental Impacts) and NFCSS codes included in the existing package of IAEA tools for modelling NESs. Short descriptions and information about these codes are given in Section 10.

9.3.5. Results of analysis — introduction of accelerator driven systems into BAU+/fast reactor scenarios

9.3.5.1. Scenario with a break-even (BR = 1.0) fast reactor

In this scenario, the FR (Section 6, Table 6.5 and Table II–5 in Annex II) has a BR of close to 1.0, so it is self-sufficient in plutonium after creation of its initial core. The introduction of ADS into the fuel cycle improves the MA accumulation problem in the moderate and high demand cases. Figures 9.1 and 9.2 show the amount of MA that would be accumulated by the year 2100 after the reprocessing of SNF from LWRs, ALWRs and FRs without ADS introduction.

In order to reduce the amount of MAs in the above mentioned scenarios, there is a need to introduce an installed capacity of about 148 GW(e) of ADSs in the high case and of about 104 GW(e) in the moderate case. These represent approximately 3% of the total installed capacity. As described in the analysis conditions, the ADSs are only introduced between 2075 and 2100, and only when a look ahead identifies sufficient MAs and plutonium feeds for the 60 year lifetime of the ADS.

Figures 9.11–9.14 illustrate the size of the installed capacities and the structure of NESs for the high and moderate cases of the BAU+ scenario with the FR break-even and ADSs.

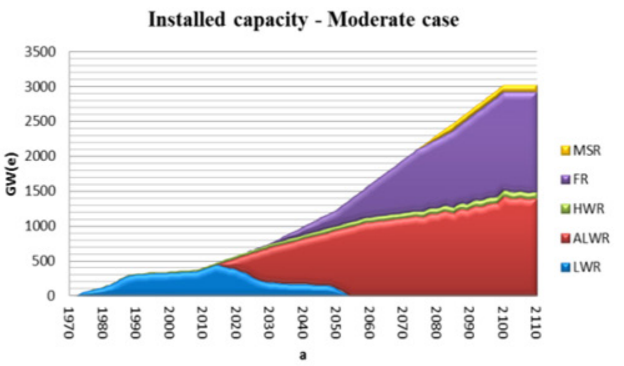


FIG. 9.11. Installed capacity for the BAU+/FR break-even and ADS scenario, moderate case.

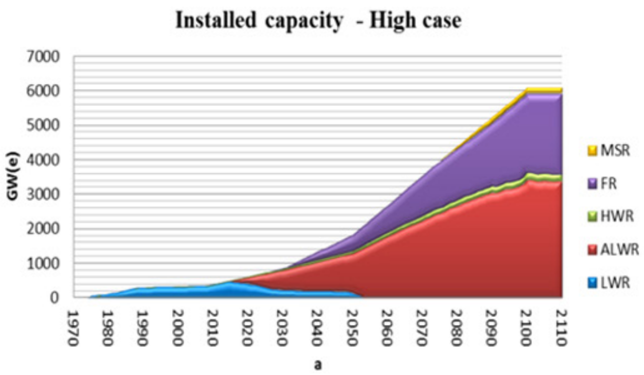


FIG. 9.12. Installed capacity for the BAU+/FR break-even and ADS scenario, high case.

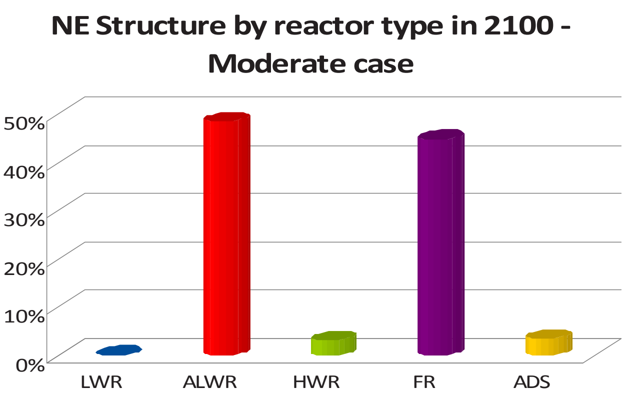


FIG. 9.13. Nuclear energy structure in 2100 for the BAU+/FR break-even and ADS scenario, moderate case.

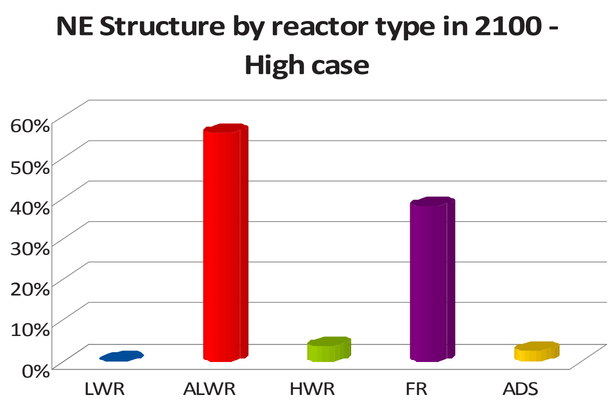


FIG. 9.14. Nuclear energy structure in 2100 for the BAU+/FR break-even and ADS scenario, high case.

The introduction of ADSs in the BAU+ scenario with the FR break-even leads to a large decrease of MA accumulation, which is illustrated in Figs 9.15 and 9.16. Without ADS introduction, the amount of MAs after reprocessing by 2100 will be around 7100 tHM in the high case and 4600 tHM in the moderate case. ADS deployment decreases this figure to around 800 tHM in the high case and to around 300 tHM in the moderate case.

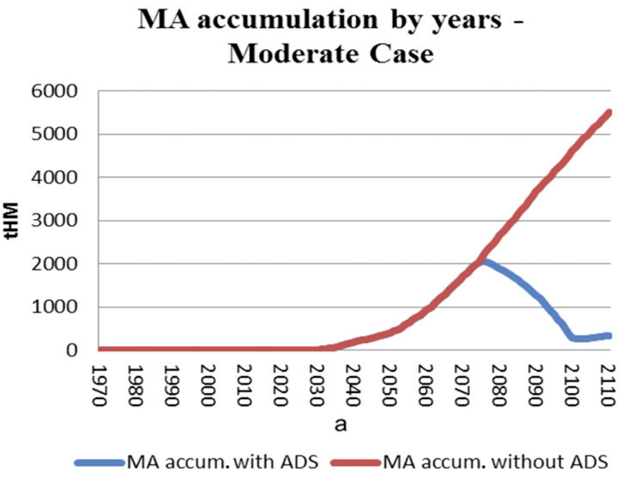


FIG. 9.15. Comparison of MA accumulation with and without ADS introduction, moderate case.

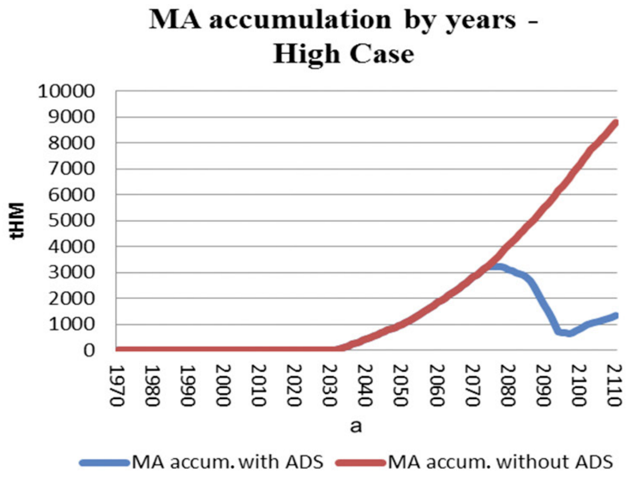


FIG. 9.16. Comparison of MA accumulation with and without ADS introduction, high case.

During the ADS building phase, MAs in the stockpile decrease partly because of MA burning in the ADS, but a larger effect is the loading of MAs in fresh ADS cores (~23.5 t MA/GW(e) capacity). However, the MA destruction rate in these ADSs after 2100 is less than the generation of MAs in other reactor systems, which makes the total MA accumulation continue to climb after 2100 (see Figs 9.15 and 9.16). The drastic decrease of MA accumulation until 2100 is, therefore, seen to be a consequence of the commissioning of new ADSs. The growing share of FR break-even reactors in this scenario (that produce more MAs per installed capacity than ALWRs), also increases the MA stockpile.

9.3.5.2. Scenario with a breeder (BR = 1.2) fast reactor

In this scenario, the FR breeder (Section 6, Table 6.6 and Table II–6 in Annex II) has a BR of approximately 1.2, so it is a net creator of plutonium over its lifetime. This allows a higher ratio of FR:ALWR to be built. The amount of MAs created by reprocessing of LWR, ALWR and FR fuel by 2100 in the moderate and high base cases, if no ADSs are built, is shown in Figs 9.3 and 9.4.

In order to reduce the amount of MAs in the above mentioned scenarios, there is a need to introduce an installed capacity of about 120 GW(e) of ADSs in the high case and about 78 GW(e) in the moderate case. These represent around 2–3% of total installed capacity in both cases. Figures 9.17–9.20 illustrate the amount of installed capacity and the structure of NESs for high and moderate cases of the BAU+ scenario with the FR breeder and ADS.

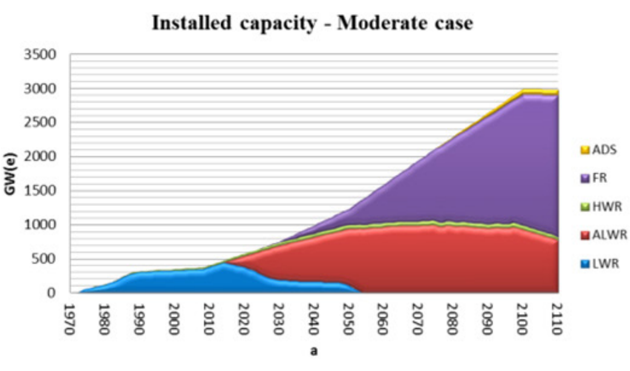


FIG. 9.17. Installed capacity for the BAU+ FR breeder and ADS scenario, moderate case.

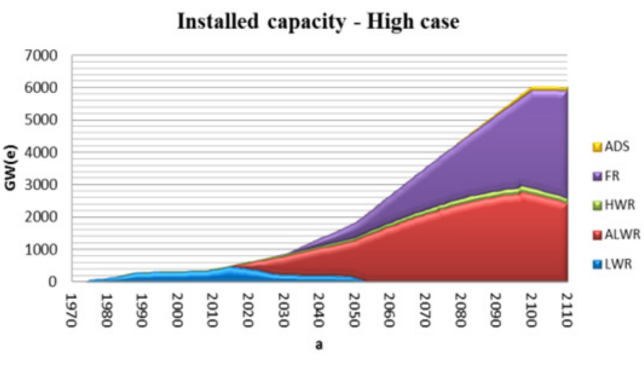


FIG. 9.18. Installed capacity for the BAU+ FR breeder and ADS scenario, high case.

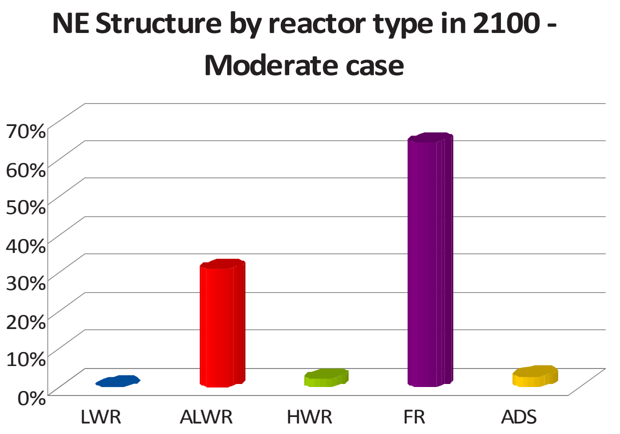


FIG. 9.19. Nuclear energy structure in 2100 for the BAU+ FR breeder and ADS scenario, moderate case.

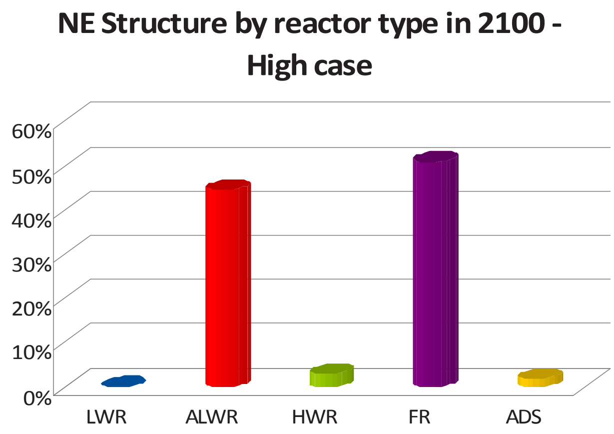


FIG. 9.20. Nuclear energy structure in 2100 for the BAU+ FR breeder and ADS scenario, high case.

The introduction of an ADS in the BAU+ scenario with the FR breeder leads to a large decrease in MA accumulation, which is illustrated in Figs 9.21 and 9.22. Without ADS introduction, the amount of MAs by 2100 will be around 5500 tHM in the high case and 3700 tHM in the moderate case. The ADS deployment decreases this figure to around 600 tHM in the high case and to around 500 tHM in the moderate case.

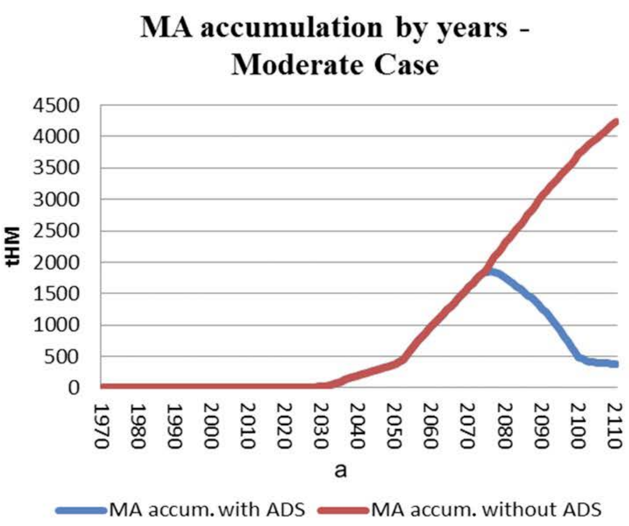


FIG. 9.21. Comparison of MA accumulation with and without ADS introduction, moderate case.

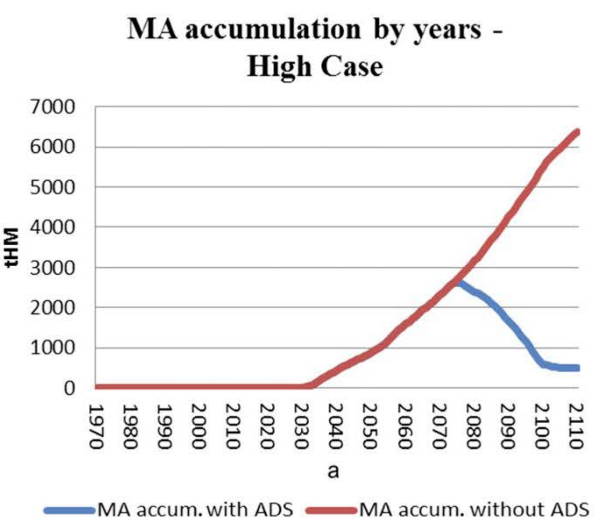


FIG. 9.22. Comparison of MA accumulation at reprocessing plant with and without ADS introduction, high case.

The same considerations apply to these cases as to the cases with the break-even FR. As before, the drop in the stockpile of MAs from 2100 to 2075 is due to the loading of fresh ADS cores. However, in these cases, a better equilibrium is found after 2100, with MA accumulation essentially stopped.

9.3.5.3. Scenario with a high burnup breeder (BR = 1.2) fast reactor

In this scenario, the FR (Section 6, Table 6.6 and Table II–6 in Annex II) has a BR of approximately 1.2, as in the previous scenario, but the fuel is taken to high burnup and has MAs added to its initial composition. The amount of MAs created from reprocessing of LWR, ALWR and FR fuel by 2100 in the moderate and high base cases, if no ADSs are built, is shown in Figs 9.5 and 9.6.

In these cases, the high burnup breeder FR is a net burner of MAs, so fewer ADSs are required to reduce the amount of MAs in the stockpile. In these cases, about 40 GW(e) of ADS installed capacity in the high case and about 25 GW(e) in the moderate case is required. These capacities account for less than 1% of total installed capacity in both cases. Figures 9.23–9.26 illustrate the amount of installed capacity and the structure of NESs for high and moderate cases of the BAU+ scenario with the FR high burnup breeder and ADS.

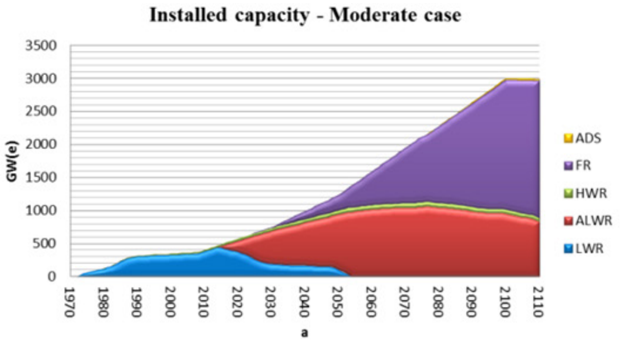


FIG. 9.23. *Installed capacity for the BAU+/FR high burnup breeder and ADS scenario, moderate case.*

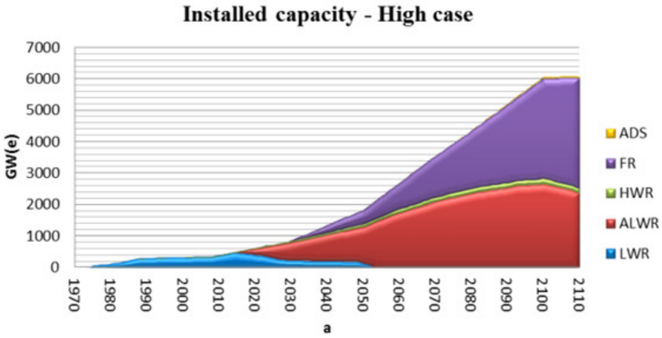


FIG. 9.24. *Installed capacity for the BAU+/FR high burnup breeder and ADS scenario, high case.*

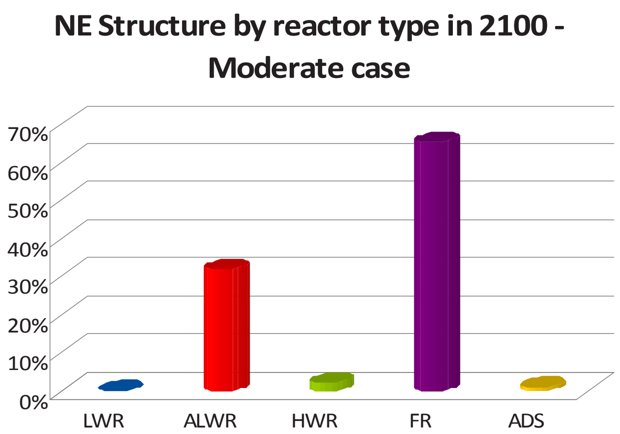


FIG. 9.25. *Nuclear energy structure in 2100 for the BAU+/FR high burnup breeder and ADS scenario, moderate case.*

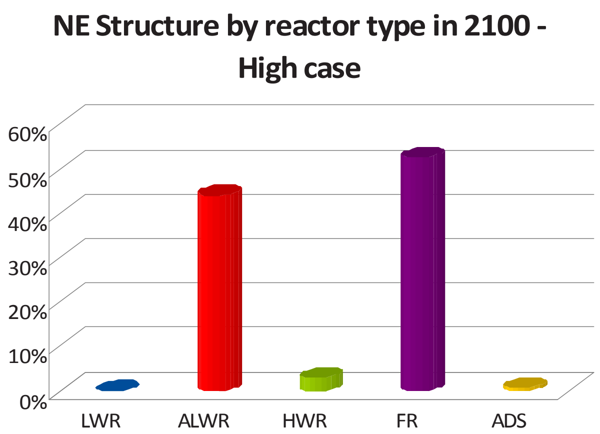


FIG. 9.26. *Nuclear energy structure in 2100 for the BAU+/FR high burnup breeder and ADS scenario, high case.*

The introduction of ADSs in the BAU+ scenario with the FR high burnup breeder leads to a decrease of MA accumulation, which is illustrated in Figs 9.27 and 9.28. Without ADS introduction, the amount of MAs by 2100 will be around 2800 tHM in the high case and 1900 tHM in the moderate case. The ADS deployment decreases this figure to around 1100 tHM in the high case and to around 800 tHM in the moderate case.

The results for these cases are qualitatively different than the preceding ones in that there is now a net decrease in MA accumulation after 2100. After 2100, the FR high burnup breeder reduces the MA accumulation alone, although at a moderate rate. The ADS accelerates the reduction of MA accumulated in the NES.

9.3.6. Conclusion

The objective of this section was to investigate incentives and the timing for introduction of ADSs into the global NES as a burner of MAs, and to identify a possible niche for the technology where it would be competitive and compatible with other prospective nuclear technologies aimed to enhance the sustainability of the NES. It was shown that the inclusion of ADSs aimed at MA burning could lead to achieving the strategy's target values and reduce the accumulation of MAs in a global NES by:

— 89% in the high GAINS scenario and 93% in the moderate GAINS scenario in the case of introduction by 2100 of about 2–3% of ADSs into the system of UOX fuelled thermal reactors and MOX fuelled FRs with a break-even core (BR: ~1) without MA burning;

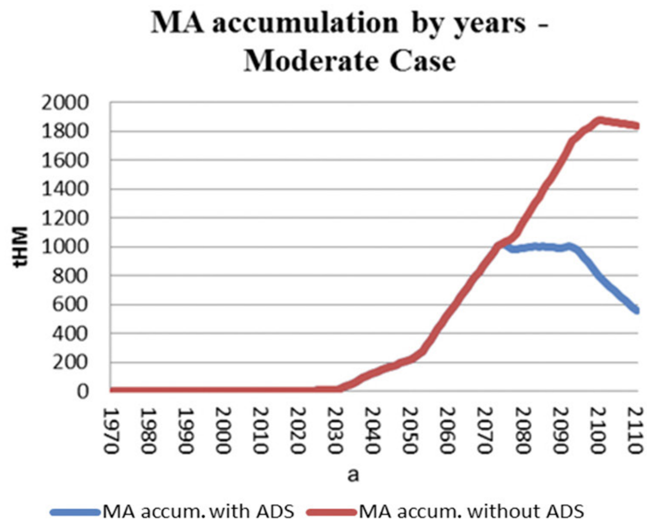


FIG. 9.27. *Comparison of MA accumulation with and without ADS introduction, moderate case.*

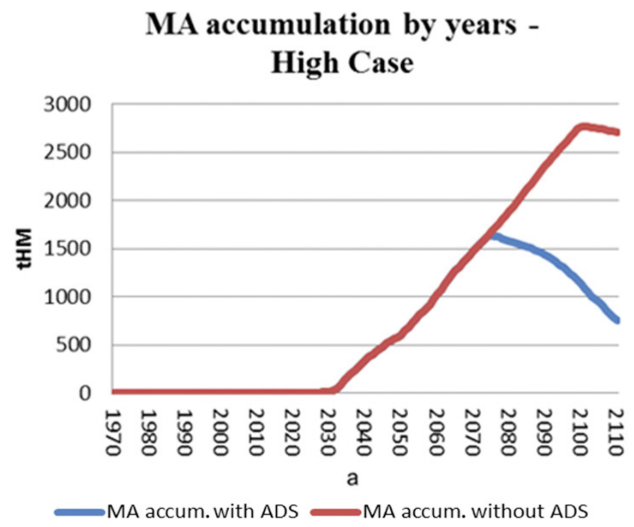


FIG. 9.28. *Comparison of MA accumulation with and without ADS introduction, high case.*

— 89% in the high scenario and 87% in the moderate scenario in the case of introduction by 2100 of about 2% of ADSs into the system of UOX fuelled thermal reactors and MOX fuelled fast breeders (BR: ~1.2) without MA burning;
— 59% in the high scenario and 57% in the moderate scenario in the case of introduction by 2100 of less than 1% of ADSs into the system based on UOX fuelled thermal reactors and high burnup fast breeders (BR: ~1.2) with MA burning.

Thus, this GAINS scenario study confirms a significant potential of the ADS as a prospective technology for MA burning, provided that technical and economic challenges are overcome. The availability of the proven ADS technology by the second half of the century would give an opportunity not to overload FRs with the function of MA burning at the first stage of their commercial introduction. This allows optimization of the operation of FRs to meet other objectives, for example, improved safety or breeding of fissile fuel.

Being a small portion of the NES, the ADS should also be considered as an integral part of future multilateral NFC centres in possible combination with other types of MA burners such as specialized FRs and MSRs.

9.4. SCENARIO: MOLTEN SALT REACTOR INTRODUCTION FOR MINOR ACTINIDE BURNING

9.4.1. Overview of the molten salt reactor

9.4.1.1. Concept

The concept of the MSR was developed in the 1950s and two small thermal-neutron-spectrum MSRs were successfully built in the 1960s. The first reactor was part of a programme to build a nuclear powered aircraft [9.4], whereas the second reactor was built to test the concept of a molten salt breeder reactor [9.5]. The programmes ended in 1976 when the USA decided to concentrate on a single breeder reactor concept — the sodium cooled FR. Today a renewed interest in MSRs exists because of renewed concerns over resource utilization, technological improvements to the component technologies and the development of fluoride salts as clean coolants.

In an MSR, the molten salt with dissolved fissile, fertile and fission isotopes flows through a reactor core. Historically, MSRs have been thermal neutron reactors in which neutrons in the reactor core were moderated by unclad graphite. Today, both thermal-MSRs and fast-spectrum MSRs are being investigated.

Compared with solid-fuel reactors, the MSR has many unique characteristics. Under emergency conditions, the liquid fuel is drained to passively cooled critically safe dump tanks. Via the use of 'freeze valves' (cooled sections of piping), and other techniques, this safety system can be passively initiated upon overheating of the coolant salt. MSRs operate at steady state conditions with no change in the nuclear reactivity of the fuel as a function of time. Fuel is added or subtracted as required. Last, FPs can be removed online and solidified. This process can minimize the radioactive inventory (accident source term) in the reactor core and can significantly reduce the hazard associated with reactor accidents.

After initial startup and operation, for some time, all of the fuel salt has only one composition — that of a high burnup fuel having poor fissile isotopics for use in weapons. If the MSR is being used for actinide burning, any batch of ‘new’ actinides is immediately diluted with the inventory of high burnup actinides upon its addition to the reactor. In contrast, in solid-fuel reactors, wide variations are present in the fissile isotopics between fuel elements and along the length of each fuel element. Therefore, in a solid-fuel reactor, if the fuel is diverted, parts of the SNF will be low burnup fuel with isotopics that are more favourable for use in weapons.

The low fissile inventories and lack of SNF reduce the MSR safeguards footprint to just the reactor site and imply that any major diversion of fissile material would shut down the reactor for power generation. Once a system is operating, there is no need for enrichment services or reprocessing of LWR fuel to provide added fissile material. Fuel reprocessing is also very straightforward in principle in an MSR, since it is already dissolved, and the handling (including separation, purification and element fabrication) of radioactive material is minimized. Such a system is also highly proliferation resistant, as fissile isotopes of plutonium would not be accessible after their introduction into the fuel. However, major challenges exist in developing reliable and economic systems to process the molten salts. The GAINS scenario is not focused on consideration of the technical or economic problems of these reactors or their solutions. The main objective is to examine what potential role such reactors could play in a future NES.

9.4.1.2. Technical data

In order to study an MSR introduction scenario, MSR design data provided by the GAINS Member States were selected and recompiled in a common data form for NFC analysis. The MSR used in the GAINS scenario calculations is a high flux MSR [9.3] with a fast-thermal[12] spectrum. This reactor has a homogeneous liquid fuel continuously circulating through the core. At the same time, the fuel is continuously fed and discharged (reprocessed). The presented MSR design implies that half of the fuel is in the core while the remainder is out of the core. The MSR is fed with Np, Am and Cm from LWR, ALWR and FR SF. The average neutron flux is 10^{15} neutrons$\cdot$cm$^{-2}\cdot$s^{-1}. The major specifications of the reactor are shown in Section 6, Table 6.11 and Table II–11 in Annex II.

9.4.2. Fuel cycle

The fuel cycle system of the scenarios is illustrated in Fig. 9.29. In this figure, the red lines indicate pure MA flows, and the line connecting the ‘MA’ box to the ‘fabrication MOX fuel’ box is there only when the FR is a high burnup breeder FR.

9.4.3. Scenario assumptions

These assumptions are the same as for the ADS introduction scenario (see Section 9.3.2) except that after 2075 MSRs, not ADSs, are introduced (consistent with plutonium and MA availability).

9.4.4. Codes used for scenario calculations

Calculation studies of MSR introduction were carried out using the MESSAGE and NFCSS (formerly VISTA) codes included in the existing package of IAEA tools for modelling NESs. Short descriptions and information about these codes are given in Section 10.

[12] ‘Fast-thermal’, in this case, means fast in the core and thermal in the reflector.

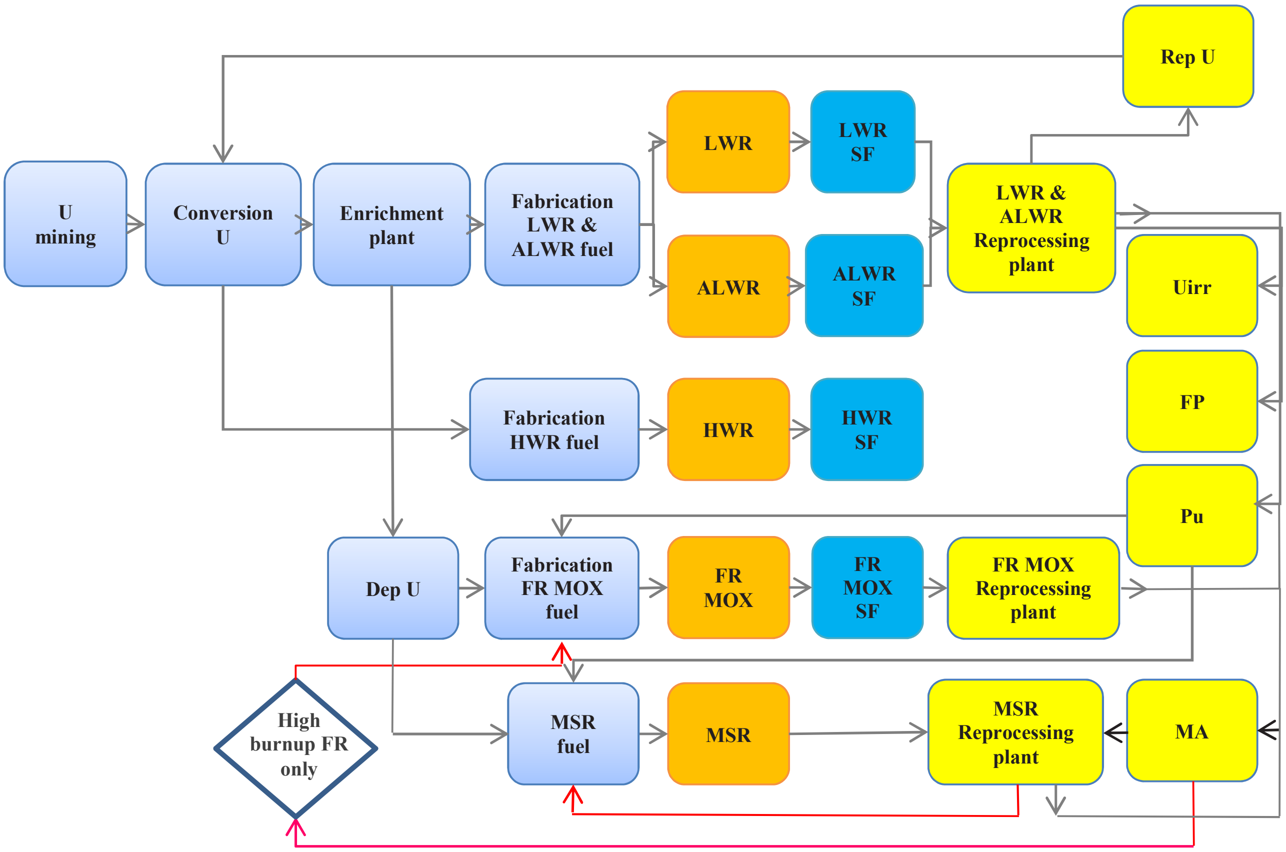


FIG. 9.29. *Fuel cycle system of the BAU+ FR break-even/FR breeder and MSR scenario.*

9.4.5. Results of analysis — introduction of molten salt reactors into BAU+ FR scenarios

9.4.5.1. Scenario with a break-even (BR = 1.0) fast reactor

The introduction of MSRs in the NFC improves the situation with the MA accumulation problem. Analysis of this issue is presented in the framework of the GAINS homogeneous model with the introduction of MSRs into the BAU+ moderate and high scenarios with the FR break-even. Figures 9.1 and 9.2 show the amount of MAs accumulated by 2100 after the reprocessing of LWR, ALWR and FR SNF without MSR introduction.

In order to reduce the amount of MAs in the above mentioned scenarios, there is a need to introduce an installed capacity of about 160 GW(e) of MSRs in the high case and about 100 GW(e) in the moderate case. This accounts for around 3% of total installed capacity in both cases.

Figures 9.30–9.33 illustrate the amount of installed capacity and the structure of NESs for the high and moderate cases of the BAU+ scenario with the FR break-even and MSRs.

The introduction of MSRs in the BAU+ scenario with the FR break-even leads to a large decrease of MA accumulation, which is illustrated in Figs 9.34 and 9.35. Without MSR introduction, the amount of MAs by 2100 will be around 7100 tHM in the high case and 4600 tHM in the moderate case. The MSR deployment decreases this figure to around 2000 tHM in the high case and to around 1200 tHM in the moderate case. It should be noted that MSR installed capacity had been introduced in the scenario by taking two aspects into consideration: (i) the MA balance needed for MSR operation during the plant lifetime (60 years) and (ii) the equilibrium share of the installed capacity of all the reactors after 2100.

9.4.5.2. Breeder (BR = 1.2) fast reactor

Analysis of the MA accumulation problem is presented in the framework of the GAINS homogeneous model with the introduction of MSRs into the BAU+ moderate and high scenario with the FR breeder. Figures 9.3 and 9.4

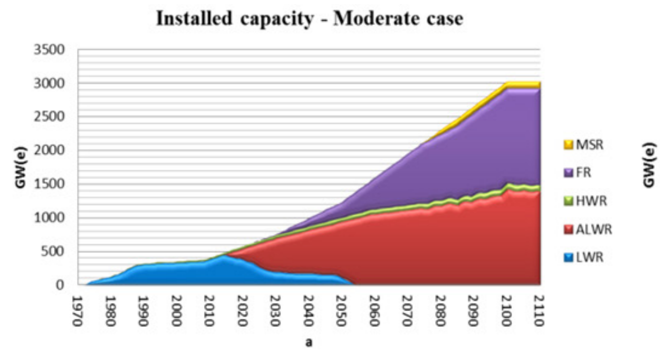


FIG. 9.30. Installed capacity for the BAU+/FR break-even and MSR scenario, moderate case.

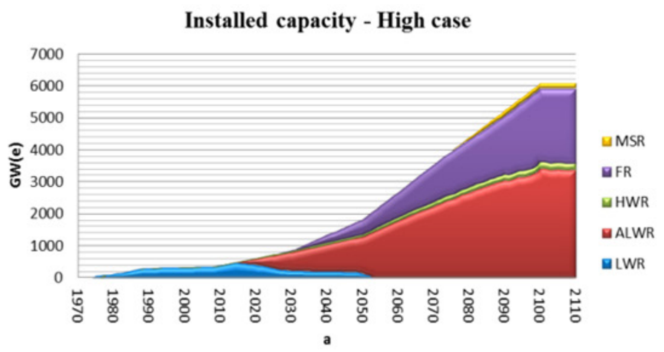


FIG. 9.31. Installed capacity for the BAU+/FR break-even and MSR scenario, high case.

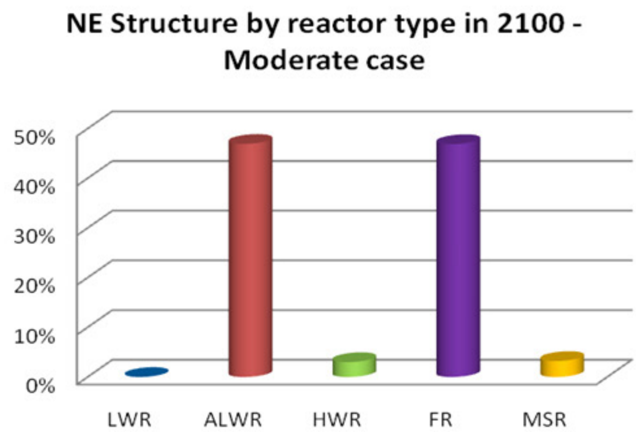


FIG. 9.32. Nuclear energy structure in 2100 for the BAU+/FR break-even and MSR scenario, moderate case.

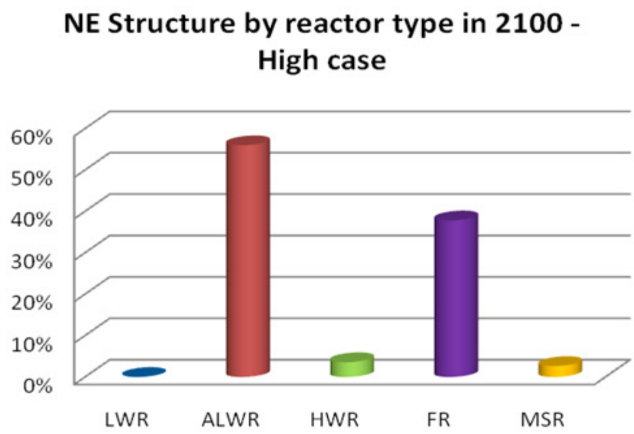


FIG. 9.33. Nuclear energy structure in 2100 for the BAU+/FR break-even and MSR scenario, high case.

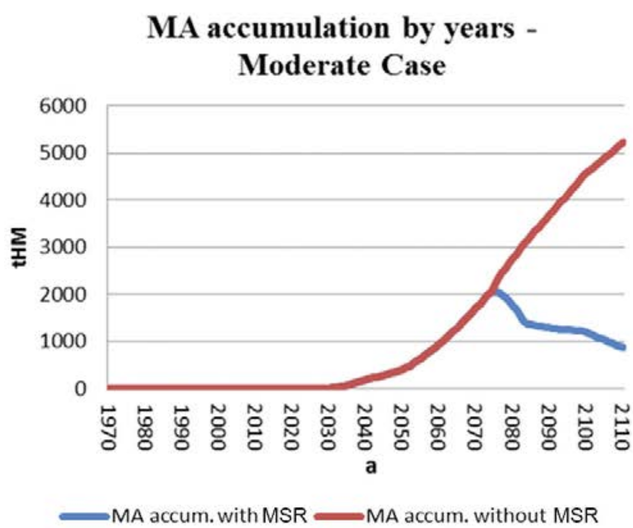


FIG. 9.34. Comparison of MA accumulation with and without MSR introduction, moderate case.

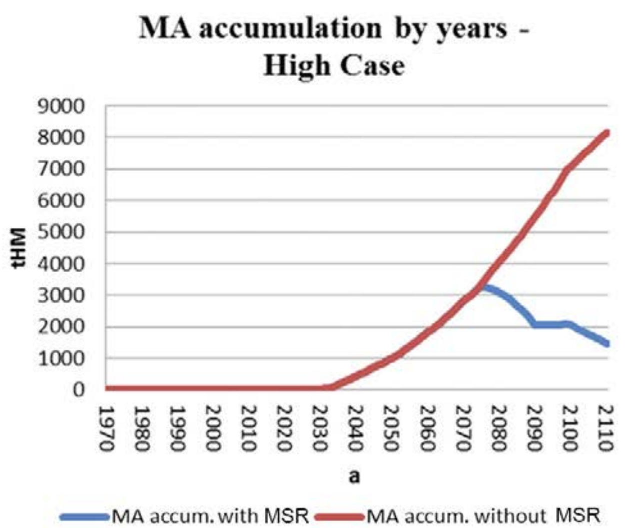


FIG. 9.35. Comparison of MA accumulation with and without MSR introduction, high case.

show the amount of MAs that are accumulated by 2100 after the reprocessing of LWR, ALWR and FR SNF without MSR introduction.

In order to reduce the amount of MAs in the above mentioned scenarios, there is a need to introduce an installed capacity of about 140 GW(e) of MSRs in the high case and about 80 GW(e) in the moderate case. This accounts for around 2% and 3%, respectively, of the total installed capacity.

Figures 9.36–9.39 illustrate the amount of installed capacity and the structure of NESs for high and moderate cases of the BAU+ scenario with the FR breeder and MSRs.

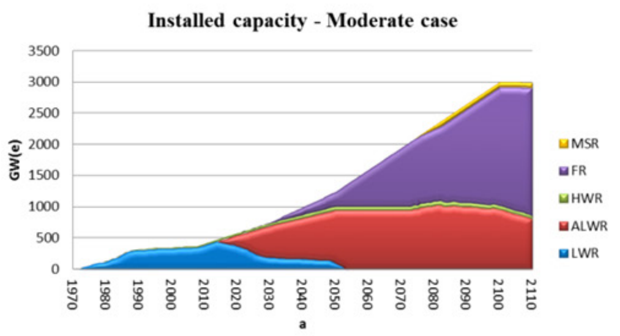


FIG. 9.36. Installed capacity for the BAU+/FR breeder and MSR scenario, moderate case

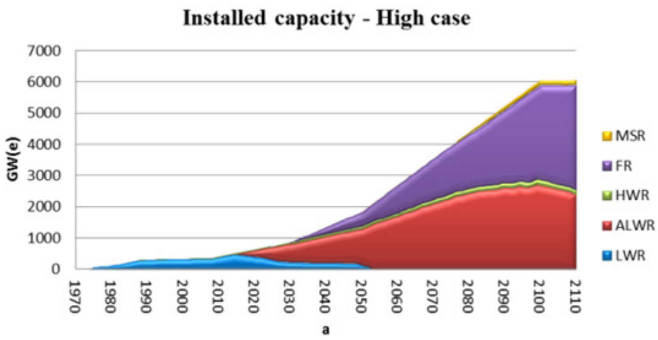


FIG. 9.37. Installed capacity for the BAU+/FR breeder and MSR scenario, high case

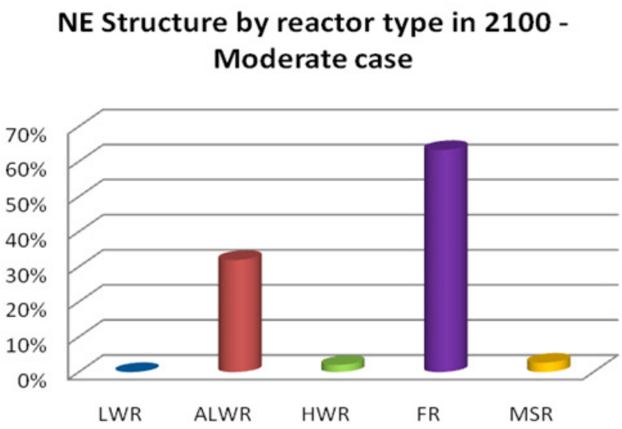


FIG. 9.38. Nuclear energy structure in 2100 for the BAU+/FR breeder and MSR scenario, moderate case

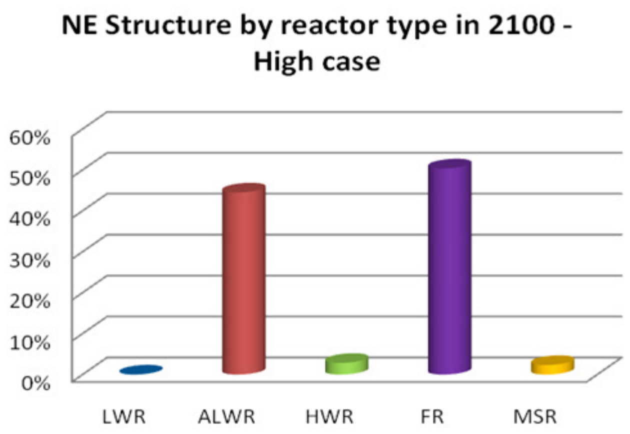


FIG. 9.39. Nuclear energy structure in 2100 for the BAU+/FR breeder and MSR scenario, high case.

The introduction of MSRs in the BAU+ scenario with the FR breeder leads to a large decrease in MA accumulation, which is illustrated in Figs 9.40 and 9.41. Without MSR introduction, the amount of MAs by 2100 will be around 5500 tHM in the high case and 3500 tHM in the moderate case. MSR deployment decreases this figure to around 1200 tHM in the high case and to around 700 tHM in the moderate case. It should be noted that MSR installed capacity had been introduced in the scenario by taking two aspects into consideration: (i) the MA balance needed for MSR operation during the plant lifetime (60 years) and (ii) the equilibrium share of the installed capacity of all of the reactors after 2100.

9.4.5.3. High burnup breeder (BR = 1.2) fast reactor

Analysis of the MA accumulation problem is presented in the framework of the GAINS homogeneous model with the introduction of MSRs into the BAU+ moderate and high scenarios with the FR high burnup breeder. Figures 9.5 and 9.6 show the amount of MAs that are accumulated by 2100 after the reprocessing of LWR, ALWR and FR SNF without MSR introduction.

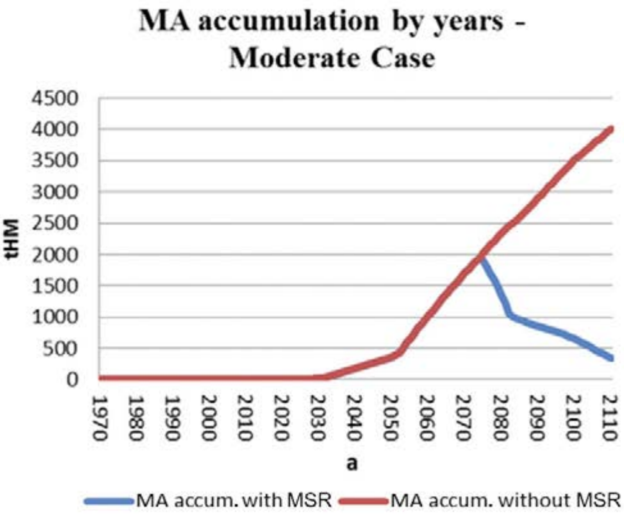


FIG. 9.40. Comparison of MA accumulation with and without MSR introduction, moderate case.

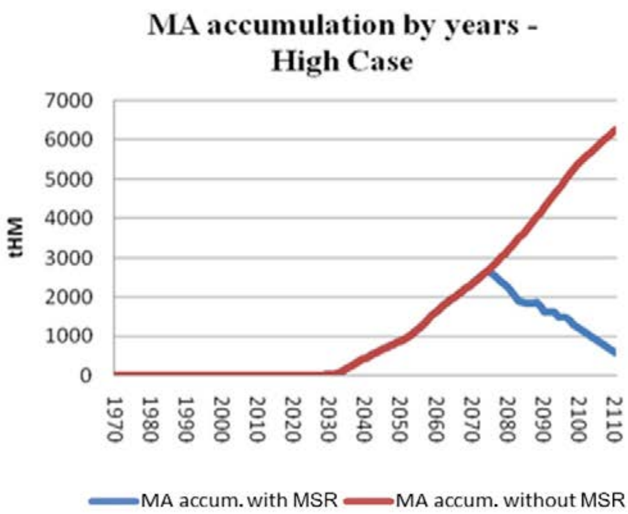


FIG. 9.41. Comparison of MA accumulation with and without MSR introduction, high case.

In order to reduce the amount of MAs in the above mentioned scenarios, there is a need to introduce an installed capacity of around 40 GW(e) of MSRs in the high case and about 25 GW(e) in the moderate case. This accounts for less than 1% of total installed capacity for both cases.

Figures 9.42–9.45 illustrate the amount of installed capacity and the structure of NESs for the high and moderate cases of the BAU+ scenario with the FR high burnup breeder and MSRs.

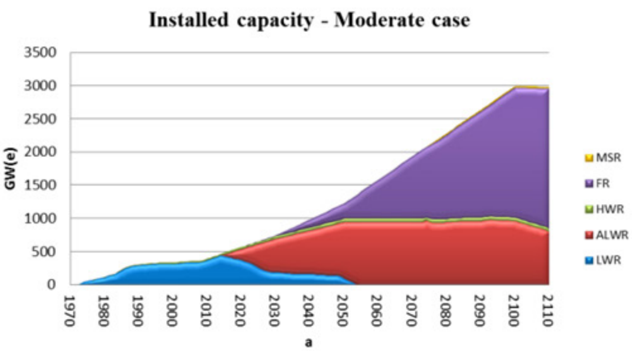


FIG. 9.42. Installed capacity for the BAU+/FR high burnup breeder and MSR scenario, moderate case.

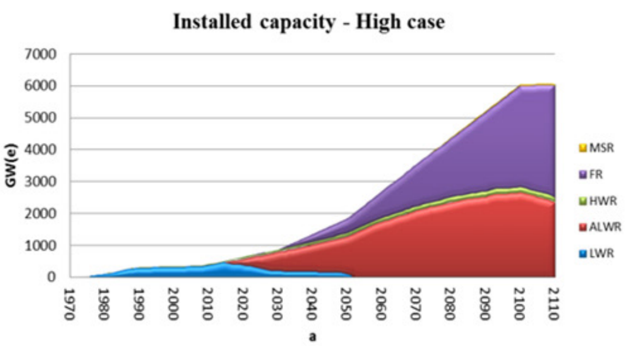


FIG. 9.43. Installed capacity for the BAU+/FR high burnup breeder and MSR scenario, high case.

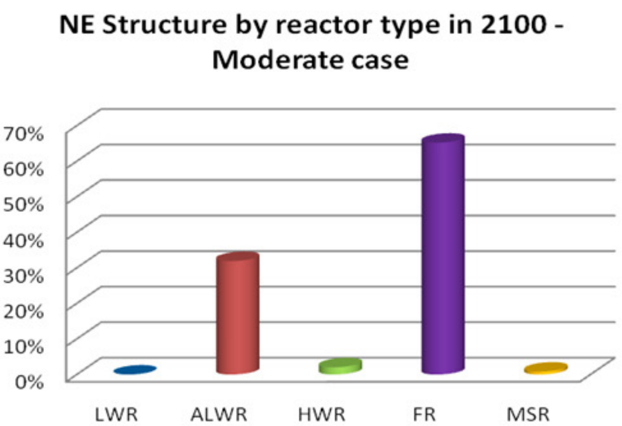


FIG. 9.44. Nuclear energy structure in 2100 for the BAU+/FR high burnup breeder and MSR scenario, moderate case.

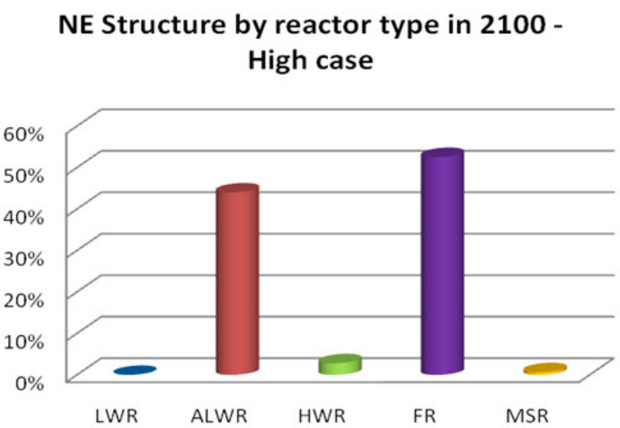


FIG. 9.45. Nuclear energy structure in 2100 for the BAU+/FR high burnup breeder and MSR scenario, high case.

The introduction of MSRs in the BAU+ scenario with the FR breeder leads to a large decrease of MA accumulation, which is illustrated in Figs 9.46 and 9.47. Without MSR introduction, the amount of MAs by 2100 will be around 2700 tHM in the high case and 1800 tHM in the moderate case. The MSR deployment decreases this figure to around 1300 tHM in the high case and to around 900 tHM in the moderate case. It should be noted that MSR installed capacity had been introduced in the scenario by taking into consideration two aspects: (i) the MA balance needed for MSR operation during the plant lifetime (60 years) and (ii) the equilibrium share of the installed capacity of all of the reactors after 2100.

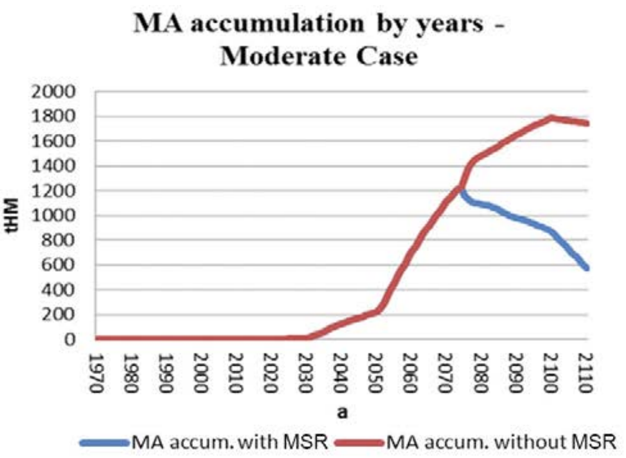


FIG. 9.46. Comparison of MA accumulation with and without MSR introduction, moderate case.

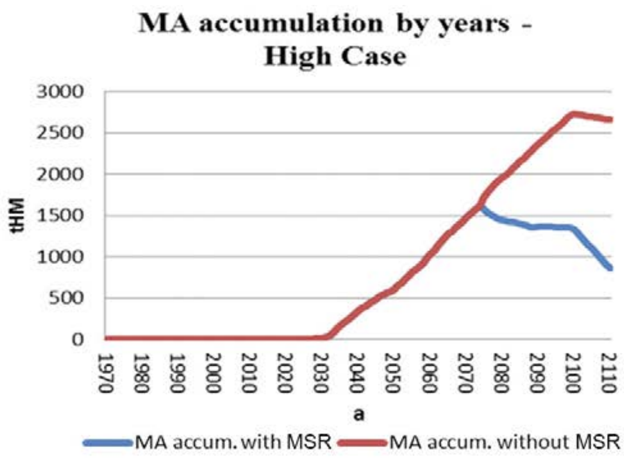


FIG. 9.47. Comparison of MA accumulation with and without MSR introduction, high case.

9.4.6. Conclusion

The objective of this section was to define incentives and the timing for introduction of MSRs into the global NES, as a burner of MAs, and to identify a possible niche for the technology where it would be compatible with other prospective nuclear technologies aimed to enhance the sustainability of the NES. It was shown that the inclusion of MSR aimed at MA burning could achieve the strategy's target values and reduce the accumulation of MAs in a global NES by:

— 71% in the high GAINS scenario and 74% in the moderate GAINS scenario with the introduction by 2100 of about 3% of MSRs into the system of UOX fuelled thermal reactors and MOX fuelled FRs with a break-even core (BR: ~1) without MA burning;
— 78% in the high scenario and 81% in the moderate scenario with the introduction by 2100 of about 2–3% of MSRs into the system of UOX fuelled thermal reactors and MOX fuelled fast breeders (BR: ~1.2) without MA burning;
— 51% in both the high and moderate GAINS scenarios with introduction by 2100 of less than 1% of MSRs into the system based on UOX fuelled thermal reactors and high burnup fast breeders (BR: ~1.2) with MA burning.

Thus, this GAINS scenario study confirms a significant potential of the MSR as a prospective technology for MA burning, provided that technical and economic challenges are overcome. The availability of the proven MSR technology by the last quarter of the century would give an opportunity to not overload FRs with the function of MAs burning at the first stage of their commercial introduction.

Being a small portion of the NES, the MSR should also be considered as an integral part of future multilateral NFC centres in possible combination with other types of MA burners such as specialized FRs and ADSs.

9.5. SCENARIO: THORIUM FUELLED HEAVY WATER REACTOR WITH NO RECYCLING

9.5.1. Description of the reactors

Three reactors are used for this scenario: the nominal GAINS LWR (Section 6, Table 6.1 and Table II–1 in Annex II), the nominal GAINS HWR (Section 6, Table 6.4 and Table II–4 in Annex II) and a thorium/plutonium fuelled HWR (Section 6, Table 6.12 and Table II–12 in Annex II).

The thorium fuelled HWR is essentially the same as a CANDU-6 with natural uranium fuel replaced by plutonium–thorium fuel bundles of similar overall size. The CANDU-6 is a 2.064 GW(th)/0.668 GW(e) HWR (32.4% efficiency), which takes natural uranium as its fuel in the form of 50 cm, 20 kg fuel bundles. The total core mass is 88.004 t (initial HM). It is refuelled while online and achieves an average burnup of ~7.500 GW·d/t U. The bundle design of the CANDU-6 consists of thirty seven 0.611 cm radius elements in four rings of elements, including a ring of one (fuelled) central element (Fig. 9.48).

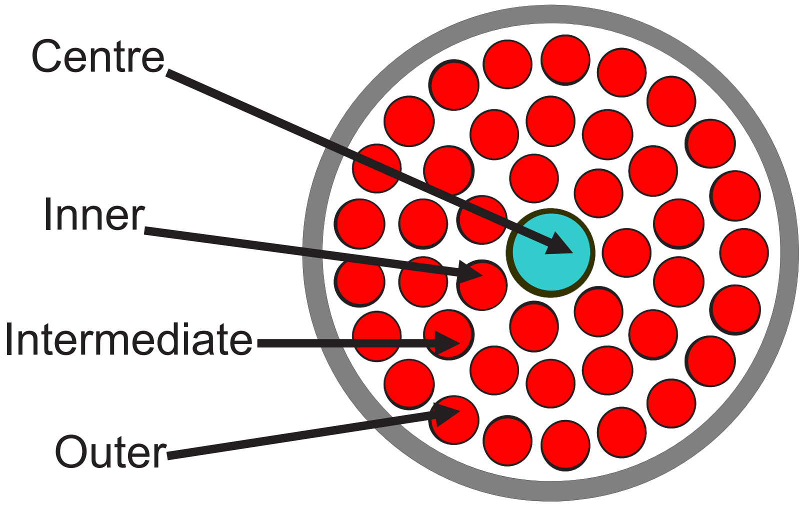


FIG. 9.48. A CANFLEX geometry bundle for Pu–ThO_2.

The thorium/plutonium fuelled HWR has a total core mass 71.391 t. Its bundles are a variant of the CANFLEX^{TM} type, consisting of 42 equal-sized elements arranged in three rings around a larger, poisoned, central pin whose function is to reduce the void reactivity inherent in the HWR design. The plutonium fraction is 3.75%. The burnup achieved is 20.3 GW·d/tHM, of which 19% comes from thorium irradiation. These reactors are referred to as 'ThPu' in this section.

The nominal HWR for GAINS (as described in Section 6, Table 6.4 and Table II–4 in Annex II) is an approximation of the CANDU-6, with 7.0 GW·d/t U exit burnup, a 30% thermal efficiency and a core mass of 83.428 t.

9.5.2. Fuel cycle of a plutonium–thorium once-through reactor (ThPu)

The high neutron fuel economy of an HWR, combined with the short fuel bundle and online refuelling allows operation with little excess reactivity and, hence, the use of a number of types of fuel with relatively low fissile content such as natural uranium or, as discussed here, thorium with a few per cent added plutonium or ^{233}U. Full utilization of the thorium fuel cycle would involve reprocessing of the ^{233}U bred into the fuel, but a likely first step towards implementation of the cycle, before the construction of a ^{233}U recycling plant, would simply be to allow ^{233}U bred in thorium bundles to extend the reactor residence time of the fuel in a once-through scenario. The initial fissile content in the thorium bundles would be plutonium (all isotopes) from a once-through LWR fuel cycle.

9.5.3. Scenario assumptions

The scenario in which the impact of thorium fuelled HWRs was analysed was the most basic BAU case, with LWRs and HWRs (but not ALWRs or FRs). Scenario assumptions were broadly similar to the standard GAINS framework, with one or two exceptions, as noted below:

- Nuclear power plant load factor and lifetime:
  - LWR and HWR: plant load factor: 85%, plant lifetime: 60 years.
  - U enrichment tails assay = 0.3 wt% throughout.
- Reprocessing:
  - LWR: cooling + reprocessing time[13]: 6 years[14].
  - HWR: not reprocessed[15].
  - ThPu: not reprocessed[16].
  - Reprocessing losses were not considered (GAINS framework: 1% losses of all material during reprocessing).
- Lead time:
  - Lead times, such as mining, conversion and fabrication process time, were not taken into account in the analysis (GAINS framework: 2 year lead time from mining to fuel fabrication).
- Reactor introduction speed:
  - Once-though ThPu are introduced starting in 2008, and are built as there is available plutonium from recycled LWR.
  - Natural uranium fuelled HWR (NU-HWR) introduction: 6% share of the GAINS medium demand scenario until 2008, but no new NU-HWR built subsequently[17]. The 60 year lifetime means that all NU-HWRs disappear by 2068.
  - LWR introduction: LWRs are built to make up the GAINS medium scenario requirement (after subtracting the HWR and ThPu share).

Two sensitivity studies were performed: evaluation of the effect of converting NU-HWRs (instead of building ThPu reactors in place of LWRs), and evaluation of the effect of the assumption that the NU-HWR fuel will not be reprocessed.

9.5.3.1. Codes used for scenario calculations

Calculation studies of the introduction of thorium fuelled HWRs were carried out using DESAE, a code included in the existing package of IAEA tools for modelling NESs. A short description of this code is given in Section 10. DESAE does not decay SF during on-site cooling, but this is not a significant problem as the cooling plus reprocessing time can be included in the description of the reactor exit fuel composition. Other limitations of DESAE (with regard to the GAINS framework) are the lack of reprocessing losses and mining/fabrication lead times, but these are smaller effects and can be neglected.

A sometimes important SF metric, not explicitly included in the GAINS framework, is the actinide decay heat, which dominates that from FPs after approximately 200 years. Decay heat from SF for long decay times is not available in DESAE, so this was calculated using a post-processing technique as follows. The decay heat was calculated by first running the programme ORIGEN-S from the SCALE 5.1 code suite [9.6] for nominal 1 t initial quantities of all the actinides tracked by DESAE 2.2, namely: ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am and ^{244}Cm. ORIGEN-S is an industry-standard isotope depletion code which can simulate irradiation and decay of nuclides from a comprehensive library. The version used for this analysis was

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[13] Cooling + reprocessing time is the time between reactor exit and the availability of the isotopes for fabrication of new fuel.

[14] This scenario was analysed with DESAE, which does decay isotopes after reprocessing, so it was not necessary to guess the length of decay time before reuse (as done in the ADS and MSR scenarios — see Sections 9.3 and 9.4).

[15] A sensitivity case was run to estimate the effect of this assumption.

[16] This spent fuel will contain a store of ^{233}U which may be recovered for a more advanced fuel cycle. This is considered in Section 9.6.

[17] A sensitivity case examines the possibility of retaining all NU-HWRs and building thorium fuelled HWRs entirely at the expense of potential LWRs.

the ORIGEN-S which comes as part of the SCALE 5.1 suite of codes. It is a point-geometry code and is usually run as the second step in a sequence in which the cross-section library is first collapsed to three energy groups using a detailed geometry description, but in this instance a pre-collapsed CANDU-specific library (distributed with SCALE 5.1) was used. The ORIGEN-S results for heat output versus time were used to develop a set of time dependent conversion factors $C_N(t) = Q(t)/M_N^{t=0}$ for each nuclide *N* and various times *t*. These conversion factors related the heat output of nuclide *N* (and its daughters) at time *t*, $Q(t)$, to the total mass of nuclide *N* initially present $M_N^{t=0}$. The total actinide decay heat at time *t* (after 2130) is then the sum of all of the heats attributable to these initial masses and can be expressed as:

$$D(t) = \sum_N C_N(t) M_N^{t=0}$$

For each scenario, the year 2130 masses of all of the actinides were exported from DESAE to an Excel spreadsheet and then pasted to another spreadsheet having the pre-calculated conversion factors for *t* = 0, 200, 400, 600, 800 and 1000 years (see Table 9.2). In this way, *D*(0), *D*(200), *D*(400) … *D*(1000 years) were found and a decay heat trend for post-irradiation long term storage was established.

TABLE 9.2. ACTINIDE DECAY HEATS

Isotope N	Heat output of isotope *N* and its daughters at time *t* is $\sum_N C_N(t) M_N^{t=0}$ when the initial quantity of isotope *N* is $M_N^{t=0}$ in tonnes and the heat output is in watts.					
	$C_N(0)$	$C_N(400)$	$C_N(400)$	$C_N(600)$	$C_N(800)$	$C_N(1000)$
^{234}U	1.79E+02[a]	1.79E+02	1.80E+02	1.81E+02	1.81E+02	1.82E+02
^{235}U	5.99E–02	6.41E–02	6.63E–02	6.84E–02	7.05E–02	7.26E–02
^{236}U	1.75E+00	1.75E+00	1.75E+00	1.75E+00	1.75E+00	1.75E+00
^{238}U	9.34E–03	1.03E–02	1.03E–02	1.03E–02	1.03E–02	1.03E–02
^{237}Np	2.01E+01	2.19E+01	2.20E+01	2.20E+01	2.20E+01	2.20E+01
^{238}Pu	5.68E+05	1.17E+05	2.42E+04	5.13E+03	1.20E+03	3.88E+02
^{239}Pu	1.93E+03	1.92E+03	1.91E+03	1.90E+03	1.89E+03	1.87E+03
^{240}Pu	7.07E+03	6.92E+03	6.78E+03	6.64E+03	6.50E+03	6.36E+03
^{241}Pu	3.29E+03	8.59E+04	6.24E+04	4.53E+04	3.29E+04	2.39E+04
^{242}Pu	1.17E+02	1.17E+02	1.17E+02	1.17E+02	1.17E+02	1.17E+02
^{241}Am	1.15E+05	8.31E+04	6.03E+04	4.38E+04	3.18E+04	2.31E+04
^{244}Cm	2.83E+06	8.16E+03	6.69E+03	6.55E+03	6.41E+03	6.28E+03

[a] ‘E+02’ means ‘$\times 10^2$’.

9.5.4. Results of analysis

9.5.4.1. Introduction of once-through thorium fuelled heavy water reactors

The division of electrical capacity between the various reactor types in this scenario is shown in Fig. 9.49. The plutonium available from spent LWR fuel allows the electricity capacity of the ThPu HWRs to rise to 350 GW(e) by the end of the century — about 11.9% of the total capacity.

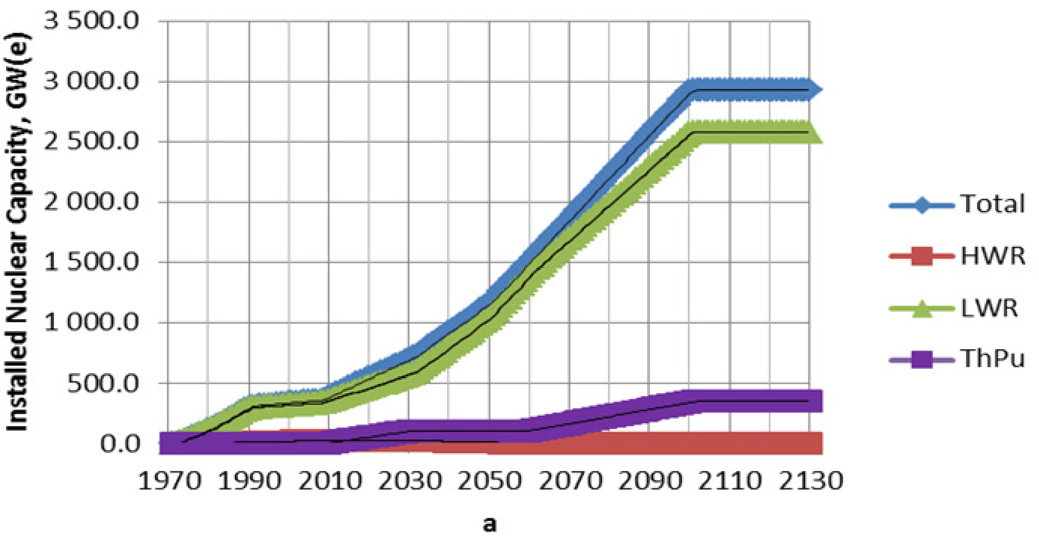

FIG. 9.49. Electrical capacity for the ThPu variant of the BAU scenario (moderate case).

The integrated natural uranium requirements (Fig. 9.50) for this scenario (41.9×10^6 t in 2130) are reduced by 9.7% over the BAU high case (46.4×10^6 t). For comparison, the estimated natural uranium requirements for a case introducing break-even FRs (BR: ~1.0) is also shown (30.5×10^6 t, or 34.3% savings over the BAU high case in 2130). The ThPu case extends the limit on easily mined natural uranium resources by about 5 years (relative to the BAU scenario, see the lower construction line in Fig. 9.50). When unconventional natural uranium resources (principally phosphates) are included, the exhaustion of resources is extended by about 10 years (from 2115 to 2125, see the higher construction line in Fig. 9.50).

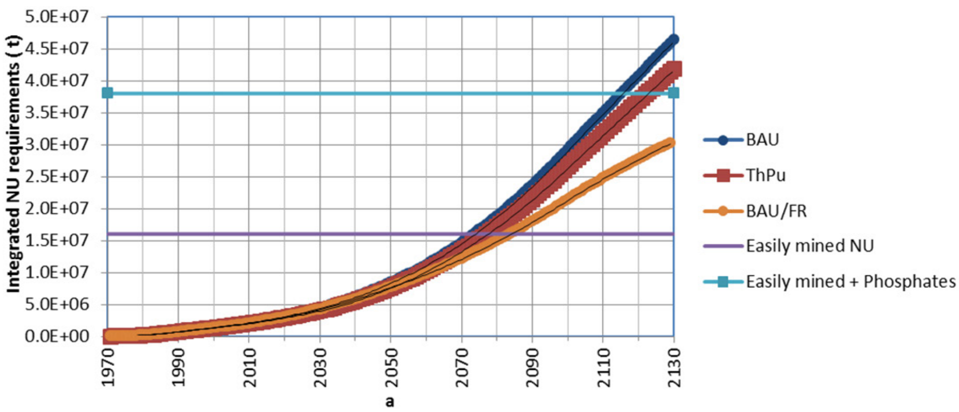

FIG. 9.50. Integrated natural uranium requirements for ThPu versus the BAU or BAU–FR scenario (high case).

Actinide decay heat was calculated using the method described in Section 9.4.4 and source terms which were the amounts of material left in the reprocessing facilities at the end of the scenario (2130). This method, thus, excludes from the calculation all fuel currently in the reactors or in the reactor cooling pools. The decay heat from spent HWR fuel was included separately.

It was found that actinide decay heat (Fig. 9.51) at 1000 years has been reduced to 0.212 GW(th) in the scenario with ThPu, about 70% of the BAU case (0.304 GW(th), see Fig. 9.52) This is a result of a number of factors. These are listed in descending order of importance:

(a) Plutonium-241 is re-irradiated and fissioned before it can decay into ^{241}Am — a major decay heat component;
(b) The irradiation of thorium, which displaces 12% of the uranium based power production in the BAU case, produces essentially no new MAs because there is no ^{238}U present;
(c) The amount of spent NU-HWR fuel, containing the major decay heat components ^{238}Pu and ^{240}Pu, and which is not reprocessed, is reduced relative to the BAU case because these reactors were phased out.

9.5.4.2. Sensitivity studies

The ThPu scenario essentially assumes that half the thorium fuelled HWRs (11.9% of electricity capacity in 2100) will be converted to NU-HWRs (6% of capacity in 2100 in the BAU scenario). If instead the thorium fuelled HWRs are built entirely at the expense of potential LWRs, then one can retain the superior natural uranium utilization of the NU-HWR in the scenario. However, the gain is at the cost of producing less plutonium in spent LWR fuel. The net gain (when one retains a 6% share of NU-HWRs throughout) is only 0.7% in total natural uranium requirements relative to the original scenario.

A scenario with plutonium recycling from HWR fuel would decrease total requirements by another 2% (to ~13.2% or 40.7 × 10^6 t in 2130) relative to the BAU scenario. This scenario did not consider the change in equilibrium of the plutonium isotopic vector which would occur if the HWR fuel plutonium were mixed with the LWR plutonium.

9.5.5. Conclusion

Recycling the spent plutonium from LWRs into the ThPu fuel cycle does not require major advances to currently available technology and would reduce natural uranium requirements by ~10% and the total decay heat of SF by around one third compared with the BAU case.

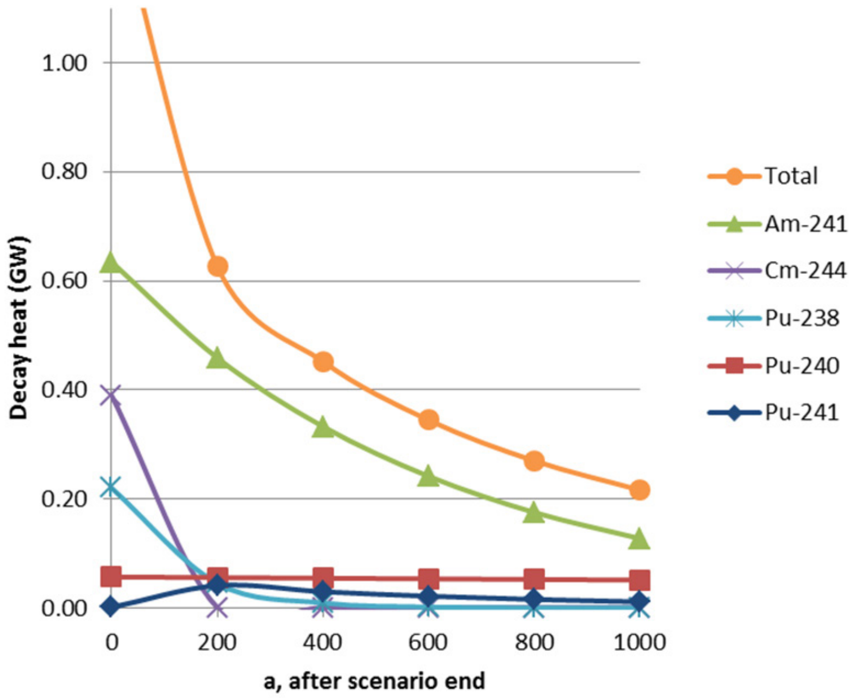

FIG. 9.51. Actinide decay heat.

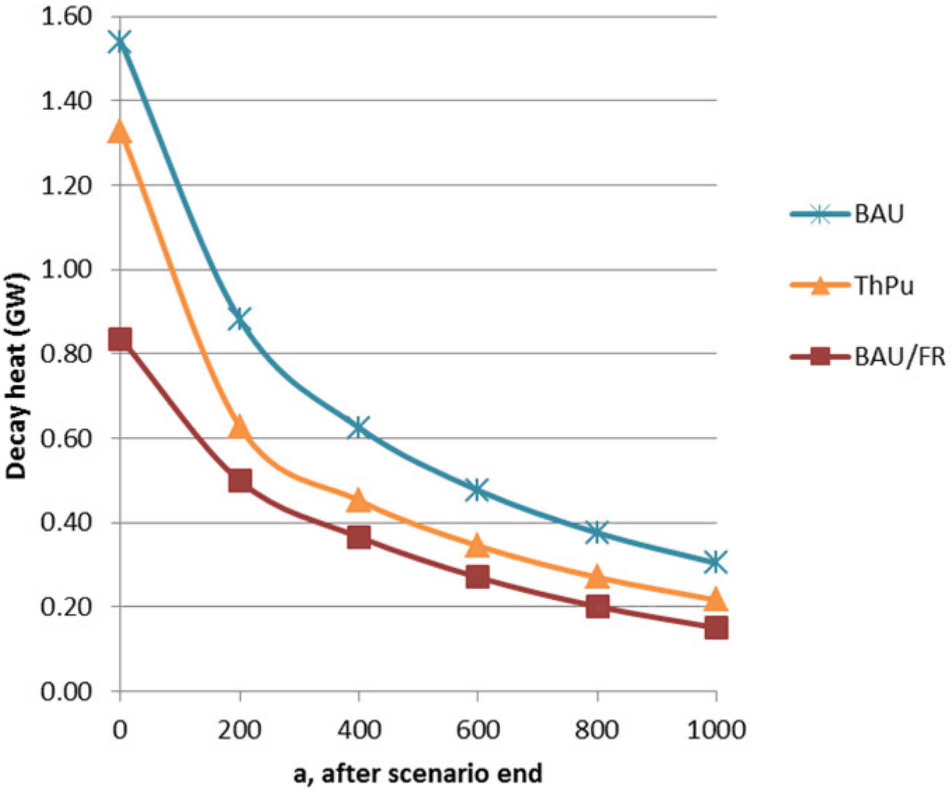

FIG. 9.52. Actinide decay heat versus the BAU and BAU break-even FR scenarios.

9.6. SCENARIO: THORIUM FUELLED HEAVY WATER REACTOR WITH ^{233}U RECYCLING (ThPuR)

9.6.1. Description of the ^{233}U recycling reactor

The reactors used in this scenario are physically identical to those described in Section 9.5.1. The ThPuR achieves a burnup of 19.85 GW·d/t, slightly smaller than the once-through ThPu reactor described in Section 9.5.1, but its fuel residence time is shorter by approximately the same proportion (810 EFPDs versus 825 EFPDs). The recycling reactors, which have plutonium in their initial fuel, will be referred to in this Section as 'ThPuR' (thorium, with initial plutonium, and recycled ^{233}U).

9.6.2. Fuel cycle of a plutonium–thorium reactor with ^{233}U recycling (ThPuR)

In this fuel cycle, graded amounts of plutonium (with the isotopic composition of LWR SF) and ^{233}U are added to a CANFLEX fuel bundle, with higher plutonium concentrations in the inner rings and higher ^{233}U concentrations in the outer rings. Uranium-233 bred in the fuel is recovered and the thorium, plutonium and FPs discarded. Any chemical separation plant which recovered ^{233}U would also recover ^{232}U, ^{234}U and ^{235}U from the SF, but these isotopes are not included in the fresh fuel inventory. Previous experience in Atomic Energy of Canada Limited (AECL) studies has been that the poisoning effect of ^{234}U is largely counterbalanced by the extra reactivity available from the ^{235}U, so that the net effect of ignoring the higher uranium isotopes is likely to be small.

9.6.3. Scenario assumptions

The scenario assumptions are the same as for the 'ThPu' reactor introduction (Section 9.5.3), except for reactor introduction rates. In order to introduce the ThPuR ^{233}U recycling reactors, an initial stockpile of ^{233}U is required. This stockpile is built up by commissioning reactors of the ThPu type, as described in Section 9.5, mid-century. The share of ThPu reactors also serves to eliminate excess plutonium from the increasing LWR population, so is gradually increased throughout the century as well. When ^{233}U is available, as many ThPuR reactors as possible are commissioned.

9.6.4. Codes used for scenario calculations

DESAE 2.2, and the ORIGEN-S + Excel spreadsheet calculation (for decay heat) were used for these calculations as for the ThPu scenario (Section 9.5.3).

9.6.5. Results of analysis

The division of electrical capacity between the various reactor types in this scenario is shown in Fig. 9.53. The division of electrical capacity in 2130 is 130 GW(e) (4.4%), 610 GW(e) (20.8%) and 2195 GW(e) (74.8%). The availability of ^{233}U is the primary restriction on the number of ThPuRs which can be built (Fig. 9.54). Approximately 13% of the available fissile plutonium is discarded from the ThPu and ThPuR reprocessing, but this plutonium is highly denatured and unsuitable for a thermal reactor.

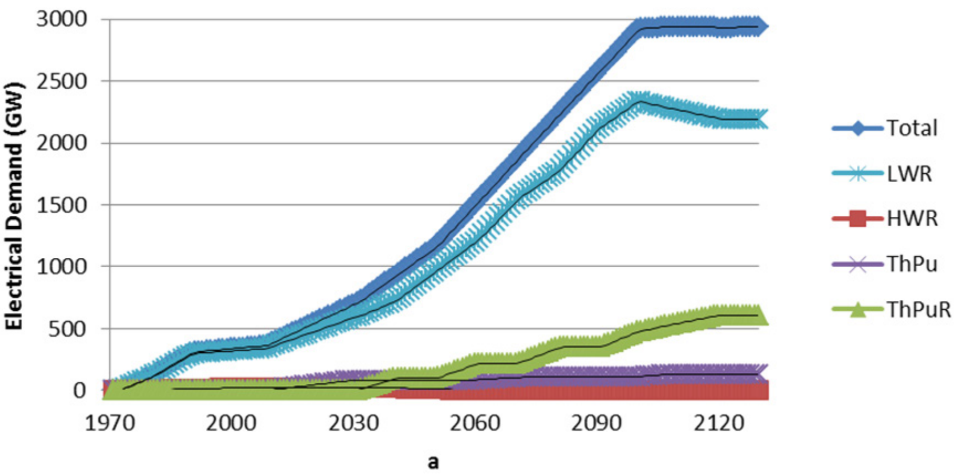

FIG. 9.53. Electrical capacity for the ThPuR variant of the BAU scenario (moderate case).

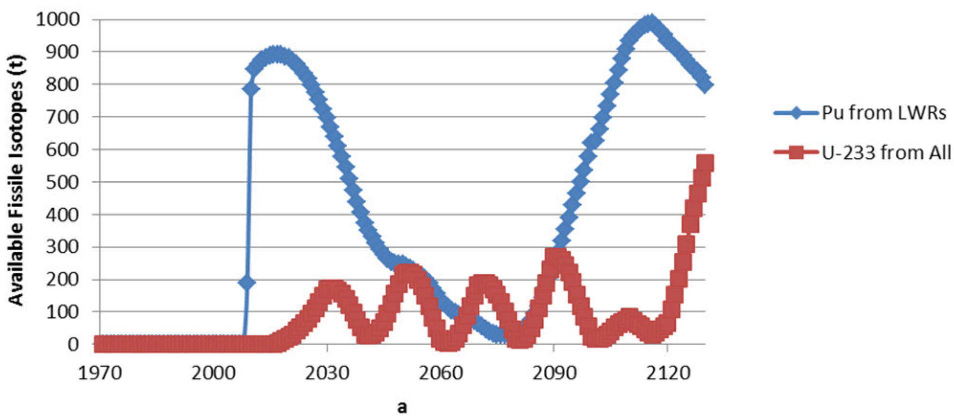

FIG. 9.54. Fissile isotopes available after reprocessing.

The integrated natural uranium requirements (Figure 9.55) for this scenario (37.8×10^6 t at 2130) are reduced by 18.7% over the BAU case (46.4×10^6 t). This is over half the potential savings of a break-even FR (BR ~ 1.0) maximum introduction case. The ThPuR case extends the limit on easily mined natural uranium resources by about 10 years (relative to the BAU scenario, see the lower construction line in Fig. 9.55). When unconventional natural uranium resources (principally phosphates) are included, the exhaustion of resources is extended by about 15 years (from 2115 to 2130, see the higher construction line in Fig. 9.55).

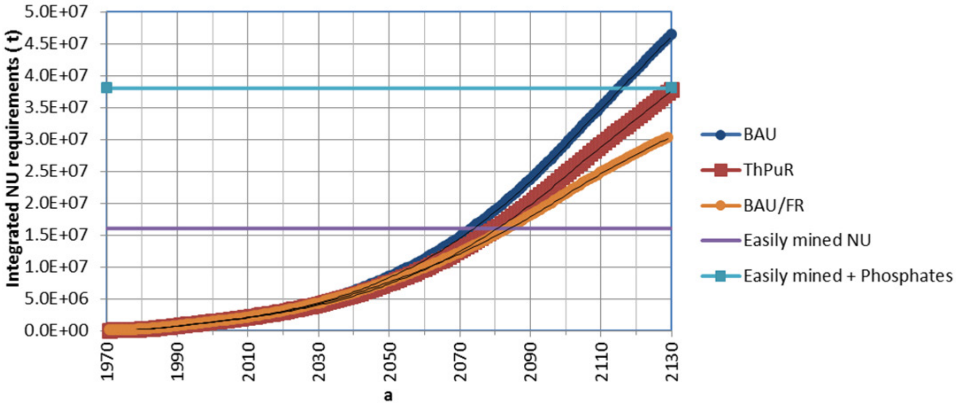

FIG. 9.55. Integrated natural uranium requirements for ThPuR versus the BAU or BAU–FR scenario (high case).

Actinide decay heat was calculated using the method described in Section 9.5.4 and source terms which were the amounts of material left in the reprocessing facilities at the end of the scenario (2130). This method, thus, excludes from the calculation all fuel currently in the reactors or in the reactor cooling pools. The decay heat from spent HWR fuel was included separately.

The total actinide decay heat was found to be ~0.182 GW(th) (Fig. 9.56), slightly more than the 0.159 GW(th) in the ThPu scenario, but still much less than the 0.304 GW(th) of the BAU scenario (Fig. 9.57).

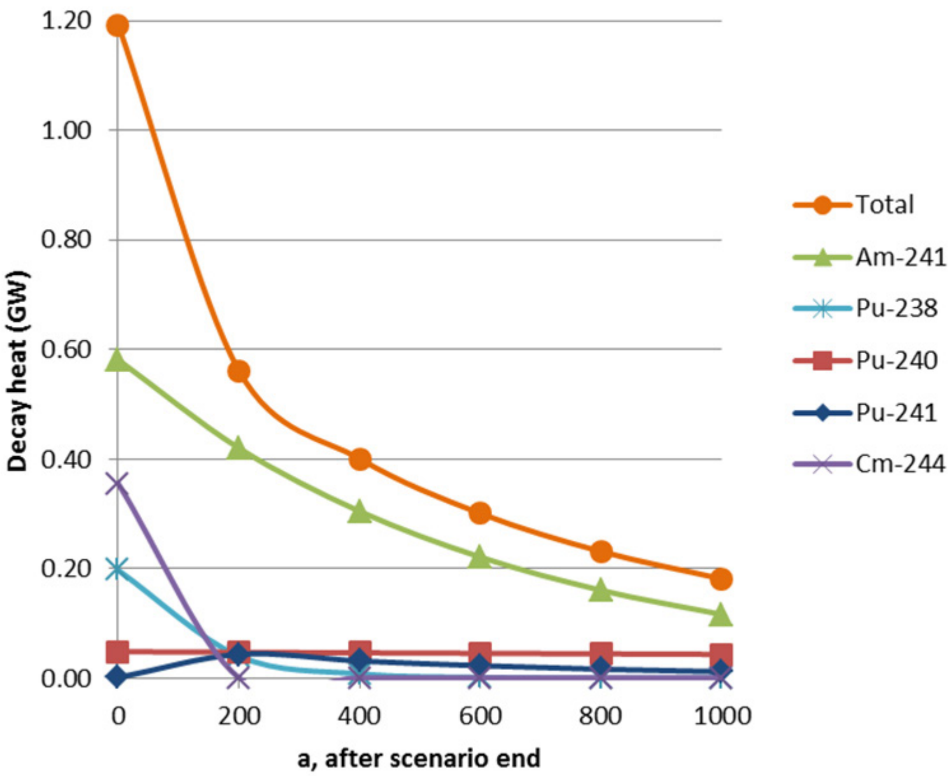

FIG. 9.56. Actinide decay heat.

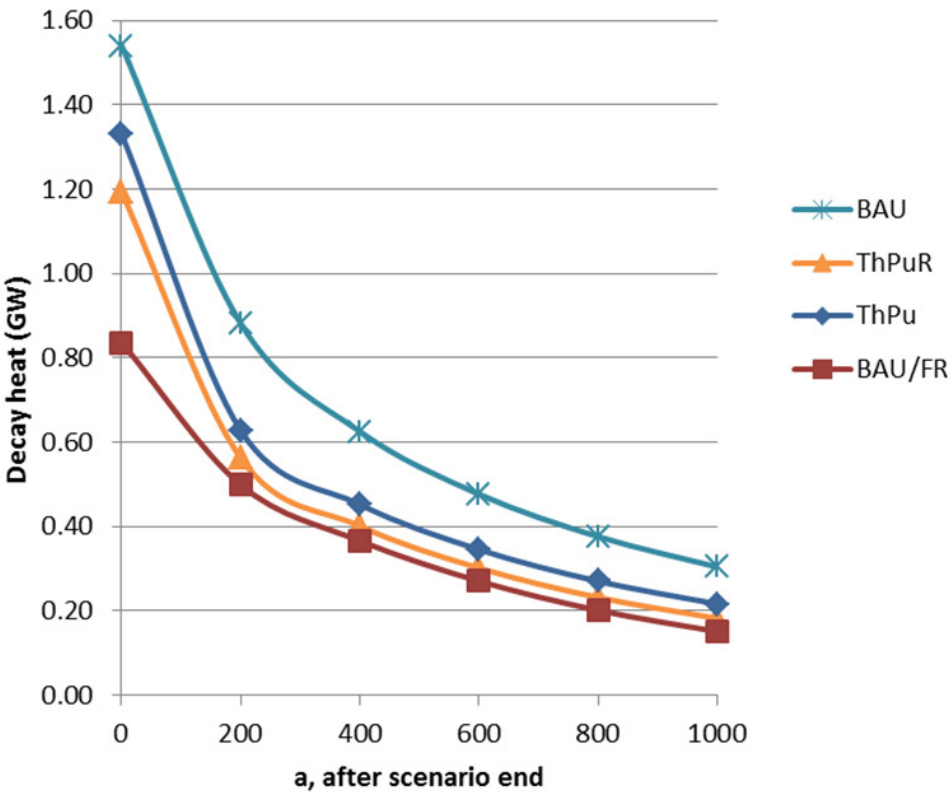

FIG. 9.57. Actinide decay heat versus the BAU and BAU break-even FR scenarios.

9.6.6. Conclusion

Building recycling thorium–plutonium reactors fuelled by plutonium from LWR SNF achieves more than half the natural uranium savings of the FR break-even case (18.7 versus 34.3%) and more than 80% of the reduction in SNF decay heat (0.182 versus 0.159 GW(th), where the BAU case has 0.304 GW(th)). It, therefore, represents a comparable technology over this time period which may be preferred by some jurisdictions which do not want to invest in the required infrastructure for FRs. However, assessment of other KIs and EPs should also be performed to provide a more informed comparison.

A significant continuing supply of natural uranium is required in this scenario, so it does not represent a solution to a natural uranium resource issue by itself. For long term resource management, a third step (after the once-through reactors and the ^{233}U recycling reactors) would be required — the establishment of the self-sufficient equilibrium thorium fuel cycle.

9.7. SOME ECONOMIC ISSUES OF DEVELOPMENT AND DEPLOYMENT OF FUTURE GLOBAL NUCLEAR ENERGY SYSTEMS

9.7.1. Economic motivation and challenges for introduction of the closed nuclear fuel cycle as an innovative component of the future nuclear energy system

Economics is an important dimension in the INPRO methodology. At the same time, the rather wide range of relative ‘margin’ between the expected costs of nuclear energy from national NESs as a consequence of differences in the input data was the main cause of the decision of the GAINS participants to address economic issues in a very preliminary manner. This section presents a few results of the study on economic aspects of the transition from the current NES overwhelmingly based on the open NFC with thermal reactors to an NES with substantial inclusion of the CNFC and a balanced fleet of thermal reactors and FRs.

The issues of economic motivation for introduction of innovative components of NESs and challenges associated with the transition period to the future NES are being discussed in many analytical studies related to the future of nuclear power. The purpose of this section is to consider how economic motivation and challenges for the introduction of a CNFC with FRs as an innovative component of NES might affect the architecture of global nuclear power.

The economic studies in GAINS were carried out with the use of the IAEA's energy model MESSAGE [9.7]. The model includes all of the major elements of the NES, both at the front end and at the back end. The major facilities for uranium conversion, enrichment and fuel fabrication at the front end, and for SF storage, reprocessing, and stockpiling of Pu, MAs and FPs can be modelled using their specific technical characteristics and associated costs. The nuclear reactor technologies that are considered for the entire twenty-first century include existing thermal reactors, advanced thermal reactors and FRs to be developed in the future together with their fuel cycles. Relevant data on the cost characteristics used in the economic calculations were taken from the IAEA databases and other reported information sources.

The results of the generating cost calculations (key indicator KI-9 in Table 4.1) have demonstrated the opportunity to decrease the dependence of the nuclear power plant electricity cost on the growth in the price of uranium in all GAINS strategy groups of the heterogeneous model under the condition that a significant share of the reactors of the global NES would use MOX fuel. This development could postpone the use of costly uranium until the end of the century, thus keeping a low steady cost of uranium fuel in the global system. The models used in the study made it possible to quantitatively evaluate the linkage between measures on uranium savings and the cost of uranium fuel in the global NES and to compare the results of calculations of the generation cost in GAINS scenarios with the results of other studies in the area.

Contrary to the assessment of the levelized generating cost, there are not many studies related to the analysis of the cost of different R&D programmes and the possible impact of their implementation on the future architecture of the global NES. However, some estimations on the R&D investments in the development of FRs in different countries and globally have appeared lately [9.8]. The role of R&D investments necessary for the development and deployment of the innovative component of the global NES (KI-10 in Table 4.1) was examined in GAINS in cooperation with experts from the IAEA [9.9] using the example of SFR technology. Some results of the analysis are briefly discussed in this section.

Most improvements to current nuclear power technologies are being financed by the nuclear industry, for example, the revolutionary pressurized water reactor by Areva, the advanced boiling water reactor by General Electric, the advanced pressure water reactor by Westinghouse, the WWER by Rosenergoatom and the advanced CANDU by AECL. The RD&D investments required for developing innovative components of NESs are beyond the R&D budgets of the vendors. The level of cost that is needed prior to commercial introduction of innovative components depends on the novelty and scope of the technological advancements to be introduced to the designs. It is a long way from the pre-conceptual studies until the first of a kind commercial unit has been constructed and operated.

Historically, most of the RD&D for nuclear technologies was carried out by individual countries with direct or indirect funding from the respective national governments. In recent years, there are many initiatives for multilateral cooperation on regional and global levels for the development and deployment of large innovative systems. Some evaluations on the scale of investments in nuclear innovation were published lately. It is indicated in Ref. [9.8] that the worldwide investments already made in the development and demonstration of SFR technology exceed \$50 billion. The cost of the USSR RD&D programme for the development of an SFR is estimated by Russian Federation experts as \$12 billion. Yet, the technology will require additional significant efforts and funding to reach the commercialization stage. Concepts such as the gas cooled or lead cooled FR will require RD&D with costs which will likely be comparable to the SFR programmes. The case presented in this section refers to the transition from the current nuclear system to a system based on thermal and FRs with the inclusion of a CNFC. However, the approach could be applicable to NESs of different kinds.

The example below demonstrates how the scale of investments needed to develop an innovative component of an NES and the size of the market for it affect the pay-back period for the component. It is assumed that RD&D investments for developing FRs and associated fuel cycle technologies would range between \$10 billion and \$40 billion, and that the FR can be built at a cost of \$2000/kW. Figure 9.58 shows the impact of the market size, i.e. the new electricity generation capacity based on the INS, on the pay-back period for the RD&D investments.

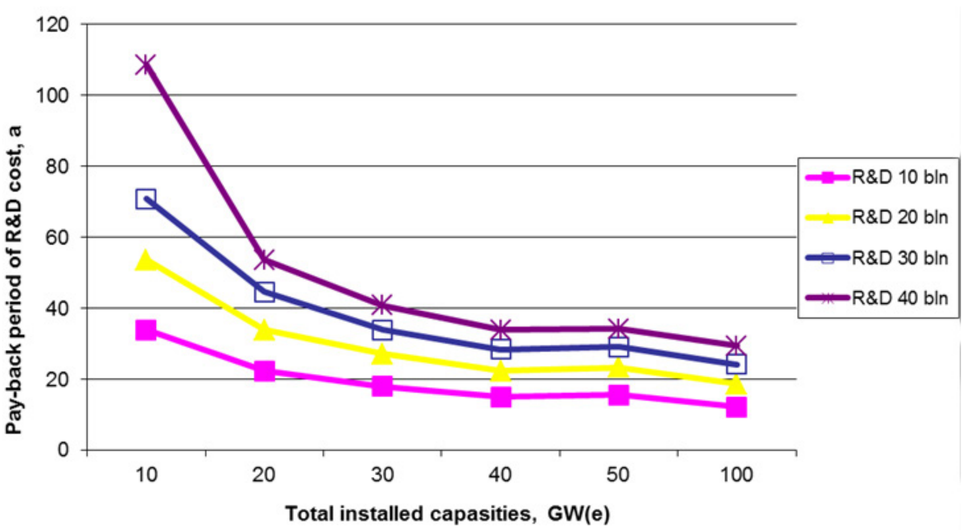

FIG. 9.58. Pay-back period for RD&D investments for the INS of different installed capacities.

In these calculations, the new capacity is commissioned at a rate of 1 GW(e)/a. The investment cost was assumed to be $2000/kW(e) and the construction time of power units 5 years. The annual return of the construction investment was calculated with 5% interest. It is also assumed that the funds provided for the implementation of RD&D have to be included in the electricity cost generated by the system and returned with zero interest.

It can be seen that only if the new capacity based on CNFC–FRs is 30 GW or more that the RD&D investments are justified in a reasonable term. They can be recovered within 20 years if the investments are $10 billion and within 40 years if they are about $40 billion. Conversely, if the market size is only around 10 GW(e), then the RD&D expenditures of about $40 billion are not justified since they will not be recovered even in a century.

This example demonstrates that for small programmes of CNFC–FR deployment the expected economic benefits from their introduction do not match the level of investment necessary for development, demonstration and deployment. Only a few countries in the world with large nuclear energy programmes (30 GW(e) or more) can bear the burden of CNFC–FR technology development. This also indicates the need for global and regional cooperation.

The above example, though simple, gives an idea of how the level of market size and the scale of RD&D investment are related. To examine the implications associated with the transition cost, a detailed analysis at the system level is needed to understand the effect of scale, timing and pace of RD&D efforts, and the buildup of new nuclear capacity based on CNFC–FRs.

9.7.2. Conclusion

The transition from the current global nuclear system that uses almost exclusively thermal reactors and once-through NFCs to an innovative system based on the CNFC with thermal reactors and FRs implies implementation of costly RD&D programmes that most likely cannot be funded by the industry and need significant governmental financial support.

For a timely return on the RD&D investments, the innovative components should be provided with a market of a certain capacity. This requirement adds specific features to the analysis of the NES economic viability, linking the issue with the ascertainment of the available room for the introduction of new INS capacity in national and global markets as well as the readiness of the markets to assimilate the advanced system.

The progress in multilateral arrangements for a global NFC can be important leverage in advancing the CNFC–FR towards early commercialization and increasing influence of the system in enhancing the sustainability of nuclear power. MNAs will help technology holders to expand their markets and, thus, provide an early return on high RD&D costs. Technology users will significantly reduce their investments in RD&D and the NFC, thus decreasing associated economic risks. Both will benefit from the economies of scale and the longer life of conventional uranium resources.

9.8. SCENARIO STUDIES ON SAFEGUARDS EFFICIENCY IMPROVEMENT

9.8.1. Background

The minimization of the total cost of the extrinsic measures implemented to increase proliferation resistance is one of the requirements of the INPRO methodology [9.10]. Safeguards is an extrinsic measure comprising legal agreements between the party having authority over the NES and a verification or control authority, binding obligations on both parties and verification using, inter alia, on-site inspections. In this report, 'safeguards' will refer to IAEA safeguards implemented under a safeguards agreement between a State and the IAEA.

The IAEA safeguards system plays an indispensable role in ensuring that nuclear material and activities are in peaceful use worldwide. The resources required for safeguards implementation grew steadily until about 1981 and have been all but constant since then. Prior to 1981, these resources increased both in absolute monetary terms and in the percentage of the IAEA regular budget allocated for nuclear verification (Fig. 9.59). Starting with 7.0% in 1968, the budgeted amount reached almost 40% by 2009. The number of facilities under safeguards increased during this period from 105 to roughly 900. Comparing the relative growth of the number of facilities under safeguards and the increase in the cost of safeguards (Fig. 9.59), a similarity in tendencies could be revealed: the growth in the number of facilities under safeguards triggered the growth in the percentage of safeguards costs within the IAEA regular budget.

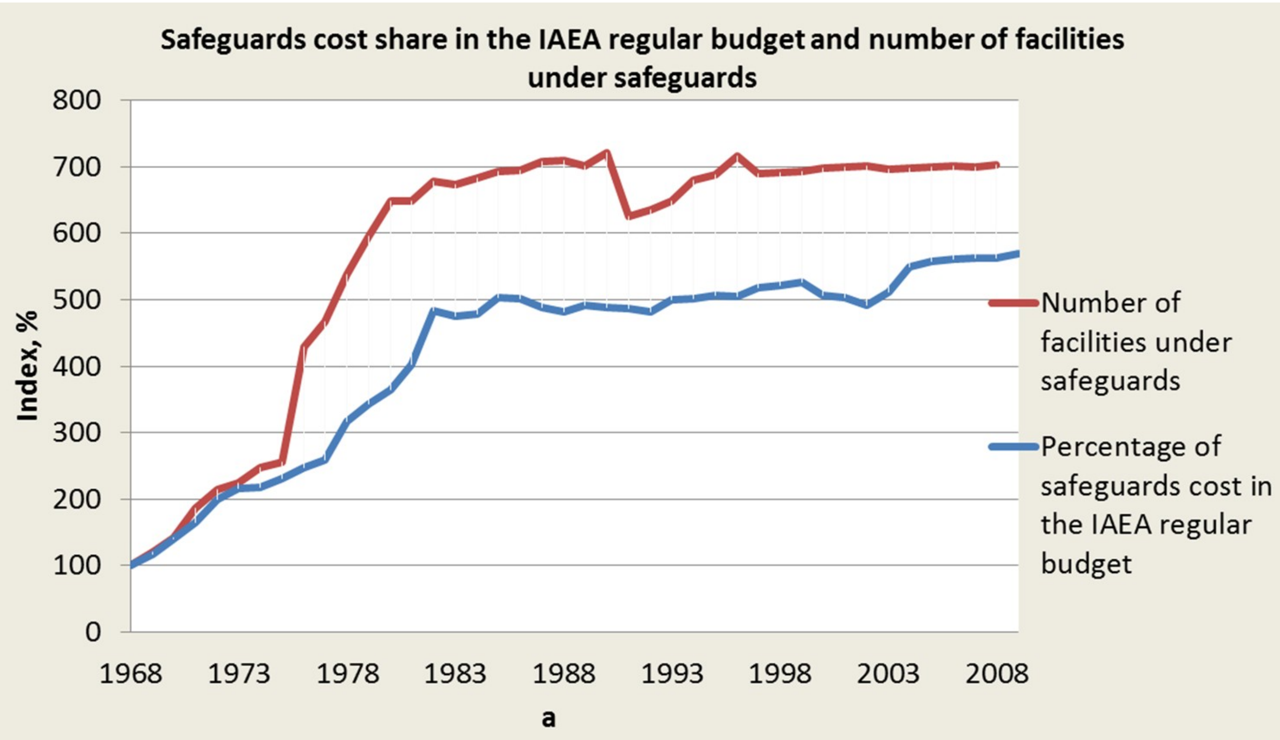

FIG. 9.59. Relative growth of safeguards costs and the number of facilities under safeguards.

According to the GAINS scenarios (Section 5), nuclear power demand will increase from the current ~300 to 2500 GW(e) (moderate case) or 5000 GW(e) (high case) by the end of the century. In order to increase the efficiency for safeguards implementation, various improvement measures should be taken.

The following measures for safeguards efficiency improvement were considered during this study:

— Integrated safeguards;
— Technical measures;
— MNAs.

The implementation of the first two measures is actively being carried out by the IAEA. The MNA is a form of arrangement of the NES global architecture. It is being built by the joint efforts of IAEA Member States and the IAEA. The scale of the current savings by the first two measures has briefly been analysed utilizing information available from the IAEA; MNA implementation development was modelled involving GAINS scenarios in order to assess potential safeguards cost reductions by cooperation and establishment of international NFC facilities.

9.8.2. Safeguards efficiency improvement by integrated safeguards

Once a State has brought both a comprehensive safeguards agreement and an additional protocol into force and a broader conclusion of the absence of undeclared nuclear material and activities has been drawn for this State, integrated safeguards could be applied. Integrated safeguards allow effectiveness to be maintained at the same level (or even strengthened) while improving the efficiency of nuclear verification activities, mainly by means of random inspections. In 2011, integrated safeguards were implemented for the whole year in 49 States. For these States, the IAEA utilized 6994 calendar days in the field for verification. 1850 person-days of inspection (i.e. the effort spent by one inspector at one facility for one day or less) were saved by virtue of integrated safeguards implementation. Integrated safeguards decreased the verification effort by roughly 21%. While there has been a reduction of inspection effort in the field, the verification and evaluation activities at the headquarters that enabled the reductions have comparably increased, so that the expenditure on inspection effort has remained largely constant.

9.8.3. Safeguards efficiency improvement by technical measures

Remote monitoring and safeguards by design are considered to be the most promising technical measures in terms of safeguards efficiency improvement. In 2011, 271 safeguards systems operating in remote monitoring mode — surveillance and radiation monitoring with remote data transmission — were implemented at 271 facilities in 21 States (compared with 258 facilities in 19 States for 2010 and 193 facilities in 17 States for 2009). Estimating the savings of this measure is difficult, but as the number of remote monitoring systems increase annually, so do the number of person-days of inspection saved.

Safeguards by design imply integration of safeguards into the design process of a nuclear facility from the initial planning (as early as possible). The concept is now under development and sound data on potential savings by this means are not available yet.

9.8.4. Safeguards efficiency improvement by multilateral nuclear approaches

Multilateral approaches to the NFC could provide the international community with a variety of significant benefits, eliminating essential concerns over nuclear power deployment. On the one hand, the idea is not novel and the global nuclear market already has different examples of multinational nuclear facilities; on the other hand, the share of these facilities is relatively low (especially comparing with the potential of the concept) and is confined only to the front end. This part of the study is dedicated to the deeper investigation of the potential benefits for safeguards costs which could be derived from boosting the multilateral dimension in the NFC.

At this stage, the study has focused on the analysis of MNA application to the following types of NFC facilities:

- Conversion;
- Fuel fabrication;
- Enrichment;
- Reprocessing;
- MOX fuel fabrication.

Two global NESs were used in the modelling: a BAU scenario based on current and advanced thermal reactors, the BAU+ scenario and the BAU+ scenario with the introduction of FRs (BAU + FR). To reflect the implications of MNA introduction, separate and synergistic options for the architecture of the NES were used:

— Separate option — nationally driven world (NDW): all States develop indigenous NFC facilities without any international cooperation — separate world;
— Synergistic option — globally driven world (GDW): maximum of international cooperation and number of multilateral facilities — synergistic world.

In order to take into account every State specific nuclear power development programme, nuclear energy demand of the major 70 States for three milestones — 2030, 2060, 2100 — has been calculated combining GAINS total nuclear energy demands (Section 5) and WNA tendencies of State-specific nuclear power deployment.

The main modelling assumptions are:

— NDW: Indigenous NFC facilities are introduced once a State reaches 10 GW(e) of nuclear power generation capacity and then according to the national requirements;
— NDW: The capacity of one indigenous NFC facility depends on the national nuclear power generation capacity with a step of 10 GW(e);
— GDW: MNA facilities are established to meet global requirements with a capacity of one plant equal to the capacity of the biggest national NFC facility existing in the world for a given year.

According to the GAINS framework, the numbers of NFC facilities in an NDW (including a breakdown for the three GAINS groups) and GDW have been calculated.

Figure 9.60 shows the number of enrichment plants in the BAU+ moderate scenario obtained using the developed framework.

According to the graph, maximum exploitation of the MNA potential could lead to a reduction by three quarters of the number of enrichment plants in the world by 2100. Whereas following the NDW scenario the international community will face roughly 100 enrichment facilities by the end of the century, with a half of them in GAINS group 3 (newcomer's group).

The results of the modelling BAU+FR scenario for reprocessing and enrichment plants are shown in Figs 9.61 and 9.62. The model suggests that in the case of fully deploying a MNA the number of reprocessing plants could be reduced by up to three times and the number of enrichment plants in a closed fuel cycle could be confined at the same level from 2030 up to 2100. This is attained by optimizing the NFC facility capacity at the global level by means of replacing numerous small national installations with high capacity installations at international centres.

Using the numbers of NFC facilities and the costs for safeguards verification activities [9.11], converted to 2008 prices in euros, the safeguards costs for NFC facilities have been calculated. Figure 9.63 shows the results for the BAU+ (once-through NFC) and the BAU+FR (closed NFC) scenarios in an NDW and a GDW. Taking certain safeguards costs in 2008 as 100%, a percentage was applied which allows analysis of the effectiveness of MNA implementation as a measure of improved safeguards efficiency.

The fact that tendencies for significant safeguards cost reduction were revealed by the model in both scenarios (regardless of the type of NFC) is particularly noteworthy and provides an argument for further investigation of MNA and future safeguards costs.

It can be concluded that multilateral approaches could effectively supplement the IAEA's efforts to reduce safeguards costs. GAINS identified a need for more comprehensive examination of the issue with the use of scenario modelling, including:

— Consideration of the other types of facility (including power reactors) and different strategies on SNF treatment.
— Introduction of additional parameters: e.g. costs of equipment and equipment maintenance, sample logistics and analysis.
— Taking specific MNA features into account: e.g. detailed analysis of scale effect, impact of international management and location in States with integrated safeguards, possible role of IAEA regional offices.
— Expansion, based on INPRO methodology, of the framework from assessing safeguards efficiency to assessing both safeguards efficiency and effectiveness.

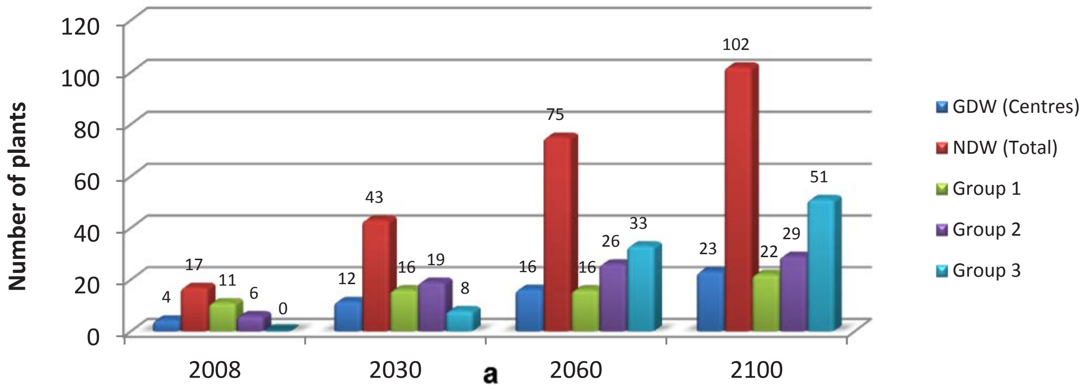

FIG. 9.60. The number of enrichment plants in the BAU+ moderate scenario based on thermal reactors.

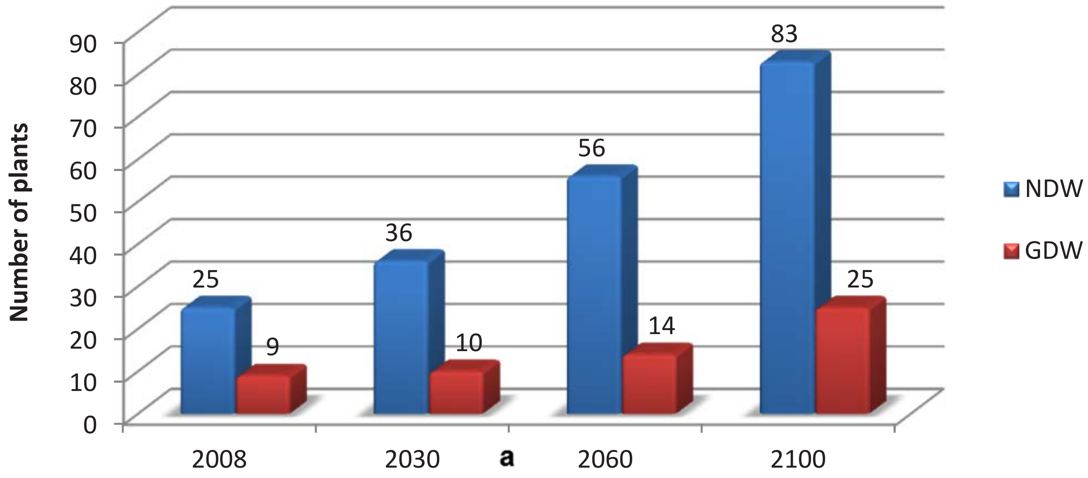

FIG. 9.61. Number of reprocessing plants in a GDW and an NDW BAU+FR scenario, moderate case.

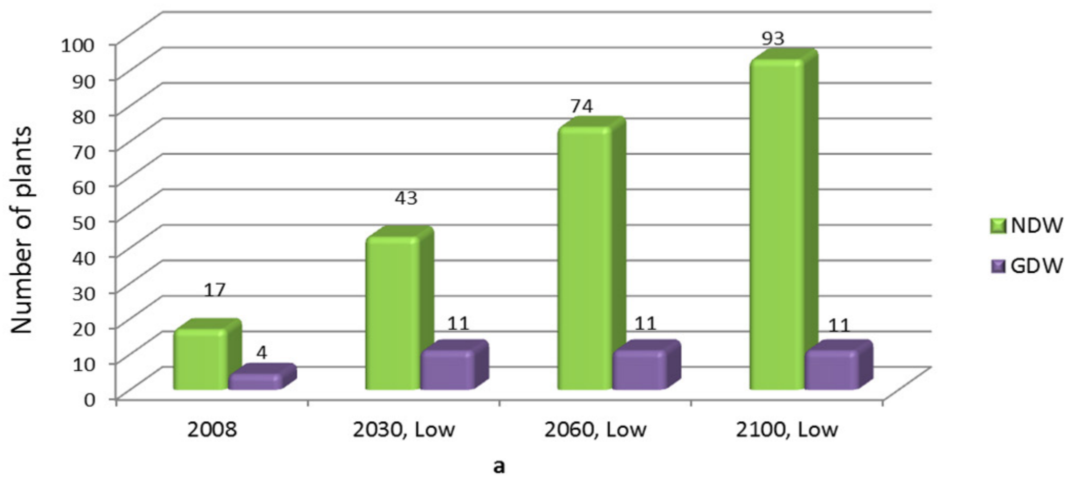

FIG. 9.62. Number of enrichment plants in a GDW and an NDW BAU+FR scenario, moderate case

9.8.5. Conclusion

Safeguards costs could essentially affect the IAEA regular budget should the global nuclear architecture expand. The work on safeguards efficiency improvement by means of integrated safeguards and technical measures is ongoing.

Multilateral approaches associated with the GAINS synergistic model could effectively supplement IAEA efforts to reduce safeguards costs.

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10. ASSESSMENT OF ANALYTICAL TOOL CAPABILITIES AND IDENTIFIED NEEDS

10.1. BACKGROUND

A number of NFC analysis codes have been developed by various organizations in many countries. The direct purpose of the codes is simply to calculate the mass flow, process throughput and work units on the operation of an NES which consists of various types of nuclear power plants and various types of fuel cycle facilities under many kinds of fuel cycle conditions. On the other hand, the principal objectives of analytical works using the codes vary widely, and the primary feature and the necessary input and output parameters are different from code to code. The participants and the IAEA have chosen their own tools to perform their analytical works for the GAINS project.

This section introduces and compares the features of the analysis codes owned by the IAEA and participants, and provides cross-check analysis results for a few samples of future NFC scenarios.

10.2. SELECTED TOOLS AND IMPROVEMENT FOR MODELLING NUCLEAR ENERGY SYSTEMS

10.2.1. Dynamics of energy systems — atomic energy

The dynamics of energy system — atomic energy (DESAE) code was developed by the United Knowledge Group in the Russian Federation as an INPRO task [10.1]. DESAE calculates the resources, both financial and material, required for a given combination of reactors to meet a specified supply of nuclear energy as a function of time (see Fig. 10.1). Thus, the user can study the practicality of a proposed system and material balances such as uranium demand as a function of time, waste generation and plutonium recycling. The code is at an early stage of development.

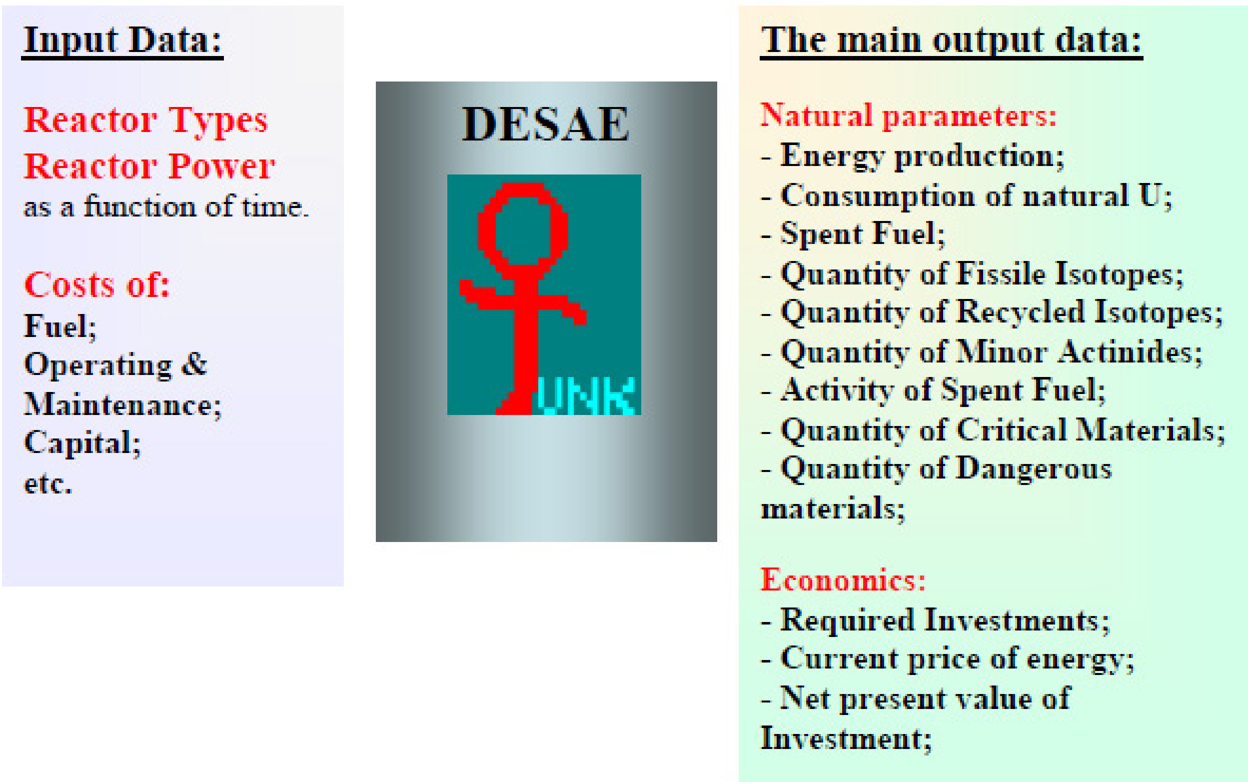

FIG. 10.1. Main input and output data of the DESAE code.

DESAE is an interactive code. The user specifies a given demand for nuclear energy — at present only nuclear electricity can be modelled — and the combination of reactor types that will be used to supply this energy, the fuel cycles to be used and the costs (e.g. overnight construction cost, fuel cost, operating costs) for each. The code then calculates a variety of parameters in near real time such as the consumption of natural uranium as a function of time, quantities of SF and other material such as actinides and recycled material, the consumption of critical material such as zirconium, the investment required and the cost of energy. The user can then seek to optimize the NES by varying the mix of reactor types and fuel cycles. The code does not use an optimization function but does provide information to the user to assist the user in the choice of alternatives.

The code performs material flow analysis based on a user defined deployment scenario of reactors and fuel cycle facilities. The code does not perform burnup or core management calculations but bases the calculations on tabled fresh and SF compositions provided by the user (databases with these characteristics are available). The tabled fuel characteristics include data for equilibrium and startup core compositions for various reactor types. The fuel composition is followed for 17 isotopes, i.e. ^{232}Th, ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{242m}Am, ^{244}Cm, ^{129}I and ^{99}Tc, with one additional group for the other FPs. The code also calculates integral and differential consumption of different material in an INS, e.g. Fe, Cu, Al and Zr.

DESAE allows modelling of seven reactor types in parallel in one simulation with all of them having any chain of fuel exchange with each of the other reactors. These fuel exchange paths need to be defined by the user. However, the fuel cycle representation in DESAE is done with only four fuel cycle facilities, without yet tracing losses in these facilities. The activity and radiotoxicity of SF is calculated but repository needs are only defined by the volume of material to be stored. Proliferation risk is assumed to be dependent on the volume of so-called relevant material, i.e. fissile plutonium.

The economic analysis within DESAE calculates the levelized cost of energy based on the capital costs for reactors and NFC facilities, the operation and maintenance costs, and the calculated fuel cycle costs as well as total investment needs to deploy a certain NES scenario. DESAE is under continuous development and is the subject of testing programmes within the INPRO community. Benchmark validation with other codes has been proposed, as well as links with the macro-economic energy market analysis code MESSAGE. The DESAE code was developed using the MATLAB-software. The code is available to all members of INPRO.

10.2.2. Model for energy supply systems and their general environmental impacts

The MESSAGE code is the most versatile and most sophisticated of all of the codes available at the IAEA and, in principle, could fulfil all the objectives of the rest of the IAEA code family of energy planning tools described below.

MESSAGE [10.2] is designed to formulate and evaluate alternative energy supply strategies consonant with user-defined constraints on new investment limits, market penetration rates for new technologies, fuel availability and trade, environmental emissions, etc. It was originally developed at the International Institute for Applied Systems Analysis. The IAEA acquired the latest version of the model and added a user interface to facilitate its applications. The underlying principle of the model is the optimization of an objective function under a set of constraints. The backbone of MESSAGE is the technical description of the modelled system. This includes the definition of the categories of energy forms considered (e.g. primary energy, final energy, useful energy) and the energy forms (commodities) actually used (e.g. coal or district heat), as well as energy services (e.g. useful space heat provided by energy).

Technologies are defined by their inputs and outputs, their efficiency and the degree of variability if more than one input or output exists, for example, the possible production patterns of a refinery or a pass-out-turbine. These energy carriers and technologies are combined to construct so-called energy chains, where the energy flows from supply to demand. The definitional limitations on supplying energy carriers are that they can belong to any category except useful energy, they have to be chosen in light of the actual problem, and limits on availability inside the region/area and on import possibilities have to be specified. The technical system provides the basic set of constraints to the model, together with demand, that is exogenous to the model. Demand must be met by the energy flowing from domestic resources and from imports through the modelled energy chain(s) (see Fig. 10.2). The model takes into account existing installations, their vintage and their retirement at the end of their useful life. During the optimization process, this determines the need to construct new capacity for various technologies. Knowing new capacity requirements permits the user to assess the effects of system growth on the economy. The

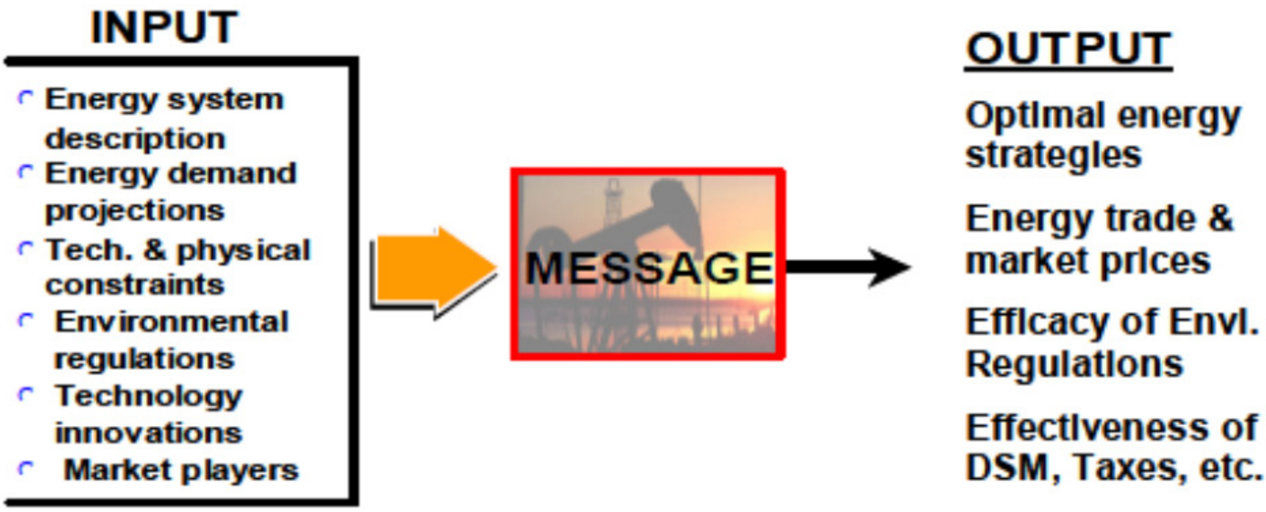

FIG. 10.2. Selected MESSAGE applications.

investment requirements can be distributed over the construction time of the plant and can be subdivided into different categories to more accurately reflect the requirements from significant industrial and commercial sectors. The requirements for basic material and for non-energy inputs during construction and operation of a plant can also be accounted for, by tracing their flow from the relevant originating industries either in monetary terms or in physical units. For some energy carriers, ensuring timely availability entails considerable cost and management effort. Electricity has to be provided by the utility at exactly the same time it is consumed. MESSAGE simulates this situation by subdividing each year into an optional number of so-called 'load regions.' The parts of the year can be aggregated into one load region according to different criteria, for example, sorted according to power requirements or aggregation of typical consumption patterns (summer/winter, day/night). The latter (semi-ordered) load representation creates the opportunity to model energy storage as the transfer of energy (e.g. from night to day, or from summer to winter). Including a load curve further improves the representation of power requirements and the use of different types of power plants.

MESSAGE is capable of simulating different NFC models with different reactor types and fuel types including non-existing fuel types (i.e. fuel with MA content). The modelling of a nuclear power system is quite flexible in MESSAGE, and the users can decide what components they would like to include in the model.

Each component can be represented in MESSAGE with necessary details such as: first loading and final unloading in reactors, cooling time for SF discharged from a reactor, lag and lead time for processes, losses and so on.

In order to demonstrate the capabilities of MESSAGE, three NFC options were simulated and a result set was obtained. They are NESs based on:

(a) LWRs and HWRs with a once-through NFC;
(b) LWRs and HWRs with a partly CNFC with plutonium recycling in LWRs in the form of MOX fuel;
(c) Thermal reactors and FRs with multi-recycling of recovered Pu.

The major assumptions and boundary conditions for NESs as well as data for thermal and fast nuclear power plants including their respective fuel cycles used in these examples were developed for the INPRO/GAINS project.

10.2.3. NFCSS (formerly VISTA)

NFCSS (formerly VISTA) [10.3] (Fig. 10.3) was developed in the context of the IAEA's NFC and reactor strategies in 1997. It has since then been further developed by, for instance, inclusion of a simplified isotopic composition calculation program (CAIN). NFCSS calculates, by year over a long period, NFC requirements for several types of reactor. Calculations could be performed for a reactor, a reactor park in a country or a worldwide nuclear reactor park. Natural uranium, conversion, enrichment and fuel fabrication quantities are estimated.

Furthermore, the quantities and qualities (isotopic composition) of SF can be evaluated to let the user apply a recycling strategy if desired. The main assumption in the model is that it is possible to simulate the NFC by taking into account the evolution of different types of reactor with time, without the precision of using a reactor by reactor database. The reactor types taken into consideration in NFCSS are PWRs, BWRs, PHWRs, advanced gas cooled reactors, gas cooled reactors, graphite moderated fuel channel reactors (RBMKs) and WWERs.

The CAIN model in NFCSS allows tracking of a set of isotopes detailed enough to grasp the main decay chains of fuel isotopes, but not overly detailing the calculation by exclusion of nuclides with very short half-lives (<8 d) or that may be considered stable (>400 a) for the scenario period of interest. In total, 13 fuel isotopes are considered, i.e. ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am, ^{242m}Am, ^{242}Cm and ^{244}Cm.

The seven different reactor types are also treated in CAIN, allowing isotopic burnup calculations based on the Bateman equations. Benchmarking with other codes and experiments has been undertaken and provided very good results, i.e. 1–3% error margin for the main uranium and plutonium isotopes. NFCSS may be used to simulate various aspects of evolving NESs, i.e. varying reactor park compositions, changing fuel cycle options, changing reactor load factors, enrichment tails assay, and others, i.e. all typical reactor and fuel cycle facility characteristics may be set to change during a simulation which allows the user to run rather realistic scenarios.

NFCSS is currently MS-Excel based. Data input is reduced to a few basic parameters in order to let non-nuclear fuel specialists develop different energy scenarios. The calculation speed of the system is quick enough to enable making comparisons of different options in a considerably short time.

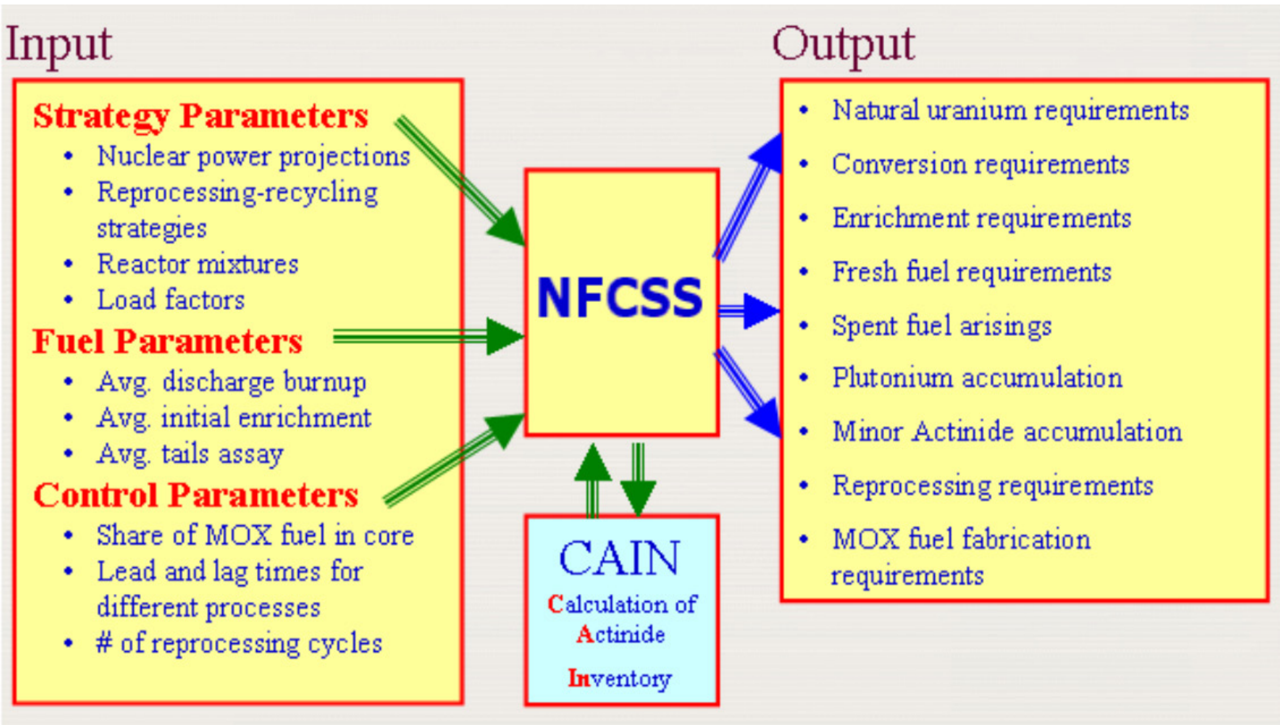

FIG. 10.3. Scheme of NFCSS.

NFCSS calculations are based on annual estimations, not cycle-wise estimations, so that the initial core loading and final core discharge must be estimated differently from the equilibrium loading and discharge. Initial core loading is calculated using the specific power density in fuel data. Although in real life fuel is loaded into the reactor in every operational cycle, for modelling purposes it is assumed that the fuel is loaded and discharged every year. Another important assumption is made in the calculation of the isotopic composition of discharged fuel. Although the composition of the initial core fuel is different from the equilibrium fuel, it is assumed that the composition of the discharged fuel is the same in the initial core and the equilibrium core. Similarly, although the average burnup level of the final core discharge is usually lower than the equilibrium fuel, it is assumed that the composition of the final core discharge is the same as that of the discharged fuel in the equilibrium core.

NFCSS is available to Member States via the IAEA.

10.2.4. COSI

COSI, developed by the French Alternative Energies and Atomic Energy Commission (CEA), France, is a code simulating a pool of nuclear electricity generating plants with their associated fuel cycle facilities [10.4]. The code has been designed to study various short, medium and long term options for the introduction of various types of nuclear reactor and for the use of associated nuclear material. COSI calculates the mass and the isotopic composition of all of the material, in each part of the nuclear park, at any time of the simulation period.

The main particularities of the COSI code are a detailed material flow accounting analysis of the NFC with the possibility to take into account the conversion, enrichment of natural and/or recovered uranium from reprocessing, and the fuel fabrication, as well as the irradiated stockpiles, reprocessing throughput and associated separated fissile/fertile material flows, wastes in the back end of the fuel cycle. All of the reactor plants and fuel cycle facilities are characterized by their unit and/or annual capacity, operating time, losses, date of commissioning, load factor, lifetime and other parameters making it possible to represent these INS components in sufficient detail for the material flow assessment.

COSI currently considers various reactor types (PWRs, SFRs, gas cooled FRs, HTRs), where the core management is user-defined according to time history, reload fuel management and type of fuel, i.e. UOX, MOX, MOX with enriched uranium, U-free fuel, MOX including MAs, HTR fuel and FR fuel. Very accurate physical models, benefiting from an extensive French benchmarking with experimental data, makes it possible to trace the material flows in great detail and accuracy, as well as to detail the isotopic composition for each batch of fuel. The isotopic composition follows a total of 250 isotopes including the main heavy nuclides (U, Pu, Am, Np, Cm, Th, Pa isotopes and their descendants) and the main FPs, making it possible to calculate the composition of SF, intermediate level waste and low level waste between 90 d of cooling until geological cooling time.

COSI allows the user to define boundaries or constraints in the operation of fuel cycle facilities, processing plant capacity in HMs and in Pu, and minimum cooling down period prior to SF processing, and the user may still define more details on fuel cycle facility operation practices, such as ‘first in’/‘first out’ management of SF and various types of dilution in reprocessing. COSI gives a detailed computation of the material balances including the computation of the Pu content or ^{235}U enrichment entering fuel fabrication based on the composition of the various batches of plutonium used, the origin of the uranium, the core management and the burnup. The computation of the fuel isotopic content in and out of the reactor at any time is given for each step in the fuel cycle. COSI can also assess an economic balance of reactors and fuel facilities, so as to obtain a levelized cost per kilowatt hour. The economic model in COSI can take account of the investment, exploitation and decommissioning costs for each of the reactors and fuel cycle facilities and their associated planning, the cost of natural material and the actualization rate.

COSI is available only through licence agreements with the CEA.

10.2.5. DANESS

DANESS (dynamic analysis of NES strategies), developed by the ANL, USA, is an integrated dynamic nuclear process model for the analysis of current and future NESs on a fuel batch, reactor and country, regional or even worldwide level [10.5]. The model allows simulating up to ten different reactor types and up to ten different fuel types in one simulation. Starting from today’s nuclear reactor park and fuel cycle situation, DANESS analyses energy-demand driven NES scenarios over time and allows the simulation of changing nuclear reactor parks and fuel cycle options. The NES may not only generate electricity but may also result in other energy vectors such as hydrogen and district heat. The energy demand is hereby given as an exogenously defined energy-demand scenario. New reactors are introduced based on the energy demand and the economic and technological ability to build new reactors. The technological development of reactors and fuel cycle facilities is modelled to simulate delays in the availability of technology.

Levelized fuel cycle costs are calculated for each nuclear fuel batch for each type of reactor over time and are combined with capital cost models to arrive at energy generation costs per reactor and, by aggregation, into a cost of energy for the whole NES. A utility sector and government policy model are implemented to simulate the decision making process for new generating assets and new fuel cycle options. The government policy model allows simulating different actions that a government may exert through, for instance, tax rates, regulation, R&D funding and others. Learning curve effects may be applied on different parameters in the simulation and may experience different learning rates. A (current) simple life cycle inventory model traces all losses in the NFC and also traces all of the main secondary material flows, such as water, energy and metals needed during the deployment, operation and decommissioning of the NES.

The material flow assessment part is based on tabled fresh and SF compositions where the isotopic composition of the fuel or high level waste is traced according to 67 isotopes, i.e. ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{237}U, ^{238}U, ^{236}Pu, ^{237}Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{244}Pu, ^{246}Pu, ^{235}Np, ^{236}Np, ^{237}Np, ^{241}Am, ^{242}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, ^{250}Cm, ^{227}Th, ^{228}Th, ^{229}Th, ^{230}Th, ^{232}Th ^{234}Th, ^{231}Pa, ^{233}Pa, ^{247}Bk, ^{249}Cf, ^{250}Cf, ^{251}Cf, ^{252}Cf, ^{253}Cf, ^{254}Cf, ^{253}Es, ^{254}Es, ^{255}Es, ^{223}Ra, ^{224}Ra, ^{225}Ra, ^{226}Ra, ^{228}Ra, ^{225}Ac, ^{227}Ac, ^{222}Rn, ^{60}Co, ^{90}Sr, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{147}Pm, ^{154}Eu, ^{155}Eu, ^{129}I, ^{99}Tc, a short lived and a long lived FP group. This decomposition allows DANESS to calculate correctly the isotopic evolution from discharge of irradiated fuel until geological disposal, whatever the fuel cycle option taken, and to calculate the decay heat to be evacuated from the repository and, thus, defining the repository space needs.

DANESS has been developed according to a flexible architecture which remains the same independent of the size of INS being assessed, i.e. from a single reactor up to multiregional NESs. A graphical user interface allows easy input and output of INS information and results, while a typical 100 year simulation only takes a few minutes on PCs. DANESS is available via licence agreements with the ANL in run-time and in a developer's version. Further developments are ongoing to include more detailed life cycle inventory models as well as to further the benchmarking of the code for various INS assessment cases.

10.2.6. VISION

10.2.6.1. Introduction and history

The verifiable fuel cycle simulation (VISION) model [10.6], developed by the Idaho National Laboratory, USA, is the primary dynamic simulation model used for system studies for the US fuel cycle technologies programme. VISION is used to perform dynamic scoping trade studies of alternative fuel cycles to obtain qualitative and quantitative comparisons of resource requirements, reactor types and mix, sequencing and timing, waste streams and geological repository requirements, with the capability to provide cost estimates of the levelized cost of electricity, and cash flow/funding requirements. The model provides a number of parameters for comparison of fuel cycle options, including repository capacity and performance, separation capacity, interim storage, energy recovery, proliferation resistance and safety. Specific waste parameters include waste mass, wasteform mass, wasteform volume, long term radiotoxicity and long term heat commitment to a repository.

In 2003, the advanced fuel cycle initiative systems analysis campaign reviewed current systems codes and adopted the existing DYMOND code developed by the ANL as the starting point for the systems code. The existing elemental mass flow model was significantly extended by a multi-laboratory group led by the Idaho National Laboratory, including tracking mass by isotope, addition of waste and economic modules, and inclusion of numerous new algorithms and indicators. The VISION name was adopted when the code exceeded the memory limits of the underlying software (Stella), and the model was reorganized and ported to PowerSim, which provided better support for the multiple arrays used to track isotopic data and other fuel cycle attributes. The current model is used by national laboratories and universities to conduct fuel cycle studies.

10.2.6.2. VISION capabilities

The full VISION implementation includes a series of Excel input and output files interacting with the core VISION model running in PowerSim. Input files include an extensive fuel library, model initialization information for the US legacy system and a primary user file that includes a large number of standard fuel cycle cases (which the user may modify) along with five user-definable cases. The core model is designed as a full systems dynamics model including feedback and logic that allows the model to forecast and make decisions to evolve all aspects of the nuclear infrastructure based on minimal high level direction such as an overall growth rate. The user may override this logic as desired via the input file, specifying everything from individual reactor builds to split fractions of material in separations. However, the underlying logic will still verify user direction and override that direction when physical constraints are exceeded (such as requesting more reactors that can be fuelled). Output Excel files provide an extensive range of parameters along with automatic graphing of both individual cases and comparisons between up to five cases. The users may develop their own output file containing specific graphs via linkage to the provided files. VISION is based on a modular view of a generic nuclear infrastructure, including reactors and front and back end facilities as depicted in Fig. 10.4. The emphasis is on the back end of the fuel cycle, as that is the focus of the US fuel cycle technologies programme's efforts.

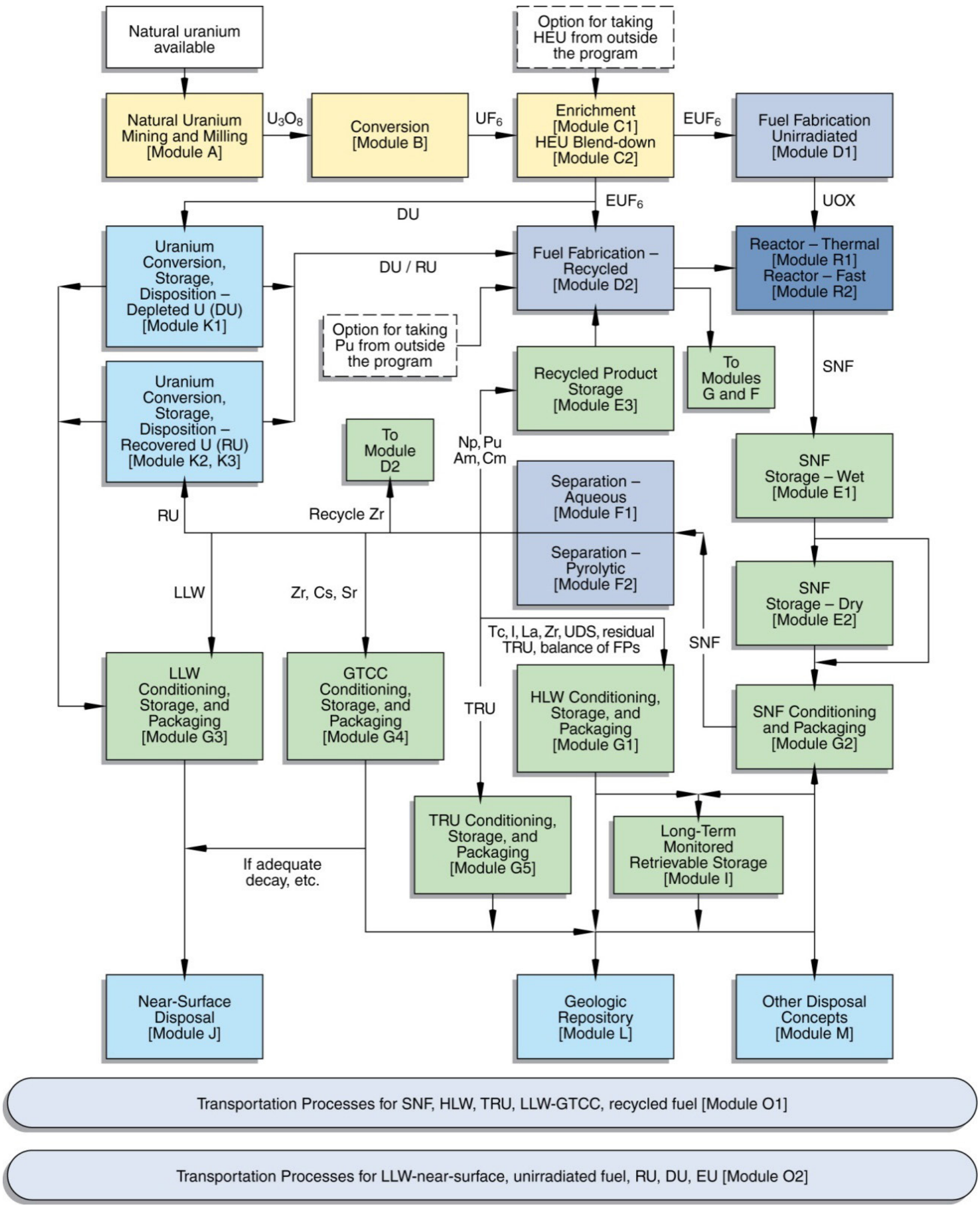

FIG. 10.4. Fuel cycle modules used in VISION.

The code logic is built around a series of backbone structures that model core features of the system. These include a fuel cycle mass flow structure, facility life cycle structures for reactors, storage, fabrication, separations and disposal facilities, and the reactor park. These structures provide quarterly and annual capacity and mass flow information that form the basis for the numerous parameters that are calculated for the user and for internal control of the model. The current version, VISION 3.4 [10.7] can model up to ten different reactor types simultaneously, supporting multigroup simulations. VISION does not explicitly model individual front end facilities other than fuel fabrication, as mining, conversion and enrichment are well established commercial ventures that operate based on market conditions, and modelling of total fleet capacity and throughputs are sufficient for calculating indicator values. Back end facilities are explicitly modelled, and the user can specify any level of control desired from only indicating when a technology is first available for use to specifying the specific capacity and construction timing of each facility. Up to ten different reprocessing facilities can be defined, each with up to eleven different product/waste streams with user defined split fractions for each waste stream for the 81 separate isotopes/isotope groups tracked by the code.

The isotopes and isotope groups tracked in VISION support calculation of multiple parameters and metrics associated with fuel value, mass flow (which includes some stable isotopes), separations and waste forms chemistry, and material properties such as decay heat and radiotoxicity. Activation products may be added in the future. The 81 isotopes/groups are:

(a) Uranium and thorium: ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{228}Th, ^{229}Th, ^{230}Th and ^{232}Th.
(b) Transuranics: ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{244}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, ^{250}Cm, ^{249}Bk, ^{249}Cf, ^{250}Cf, ^{251}Cf and ^{252}Cf.
(c) Fission and decay products : ^{3}H, ^{4}He, ^{14}C, C-other, ^{79}Se, ^{81}Kr, ^{85}Kr, inert gas other (Kr, Xe), Rb, ^{90}Sr (including ^{90}Y decay energy), Sr-other, ^{93}Zr (including ^{93m}Nb decay energy), ^{95}Zr (including ^{95m}Nb decay energy), Zr-other, ^{99}Tc, Tc-other, ^{106}Ru (including ^{106}Rh decay energy), ^{107}Pd, Mo-Ru-Rh-Pd-other, ^{113m}Cd, ^{125}Sb (including ^{125m}Te decay energy), ^{126}Sn (including ^{126m}Sb and ^{126}Sb decay energy), transition metals-other (^{9}Be, Li, Co-Se, Nb, Ag-Te), ^{129}I, halogen-other (Br, I), ^{134}Cs, ^{135}Cs, ^{137}Cs, Cs-other, Ba, ^{144}Ce (including ^{144m}Pr and ^{144}Pr decay energy), ^{147}Pm, ^{146}Sm, ^{147}Sm, ^{151}Sm, ^{154}Eu, ^{155}Eu, ^{166m}Ho, lanthanide-other (including Y), ^{206}Pb, ^{207}Pb, ^{208}Pb, ^{210}Pb, ^{209}Bi, ^{226}Ra, ^{228}Ra, ^{227}Ac and ^{231}Pa.

VISION uses a fuel cycle library reference to calculate transmutation performance for each reactor/fuel combination. The model adjusts the pre-calculated fuel 'recipes' to reflect the actual isotopic and fissile content of feed material. The model logic supports both one-tier and two-tier recycle systems, with different fuel, reactor and separation technologies for each stage (e.g. thermal reactors with first-run material, thermal reactors with recycled material, FRs).

The user can specify how many times fuel is recycled at each stage (from zero to unlimited). Fuel recycle compositions are modelled reflecting evolution of isotopic content through up to five cycles, at which point the fuel is considered to be at an equilibrium composition. At this time, startup and shutdown fuel loads for individual reactors are modelled explicitly for mass, but not for any difference in isotopic content versus normal fuel reloads. Discharged material is decayed to reflect isotopic changes while in storage prior to disposal or reuse.

10.2.7. TEPS

TEPS (tool for energy planning studies), developed by the Bhabha Atomic Research Centre, India, has been designed as a tool for energy planning studies as relevant for electricity producing nuclear reactor systems. The current version of TEPS can handle up to twenty reactor types and twenty material types, and systematic upgrading is in progress for accommodating more. Within a typical NES, there is no fundamental limit on the number of reactors that can be installed. The input to the code consists of metadata consisting of overall information such as number of reactor types involved in the analysis, amount of material to be tracked, the target energy demand curve, material availability information, the user-defined reactor deployment priority and prior installation history (so as to be able to start an analysis from some well defined point in time). The code additionally requires information on material requirements (annualized flows in the current version) for each of the nuclear reactor types that are going to be employed in the analysis. These flows typically contain the initial core requirements, annual reload requirements, annual discharges from the reactor and the quantity that will be discharged at the time of decommissioning of the reactor. Further, the reactor rated power and lifetime load factor, the typical time for the reusable components from the SF bay to the re-fabrication of fresh fuel, reprocessing losses, construction period and reactor lifetime are to be provided. The TEPS code uses the information to calculate the material flows needed to achieve the target demand curve. If it so happens that the target demand curve is non-commensurate with the material availability information provided, TEPS will automatically readjust and calculate the maximum installations possible.

Fundamentally, TEPS is an optimization code that tries to maximize reactor installations subject to material and priority constraints. The material flows provided as output by TEPS are completely user specific and non-radiological (decay of radioisotopes is not considered). The user specificity is useful. When, for example, users are interested in obtaining the MA load arising out of an NES, they may track such MAs that are essential to the analysis. Similarly, a user analysing the requirements for infrastructural material may choose to monitor the amount of steel needed for an NES. The material flows that need to be mandatorily tracked are for fissile and fertile material such as ^{233}U, ^{235}U, ^{238}U, plutonium and thorium. An interesting feature that is employed in TEPS is to account for lifetime requirements of material for a given reactor type. To consider this, TEPS employs a forward looking calculation that inhibits reactor installations that would result in negativity of material flow at some future point of time. It is emphasized here that the lifetime requirement of a reactor is not allocated at the outset, but merely the lifetime requirement is factored in as a constraint to the optimization problem. Additional constraints can also be posed to the code using the construction period limits by formulating either a cap on the maximum reactors that can be constructed in the period or on the total number of reactors of any one kind.

10.3. COMPARATIVE TABLE OF CAPABILITIES OF CODES

As described in Section 10.1, GAINS members have different fuel cycle analysis codes. According to the original purposes of their code development, the key functions and the key output parameters may be different.

Table 10.1 compares the major functions and the methods of member's codes from the viewpoint of the implementation of GAINS transition scenario studies.

10.4. CROSS-CHECK CALCULATION AND COMPARISON BETWEEN CODES

In the GAINS project, members have studied about many NFC scenarios by using their own analysis codes. Therefore, in order to ensure the credibility of their analysis results quantitatively, cross-check calculations were proposed and performed by some members and with different codes. The codes participating in one or more of the cross-check and the members who performed the calculations are:

— NFCSS: IAEA.
— MESSAGE: IAEA.
— DESAE: Belgium.
— FAMILY: Japan.
— DANESS: Republic of Korea.
— TEPS: India.

Some members did not participate in the cross-check activity, especially when their analysis codes had already been benchmarked against other codes. These past benchmarking efforts include:

— MIT [10.8]: which included CAFCA (MIT code), COSI, DANESS and VISION.
— NEA [10.9]: which included COSI, DANESS, DESAE, FAMILY and VISION.

The cross-check calculation was performed for three scenarios, two cases using a once-through fuel cycle with only thermal reactors and one plutonium recycle scenario based on thermal reactors and a break-even FR which has a BR of 1.0. The analysis conditions are the same as those discussed in Sections 6 and 8, with the cross-check calculations using the high case of nuclear power growth curve and the homogeneous global model.

TABLE 10.1. COMPARISON OF NFC CODES ON THE FUNCTION AND THE METHODS

		NFCSS (IAEA)	MESSAGE (IAEA)	DESAE2.2 (IAEA)	FAMILY (Japan)	COSI (France)	VISION (USA)	DANESS (USA)	TEPS (India)
Reactor and fuel data	Reactor and fuel database	+	–	+	+	+	+	+	–
	Depletion calculation (composition of discharged fuel)	+	–	–	+	+	–	–	–
	External refuelling data providing (fresh fuel, discharged fuel and after cooling)	+	+	+	+	+	+	+	+
	Decay correction (after discharge to reprocessing, certain period)	±	±	+	+	+	+	–	–
	Model of initial core loading and full core discharge at retirement	+	+	+	+	±	±	±	+
	Pu multi-recycle method (1) (Pu-total or Pu-fissile preservation)	Pu-total	Pu-total	Pu-fissile	Pu-fissile	TRU-total Pu-total Pu-fissile ^{239}Pu, etc.	TRU-total Pu-total Pu-fissile ^{239}Pu, etc.	Pu-total	Pu-total
	Pu multi-recycle method (2) (Pu enrichment adjustment and discharged fuel composition correction)	–	–	–	+	+	–	–	–
Recycle option	Pu recycle	+	+	+	+	+	+	+	+
	U recycle	+	+	+	+	+	+	+	+
	MA recycle	+	+	+	+	+	+	+	+
	Th recycle	–	+	+	–	+	–	–	+

TABLE 10.1. COMPARISON OF NFC CODES ON THE FUNCTION AND THE METHODS (cont.)

		NFCSS (IAEA)	MESSAGE (IAEA)	DESAE2.2 (IAEA)	FAMILY (Japan)	COSI (France)	VISION (USA)	DANESS (USA)	TEPS (India)
Recycle option	Reprocessing capacity	Manual	+	Manual	LWR: manual FR: +	Manual or +	+	±	Manual
	Modelling of delayed use of SF	–	+	+	+	+	+	–	+
	Optimization FR share according to Pu availability	Manual	+	Manual	+	Manual	+	Manual	+
	Cooling time in nuclear power plant storage	+	+	+	+	+	+	+	+
	Lead/lag time modelling	–	+	–	+	+	+	+	+
	Option of SF, Pu intial condition	+	+		–	+	+	+	+
	HM loss in process (%)	+	+	–	+	+	+	+	+
General	Nuclear power plant capacity	Input	+	Input	Input	Input	+	+	Input
	Commissioning capacity	Input	+	+	LWR: input FR: +	Input	+	+	Manual
General	Decommissioning capacity	Input	±	+	LWR: input FR: +	±	+	+	Manual
Front end	Natural uranium consumption (ktHM)	+	+	+	+	+	+	+	+
	Separative work (SWU)	+	+	+	+	+	+	+	–
	Depleted uranium balance (ktHM)	+	+	+	+	+	+	+	+
	Fuel fabrication load (ktHM)	+	+	+	+	+	+	+	+

TABLE 10.1. COMPARISON OF NFC CODES ON THE FUNCTION AND THE METHODS (cont.)

		NFCSS (IAEA)	MESSAGE (IAEA)	DESAE2.2 (IAEA)	FAMILY (Japan)	COSI (France)	VISION (USA)	DANESS (USA)	TEPS (India)
Back end	Annual SF generation	+	+	+	+	+	+	+	Manual
	Annual reprocessed Pu available	+	+	+	+	+	+	+	Manual
	SF nuclear power plant storage	+	+	+	+	+	+	+	Manual
	SF long term storage	–	+	+	+	+	+	+	+
Back end	Reprocessing capacity	Manual	+	Manual	LWR: manual FR: +	Manual or +	+	±	Manual
	Pu, MA, U, FP in SF long term storage	–	–	±	±	+	+	+	+
	Reprocessed Pu, MA, U, FP stock	+	+	+	±	+	+	+	+

Note: +: yes, –: no, ±: limited function, input: input needed, manual: no optimization function.

The cross-check calculations for the two once-through cycle scenarios used the BAU and BAU+ scenarios. As previously mentioned in Section 6, in the BAU scenario, the nuclear power system consists of only LWRs (L1) and HWRs (H1), while in the BAU+ scenario ALWRs (L2) replace LWRs (L1) from 2015. The parameters for the enrichment tails assay and the model of the initial core are different between the two scenarios. In the BAU scenario, the tails assay from uranium enrichment is assumed to be constant at 0.3% for the whole time period and the uranium enrichment of the initial loading core was modelled to be equal to that of the equilibrium cores. On the other hand, in the BAU+ scenario, the tails assay is assumed to change from 0.3% to 0.2% in 2015, accompanied with ALWRs (L2) replacing LWRs (L1) for new reactors at the same time.

The cross-check calculation for the FR introduction scenario used the break-even FR (F1) introduction scenario and the high case. The cycle conditions of the scenario followed the same approach as in the sensitivity analysis in Section 8, in which the out of reactor time of LWRs and ALWRs is six years, and of FRs three years. The introduction speed of FRs from 2021 to 2050 is fixed in all code calculations as mentioned in Section 8. However, after 2050, each member was to aim at maximum FR introduction within plutonium availability. No limitation of reprocessing facility capacity was provided in the cross-check specification, so some members assumed 100% reprocessing of historical LWR SF prior to FR introduction, while other members assumed a reprocessing load of LWR SF and ALWR SF based on the demand for plutonium supply for FRs to keep the annual plutonium balance zero.

10.4.1. BAU scenario

10.4.1.1. Nuclear power growth curve

The cross-check calculations were performed only on the high nuclear power growth case. Figure 10.5 shows the nuclear power capacity growth for the calculation. The load factor of all nuclear power plants was assumed as 80% in the analysis. The curve was the input condition for the calculation, so the curve must be the same for all calculation codes, as shown in the figure.

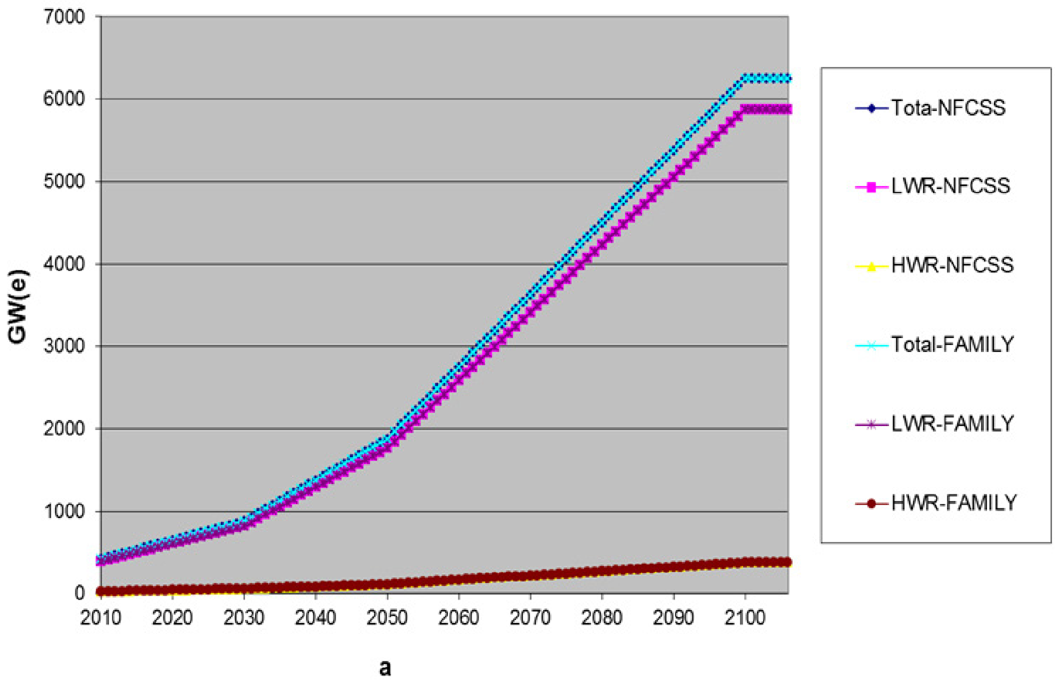

FIG. 10.5. Nuclear power growth curve of the BAU scenario, high case.

10.4.1.2. Comparison of results

10.4.1.2.1. Annual and cumulative natural uranium demand

Figures 10.6 and 10.7 show the annual and cumulative natural uranium demand, respectively. As shown in the two figures, the agreement between codes is rather good. DESAE2.2 gives the highest value and TEPS the lowest value. The value of TEPS is around 6% lower than the average of the others in 2100. The result of DANESS is somewhat higher than the others.

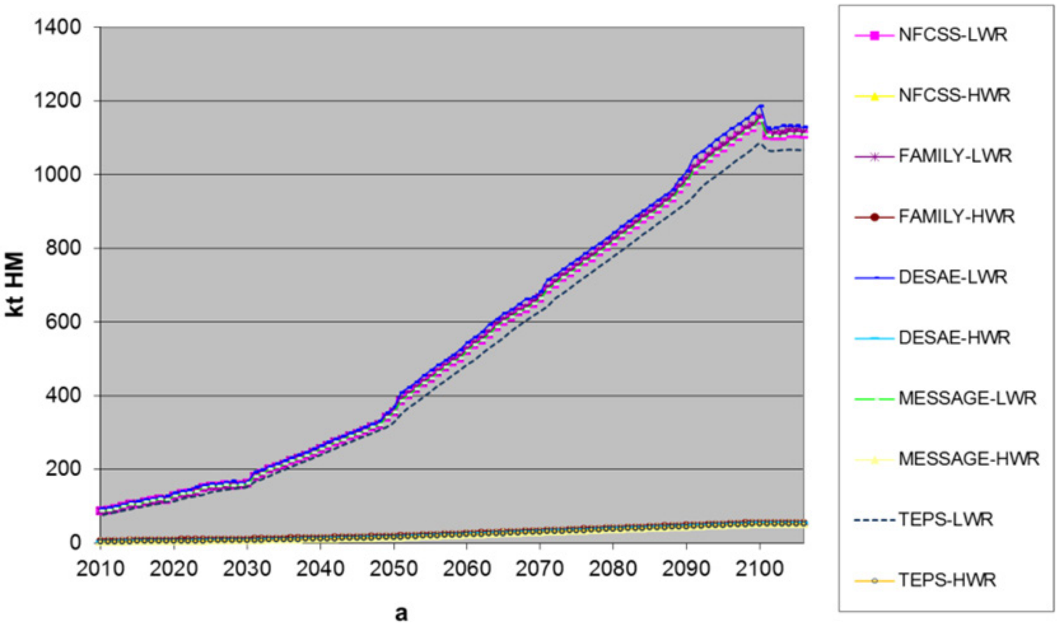

FIG. 10.6. Annual natural uranium demand in the BAU scenario, high case.

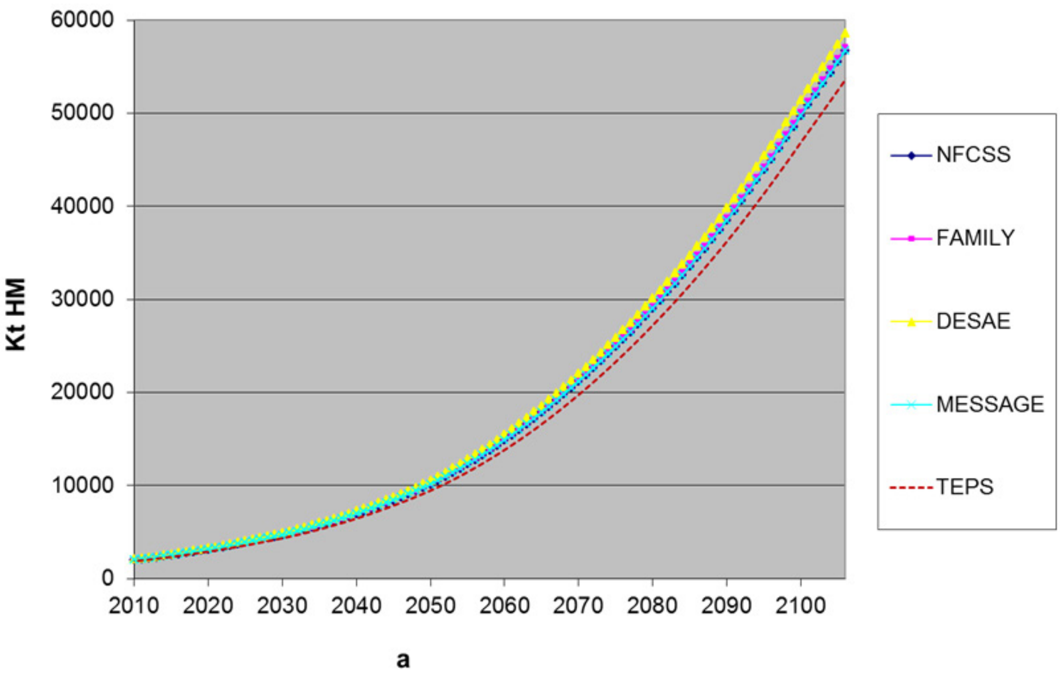

FIG. 10.7. Cumulative natural uranium demand in the BAU scenario, high case.

10.4.1.2.2. Uranium separative work

Figure 10.8 shows the annual uranium separative work in the BAU scenario, high case. In the same manner as the natural uranium demand, the agreement is good. Although the DESAE code gives the highest value, the difference between other codes is within 3%. Table 10.2 gives the summary of comparison of BAU results by different codes on some major time points.

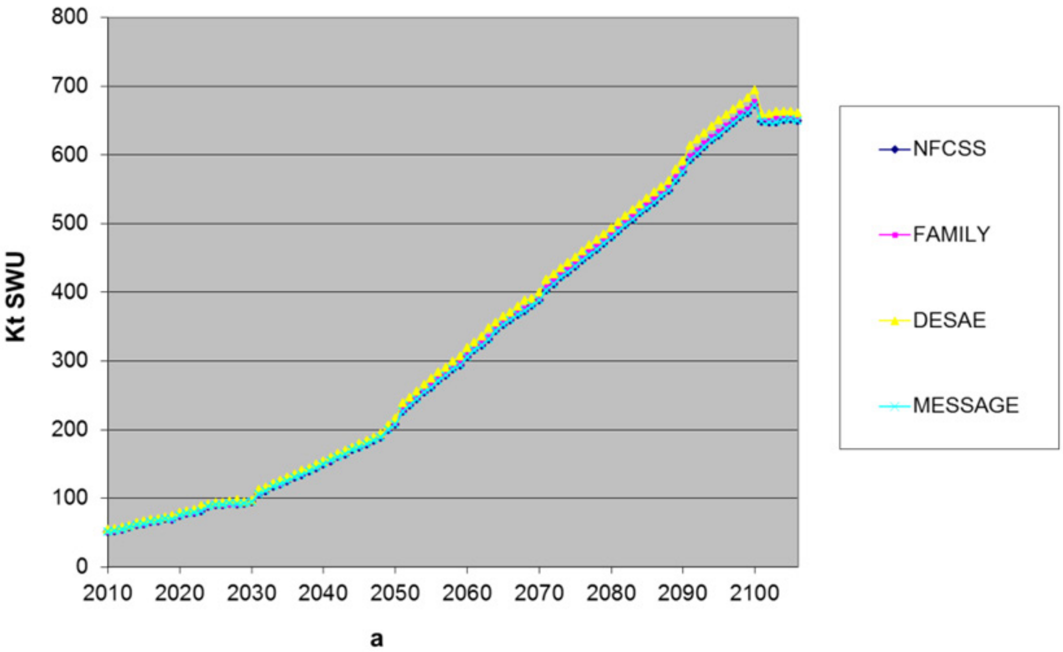

FIG. 10.8. Uranium separative work in the BAU scenario, high case.

10.4.2. BAU+ scenario

10.4.2.1. Nuclear power growth curve

The total nuclear power growth curve of the BAU+ scenario was the same as the BAU scenario. As shown in Fig. 10.9, in the BAU+ scenario, ALWRs (L2) are introduced replacing LWRs (L1) from 2015 and LWRs disappear in 2054 completely based on the plant lifetime assumption of 40 years.

10.4.2.2. Comparison of results

10.4.2.2.1. Annual and cumulative natural uranium demand

The results of natural uranium demand agreed well in all codes (see Fig. 10.10). The MESSAGE code gives the highest value among the codes in this case. The reason for this overestimation comes from the uranium enrichment model for the initial loading core. As shown in Table 10.3, because of a lack of function for adjusting the initial core uranium enrichment, MESSAGE overestimates natural uranium demand by around a few per cent compared to NFCSS or FAMILY.

10.4.2.2.2. Uranium separative work

The tendency of the agreement among codes is the same also on the uranium separative work for LWRs and ALWRs in the BAU+ scenario. Owing to the enrichment model for the initial loading core, MESSAGE gives a few per cent higher value of uranium separative work than other codes (see Fig. 10.11).

TABLE 10.2. SUMMARY OF THE COMPARISON OF THE BAU SCENARIO, HIGH CASE

Code		NFCSS (IAEA)	MESSAGE (IAEA)	DESAE2.2 (SCKCEN)	FAMILY (Japan)	DANESS (ROK)	TEPS (India)
Calc. conditions	Calculation period (CY)	1970 - 2130	1970 - 2130	1970 - 2130	1970 - 2130	2000 - 2130	1970 - 2130
	Tails assay in uranium enrichment (%)	0.3	0.3	0.3	0.3	0.3	0.3
	Note					Use Initial value at 2000	
LWR data	Initial loading enrichment (%)	4.0	4.0	4.0	4.0	4.0	4.0
	Equilibrium loading enrichment (%)	4.0	4.0	4.0	4.0	4.0	4.0
	Note						
Calculation Results	Annual natural uranium demand total at 2100 (kt HM)	1199	1199	1238	1206	1234	1137
	Annual natural uranium demand LWR at 2100 (kt HM)	1147	1147	1184	1154	1184	1084
	Annual natural uranium demand HWR at 2100 (kt HM)	52	52	54	52	51	52
	Cumulative natural uranium demand (at 2008) (kt HM)	1848	1848	1911	1860	1815	1736
	Cumulative natural uranium demand (at 2100) (kt HM)	49754	49754	51479	50075	54083	46888
	Annual uranium separative work (at 2100) (10^6 SWU)	672	672	694	676	666	-
	Depleted U stock at 2100 (kt HM)	42339	42338	43798	42596	42264	39488
	HWR; cumulative amount of SF at 2100 (kt, HM+FP)	2072	2072	2155	2092	1946	1819
	LWR; cumulative amount of SF at 2100 (kt, HM+FP)	4829	4827	5012	4829	4893	4127
	Note						

TABLE 10.3. SUMMARY OF THE COMPARISON OF THE BAU+ SCENARIO, HIGH CASE

Code		**NFCSS (IAEA)**	**MESSAGE (IAEA)**	**DESAE2.2 (SCKCEN)**	**FAMILY (Japan)**	**DANESS (ROK)**	**TEPS (India)**
Cconditions	Calculation period (CY)	1970 - 2130	1970 - 2130	1970 - 2130	1970 - 2130	2000 - 2130	1970 - 2130
	Tails assay in uranium enrichment (%)	1970-2014 : 0.3% 2015 - : 0.2%	1970-2014 : 0.3% 2015 - : 0.2%	1970-2014 : 0.3% 2015 - : 0.2%	1970-2014 : 0.3% 2015 - : 0.2%	1970-2014 : 0.3% 2015 - : 0.3%	1970-2014 : 0.3% 2015 - : 0.2%
	Note					use initial value at '00	
LWR data	Initial loading enrichment (%)	2.40	*4.00*	2.40	2.40	*4.00*	2.40
	Equilibrium loading enrichment (%)	4.00	4.00	4.00	4.00	4.00	4.00
	Note						
ALWR data	Initial loading enrichment (%)	3.40	*4.95*	3.39	3.40	*4.95*	3.40
	Equilibrium loading enrichment (%)	4.95	4.95	4.93	4.95	4.95	4.95
	Note						
Calculation Results	Annual natural uranium demand total at 2100 (kt HM)	887	920	905	885	1249	900
	Annual natural uranium demand A+LWR at 2100 (kt HM)	835	868	851	845	1198	848
	Annual natural uranium demand HWR at 2100 (kt HM)	52	52	54	40	51	52
	Cumulative natural uranium demand (at 2008) (kt HM)	1741	1848	1802	1742	1815	1736
	Cumulative natural uranium demand (at 2100) (kt HM)	37839	40120	38892	37766	55069	38304
	Annual uranium separative work (at 2100) (10^6 SWU)	776	816	791	786	917	-
	Depleted U stock at 2100 (kt HM)	31613	33893	32462	32010	44180	32030
	HWR; cumulative amount of SF at 2100 (kt, HM+FP)	2072	2072	2153	1806	1946	1819
	A+LWR; cumulative amount of SF at 2100 (kt, HM+FP)	3595	3595	3642	3613	3588	3182

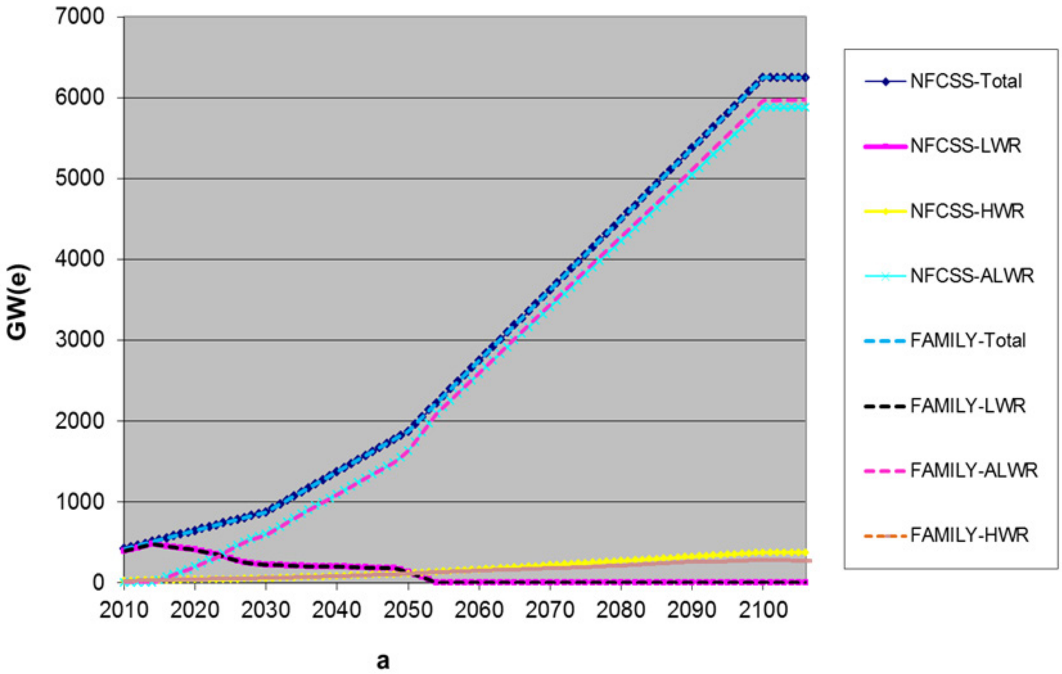

FIG. 10.9. Nuclear power growth curve in the BAU+ scenario, high case.

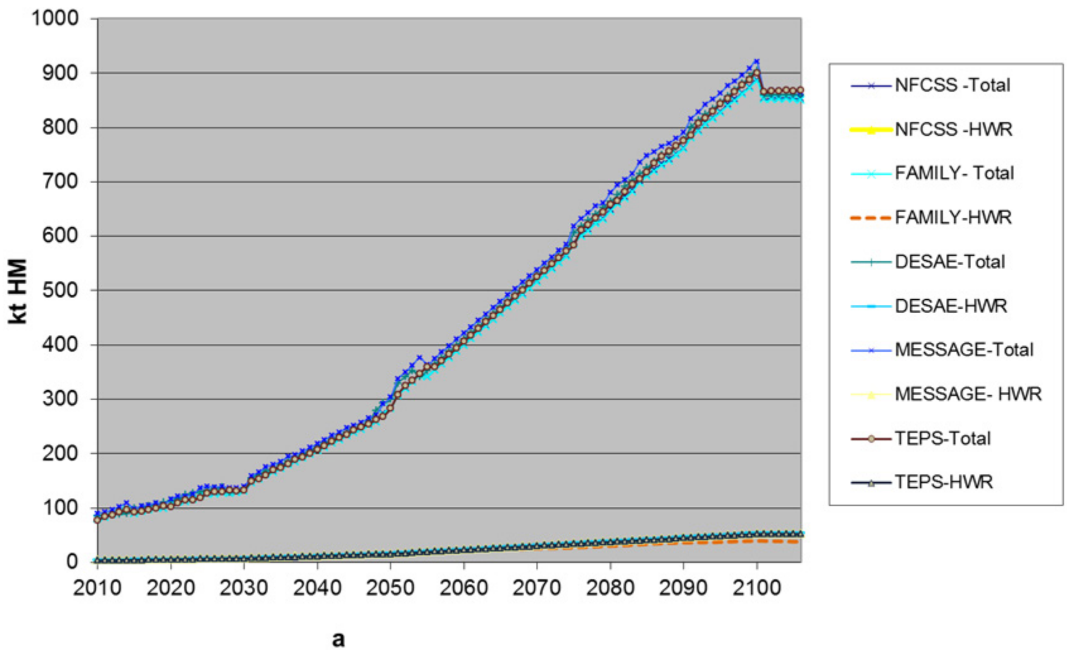

FIG. 10.10. Natural uranium annual demand in the BAU+ scenario, high case.

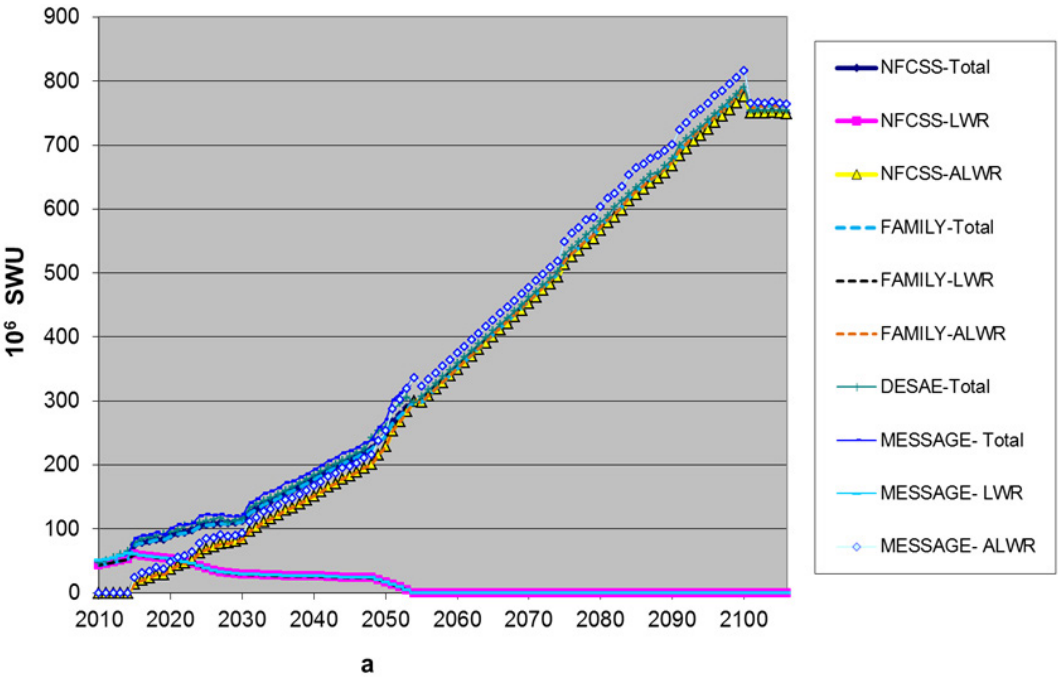

FIG. 10.11. Uranium separative work in the BAU+ scenario, high case.

10.4.3. Maximum fast reactor introduction scenario

10.4.3.1. Methods for plutonium multi-recycle calculation

As mentioned in the previous section, the agreement of mass flow calculation results among NFC codes is very good concerning the scenarios of once-through cycle systems with thermal reactors, because in the once-through cycle scenarios the uncertainty of the composition change of discharged fuel does not affect the mass flow calculation results focused on fuel material needs and separative work. Therefore, given the nuclear power growth curve and load factors of nuclear power plants and the compositions of fresh fuel and its residence time in the core of nuclear power plants for the scenarios, the necessary fuel material supply is uniquely decided. So, the fuel demand, such as natural uranium demand and uranium separative work, is easily calculated, and differences between codes perhaps come from how they deal with such matters as lead time or process time counting.

On the other hand, when we try to study recycling scenarios, it is expected that the calculation method for treating the isotopic composition changes of uranium, plutonium and MAs of fresh fuel and discharged fuel during cooling, reprocessing and fabrication taking account of various kinds of fuel and burnup performances affect the calculation results considerably.

One typical big issue is the method for the modelling of plutonium multi-recycling. As mentioned in Section 6, mass flow analyses are able to be started by using the fresh fuel composition and the discharged fuel composition provided from fuel core designs. Usually, the isotopic compositions of fresh fuel and discharged fuel provided by a design work are based on some assumptions on the actual situation of a particular country and a certain time period. However, in actual situations, the isotopic composition of available plutonium varies from time to time depending on the actual load factors of existing nuclear power plants and the structure of the NES of a certain country or region and a certain time period. Basically, most FRs can cope with a very wide range of plutonium isotopic compositions by adjusting the plutonium enrichment with neutronic calculations on the criticality and the burnup reactivity swing changes of the FR, etc. However, once we adjust the plutonium enrichment of fresh fuel for the FR, we have to recalculate the isotopic composition of discharged fuel, which varies from the composition provided by the design. So, if we try to treat the matter in the same manner as real reactor core management, it will be a huge effort and not practical for long term fuel cycle analyses. Instead, approximation methods are used.

The available methods for long term fuel cycle calculations for plutonium multi-recycle scenarios are considered as follows.

10.4.3.1.1. Method of Pu-total preservation

The amount of Pu-total for fresh fuel and discharged fuel is preserved the same as in the design data. It is expected that the error of this method will become small if actual available plutonium in the scenario study has an isotopic composition close to the one used in design work and then the isotopic composition of the discharged fuel is not far from that of fresh fuel. The codes NFCSS and MESSAGE use the method of Pu-total preservation as shown in Table 10.1.

10.4.3.1.2. Method of Pu-fissile preservation

The amount of Pu-fissile for fresh fuel and discharged fuel is preserved the same as in the design data. Usually, the definition of Pu-fissile is the same as that of thermal reactors, in which ^{239}Pu and ^{241}Pu are regarded as Pu-fissiles. The accuracy of this method depends on the definition in which Pu-fissiles are dominant to the criticality of reactor cores using plutonium fuel, though Pu-fertiles, such as ^{238}Pu and ^{240}Pu, make a considerable contribution to the criticality and burnup reactivity swing. The DESAE code uses the method of 'Pu-fissile preservation' as shown in Table 10.1.

10.4.3.1.3. Method with equivalent Pu-fissile coefficient and depletion transition matrix

The most detailed method for a fuel cycle analysis code is to simulate actual reactor core managements. The plutonium enrichment of fresh fuel for FRs is usually adjusted taking account of the criticality change of the beginning of the operation cycle and the change of burnup reactivity loss due to the plutonium isotopic composition change. In order to simulate real core management, some fuel cycle calculation codes have an 'equivalent Pu-fissile coefficient' table and a 'depletion transition matrix'. The former is for adjusting plutonium enrichment of fresh fuel based on the plutonium isotopic composition available for the fuel fabrication, and the latter is for evaluating the composition of discharged fuel. In this detailed method, the information of all isotopic compositions of fuel material is preserved and managed simulating the fuel cycle scenario. The cross-check calculation result of the FAMILY code was based on this 'method with equivalent Pu-fissile coefficient and depletion transition matrix', though the coefficient and the matrix are to be unique for each particular core design.

10.4.3.2. Reactor data and analysis conditions

The cross-check calculation was completed for the maximum FR introduction scenario of break-even FR (F1) and the high case with the same reactor data and the analysis conditions as mentioned in Section 8.6.

10.4.3.2.1. Fast reactor introduction speed

- 2021–2030: 1 GW(e) FR demand growth a year (total demand: 10 GW(e) in 2030).
- 2031–2050: 19.5 GW(e) FR demand growth a year (total demand: 400 GW(e) in 2050).
- After 2051: adjust to maximum FR introduction following plutonium availability.

As mentioned above, the FR introduction speed from 2021 to 2050 was fixed. On the other hand, the introduction speed after 2051 is to be maximized depending on plutonium availability.

10.4.3.2.2. Spent fuel reprocessing strategy

To evaluate the maximum FR introduction speed, it was assumed that LWR or ALWR SF should be reprocessed as needed to cope with the requirement to provide sufficient plutonium for FR introduction. In order to achieve zero plutonium balance, the limitation of reprocessing capacity was not taken into account in the analysis. The FR SF is all reprocessed in principle.

10.4.3.3. Comparison result

10.4.3.3.1. Maximum break-even fast reactor power share

Figure 10.12 shows the calculation results by four analysis codes for maximum break-even FR (F1) introduction in the high growth case. As shown in the figure, NFCSS and MESSAGE give almost equivalent results, because both codes use the same method of 'Pu-total preservation'. As summarized in Table 10.3, the FR power share of the MESSAGE result is a little larger than that of NFCSS, perhaps because of not accounting for the 1% plutonium loss in reprocessing.

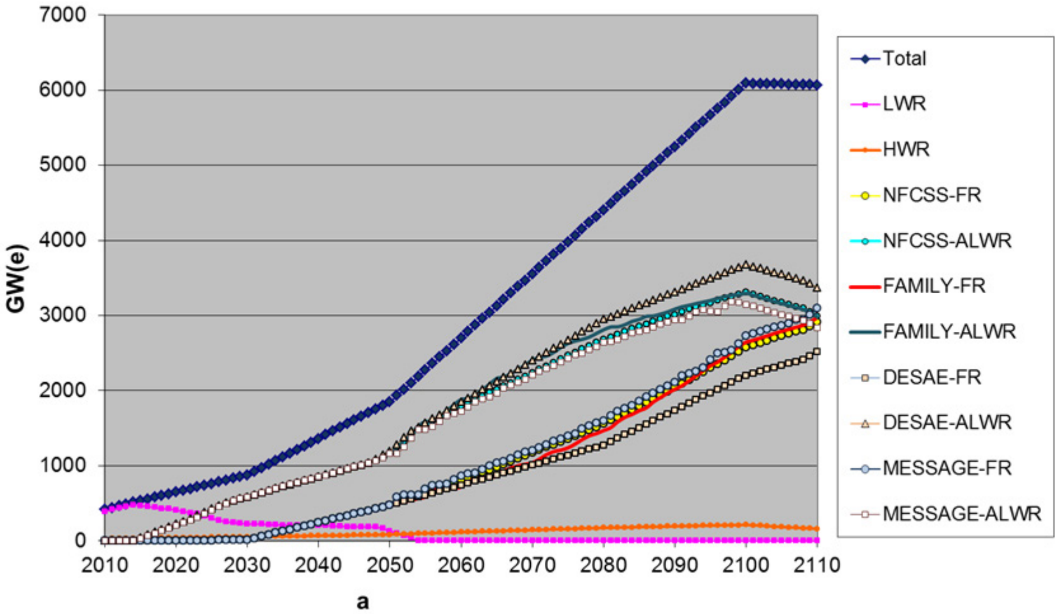

FIG. 10.12. Maximum fast reactor power share after 2050, high case.

The detailed calculation of the FAMILY code with an 'equivalent Pu-fissile coefficient and depletion transition matrix' gives lower FR power share during the time period around 2070–2090 than NFCSS, but gives higher FR power share after 2090. As a result, FAMILY gives a higher power share in 2100 than NFCSS.

The DESAE code based on 'Pu-fissile preservation' gives the most conservative result of the three codes. It resulted in lower FR power share than other codes due to the Pu-fissile counting by the code. As summarized in Table 10.3, the FR power share in 2100 by DESAE is 37% in comparison with the other three results of 44–46%.

10.4.3.3.2. Reprocessing load for spent fuel from each reactor

Figure 10.13 shows the necessary reprocessing load for SF from each reactor. Regarding the reprocessing strategy, DESAE and FAMILY used different assumptions from NFCSS and MESSAGE. The analyses of the first two codes used a simple assumption of a virtually huge load of reprocessing in one or two years for the simplification of decay correction of reprocessed plutonium isotopic compositions. Although the special reprocessing load for old LWR SF scales over for DESAE and FAMILY results, DESAE assumes 137 ktHM of reprocessing load for LWR SF in 2008, and FAMILY 114 and 106 ktHM of reprocessing load for LWR SF in 2010 and 2011, respectively. On the other hand, the reprocessing strategy used in the analyses of NFCSS and MESSAGE are closer to reality than the analyses of the DESAE and FAMILY codes, although the decay correction for the composition of historical LWR SF is simplified as one composition based on 30 years cooling. In the latter two analyses, the reprocessing load for LWR SF and ALWR SF is adjusted to maintain the reprocessed Pu-total balance close to zero in each year (no stored inventories of separated Pu).

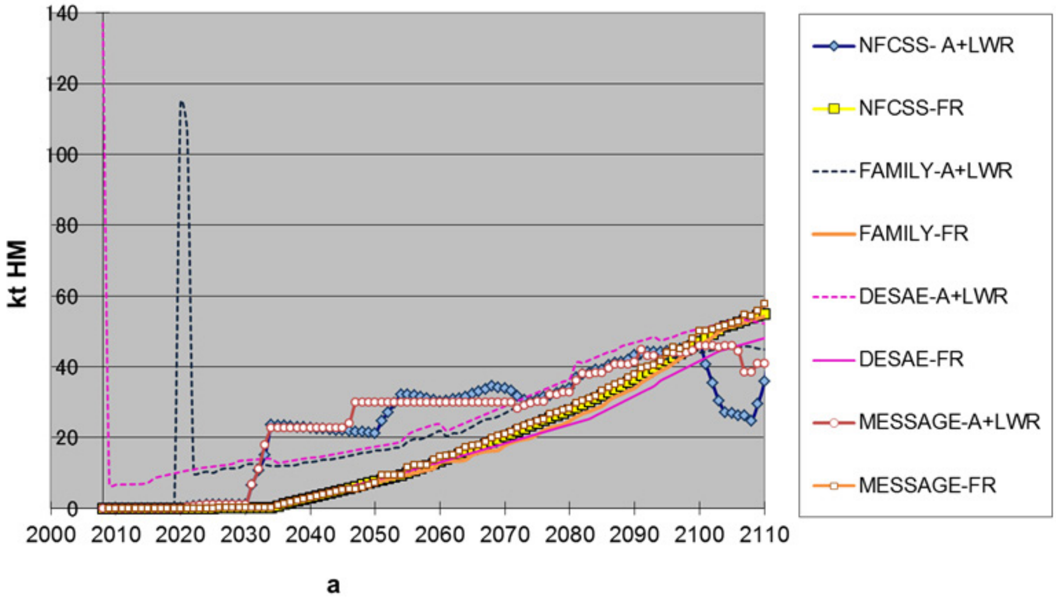

FIG. 10.13. Reprocessing load for each reactor spent fuel, high case.

10.4.3.3.3. Reprocessed Pu-total stock of each reactor

Figure 10.14 shows the change of reprocessed Pu-total stock of each reactor. As shown in the figure, the Pu-total stock from FRs was negative in the whole period and the value in 2100 becomes around 25 000 t, which is balanced with the positive stock from ALWRs across the whole calculation period in the NFCSS and MESSAGE analyses. Although the reprocessing strategies for the FR introduction are different, the results from around 2060 to 2100 agree very well between FAMILY and the NFCSS and MESSAGE codes. Regarding the result from DESAE, because of the Pu-fissile preservation method, the reprocessed Pu-total stock does not come close to zero.

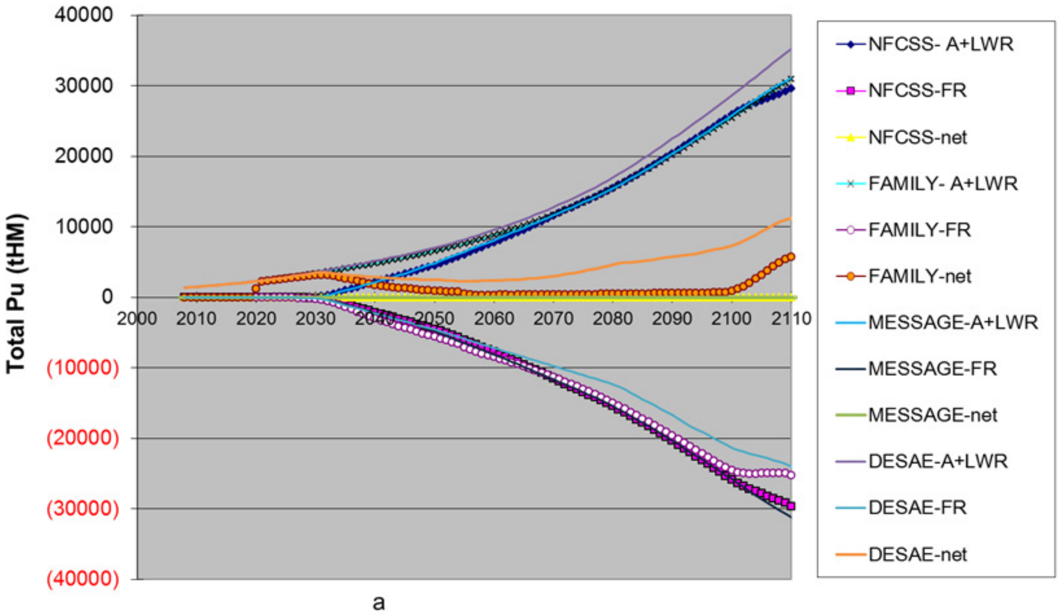

FIG. 10.14. Change of reprocessed Pu-total stock, high case (reprocessed plutonium stock: cumulative value of annual reprocessed plutonium balance).

10.4.3.3.4. Cumulative natural uranium demand

Figure 10.15 shows the cumulative natural uranium demand results. As shown in the figure, the difference between NFCSS, MESSAGE and FAMILY is not visible because of the very close FR power share (i.e. very close ALWR power share) of all three results. On the other hand, DESAE's result lies apart from the others with around a 10% higher value. It is necessary to clarify whether 'Pu-total preservation' or 'Pu-fissile preservation' is suitable for the long term plutonium multi-recycle analysis by undertaking further investigations, although the 'Pu-total preservation' method gives close results to a detailed calculation (FAMILY) in the present cross-checking.

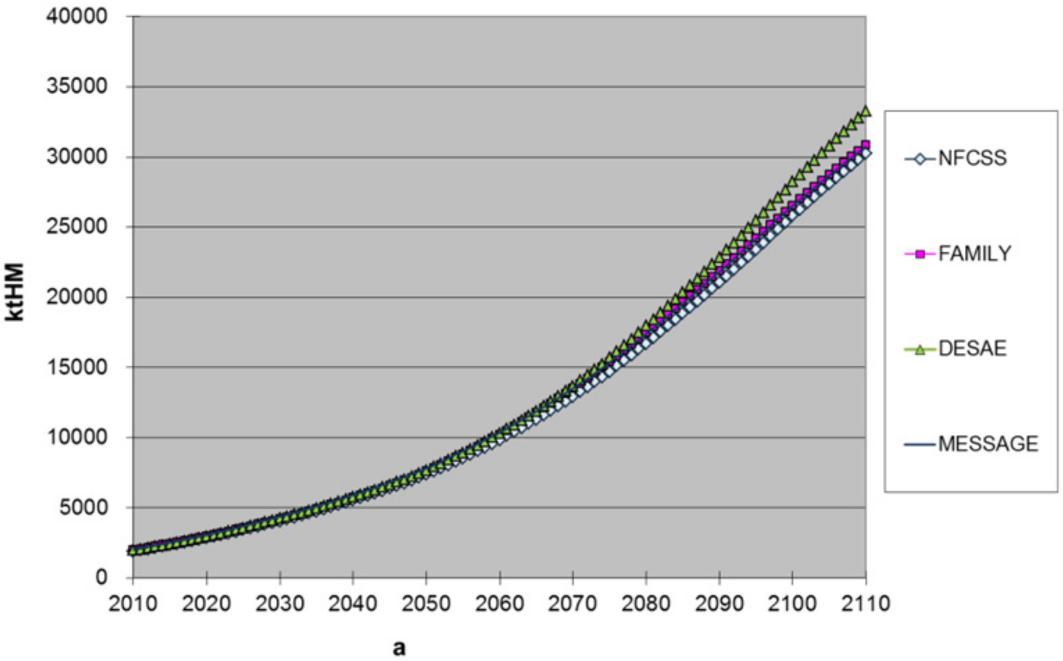

FIG. 10.15. Cumulative natural uranium demand of maximum break-even fast reactor introduction, high case.

10.4.4. Cross-check results

10.4.4.1. Results for the BAU and BAU+ scenarios

Except for the differences due to counting fuel enrichment changes between the initial core and the equilibrium core, there is no significant difference between the codes. The DANESS code shows the most discrepancy. It should be noted that the developers of DANESS did not participate in the cross-check and familiarity of the code by the users may be a factor in the results obtained.

10.4.4.2. Maximum break-even fast reactor introduction scenarios

Different results may be obtained between the codes depending on how they treat the approximation of the isotopic vectors of available Pu (see Table 10.4). This may result in significant differences in FR power share in short term and localized scenarios.

In the multi-Pu recycle calculation, the effect from the plutonium vector treatment for fresh and discharged fuel has to be checked and reflected in analysis code development to obtain accurate quantitative results.

On the other hand, in long term scenario studies, the effects on the critical indicators, such as possible FR share and cumulative uranium demand, become small and do not produce significant differences.

TABLE 10.4. SUMMARY OF COMPARISON OF MAXIMUM BREAK-EVEN FAST REACTOR INTRODUCTION, HIGH CASE

Code		NFCSS (IAEA)	MESSAGE (IAEA)	DESAE2.2 (SCKCEN)	FAMILY (Japan)
Cond.	Calculation period (CY)	1970 - 2130	1970 - 2130	1970 - 2130	1970 - 2150
	Tails assay in uranium enrichment (%)	1970-2014 : 0.3% 2015 - : 0.2%	1970-2014 : 0.3% 2015 - : 0.2%	1970-2014 : 0.3% 2015 - : 0.2%	1970-2014 : 0.3% 2015 - : 0.2%
LWR	Initial loading enrichment (%)	2.40	*4.00*	2.40	2.40
	Equilibrium loading enrichment (%)	4.00	4.00	4.00	4.00
ALWR	Initial loading enrichment (%)	3.40	*4.95*	3.39	3.40
	Equilibrium loading enrichment (%)	4.95	4.95	4.93	4.95
FR	Initial loading Pu enrichment (whole core averaged) (%)	11.37	11.80	Adjust by Pu-fissile	Adjust by equivalent Pu-fissile coefficient
	Equilibrium loading Pu enrichment (whole core averaged) (%)	11.80	11.80	Adjust by Pu-fissile	Adjust by equivalent Pu-fissile coefficient
Calculation results	Annual natural uranium demand total at 2100 (kt HM)	492	446	555	454
	Annual natural uranium demand A+LWR at 2100 (kt HM)	462	419	521	434
	Annual natural uranium demand HWR at 2100 (kt HM)	29	28	34	20
	Cumulative natural uranium demand (at 2008) (kt HM)	1741	1848	1802	1812
	Cumulative natural uranium demand (at 2100) (kt HM)	25803	26510	28223	26544
	Annual uranium separative work (at 2100) (10^6 SWU)	431	393	480	407
	Depleted U stock at 2100 (kt HM)	20191	21067	20956	22494
	FR power share at 2100 (%)	43.7	46.4	37.4	45.0
	Reprocessed U stock from A+LWR at 2100 (kt HM)	1911	2064	2230	1772
	Pu stock(net) from FR at 2100 (tHM)	-25909	-25810	-21322	-24543
	Pu stock from A+LWR at 2100 (tHM)	25909	25811	28652	25442
	Note	Total Pu balance. 1 % of Pu loss counted.	Total Pu balance. Pu loss not counted.	Pu-fissile balance. Pu loss not counted.	Adjusted based on Pu vector with trans. matrix. 1 %Pu loss.

10.5. SUMMARY

NFC analysis codes of Member States, which could be used in the GAINS scenario studies, were introduced and their features, such as available functions and their methods, compared.

In order to ensure the credibility of analysis results in GAINS, three basic scenarios were analysed and cross-checked by several codes.

According to the cross-check results, in the case of plutonium multi-recycle scenarios, considerable differences in the plutonium balance may appear between the codes due to the method of plutonium isotopic vector treatment on short term mass flow and possible FR introduction speed. On the other hand, the uncertainty of long term assessment, such as cumulative natural uranium consumption, may become small. Further detailed investigations of the isotopic vector treatment of plutonium multi-recycle may be warranted as part of specific scenario studies in future work.

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11. OVERALL CONCLUSION: PROJECT ACCOMPLISHMENTS AND OBSERVATIONS

11.1. BACKGROUND

The United Nations general concept of sustainability and considerations specific to the concept of sustainable energy were incorporated into the INPRO objectives and integrated into the INPRO methodology [11.1]. Various options for enhancing sustainability features of NESs were analysed at both the national and international level. GAINS is an IAEA CP that has demonstrated the broad interest of IAEA Member States in the comprehensive consideration of pathways for sustainable nuclear development. This interest and active participation in the project can be explained by the growing understanding of an increasing role of the multilateral approach to building future NESs in a sustainable manner. The objectives and organization of the project are presented in Section 1. Section 2 collects basic information on a number of international studies where there might be an interaction with the goals set out for the GAINS CP. This overview helps to avoid duplication of work and focus work on the GAINS project.

The primary outputs of the GAINS project are: (i) development of a standard analytical framework (methodological platform, assumptions and boundary conditions) for assessing global NES architectures; and (ii) results from sample analyses for scenarios involving transition from the present NES architecture to future architectures including innovative NESs. At the same time, the participants of the project did not try to cover all aspects of the NES sustainability concept set forth in Refs [11.1, 11.2] and focused on key issues in several dimensions of sustainable development of energy supply that can be most appropriately addressed within the scenario approach such as: preservation of nuclear material resources, management options for nuclear waste capable of restraining the accumulation of used fuel, ways for enhancing proliferation resistance of nuclear technologies or the possible impact of the nuclear architecture arrangement on the economics of global NESs. The main results of the project in creating a standard analytical framework for vision and scenario studies, conclusions on the simulation analysis of the sample case studies, and some observations with regard to future activities in the area are presented in this section.

11.2. DEVELOPMENT OF AN ANALYTICAL FRAMEWORK FOR ASSESSING GLOBAL NUCLEAR ENERGY SYSTEM ARCHITECTURES IN TERMS OF SUSTAINABLE DEVELOPMENT

The IAEA/INPRO activities on global vision and transition scenarios to a sustainable nuclear future were originated within two organizational modes. The experience accumulated by several Member States in the area was generalized and published in 2010 in the IAEA Nuclear Energy Series [11.3]. The results and conclusions of the publication are based on input data and different national simulation tools that are not always available to an interested reader.

GAINS was initiated as a CP of IAEA Member States in which the joint work on the development/updating of the methodological platform for dynamic modelling of innovative global NES architectures (assumptions, boundary conditions, reactor database and tools) and the joint assessment of the results were an essential feature. As a result of efforts carried out within the GAINS project, a unique analytical framework for assessing future NESs taking into account sustainable development was developed. The main elements of the framework are briefly outlined below.

11.2.1. Homogeneous and heterogeneous models of the global nuclear energy system

An important task in developing the GAINS framework was the creation of a model to examine the potential synergy between current and innovative NESs, as well as the synergy between groups of countries with different preferences in reactor types and fuel cycle options.

It was agreed by the project members that some tasks of the work plan can be successfully implemented with the use of the homogeneous model that is widely used in many scenario studies. However, this model assumes no differences in the nuclear development strategies of different countries and has very limited abilities to reflect essential features of the advanced architecture that has to include cooperation of States with different levels of nuclear technology development. The participants of the GAINS project recognized the advantages of the heterogeneous model originally used for simulation of nuclear fresh and SF flows between several European countries [11.4]. The heterogeneous model of the world developed in GAINS comprises different NGs of countries based on the SNF management strategy being pursued for the back end of the fuel cycle at the end of the simulation period (Section 3). The heterogeneous model was found to be a flexible and powerful tool for comprehensive examination of a global NES allowing analysis of the variants of the global architecture. Combined use of homogeneous and heterogeneous models allowed the modelling of three story lines for global NES development: a convergent homogeneous world without any differences in NES development strategies, a heterogeneous separate world based on self-reliance and preservation of local identities, and a heterogeneous synergistic world with rapid changes towards global solutions. It was understood that including some more sophisticated functions in the heterogeneous model, such as improved modelling of nuclear material transport between different groups, limitations from nuclear and non-nuclear industries and some institutional barriers would be desirable for further improvement of the model.

11.2.2. Estimation of nuclear energy needs at the global level and in the identified groups of countries

Specification of nuclear energy needs (Section 5) is an essential part of modelling scenarios of future NES development. For the simulation of the world with the use of homogeneous models, a global nuclear energy demand profile is required. The heterogeneous models require not only a global nuclear energy demand profile but also specification of the nuclear energy demand in every NES strategy group of countries used in the model. The procedure for preparation of the nuclear energy demand for each GAINS strategy group may be considered an added value of the project to the scenario studies' activities.

The GAINS approach consists of a combination of assessments made by competent energy agencies with the use of macro-models and States' expectations for the nuclear power deployment compiled by the IAEA. The story lines of macro-models, based on different assumptions of demographic, social, economic, technological and environmental developments, provide long term but rather divergent trends of nuclear power's share in total energy production. A serious deficiency of the pure theoretical macro projections is not accounting for the energy plans and aspirations of the countries based on their vision of opportunities and challenges.

Taking advantage of interagency cooperation, GAINS consolidated the macro projections on nuclear energy deployment of the different groups and organizations (top-down assessments) and estimations of nuclear growth made within related activities of the IAEA (bottom-up assessments) (Section 5). In order to comprehend the uncertainty of the scale of future nuclear power growth, a 'high' case and 'moderate' case were set in the project. The combination of the top-down and bottom-up approaches allowed building high and moderate global nuclear energy scenarios and identifying high and moderate nuclear energy scenarios in three GAINS groups. The global scenarios developed at the first stage of GAINS were taken into account in the study in Ref. [11.3] and in other INPRO CPs related to programme area B [11.5].

The recommendations of the first GAINS consultancy meeting included a proposal that the IAEA increase the time range of the IAEA estimations from about 20 years to at least 40 years. This request was recently implemented. At present, IAEA Member States present their high and low estimations on nuclear power deployment until 2030 and expected trends until 2050. This is a significant development and a valuable contribution to the reduction of uncertainties in modelling the nuclear future and for preparation of more definite grounds for decision making.

11.2.3. Set of nuclear energy systems adopted for analysis in GAINS and related database

Compiling, assessing and tabulating characteristics and parameters of current and innovative NESs (reactors and associated fuel cycles) selected for the simulation within GAINS scenarios was a necessary and important task of the project implementation plan. Prior to the systematic survey of future global nuclear architectures, a set of reactor design parameters of various kinds of nuclear power plant and basic analytical conditions of fuel cycle systems have been settled (Section 6). For the nuclear reactor concepts, the burnup performances and refuelling data of each reactor concept linked to NFC analysis were requested from the relevant project participants as input to the study. Regarding the NFC systems, some important conditions which affect mass flow analysis results were also requested.

The GAINS database built in the course of a laborious process covers characteristics and parameters of several types of current and innovative systems, for example, LWRs, FRs, MSRs and ADSs. Important variations within a given system type are also taken into account, for example, LWR variations based on fuel burnup, or FR variations based on fuel burnup or BR. In accordance with the philosophy of the project, most innovative concepts are being developed and most innovative systems operate in countries of the first GAINS NG (NG1). These countries bear high expenses connected with RD&D costs and financial risks. As contrasted to NG1, countries in NG3 use proven nuclear technologies in an optimal commercial regime with minimal nuclear infrastructure and minimal financial risks. NG2 includes once-through fuel cycle facilities (e.g. fuel fabrication, waste repository) and may or may not include innovative reactor systems, but does not include fuel reprocessing infrastructure and associated costs.

The GAINS database has essentially extended the IAEA database for scenario studies available to IAEA Member States. Previously, it contained only data on a few types of current reactors that could be simulated with the use of the IAEA code NFCSS [11.6]. It is now possible to simulate material flows in architectures based on a wide range of reactors and associated fuel cycles with different technical maturities, ranging from operating systems generating most of the currently available nuclear electricity, to very innovative concepts currently in various stages of research and development [11.7]. GAINS has demonstrated the benefit of the coordinated efforts of specialists from Member States and IAEA experts in creating a unified database in support of the development of long term nuclear strategies.

Emphasizing this progress, the project participants note the value in making further progress on an enhanced database for scenario studies. The INPRO methodology applies a holistic approach to the assessment of the sustainability features of NESs. GAINS tentatively addressed some issues of economics and proliferation resistance that significantly impact the vision of the future architecture of global nuclear power. A lack of reliable input data in these areas and others (e.g. safety) prevents drawing conclusions with the same level of certainty as in the case of material flows analysis.

GAINS participants consider it expedient to include into the future IAEA/INPRO action plans activities for further extension of the database to support strategic decision making. Detailed radiation and heat characteristics of nuclear waste for the development of waste management strategies or assessed data on the construction, operation and fuel costs of the facilities of NES for improved economic studies could be among near steps in the creation of such a database.

11.2.4. Key indicators, tools and template for assessing sustainability of nuclear energy systems in GAINS analysis

KIs and EPs calculated for each NES form the main basis for comparing the different options and results (Section 4). Definitions and concepts listed in IAEA publications and the INPRO methodology document served as background for the work. Ten 'GAINS key indicators' were identified by screening the more than one hundred indicators of the INPRO methodology. These indicators (KIs) depict nuclear power production, material resources, discharged fuel, radioactive waste and MAs, fuel cycle services, system safety, and costs and investments of NESs. EPs were defined and included to obtain quantitative values and other information, in particular on the sustainability of a given NES. The indicators and parameters were chosen based on the participants' experience, availability of resources, and capabilities of methodological and computational tools at the disposal of the participants. For a given indicator or parameter, the range of uncertainty levels was defined depending on the maturity level of the specific NES under consideration.

After reaching a consensus on evaluation criteria, some project participants calculated KIs and EPs using different analytical tools and methods (Section 10), including DANESS (Republic of Korea), DESAE (Belgium, Russian Federation), COSI (France), FAMILY (Japan), TEPS (India) and VISION (USA). The MESSAGE and NFCSS codes disseminated by the IAEA were also used. In the homogeneous model of a global NES, KIs and EPs were calculated for the whole world. In the heterogeneous model, the KIs and EPs were also calculated for each GAINS strategy group of countries.

A special template was developed for facilitating joint analysis of KIs and EPs of an NES under consideration (Section 4 and Annex III). The development of a template for the evaluation of KIs and EPs of a global NES scenario provides essential progress in harmonizing the analytical tools of IAEA Member States which can be used to support decision making related to a long term nuclear strategy and energy planning. The definition of the range of uncertainty levels of KIs made in the project based on the maturity level of the specific NES is an interesting proposal that is useful for sensitivity analysis of the assessment results and for identifying a vector for future work on enhancing the reliability of the conclusions of subsequent studies of this sort.

When exploring results of cross-check assessments, comparing the results of different codes using the standard template, it was found that both national tools and tools disseminated by the IAEA provided fairly consistent results related to the calculation of indicators in the area of fresh and discharged fuel flows and waste flows. The accuracy of the calculation permits drawing reliable conclusions on the trends in the consumption of uranium, the accumulation of discharged fuel and fissile material and the main components of radioactive waste for the selected scenarios. Additional cross-check comparisons for multi-group (heterogeneous) synergistic scenarios can be undertaken in the future. In the current GAINS project, two different codes were used to assess variations of a synergistic scenario, and provided an indirect means to assist in assessing validity.

At the same time, a comparison of the calculation tools assigned for operational routine analysis with the more mature and capable tools of some IAEA Member States identified several directions for the development of the former. A few possible improvements of the IAEA operational tools that were discussed by the project participants are to:

- Provide the ability to model isotopic decay of stored material, such as SF, over a long time interval after discharge for the NFCSS and MESSAGE codes, and to treat isotopic vectors for better taking account of plutonium and MA enrichment levels in recycled fuel, proliferation indicators and other factors.
- Enlarge the list of radionuclides in the NFCSS, MESSAGE and DESAE codes for more precise evaluation of the radiological protection level for human health and the environment when comparing long term waste management strategies.
- Include additional options and outputs in the codes disseminated by the IAEA. An example of an additional option is the possibility of calculating for MSRs, ADSs and the thorium fuel cycle. It is desired that examples of additional refuelling data options and the Th chain be provided in the NFCSS code.

GAINS identified a few directions for further development of the template for the assessment of scenarios and vision studies to be implemented. One direction would be to extend the template developed in the project from the area of fissile material and waste flow assessment to a wider scope covering more dimensions of sustainability defined in the INPRO methodology. In particular, the advanced template might include a set of KIs and EPs for comparing and assessing the economic competitiveness and proliferation resistance of various NESs. Another direction is to enhance the template for heterogeneous scenarios through inclusion of side-by-side comparison charts for individual groups and the global system.

On the whole, GAINS developed an important segment of a standard framework for assessing future NESs taking into account sustainable development criteria. Further progress in developing the framework for the analysis of the sustainability of NESs and ranking of the alternatives depends on incentives and activities of IAEA Member States in transition to a synergistic global NES based on beneficial cooperation between nuclear power technology holders and users. Practical steps in the implementation of a multilateral approach to enhancing sustainability of global nuclear power will define further needs and priorities of the involved stakeholders in the development of a relevant methodological platform.

11.3. RESULTS OBTAINED FROM SAMPLE SCENARIO STUDIES

11.3.1. Issues addressed in the sample analysis and nuclear energy system architectures under consideration

Validation of simulation results obtained for different configurations of a global NES with use of the methodological platform, reactor database, assumptions and boundary conditions developed and agreed by the project participants was an essential part of the overall objective of GAINS. There are many issues on the agenda of the sustainable development of a global NES directly linked with the architecture of the system. GAINS sample case studies (Sections 7 and 8) addressed some of them via assessing KIs and EPs selected by project participants. KIs and EPs calculated for each type of NES form the main basis for comparing the different options and results. This concluding section summarizes only the main findings of the study related to the role of nuclear architecture in addressing such concerns as assurance of nuclear material resources, fissile material and high level radioactive waste inventories and investment barriers to the introduction of an NES in a commercial manner.

The architecture of an NES under consideration comprises different types of reactor system and associated NFCs and various schemes of interaction between elements of an NES. To estimate the role of technical innovations, advanced components of NESs such as FRs and MSRs and associated fuel cycles, ADSs and the thorium fuel cycle were successively introduced into the current system. In compliance with these introductions, architectures of four types of NES with a gradually increased level of innovations were examined (Section 6):

(a) A homogeneous BAU scenario based on PWRs (94% of power generation) and HWRs (6%) operated in a once-through fuel cycle in which the world was modelled as a single NG. A variant of this scenario included the introduction of an advanced PWR replacing conventional PWR technology (termed the 'BAU+' scenario).
(b) Homogeneous (single-group) scenarios for a closed cycle using thermal reactors and FRs for comparison with the above scenarios. Some of these fuel recycle scenarios included HWRs (6%) operated in a once-through mode.
(c) A hybrid heterogeneous architecture scenario comprising a once-through fuel cycle strategy in NG2, a closed fuel cycle strategy in NG1, and use of thermal reactors in a once-through mode in NG3. Both synergistic and non-synergistic cases were analysed for this scenario. In the synergistic case, NG3 receives fresh fuel from NG2 and NG1, and returns associated SNF to those groups.
(d) Other innovative NES scenarios using the homogeneous world model including:
    (i) Construction of fast-spectrum reactors or thermal-spectrum HWRs using the thorium fuel cycle for the reduction of natural uranium requirements;
    (ii) Reduction of MAs using ADSs or MSRs.

The project philosophy implies that technological innovations will differ between the GAINS NGs of countries. Enhancement of the cooperation between these groups by means of strengthening the material and financial flows was considered in GAINS as part of institutional innovations. The impact of different configurations of the future global nuclear architecture on the sustainability of an NES were examined in a 'what if' manner, including the assumption of no interactions between groups (heterogeneous separate world) to the assumption of no restrictions on multilateral cooperation.

The heterogeneous model of the global NES architecture described in Section 3 was the principal instrument for estimating the institutional innovations in the global nuclear infrastructure. The heterogeneous description of the world is more realistic and helps to reveal essential peculiarities of nuclear energy deployment. A focus of the GAINS project is to consider synergetic global NES development with interactions between GAINS groups in order to understand how synergy impacts the KIs and EPs such as uranium cumulative demand, used fuel amount, plutonium availability, SWU and fuel cycle services.

11.3.2. Cumulative demand of natural uranium and thorium

Preservation of natural uranium resources is one of the examples which illustrates the coherent effect of technical and institutional innovations. The EP 'Cumulative demand of natural uranium' (EP-2.1 from Table 4.1) is used in this section in order to demonstrate some results of the study related to the assurance of nuclear fuel supply as an important dimension of nuclear energy sustainability.

As noted in Section 7 (Fig. 7.7), identified and undiscovered conventional natural uranium resources estimated in the Red Book [11.8] as available at economical prices, ensure uranium fuel supply for the BAU system to the middle of the century both for the moderate and high growth GAINS scenarios. At the same time, demand for natural uranium for the BAU system by 2100 would significantly exceed the estimated resource for these scenarios. Even involvement of unconventional uranium resources, such as phosphate rocks or non-ferrous ores (excluding that available in sea water), does not ensure a sustainable fuel supply for the BAU scenario in 2100 (Fig. 7.7).

Simulating scenarios with FRs (BAU with FRs) and a break-even CR (CR: ~1) using the homogeneous model showed (Fig. 7.22) that introduction of recycling and a break-even FR significantly reduces consumption of uranium in the global system. Although total uranium usage exceeds projected resources for the high case nuclear growth, it becomes roughly equal to the currently identified total projected uranium resource in the moderate growth scenario. It is realized that this result is credible only within the domain of applicability of the homogeneous model and assumes that plutonium from SF of all thermal reactors is available for use in FRs and that many FRs are in operation all over the world. However, currently only a few countries have plans and programmes for their deployment.

The heterogeneous model has given more flexibility to take into account preferences of countries developing nuclear energy technology and different options for multinational cooperation. When estimated with the use of the separate heterogeneous model, uranium savings by the end of the century due to the introduction of FRs with a break-even CR (CR: ~1) and SF reprocessing is about 10 Mt of natural uranium for the case of high nuclear energy growth and 6 Mt for the case of moderate growth. This is approximately a quarter of total consumption by the BAU system in each case.

Application of the synergistic heterogeneous model for estimation of the cumulative demand of natural uranium by introduction of FRs has identified a strong dependence of the demand on the architecture of the global NES and the intensity of material flows between strategy groups. For the framework base case (Section 7), in which the NG3 group with minimal fuel cycle infrastructure obtains front and back end fuel cycle services from NG1 and NG2, the synergistic effect on global uranium demand is very small. In contrast, the analysis of alternate NES architecture configurations examined within sensitivity studies reveals variants in which this effect becomes essential.

For instance, when all SF from NG2 and NG3 groups is recycled in NG1 and all plutonium from this SF is used, the uranium saving due to the introduction of FRs redoubles compared to the non-synergistic heterogeneous case with the same types of FRs. As can be seen in Fig. 11.1, the moderate GAINS demand scenario, assuming energy generation of 2500 GW(e)/a by the end of the century, could be realized under the constraint of 16 Mt of natural uranium consumption when all reprocessed plutonium from SF of the NG2 and NG3 groups is used in NG1 in FRs with a BR of 1.2. This example shows that synergy can provide significant uranium savings without introduction of recycling and FR technologies in the countries of GAINS strategy groups NG2 and NG3 which would not like to incur the economic and other risks associated with development of these technologies.

For the high demand scenario (5000 GW(e)/a by the end of the century), Fig. 11.1 predicts that the same BAU+FR system would need more than 16 Mt of uranium. Thus, modelling the architectures based on the BAU+FR system, when the FR BR is in the range 1–1.2, does not exclude a possible shortage of the estimated uranium resource after the third quarter of the century unless more advanced fuel cycles are adopted. This opinion is shared by some of the GAINS participants. In a few countries, additional technical innovations are being considered, such as increasing the BR of FRs [11.9] or introducing the Th fuel cycle [11.7].

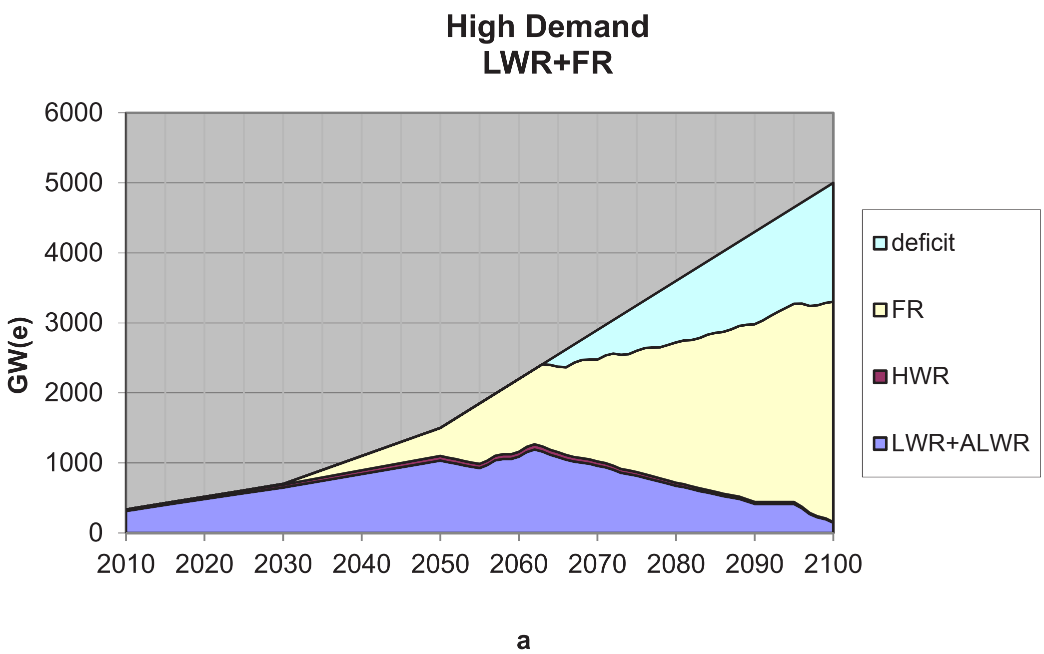

FIG. 11.1. Nuclear power structure for the BAU+ scenario with an FR system and an FR breeding ratio of 1.2.

Switching to thorium reactors, which do not require natural uranium, would be an alternative to the adoption of FRs with a low BR (BR: ~1.0). Sections 9.4–9.6 have examined two possibilities for the implementation of the thorium cycle in a thermal-spectrum HWR (such as a CANDU) — a once-through fuel cycle and a fuel cycle with ^{233}U recycling. The results of R&D studies carried out in some countries participating in GAINS showed that recycling the spent plutonium from LWRs into the ThPu fuel cycle, which might be easier to implement in the near term, would reduce natural uranium requirements although not as much as fuel cycles with break-even or breeder FRs. Switching to thorium, estimated to be 3–4 times more common in the Earth's crust than uranium, may be a complementary option for the increased supply of nuclear fuel resource, especially in those countries where uranium resources are limited or which are not considering the introduction of systems based on a U/Pu CNFC.

R&D programmes on achieving both a high BR (BR > 1.2) and high burnup of FRs or introducing the Th fuel cycle can require investment across the entire fuel cycle, including fuel performance, reactor systems and reprocessing technology development. Not all GAINS participants share the conviction on the expediency of these efforts in the existing economic environment and that expected in the medium term. Some participants support the view that more uranium resources (maximum: ~38 Mt) will be identified in the future. Although the differing views regarding the degree to which uranium resources will prove constraining goes beyond the scope of the GAINS ToR, the project contributed to the development of a reliable instrument for assessing the scale of uranium consumption under different nuclear energy demands and nuclear architecture configurations. This instrument enables one to compare uranium and thorium consumption versus their resources specified at global or national level and, thus, support decision making in providing assurance of nuclear fuel resources for long term nuclear strategies. It follows from the study implemented by GAINS participants that MNAs might have an essential part in the assurance of nuclear material resources in the twenty-first century.

11.3.3. Discharged fuel inventories

The management of discharged fuel from nuclear power reactors (also referred to as used fuel or SNF) is an important concern related to the use of nuclear energy. It is shown in the study that synergistic variants of a global NES architecture might lead to efficient long term SF management strategies. These strategies can facilitate SF management for a specific group of countries or globally.

It was assumed in the study for the framework base cases that the GAINS NG NG3 follows a strategy to limit infrastructure investments by only building reactors and obtaining fuel cycle services from NG1 (recycling group) and NG2 (once-through fuel cycle group). In this scenario, any waste generated by reprocessing of NG3 SF for use in reactors in NG1 is kept in NG1. As scrutinized in Section 7 and illustrated in related figures, the global impacts on most of the KIs and performance parameters of NG1 and NG2, including those related to discharged fuel, are very small.

At the same time, benefits are significant for all groups. NG3 benefits by not having to develop, site and construct NFC facilities, including those related to the disposition of highly radioactive SF. NG1 and NG2 must augment their fuel cycle infrastructures to support this strategy. In return, NG1 gains a source of additional used LWR fuel to support its strategy of transitioning to FRs. Benefits to NG2 are also common to other groups, including supporting the global growth of nuclear power for economic development and reduced greenhouse gas emissions while seeking to reduce proliferation risks.

Potentially, the synergistic approach might provide more scaled options for decreasing discharged fuel inventories. Assuming no 'physical' limitations to the reprocessing of SF of all groups in NG1, the recovered plutonium (and any recovered uranium) could be used in fuel of FRs and thermal reactors. This speculative case was briefly outlined in Section 11.3.2 with regard to uranium resource savings but it can also be of interest for the management of discharged fuel and plutonium.

11.3.4. Plutonium inventories and plutonium management options

The national strategy on SF management substantially depends on attitudes towards plutonium accumulated in used nuclear fuel. A gradual transition from managed storage of used fuel in the BAU option to plutonium use in the BAU with FR system, was simulated using separated and synergistic approaches. Simulation of plutonium management options in GAINS scenarios indicated a high sensitivity of the related KIs to innovations in nuclear technologies and to innovations in the arrangement of the global nuclear architecture.

Modelling of FR introduction with a CNFC (CNFC–FR) into the global NES has shown positive effects on many GAINS indicators; however, one of the limitations on the deployment rate of FRs in the long term is the availability of plutonium. In the case of groups operating separately, the reprocessed plutonium for FRs of the NG1 was supposed to be provided by thermal reactors from the same group. In the synergistic case, that assumes reprocessing of SF of all groups in NG1 and the FRs of NG1 are supplied with the reprocessed plutonium from SF of all GAINS groups.

Figure 11.2(a) illustrates the potential of FR deployment in these two cases. The synergistic model demonstrates that the fleet of FRs in the synergistic case could be doubled compared to the separate case. It can be concluded that in the second half of the century, the plutonium inventory could be kept to a minimum through intensive introduction of MOX fuelled FRs in NG1 (the nuclear power share of NG1 in the world is set as 50% in this case study), and the arrangement of fresh and used thermal reactor fuel flows between the three groups of countries.

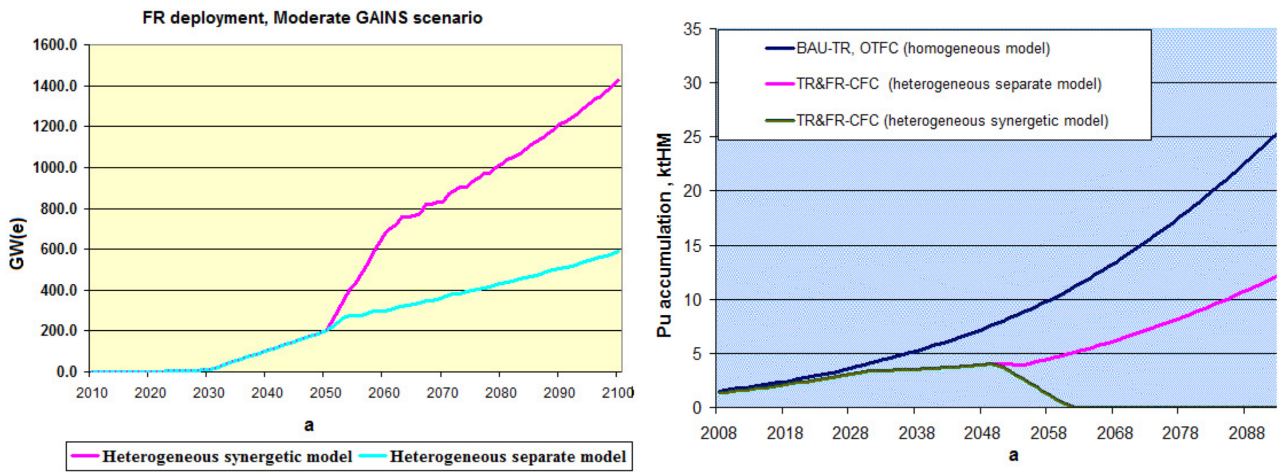

FIG. 11.2. (a) Potential for fast reactor deployment; (b) Plutonium accumulation in SNF after short term cooling in GAINS separate and synergistic heterogeneous cases (assumes recycle of SNF from all nuclear strategy groups).

Since plutonium is a long lived hazardous radioactive element, and, at the same time, it is a direct use material (KI-3) which can be suitable for manufacturing a nuclear explosive device, managing the plutonium inventory is very important both for nuclear waste management and for the reduction of potential proliferation risks. Keeping plutonium and other fissile material to the operational minimum, as shown in Figure 11.2(b), could contribute to assurances of global non-proliferation, but only when this principle is put into practice worldwide. The heterogeneous synergistic model indicates that innovative nuclear technologies based on U/Pu (or ThPu) NFCs are technical instruments for managing the plutonium balance in a global NES with reduced plutonium inventories achieved compared with the non-synergistic case. A few industrial limitations to managing inventories of used fuel and plutonium were identified in the study; for example, the idealized case of reprocessing ‘on demand’ leads to sharp reprocessing capacity spikes which are not compatible with industrial and economic realities. The necessity to operate reprocessing plants at the maximum capacity until the end of life is a limitation for the deployment rate of FRs.

11.3.5. Fuel cycle services

The movement of material between strategy groups in synergistic cases results in improving the ability of each group to follow their selected fuel cycle strategies. The heterogeneous model and related GAINS KIs (KI-6 and KI-7) help to understand the character and scale of necessary adjustments. Section 7 demonstrates how a minimal cycle infrastructure in NG3 might require additional fuel cycle infrastructure in NG1 and NG2 to provide front end services of mining, converting and enriching uranium and fabricating fresh LWR fuel, and back end services of taking back used LWR fuel. An assessment of the modified reprocessing rates for NG1 to account for the additional SF by NG3 is also provided in the section.

Although scenario studies implemented in GAINS demonstrate some technical and institutional opportunities for managing SF and plutonium inventories, many realities and limitations were not taken into account. Fuel transport is one of the issues to be comprehensively considered in the future as an important component of the global NES architecture. The evaluation of costs for SF and plutonium management would also provide valuable information for key milestones and transition phases from SF storage over long periods of time to advanced SF and plutonium management strategies.

Annex I illustrates some issues of a long range national nuclear energy strategy in Ukraine as a typical country of NG2. This example demonstrates a set of challenges and choices associated with searching for the balance between national and multinational solutions in NFC. Many countries are addressing or will address similar problems.

11.3.6. Other innovative nuclear energy system scenarios

When simulating the global NES over a time horizon of about 100 years, it was found reasonable to include in the examination innovative nuclear designs and components of a future NES that are now in the stages of pre-conceptual or conceptual development. Some of these concepts are being considered as the ‘second stratum’ of a global NES dedicated to transmutation of MAs. Characteristics of the lead cooled EFIT fuelled with MAs (U-free fuel) were used in GAINS to supply reference data for an ADS. Characteristics of the high flux MSR (HFMSR) with a fast spectrum considered in the joint Russian Federation–Czech Republic study represented a reference design for the MSR. A separate introduction of these systems into the global NES architecture was simulated after 2075.

The preliminary analysis of results indicated many common features for these two systems. It was shown that the inclusion of the MA dedicated systems could provide a reduction in MA accumulation (Figs 9.15 and 9.16, and 9.34 and 9.35) when employing a very small portion of ADSs or MSRs in the global NES (2–3%). Availability of the proven technology of MA transmutation by the last third of the century would give an opportunity to avoid overloading FRs with the function of MA burning at the first stage of their commercial introduction. Being a small portion of the NES, MSRs or ADSs may also be considered an integral part of future multilateral NFC centres in possible combination with MA-dedicated FRs.

11.4. SYNERGISTIC APPROACH TO BUILDING A GLOBAL ARCHITECTURE FOR SUSTAINABLE NUCLEAR ENERGY SYSTEMS

Sample analysis of NES scenarios using the GAINS framework has shown quantitatively that a synergistic approach based on technological and institutional innovations could provide significant potential for win–win collaboration between all NGs. This approach can enhance nuclear power production, preserve material resources, reduce inventories of SF and direct use material, and provide potential benefits in other areas, such as economics.

The introduction of innovative nuclear technologies in some strategy groups serves as a driving force for enhancing the sustainability features of a global NES architecture, while multilateral approaches can amplify the positive effects of the innovations. The higher the nuclear energy demand, the higher the positive effect of technical and institutional innovations. Only synergistic NES architectures are capable of providing a global response to global challenges related to a large scale deployment of nuclear power in the world.

Although the scenario study implemented in GAINS has demonstrated some technical and institutional opportunities for managing SF and plutonium inventories, many realities and limitations from industry, infrastructure and legislation were not taken into account. Fuel transport is one of the issues to be comprehensively considered in the future as an important component of the global NES architecture. The evaluation of costs for SF and plutonium management would also provide valuable information for key milestones and transition phases from postponed decisions such as SF storage to advanced SF and plutonium management strategies. It is also understood that further progress in the legislative basis and public consensus are crucial issues for realization of the synergistic back end architecture.

Identification of specific options for, and practical steps in realizing, innovative synergistic global NES architectures based on beneficial cooperation between nuclear power technology holders and users requires feedback from interested IAEA Member States on their demand-and-supply potentials, the industrial limitations and institutional challenges, acceptable cost of nuclear services, and desirable links to a global infrastructure. GAINS has made an important contribution by developing a standard analytical framework for assessing such options. Pathways towards sustainable nuclear power as part of synergistic global architectures with minimal financial, environmental and political risks can be further defined and evaluated via new CPs.

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Annex I

EXAMPLE OF ISSUES AND CHOICES UNDER DEVELOPMENT OF A LONG RANGE NATIONAL NUCLEAR ENERGY STRATEGY (CASE OF UKRAINE)

Although Ukraine is not a developer of nuclear technologies, it has approximately 30 years of experience in using nuclear energy. Pursuant to the Global Architecture of Innovative Nuclear Energy Systems Based on Thermal and Fast Reactors Including a Closed Fuel Cycle (GAINS) project classification, Ukraine belongs to the second non-geographic country group: countries having significant experience in the use of nuclear energy and most of the necessary infrastructure available, but not so clear readiness to incorporate rapidly the most advanced nuclear energy system (NES) from the moment of its commercial availability.

I–1. UKRAINE'S ENERGY SYSTEM

The key energy sources in Ukraine are: thermal power plants with a total installed capacity of 33 GW: nuclear power plants — 13.8 GW, hydro power plants — 5.4 GW and others — 0.0943 GW [I–1]. Taking into account the obsolescence and physical wear-out of the thermal power plants' equipment and the substantial production cost of the electricity they produce, the distribution of the total electricity production in 2010 by generation types was as follows: 47.4% — nuclear power plants, 45.7% — thermal power plants, 6.9% — hydro power plants.

Ukrainian nuclear power plants operate water cooled, water moderated power reactors (WWERs) of different capacities: 2 power units are WWER-440 and 13 units are WWER-1000 (with insignificant design variances from the standpoint of system modelling). The four RBMK-1000 reactors at the Chernobyl nuclear power plant have recently been shut down and are undergoing decommissioning.

I–2. URANIUM RESOURCES IN UKRAINE

Ukraine has substantial uranium ore deposits. The ore is extracted mostly by mining and partially by in situ leaching. The majority of ore deposits are base uranium ore. The reasonably assured uranium deposits in Ukraine at a price of less than $130/kg uranium amount to 105 kt. The expert estimate of uranium ore deposits at a price less than $260/kg uranium amount to 250 kt [I–2]. The design capacity of the existing hydrometallurgical facilities amounts to 1500 t of uranium concentrate per year, the actual is 800 t/a and there are plans of increasing the capacity to 4000 t/a by 2030 [I–3].

I–3. RADIOACTIVE WASTE MANAGEMENT

The radioactive waste management strategy is implemented based on the deferred decision concept:

- Zaporizhzhya nuclear power plant (6 operating units, total installed capacity of 6 GW) operates an on-site dry storage facility with a design capacity of 380 containers with 24 spent fuel assemblies in each;
- The remaining nuclear power plants (9 power units, total installed capacity of 7.8 GW) temporarily store the spent nuclear fuel (SNF) in spent fuel pits with plans for subsequent transport to, and storage in, the Centralized SNF Dry Storage Facility, that was recently commissioned;
- SNF from the decommissioned Chernobyl nuclear power plant is planned to be stored in a dedicated dry storage facility using horizontal containers 'NUHOMS' (Framatome) with a design capacity of 21 356 fuel assemblies of spent fuel;
- There are no SNF processing facilities in Ukraine.

I–4. ELECTRICITY CONSUMPTION FORECAST FOR UKRAINE UNTIL 2100

According to the Organization of the Petroleum Exporting Countries forecast, the average annual gross domestic product growth rate in countries with a transitional economy will amount to 2.8% in the period from 2010 to 2030 [I–4]. For a macroeconomic forecast for Ukraine until 2100, see Fig. I–1. Ukraine's population is expected to decrease by 10% by 2030, with a subsequent reduction in the rate. The expected population for 2100 is approximately 37.5 million.

From the standpoint of the energy consumption forecast, two main scenarios have been considered: 'economy as it is' and economic development with extensive use of renewable energy sources. For an energy consumption forecast until 2100, see Fig. I–2. Given substantial use of non-conventional and renewable energy sources, the highest development rate is expected for wind and solar energy systems. The consumption estimate variance for both scenarios until 2100 is under 1% and is inconsequential for the long term forecast.

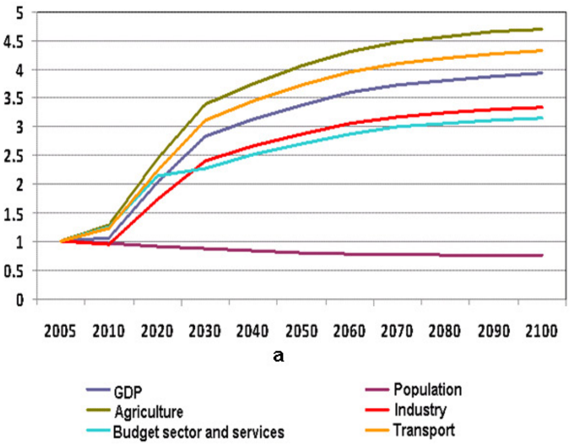


FIG. I–1. Ukraine's macroeconomic forecast against 2005 values.

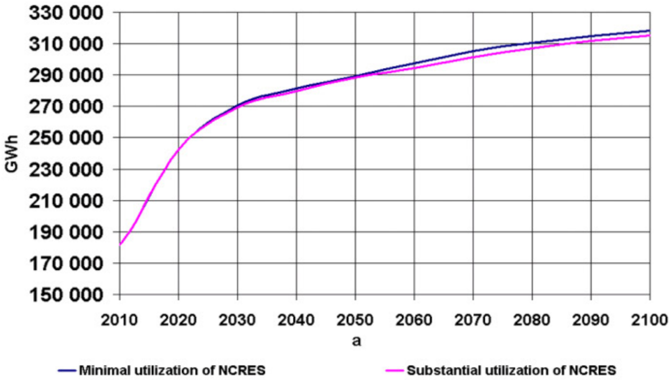


FIG. I–2. Electricity consumption forecast for Ukraine.

In the initial phase, covering the period until 2030, the energy consumption forecast for Ukraine is almost proportionate to the GAINS forecast used to model a moderate scenario for the second country group. On this basis, it is reasonable to presume that if a nuclear share in the energy pie of Ukraine were retained the results derived for the second group would, in the middle term, qualitatively correspond to the modelling results for the domestic forecast. From 2030 on, the energy consumption forecast for Ukraine shows noticeable saturation resulting from the expected increased efficiency and decreased energy intensity of the economy. As a result, the expected nuclear energy development scenario for Ukraine could be different from that for the second country group in the long term.

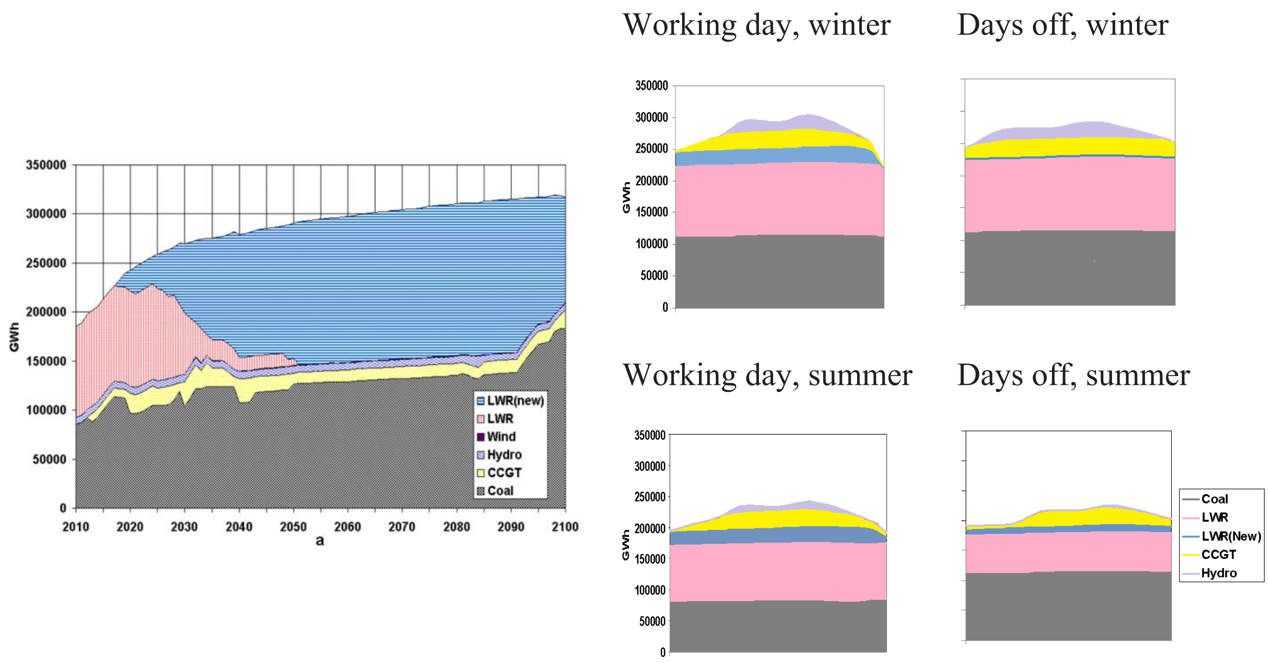
I–5. PROSPECTS OF NUCLEAR GENERATION DEVELOPMENT IN UKRAINE UNTIL 2100

To assess nuclear generation development prospects and structure in Ukraine, the MESSAGE code was used with a single-region energy system model of Ukraine. The modelling employed two scenarios — one assuming the current nuclear generation limitation to approximately 50% of the total electricity production, and the other – with no such limitation. It was assumed that the amount of uranium consumed by nuclear power plants is limited to reserves available in Ukraine.

To permit comparative assessment of the generation structure, both an open nuclear fuel cycle (NFC) with thermal neutron reactors and a closed NFC with thermal and fast neutron reactors were considered. Conversion, uranium enrichment and fuel fabrication technologies were addressed as services. To fabricate fuel for fast reactors (FRs), depleted uranium after enrichment and plutonium recovered from SNF of light water reactors (LWRs) and

FRs are used. Uranium generated during SNF reprocessing will not be used as nuclear fuel within the considered time span. Radioactive waste disposal was not addressed in the model.

A forecast for the generation structure in an open NFC with a nuclear share limited to 50% is given in Fig. I–3. The main electricity production is ensured by thermal (coal-fired) and nuclear power plants, with a small share of electricity production covered by hydro and gas-fired plants. The nuclear share in electricity production will not decrease until 2090. From then on, the nuclear share will decline as a result of a depletion of uranium reserves and a subsequent decrease in the nuclear power plants' total installed capacity. Given the forecasted electricity consumption rate and adopted limitations for nuclear generation, the total installed nuclear power plant capacity is not expected to go beyond 23 GW by 2100.



a b

FIG. I–3. Generation structure forecast given a nuclear share limited to 50% and an 'open' NFC; (a) annual average; (b) day and seasonal fluctuation.

Extensive use of renewable sources (hydro and wind power) does not substantially alter the electricity generation structure. Wind power is expected starting from 2030 and its share in the total generation will amount to 3% by 2100.

When the 'under 50%' limitation is lifted for nuclear generation, the electricity production at nuclear power plants will dominate over other generation types (see Fig. I–4). Starting from 2070, the electricity production at nuclear power plants will be declining until 2090, when it will be phased out since the domestic uranium reserves will have been exhausted. Under this scenario, nuclear power plants are expected to be involved in the regulation of the energy system parameters.

The share of renewable energy sources will not substantially impact the generation structure, just as in the scenario with a limitation of the nuclear power plant capacity.

A closed NFC was modelled for FRs with a breeding ratio of 0.98. Two scenarios were modelled: optimization in terms of economic efficiency and assigning priority to FRs starting from 2050. The generation structure forecast for both scenarios of a closed NFC model is given in Fig. I–5.

According to the moderate scenario used in the GAINS project, starting from 2020, FRs will be commissioned in the first country group (technology developers) at a rate of 1–2 reactors per year. Starting from 2030, the rate of FR commissioning could be increased up to 13 power units per year to end up with approximately 260 fast

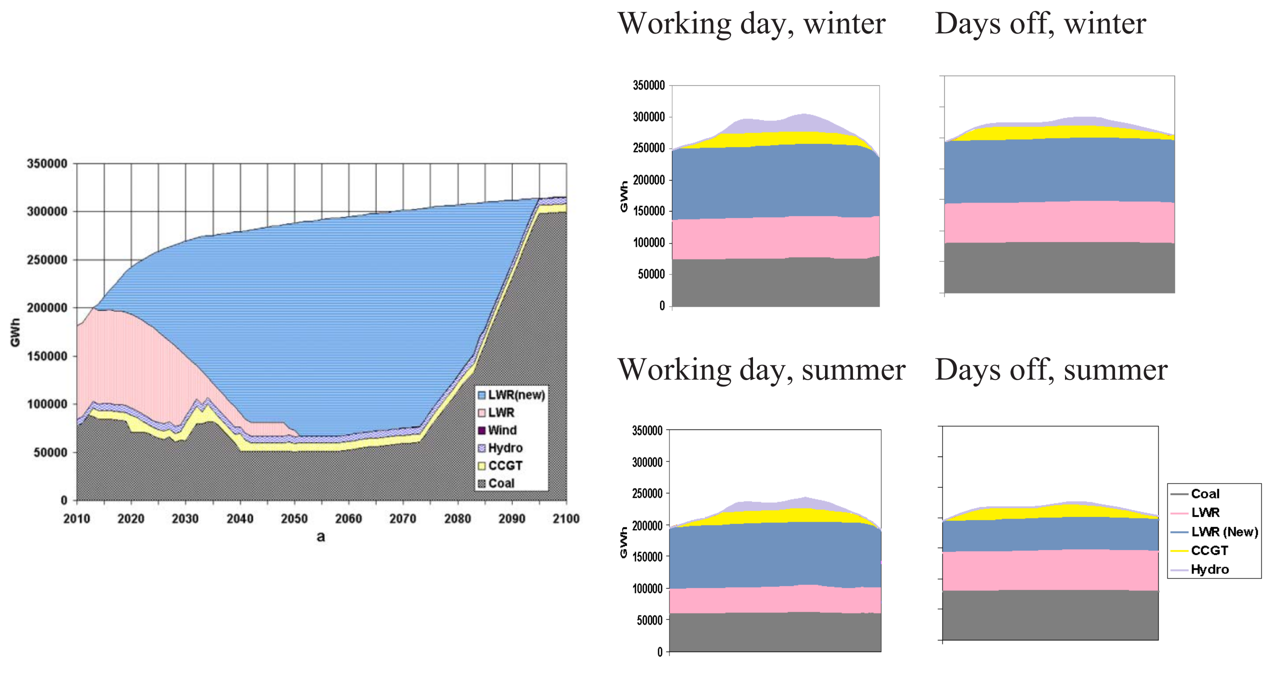


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FIG. I–4. Generation structure forecast with no limitations for a nuclear share in an 'open' NFC: (a) annual average; (b) day and seasonal fluctuations.

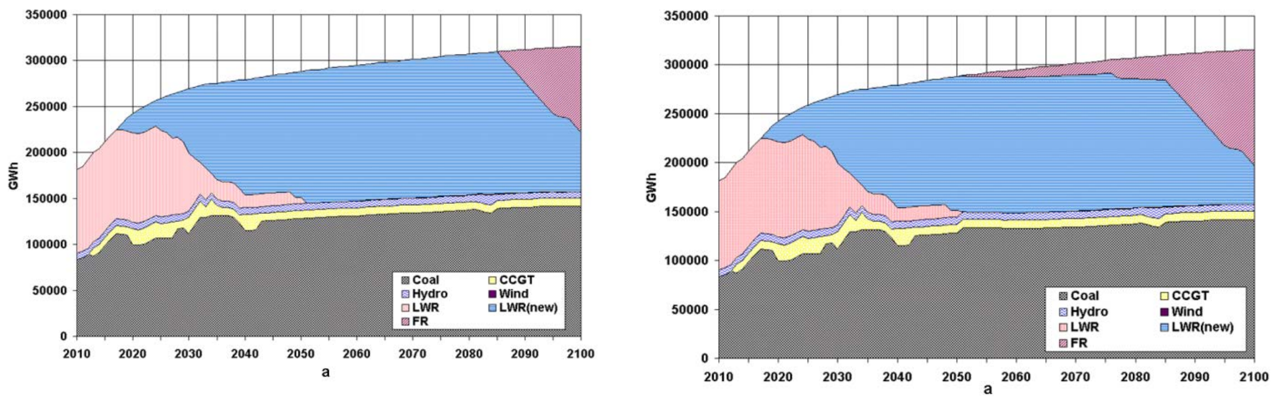

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FIG. I–5. Electricity generation structure forecast for a closed NFC, optimized in terms of (a) economic efficiency and (b) prioritization of fast reactors.

breeder reactors by 2050. The construction rate and widespread distribution of fast reactors prompt a reasonable expectation of their availability to the second country group by the middle of the twenty-first century, provided specific non-proliferation requirements are met (e.g. radioactive waste reprocessing and fuel fabrication facilities for FRs could be concentrated in the first country group).

The economic optimization model implies FR commissioning in Ukraine starting from 2080, which most likely will be preconditioned by depletion of domestic natural uranium resources. Commissioning of FRs is not the only scenario for the development of the energy system of Ukraine after 2080. Further energy production is possible with thermal reactors and uranium import. From the standpoint of the security of supply, each of these scenarios implemented separately could be vulnerable; therefore, in the future, a mixed energy system could be preferred using both FRs and thermal reactors.

At the same time, a scenario with prioritization of FRs starting from 2050 is preferred due to the following. A stepwise programme of FR construction will permit mitigation of the economic burden of construction capital costs, ensure stepwise infrastructure preparation and personnel training, accumulation of experience of FR construction and operation, and related production. Since the 'deferred decision' concept is temporary in nature, it requires a solution for SNF management and high level radioactive waste disposal. Introduction of FRs will make it possible to substantially decrease SNF accumulation and solve the issues related to its long term storage. Introduction of the closed NFC with FRs will permit more efficient use of the uranium potential due to ^{235}U and, as a result, the optimized use of natural uranium resources.

The impact of the closed NFC on the SNF inventory is demonstrated in Fig. I–6. Introduction of FRs starting from 2080 will permit a reduction in the LWR SNF inventory by approximately one third towards the end of the century. If FRs are commissioned starting from 2050, the SNF inventory could be reduced by a further 15%.

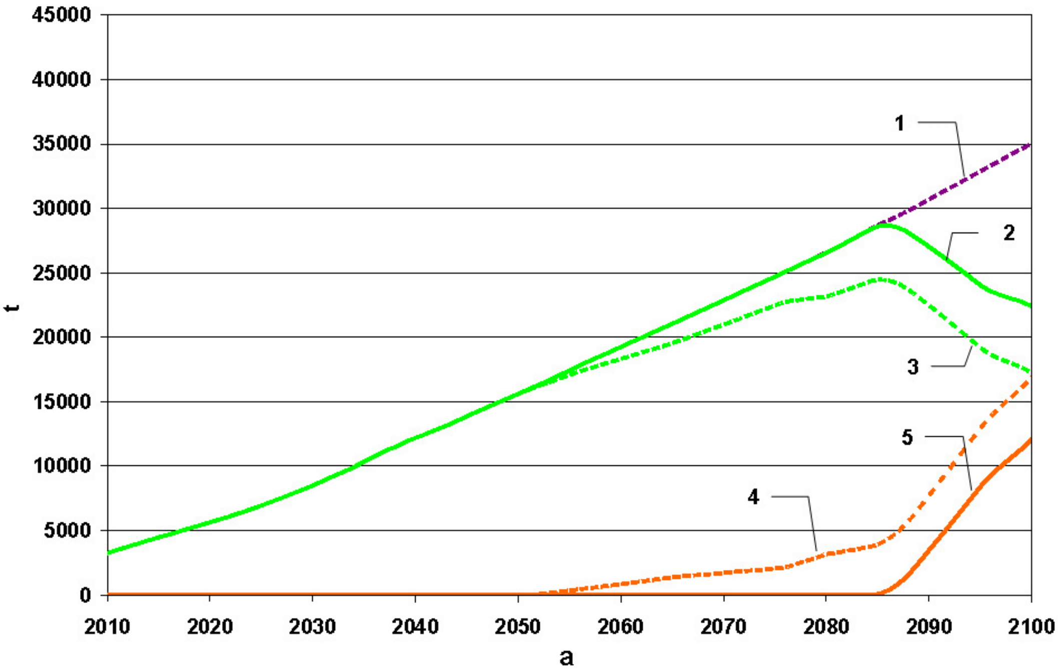

FIG. I–6. Accumulation forecast for LWR SNF (1–3) and reprocessed uranium (4, 5) for the following scenarios: without FR — 1; assumed FR commissioning starting from 2080 — 2, 5; FR priority commissioning starting from 2050 — 3, 4.

The system was modelled based on the assumption that only depleted uranium is used for fabrication of FR fuel; therefore, this model shows accumulation of reprocessed uranium in the respective amount. The issue of using reprocessed uranium was not addressed in the current study.

In general, the results obtained demonstrate the feasibility of changing over to FR technology in the second half of the century for reasons of natural uranium reserve depletion and due to the need to reduce the SNF inventory, implying costs for long term storage and disposal.

I–6. INTERNATIONAL COOPERATION WITH REGARD TO THE NUCLEAR FUEL CYCLE OF UKRAINE

Taking into account the available infrastructure and Ukraine's plans for NFC development until 2100, one can speak of a gradual depletion of domestic natural uranium reserves and accumulation of substantial amounts of SNF should the 'deferred decision' concept continue.

Calculations of natural uranium consumption demonstrate that if uranium is consumed to supply only domestic needs, uranium ore reserves will have been exhausted by the end of the century. On the assumption of a 60 year design service life for thermal reactors, introduction of this reactor type in Ukraine starting from

2040–2050 holds little promise. This testifies to the prospects of innovative technology deployment after 2050 with nuclear fuel breeding and burning of minor actinides.

Ukraine does not possess the required technologies and expertise in SNF reprocessing. Development and implementation of these technologies would require substantial capital investments. These problems can only be solved within international cooperation on the establishment of a closed NFC using FRs.

I–7. PROPOSALS FOR INTERNATIONAL COOPERATION DOMAINS

The international cooperation scheme, terms of interaction and responsibility distribution must be more extensively investigated in the following areas:

- NFC development and modelling with account of international cooperation in the establishment of regional centres for SNF reprocessing to reduce spent nuclear fuel amounts given the existing generating structure;
- Comparative economic assessment of existing and innovative NFCs using fast neutron technologies;
- Development of cooperation to secure sustainability of different NFC types;
- Development of an NES to permit an economically feasible transition from an open to a closed NFC using FRs.

This will permit an assessment of the resource base, technical and economic potential of the stakeholders, legal aspects, issues of diversification of supplies and the required capital investments.

REFERENCES TO ANNEX I

[I–1] NATIONAL POWER COMPANY UKRENERGO, Changes of the Installed Capacity of the Unified Energy System of Ukraine in 2010, NEC “Ukrenergo”, Kiev (2011).

[I–2] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, NUCLEAR ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Uranium 2009: Resources, Production and Demand, OECD, Paris (2010).

[I–3] CABINET OF MINISTERS OF THE UKRAINE (CMU), Decree #1004 of 23.09.2009, National Programme “Nuclear Fuel of Ukraine”, Kiev (2009).

[I–4] ORGANIZATION OF PETROLEUM EXPORTING COUNTRIES, World Oil Outlook, OPEC, Vienna (2010).

Annex II

FUEL COMPOSITION DATA OF EACH REACTOR SYSTEM

Fuel composition data for each of the reactor systems considered in the scenario studies of the Global Architecture of Innovative Nuclear Energy Systems Based on Thermal and Fast Reactors Including a Closed Fuel Cycle (GAINS) project are provided in Tables II–1 to II–13:

— Table II–1: Low burnup light water reactor (‘L1’).
— Table II–2: High burnup light water reactor (‘L2’) (advanced light water reactor).
— Table II–3: Medium burnup light water reactor (‘L3’).
— Table II–4: Heavy water reactor (‘H1’).
— Table II–5: Break-even fast reactor (‘F1’).
— Table II–6: Medium breeding ratio, medium burnup, breeder fast reactor (‘F2’).
— Table II–7: Medium breeding ratio, high burnup, breeder fast reactor (‘F3’).
— Table II–8: Burner fast reactor (‘F4’).
— Table II–9: Lead cooled fast reactor (‘F5’).
— Table II–10: Minor actinide burning lead cooled accelerator driven system (‘A1’).
— Table II–11: Minor actinide burning molten salt reactor (‘M1’).
— Table II–12: ThO_2 and PuO_2 CANDU (‘H2’).
— Table II–13: ThO_2, ^{233}U and PuO_2 CANDU (‘H3’).

In the tables, reload and discharge correspond to an equilibrium refuelling cycle.

TABLE II–1. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF LOW BURNUP LIGHT WATER REACTORS (‘L1’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-235	1.888E+03	2.400E+00	7.865E+02	4.000E+00	1.566E+02	7.965E-01	6.265E+02	7.965E-01
U-236					1.020E+02	5.189E-01	4.081E+02	5.189E-01
U-238	7.677E+04	9.760E+01	1.888E+04	9.600E+01	1.827E+04	9.290E+01	7.307E+04	9.290E+01
Np-237					1.365E+01	6.940E-02	5.459E+01	6.940E-02
Pu-238					5.040E+00	2.563E-02	2.016E+01	2.563E-02
Pu-239					1.063E+02	5.405E-01	4.251E+02	5.405E-01
Pu-240					4.133E+01	2.102E-01	1.653E+02	2.102E-01
Pu-241					3.645E+01	1.854E-01	1.458E+02	1.854E-01
Pu-242					1.538E+01	7.820E-02	6.151E+01	7.820E-02
Am-241					1.233E+00	6.270E-03	4.932E+00	6.270E-03
Am-242m					2.949E-02	1.500E-04	1.180E-01	1.500E-04
Am-243					3.604E+00	1.833E-02	1.442E+01	1.833E-02
Cm-242					4.306E-01	2.190E-03	1.723E+00	2.190E-03
Cm-244					1.262E+00	6.420E-03	5.050E+00	6.420E-03
Total FP					9.122E+02	4.639E+00	3.649E+03	4.639E+00
Total HM & FP	78653.193	100.000	19663.298	100	19663.298	100.000	78653.193	100.000
Total U	78653.193	100.000	19663.298	100	18526.467	94.219	74105.867	94.219
Total Pu	0.000	0.000	0.000	0	204.467	1.040	817.867	1.040
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0	20.206	0.103	80.824	0.103

TABLE II–2. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF HIGH BURNUP LIGHT WATER REACTORS (‘L2’) (ADVANCED LIGHT WATER REACTORS)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-232					6.140E-05	1.905E-07	1.027E-04	7.968E-08
U-233					1.097E-04	3.404E-07	3.756E-04	2.914E-07
U-234	4.315E+01	3.348E-02	1.434E+01	4.450E-02	6.898E+00	2.141E-02	3.978E+01	3.086E-02
U-235	4.383E+03	3.400E+00	1.595E+03	4.950E+00	2.523E+02	7.831E-01	2.508E+03	1.946E+00
U-236					2.142E+02	6.648E-01	6.563E+02	5.092E-01
U-238	1.245E+05	9.657E+01	3.062E+04	9.501E+01	2.929E+04	9.088E+01	1.193E+05	9.255E+01
Np-237					2.821E+01	8.755E-02	6.520E+01	5.058E-02
Pu-238					1.473E+01	4.571E-02	2.560E+01	1.986E-02
Pu-239					2.053E+02	6.371E-01	7.550E+02	5.857E-01
Pu-240					9.926E+01	3.080E-01	2.570E+02	1.994E-01
Pu-241					6.290E+01	1.952E-01	1.609E+02	1.248E-01
Pu-242					3.337E+01	1.035E-01	5.951E+01	4.616E-02
Am-241					2.405E+00	7.462E-03	5.302E+00	4.113E-03
Am-242m					3.872E-02	1.202E-04	8.387E-02	6.507E-05
Am-243					8.775E+00	2.723E-02	1.355E+01	1.051E-02
Cm-242					1.052E+00	3.263E-03	1.896E+00	1.471E-03
Cm-244					4.388E+00	1.362E-02	5.871E+00	4.555E-03
Total FPs					2.006E+03	6.224E+00	5.043E+03	3.913E+00
Total HM & FPs	128900.000	100.000	32225.000	100	32224.666	100.000	128899.577	100.000
Total U	128900.000	100.000	32225.000	100	29758.530	92.347	122506.387	95.040
Total Pu	0.000	0.000	0.000	0	415.544	1.290	1257.971	0.976
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0	44.869	0.139	91.903	0.071

TABLE II–3. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF MEDIUM BURNUP LIGHT WATER REACTORS (‘L3’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-234	3.00E+01	3.44E-02	1.00E+01	3.44E-02	5.002E+00	1.719E-02	1.501E+01	1.719E-02
U-235	3.75E+03	4.300E+00	1.251E+03	4.30E+00	2.226E+02	7.649E-01	6.677E+02	7.649E-01
U-236					1.660E+02	5.706E-01	4.981E+02	5.706E-01
U-238	8.35E+04	9.567E+01	2.784E+04	9.57E+01	2.680E+04	9.209E+01	8.039E+04	9.209E+01
Np-237					1.802E+01	6.192E-02	5.406E+01	6.192E-02
Pu-238					8.536E+00	2.933E-02	2.561E+01	2.933E-02
Pu-239					1.791E+02	6.154E-01	5.372E+02	6.154E-01
Pu-240					8.456E+01	2.906E-01	2.537E+02	2.906E-01
Pu-241					5.116E+01	1.758E-01	1.535E+02	1.758E-01
Pu-242					2.514E+01	8.639E-02	7.542E+01	8.639E-02
Am-241					1.875E+00	6.443E-03	5.625E+00	6.443E-03
Am-242m					2.483E-02	8.534E-05	7.450E-02	8.534E-05
Am-243					5.779E+00	1.986E-02	1.734E+01	1.986E-02
Cm-242					7.577E-01	2.604E-03	2.273E+00	2.604E-03
Cm-244					2.497E+00	8.579E-03	7.490E+00	8.579E-03
Cm-245					1.665E-01	5.722E-04	4.995E-01	5.722E-04
Total FPs					1.532E+03	5.264E+00	4.595E+03	5.264E+00
Total HM & FPs	87301.172	100.000	29100.391	100	29100.332	100.000	87300.995	100.000
Total U	87301.172	100.000	29100.391	100	27190.991	93.439	81572.974	93.439
Total Pu	0.000	0.000	0.000	0	348.472	1.197	1045.417	1.197
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0	29.120	0.100	87.359	0.100

TABLE II–4. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF HEAVY WATER REACTORS (‘H1’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-235	5.932E+02	7.110E-01	5.932E+02	7.110E-01	1.982E+02	2.375E-01	1.982E+02	2.375E-01
U-236					5.933E+01	7.111E-02	5.933E+01	7.111E-02
U-238	8.284E+04	9.929E+01	8.284E+04	9.929E+01	8.225E+04	9.858E+01	8.225E+04	9.858E+01
Np-237					2.161E+00	2.590E-03	2.161E+00	2.590E-03
Pu-238					2.753E-01	3.300E-04	2.753E-01	3.300E-04
Pu-239					2.218E+02	2.658E-01	2.218E+02	2.658E-01
Pu-240					7.984E+01	9.570E-02	7.984E+01	9.570E-02
Pu-241					1.513E+01	1.813E-02	1.513E+01	1.813E-02
Pu-242					3.279E+00	3.930E-03	3.279E+00	3.930E-03
Am-241					1.168E-01	1.400E-04	1.168E-01	1.400E-04
Am-242m								
Am-243					1.001E-01	1.200E-04	1.001E-01	1.200E-04
Cm-242					4.171E-02	5.000E-05	4.171E-02	5.000E-05
Cm-244					8.343E-03	1.000E-05	8.343E-03	1.000E-05
Total FPs					6.021E+02	7.217E-01	6.021E+02	7.217E-01
Total HM & FPs	83428.581	100.000	83428.581	100	83428.573	100.000	83428.573	100.000
Total U	83428.581	100.000	83428.581	100	82503.792	98.892	82503.792	98.892
Total Pu	0.000	0.000	0.000	0	320.291	0.384	320.291	0.384
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0	2.428	0.003	2.428	0.003

TABLE II–5. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF BREAK-EVEN FAST REACTORS (‘F1’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-234					3.863E-03	4.951E-05	7.944E-03	3.271E-05
U-235	6.458E+01	2.659E-01	2.065E+01	2.646E-01	1.932E+01	2.476E-01	6.668E+01	2.745E-01
U-236					1.695E+00	2.173E-02	4.017E+00	1.654E-02
U-238	2.146E+04	8.836E+01	6.862E+03	8.794E+01	6.537E+03	8.377E+01	2.073E+04	8.534E+01
Np-237					1.037E+00	1.329E-02	2.262E+00	9.312E-03
Pu-238	1.381E+01	5.685E-02	4.602E+00	5.898E-02	3.522E-01	4.514E-03	5.661E-01	2.331E-03
Pu-239	1.657E+03	6.822E+00	5.523E+02	7.078E+00	5.767E+02	7.390E+00	1.762E+03	7.253E+00
Pu-240	6.766E+02	2.786E+00	2.255E+02	2.890E+00	2.459E+02	3.151E+00	7.280E+02	2.997E+00
Pu-241	3.010E+02	1.239E+00	1.003E+02	1.286E+00	7.410E+01	9.496E-01	2.463E+02	1.014E+00
Pu-242	1.132E+02	4.662E-01	3.774E+01	4.837E-01	4.006E+01	5.134E-01	1.193E+02	4.913E-01
Am-241					3.926E+00	5.031E-02	8.531E+00	3.512E-02
Am-242m					8.594E-02	1.101E-03	1.455E-01	5.990E-04
Am-243					2.960E+00	3.793E-02	6.071E+00	2.500E-02
Cm-242					2.694E-01	3.452E-03	4.793E-01	1.973E-03
Cm-244					3.094E-01	3.966E-03	4.930E-01	2.030E-03
Cm-245					1.039E-02	1.331E-04	1.425E-02	5.868E-05
Total FPs					2.997E+02	3.841E+00	6.166E+02	2.539E+00
Total HM & FPs	24288.257	100.000	7803.086	100.000	7803.086	100.000	24288.257	100.000
Total U	21526.758	88.630	6882.586	88.203	6557.715	84.040	20797.868	85.629
Total Pu	2761.499	11.370	920.500	11.797	937.062	12.009	2855.758	11.758
Total MAs (Np + Am + Cm)	13.807	0.057	0.000	0.000	8.598	0.110	17.996	0.074

TABLE II–6. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF MEDIUM BREEDING RATIO, MEDIUM BURNUP, BREEDER FAST REACTORS (‘F2’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-232					8.174E-07	1.129E-08	1.585E-06	5.800E-09
U-233					4.955E-07	6.843E-09	1.385E-06	5.069E-09
U-234	3.635E-05	1.330E-07	9.689E-06	1.338E-07	2.153E-03	2.973E-05	5.177E-03	1.895E-05
U-235	6.257E+01	2.290E-01	1.625E+01	2.243E-01	1.198E+01	1.654E-01	4.861E+01	1.779E-01
U-236	2.497E-05	9.139E-08	5.899E-06	8.143E-08	1.039E+00	1.434E-02	3.433E+00	1.257E-02
U-238	2.528E+04	9.251E+01	6.566E+03	9.064E+01	6.308E+03	8.711E+01	2.447E+04	8.956E+01
Np-237					7.297E-01	1.008E-02	1.950E+00	7.135E-03
Pu-238					2.282E-01	3.151E-03	4.452E-01	1.629E-03
Pu-239	1.363E+03	4.987E+00	4.543E+02	6.271E+00	4.974E+02	6.869E+00	1.678E+03	6.142E+00
Pu-240	4.894E+02	1.791E+00	1.631E+02	2.252E+00	1.786E+02	2.466E+00	5.310E+02	1.943E+00
Pu-241	1.051E+02	3.845E-01	3.502E+01	4.835E-01	3.112E+01	4.297E-01	9.646E+01	3.530E-01
Pu-242	2.708E+01	9.911E-02	9.028E+00	1.246E-01	1.036E+01	1.431E-01	2.992E+01	1.095E-01
Am-241					1.927E+00	2.661E-02	4.217E+00	1.543E-02
Am-242m					3.587E-02	4.953E-04	6.205E-02	2.271E-04
Am-243					8.079E-01	1.116E-02	1.707E+00	6.246E-03
Cm-242					1.153E-01	1.592E-03	2.077E-01	7.602E-04
Cm-244					1.711E-01	2.362E-03	2.845E-01	1.041E-03
Cm-245					8.287E-03	1.144E-04	1.197E-02	4.380E-05
Total FPs					1.987E+02	2.744E+00	4.554E+02	1.667E+00
Total HM & FPs	27328.107	100.000	7243.872	100.000	7241.501	100.000	27325.399	100.000
Total U	25343.780	92.739	6582.430	90.869	6321.330	87.293	24525.374	89.753
Total Pu	1984.328	7.261	661.443	9.131	717.673	9.911	2336.169	8.549
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0.000	3.795	0.052	8.440	0.031

TABLE II–7. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF MEDIUM BREEDING RATIO, HIGH BURNUP, BREEDER FAST REACTORS (‘F3’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-235	3.984E+02	2.765E-01	9.960E+01	2.765E-01	6.150E+01	1.708E-01	2.921E+02	2.028E-01
U-236					8.020E+00	2.227E-02	2.296E+01	1.594E-02
U-238	1.324E+05	9.190E+01	3.310E+04	9.190E+01	3.080E+04	8.553E+01	1.265E+05	8.781E+01
Np-237	5.608E+01	3.893E-02	1.402E+01	3.893E-02	1.136E+01	3.154E-02	4.842E+01	3.361E-02
Pu-238	1.233E+02	8.557E-02	3.082E+01	8.557E-02	3.715E+01	1.032E-01	1.401E+02	9.723E-02
Pu-239	6.062E+03	4.208E+00	1.515E+03	4.208E+00	1.857E+03	5.158E+00	6.979E+03	4.845E+00
Pu-240	3.597E+03	2.497E+00	8.992E+02	2.497E+00	8.661E+02	2.405E+00	3.506E+03	2.434E+00
Pu-241	4.818E+02	3.345E-01	1.205E+02	3.345E-01	1.315E+02	3.651E-01	5.206E+02	3.614E-01
Pu-242	4.370E+02	3.033E-01	1.093E+02	3.033E-01	1.002E+02	2.782E-01	4.126E+02	2.864E-01
Am-241	2.243E+02	1.557E-01	5.608E+01	1.557E-01	4.418E+01	1.227E-01	1.890E+02	1.312E-01
Am-242m					3.620E+00	1.005E-02	1.169E+01	8.115E-03
Am-243	1.122E+02	7.785E-02	2.804E+01	7.785E-02	2.873E+01	7.978E-02	1.146E+02	7.953E-02
Cm-242					3.080E+00	8.552E-03	1.316E+01	9.135E-03
Cm-244	1.122E+02	7.785E-02	2.804E+01	7.785E-02	3.270E+01	9.080E-02	1.246E+02	8.652E-02
Cm-245					5.480E+00	1.522E-02	1.586E+01	1.101E-02
Total FPs	5.860E+01		1.465E+01		2.020E+03	5.610E+00	5.165E+03	3.586E+00
Total HM & FPs	144065.480	100.000	36016.370	100.000	36013.240	100.000	144055.150	100.000
Total U	132801.400	92.181	33200.350	92.181	30871.440	85.722	126813.710	88.031
Total Pu	10700.760	7.428	2675.190	7.428	2992.280	8.309	11558.810	8.024
Total MAs (Np + Am + Cm)	504.720	0.350	126.180	0.350	129.150	0.359	517.360	0.359

TABLE II–8. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF BURNER FAST REACTORS (‘F4’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-232	8.557E-04	2.232E-06	1.370E-04	2.232E-06	4.291E-04	6.994E-06	2.680E-03	6.994E-06
U-233					1.227E-04	2.000E-06	7.664E-04	2.000E-06
U-234	2.251E+01	5.873E-02	3.604E+00	5.873E-02	3.461E+00	5.640E-02	2.162E+01	5.640E-02
U-235	1.636E+01	4.269E-02	2.620E+00	4.269E-02	1.589E+00	2.589E-02	9.923E+00	2.589E-02
U-236	1.554E+01	4.055E-02	2.488E+00	4.055E-02	2.479E+00	4.041E-02	1.549E+01	4.041E-02
U-238	3.016E+04	7.870E+01	4.829E+03	7.870E+01	4.293E+03	6.997E+01	2.682E+04	6.997E+01
Np-237	1.282E+02	3.345E-01	2.053E+01	3.345E-01	1.255E+01	2.046E-01	7.841E+01	2.046E-01
Pu-238	1.987E+02	5.185E-01	3.182E+01	5.185E-01	2.888E+01	4.706E-01	1.804E+02	4.706E-01
Pu-239	3.968E+03	1.035E+01	6.353E+02	1.035E+01	5.764E+02	9.394E+00	3.600E+03	9.394E+00
Pu-240	2.359E+03	6.156E+00	3.777E+02	6.156E+00	3.544E+02	5.775E+00	2.213E+03	5.775E+00
Pu-241	4.120E+02	1.075E+00	6.596E+01	1.075E+00	5.453E+01	8.888E-01	3.406E+02	8.888E-01
Pu-242	5.260E+02	1.372E+00	8.421E+01	1.372E+00	7.576E+01	1.235E+00	4.733E+02	1.235E+00
Am-241	2.059E+02	5.372E-01	3.296E+01	5.372E-01	2.545E+01	4.147E-01	1.589E+02	4.147E-01
Am-242m	1.272E+01	3.318E-02	2.036E+00	3.318E-02	1.795E+00	2.926E-02	1.121E+01	2.926E-02
Am-243	1.663E+02	4.340E-01	2.663E+01	4.340E-01	2.474E+01	4.033E-01	1.546E+02	4.033E-01
Cm-242	1.372E+00	3.580E-03	2.197E-01	3.580E-03	1.458E+00	2.376E-02	9.108E+00	2.376E-02
Cm-244	1.044E+02	2.725E-01	1.672E+01	2.725E-01	1.730E+01	2.819E-01	1.081E+02	2.819E-01
Cm-245	2.663E+01	6.948E-02	4.263E+00	6.948E-02	4.426E+00	7.214E-02	2.765E+01	7.214E-02
Total FPs					6.576E+02	1.072E+01	4.108E+03	1.072E+01
Total HM & FPs	38328.385	100.000	6136.002	100	6135.671	100.000	38326.316	100.000
Total U	30218.263	78.840	4837.650	78.840	4300.397	70.088	26862.319	70.088
Total Pu	7464.493	19.475	1194.993	19.475	1089.904	17.763	6808.061	17.763
Total MAs (Np + Am + Cm)	645.630	1.684	103.359	1.684	87.721	1.430	547.945	1.430

TABLE II–9. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF LEAD COOLED FAST REACTORS (‘F5’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-234	1.289E+00	2.507E-03	1.289E+00	2.507E-03	6.827E+00	1.336E-02	6.827E+00	1.336E-02
U-235	1.736E+02	3.377E-01	1.736E+02	3.377E-01	1.002E+02	1.960E-01	1.002E+02	1.960E-01
U-236	4.297E+00	8.358E-03	4.297E+00	8.358E-03	2.111E+01	4.131E-02	2.111E+01	4.131E-02
U-238	4.279E+04	8.323E+01	4.279E+04	8.323E+01	3.975E+04	7.778E+01	3.975E+04	7.778E+01
Np-237	0.000E+00	0.000E+00	0.000E+00	0.000E+00	8.408E+00	1.645E-02	8.408E+00	1.645E-02
Pu-238	1.969E+02	3.829E-01	1.969E+02	3.829E-01	1.391E+02	2.722E-01	1.391E+02	2.722E-01
Pu-239	4.801E+03	9.338E+00	4.801E+03	9.338E+00	5.034E+03	9.851E+00	5.034E+03	9.851E+00
Pu-240	2.279E+03	4.433E+00	2.279E+03	4.433E+00	2.426E+03	4.748E+00	2.426E+03	4.748E+00
Pu-241	5.153E+02	1.002E+00	5.153E+02	1.002E+00	3.841E+02	7.517E-01	3.841E+02	7.517E-01
Pu-242	6.494E+02	1.263E+00	6.494E+02	1.263E+00	5.973E+02	1.169E+00	5.973E+02	1.169E+00
Am-241					8.322E+01	1.629E-01	8.322E+01	1.629E-01
Am-242m					2.942E+00	5.757E-03	2.942E+00	5.757E-03
Am-243					5.864E+01	1.147E-01	5.864E+01	1.147E-01
Cm-242					2.568E+00	5.026E-03	2.568E+00	5.026E-03
Cm-244					1.073E+01	2.099E-02	1.073E+01	2.099E-02
Cm-245					6.920E-01	1.354E-03	6.920E-01	1.354E-03
Total FPs					2.477E+03	4.847E+00	2.477E+03	4.847E+00
Total HM & FPs	51413.203	100.000	51413.203	100.000	51104.932	100.000	51104.932	100.000
Total U	42971.814	83.581	42971.814	83.581	39879.504	78.035	39879.504	78.035
Total Pu	8441.389	16.419	8441.389	16.419	8581.012	16.791	8581.012	16.791
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0.000	167.204	0.327	167.204	0.327

TABLE II–10. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF MINOR ACTINIDE BURNING LEAD COOLED ACCELERATOR DRIVEN SYSTEMS (‘A1’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
U-234					8.432E+00	1.260E-01	8.432E+00	1.260E-01
U-235					3.699E-01	5.527E-03	3.699E-01	5.527E-03
U-236					5.234E-01	7.820E-03	5.234E-01	7.820E-03
U-238					3.294E-03	4.921E-05	3.294E-03	4.921E-05
Np-237	1.385E+02	2.096E+00	1.385E+02	2.096E+00	1.247E+02	1.863E+00	1.247E+02	1.863E+00
Pu-238	1.140E+02	1.725E+00	1.140E+02	1.725E+00	3.618E+02	5.405E+00	3.618E+02	5.405E+00
Pu-239	1.422E+03	2.153E+01	1.422E+03	2.153E+01	1.043E+03	1.558E+01	1.043E+03	1.558E+01
Pu-240	1.049E+03	1.588E+01	1.049E+03	1.588E+01	1.005E+03	1.502E+01	1.005E+03	1.502E+01
Pu-241	1.187E+02	1.797E+00	1.187E+02	1.797E+00	1.057E+02	1.579E+00	1.057E+02	1.579E+00
Pu-242	3.674E+02	5.561E+00	3.674E+02	5.561E+00	4.052E+02	6.054E+00	4.052E+02	6.054E+00
Am-241	2.737E+03	4.143E+01	2.737E+03	4.143E+01	2.087E+03	3.119E+01	2.087E+03	3.119E+01
Am-242m	9.234E+00	1.398E-01	9.234E+00	1.398E-01	8.503E+01	1.270E+00	8.503E+01	1.270E+00
Am-243	5.868E+02	8.881E+00	5.868E+02	8.881E+00	4.876E+02	7.284E+00	4.876E+02	7.284E+00
Cm-242	2.047E+00	3.098E-02	2.047E+00	3.098E-02	4.280E+01	6.395E-01	4.280E+01	6.395E-01
Cm-244	5.249E+01	7.944E-01	5.249E+01	7.944E-01	1.628E+02	2.432E+00	1.628E+02	2.432E+00
Cm-245	9.240E+00	1.399E-01	9.240E+00	1.399E-01	3.946E+01	5.896E-01	3.946E+01	5.896E-01
Total FPs					7.332E+02	1.095E+01	7.332E+02	1.095E+01
Total HM & FPs	6606.989	100.000	6606.989	100.000	6693.231	100.000	6693.231	100.000
Total U	0.000	0.000	0.000	0.000	9.329	0.139	9.329	0.139
Total Pu	3071.577	46.490	3071.577	46.490	2920.986	43.641	2920.986	43.641
Total MAs (Np + Am + Cm)	3535.411	53.510	3535.411	53.510	3029.730	45.266	3029.730	45.266

TABLE II–11. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF MINOR ACTINIDE BURNING MOLTEN SALT REACTORS (‘M1’)

Isotope	Initial loading (kg)		Reload (kg/a)		Discharge (kg/a)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
Th-230	3.080E-01	8.738E-04					3.080E-01	8.738E-04
Th-232	1.000E-03	2.837E-06					1.000E-03	2.837E-06
U-232	1.540E-01	4.369E-04					1.540E-01	4.369E-04
U-233	1.510E-01	4.284E-04					1.510E-01	4.284E-04
U-234	1.711E+03	4.853E+00					1.711E+03	4.853E+00
U-235	4.723E+02	1.340E+00					4.723E+02	1.340E+00
U-236	5.192E+02	1.473E+00					5.192E+02	1.473E+00
U-238	6.660E-01	1.889E-03					6.660E-01	1.889E-03
Np-237	5.066E+03	1.437E+01	3.947E+02	4.413E+01			5.066E+03	1.437E+01
Pu-238	9.170E+03	2.601E+01					9.170E+03	2.601E+01
Pu-239	3.089E+03	8.762E+00					3.089E+03	8.762E+00
Pu-240	3.459E+03	9.813E+00					3.459E+03	9.813E+00
Pu-241	5.393E+02	1.530E+00					5.393E+02	1.530E+00
Pu-242	2.574E+03	7.302E+00					2.574E+03	7.302E+00
Am-241	5.151E+03	1.461E+01	4.396E+02	4.914E+01			5.151E+03	1.461E+01
Am-242m	1.705E+02	4.836E-01	3.777E-01	4.222E-02			1.705E+02	4.836E-01
Am-243	1.440E+03	4.084E+00	5.280E+01	5.903E+00			1.440E+03	4.084E+00
Cm-242	2.168E+02	6.151E-01	9.943E-05	1.111E-05			2.168E+02	6.151E-01
Cm-244	1.280E+03	3.632E+00	6.367E+00	7.117E-01			1.280E+03	3.632E+00
Cm-245	3.910E+02	1.109E+00	7.003E-01	7.829E-02			3.910E+02	1.109E+00
Total FPs					9.898E+02	1.000E+02	0.000E+00	0.000E+00
Total HM & FPs	35250.318	100.000	894.589	100.000	989.785	100.000	35250.318	100.000
Total U	2703.099	7.668	0.000	0.000	0.000	0.000	2703.099	7.668
Total Pu	18831.332	53.422	0.000	0.000	0.000	0.000	18831.332	53.422
Total MAs (Np + Am + Cm)	13715.578	38.909	894.589	100.000	0.000	0.000	13715.578	38.909

TABLE II–12. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF ThO_2 AND PuO_2 CANDU (‘H2’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
Th-232	6.871E+04	9.624E+01	6.871E+04	9.624E+01	6.777E+04	9.492E+01	6.777E+04	9.492E+01
U-233					6.211E+02	8.700E-01	6.211E+02	8.700E-01
U-234					3.606E+01	5.051E-02	3.606E+01	5.051E-02
U-235					2.775E+00	3.887E-03	2.775E+00	3.887E-03
U-236					1.527E-01	2.139E-04	1.527E-01	2.139E-04
U-238					3.120E-03	4.370E-06	3.120E-03	4.370E-06
Np-237					2.981E-01	4.175E-04	2.981E-01	4.175E-04
Pu-238	6.716E+01	9.407E-02	6.716E+01	9.407E-02	4.015E+01	5.623E-02	4.015E+01	5.623E-02
Pu-239	1.456E+03	2.039E+00	1.456E+03	2.039E+00	3.106E+02	4.350E-01	3.106E+02	4.350E-01
Pu-240	6.394E+02	8.956E-01	6.394E+02	8.956E-01	6.229E+02	8.726E-01	6.229E+02	8.726E-01
Pu-241	3.385E+02	4.741E-01	3.385E+02	4.741E-01	1.558E+02	2.183E-01	1.558E+02	2.183E-01
Pu-242	1.827E+02	2.559E-01	1.827E+02	2.559E-01	2.640E+02	3.698E-01	2.640E+02	3.698E-01
Am-241					5.521E+01	7.734E-02	5.521E+01	7.734E-02
Am-242m					5.521E+01	7.734E-02	5.521E+01	7.734E-02
Am-243					1.192E-01	1.670E-04	1.192E-01	1.670E-04
Cm-242					3.957E+01	5.543E-02	3.957E+01	5.543E-02
Cm-244					1.552E-03	2.175E-06	1.552E-03	2.175E-06
Cm-245					6.458E+00	9.046E-03	6.458E+00	9.046E-03
Total FPs					1.413E+03	1.980E+00	1.413E+03	1.980E+00
Total HM & FPs	71390.664	100.000	71390.664	100	71390.664	100.000	71390.664	100.000
Total U	0.000	0.000	0.000	0.000	660.065	0.925	660.065	0.925
Total Pu	2683.664	3.759	2683.664	3.759	1393.552	1.952	1393.552	1.952
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0	156.871	0.220	156.871	0.220

TABLE II–13. COMPOSITION DATA OF FRESH AND DISCHARGED FUEL OF ThO_2, ^{233}U AND PuO_2 CANDU (‘H3’)

Isotopes	Initial loading (kg)		Reload (kg)		Discharge (kg)		Full core discharge at retirement (kg)	
	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)	Weight (kg)	(%)
Th-232	6.958E+04	9.747E+01	6.958E+04	9.747E+01	6.839E+04	9.581E+01	6.839E+04	9.581E+01
U-233	1.031E+03	1.445E+00	1.031E+03	1.445E+00	1.031E+03	1.445E+00	1.031E+03	1.445E+00
U-234					1.081E+02	1.514E-01	1.081E+02	1.514E-01
U-235					1.070E+01	1.500E-02	1.070E+01	1.500E-02
U-236					9.038E-01	1.266E-03	9.038E-01	1.266E-03
U-238					9.889E-04	1.385E-06	9.889E-04	1.385E-06
Np-237					9.707E-02	1.360E-04	9.707E-02	1.360E-04
Pu-238	1.940E+01	2.718E-02	1.940E+01	2.718E-02	1.077E+01	1.508E-02	1.077E+01	1.508E-02
Pu-239	4.206E+02	5.892E-01	4.206E+02	5.892E-01	5.505E+01	7.711E-02	5.505E+01	7.711E-02
Pu-240	1.847E+02	2.587E-01	1.847E+02	2.587E-01	1.558E+02	2.183E-01	1.558E+02	2.183E-01
Pu-241	9.778E+01	1.370E-01	9.778E+01	1.370E-01	4.395E+01	6.156E-02	4.395E+01	6.156E-02
Pu-242	5.277E+01	7.393E-02	5.277E+01	7.393E-02	8.345E+01	1.169E-01	8.345E+01	1.169E-01
Am-241					1.532E+01	2.146E-02	1.532E+01	2.146E-02
Am-242m					3.291E-02	4.611E-05	3.291E-02	4.611E-05
Am-243					1.417E+01	1.985E-02	1.417E+01	1.985E-02
Cm-242					4.894E-04	6.856E-07	4.894E-04	6.856E-07
Cm-244					2.321E+00	3.251E-03	2.321E+00	3.251E-03
Total FPs					1.460E+03	2.045E+00	1.460E+03	2.045E+00
Total HM & FPs	71385.517	100.000	71385.517	100	71385.517	100.000	71385.517	100.000
Total U	1031.254	1.445	1031.254	1.445	1150.970	1.612	1150.970	1.612
Total Pu	775.281	1.086	775.281	1.086	349.045	0.489	349.045	0.489
Total MAs (Np + Am + Cm)	0.000	0.000	0.000	0	31.935	0.045	31.935	0.045

Annex III

SCENARIO CASES AND DENOTATION

The members of GAINS project have been studying various scenarios using their own codes. Scenarios can be defined in general by the type and timing of nuclear energy system (NES) deployment, and more specifically by the complete set of assumptions used in their calculation or code case.

To document the used assumption, input data and the results obtained by each GAINS participant, a naming convention was developed to help identifying, and distinguishing between, analytical cases. The alpha-numeric naming convention consists of five fields, separated by hyphens, and is best described using an example:

NG0Syn-High-L1L2H1F1-FC2-N (III–1)

The first field in (III–1) denotes the analysis model. Current options are:

— HG0: Homogeneous global model.
— NG0Sep: Nuclear strategy group (NG) global model, separate case.
— NG0Syn: NG group global model, synergy case (‘0’ means global, if it is desired to extract each NG group or calculate only each NG group, it can be expressed with NG1Sep, NG2Syn, and so on).

The second field in (III–1) denotes the global nuclear power demand scenarios along the lines described in Section 5. Current options corresponding to the profiles discussed in Section 5 are:

— High: high global demand;
— Mod: moderate global demand.

The third field in (III–1) denotes the reactor/transmutation system types used in a particular scenario by listing their designators in Section 6. Current options include:

— L1: low burnup (45 GW·d/t) light water reactor.
— L2: high burnup (60 GW·d/t) light water reactor.
— L3: medium burnup (51 GW·d/t) light water reactor.
— H1: typical current heavy water reactor.
— H2: ThO_2 and PuO_2 CANDU.
— H3: ThO_2 and ^{233}U and PuO_2 CANDU.
— F1: ‘break-even’ (BR: ~1.0) fast reactor.
— F2: medium breeding ratio (BR: ~1.2), medium burnup (~31 GW·d/t) fast reactor.
— F3: medium breeding ratio (BR: ~1.2), high-burnup (~54 GW·d/t) fast reactor.
— A1: A driven system for minor actinide burning.
— M1: Molten salt reactor for minor actinide burning.

The fourth field in (III–1) denotes a set of analysis conditions related to the fuel cycle (FC), including any changes in load factor, tails assay, lead time, timing and rate of deployment of systems, and other assumptions or alterations made which uniquely define the scenario. This field is displayed ‘FC’ followed by a number. Specific designators for the primary fuel cycle conditions and a summary of their meanings/assumptions are:

— FC1: Initial set of fuel cycle assumptions using a single-light water reactor (LWR) model with a 0.3% tails assay, and a LWR:heavy water reactor (HWR) ratio of 94:6;
— FC2: Revised set of fuel cycle assumptions for base cases using a single-LWR model, with changes to load factor, lifetime and the fraction of power production provided by HWRs revised from FC1;
— FC3: Set of fuel cycle assumptions used in modelling advanced LWRs (ALWRs) separately from LWRs, with the tails assay change of 0.3 to 0.2%, (LWR+ALWR):HWR ratio of 94:6.

The detailed descriptions of all fuel cycle conditions are listed in Table III–1.

The fifth and last field in (III–1) is used to identify the fuel cycle code and version used to calculate the scenario. The first digit denotes the name of the code, and the next two digits the version number if necessary. Current options include:

— C: COSI.
— D: DESAE2.2.
— F: FAMILY.
— M: MESSAGE.
— N: NFCSS.
— S: DANESS.
— T: TEPS.
— V: VISION.

So, the case of BAU explained in Section 6.2 can be designated as HG0-High-L1H1-FC2-V, if the mass flow analysis is done for the high demand case and with the VISION code.

The contents of the attached CD-ROM document the input and output data of each analytical case considered within the GAINS study. The input and output data in each of the files (templates) on the CD-ROM follow the output formats of the NFCSS code. The designations of the cases follow the pattern described above and are used to name the corresponding templates (files).

The templates do not provide for automatic input to any of the used codes, i.e. manual input data transfer will be needed to repeat the calculations performed by GAINS participants using the same or different codes. The outputs are only documented in graphic form.

Table III–2 lists the GAINS templates provided by members and made available on the CD-ROM, following the denotations introduced above.

The data contained on the CD-ROM can be used to repeat, or elaborate upon, each of the analytical cases considered within GAINS, as well as in further studies of global architectures and scenarios involving present day and future NESs.

III–1. GAINS TEMPLATE

A GAINS template has been developed in order to visualize key indicators, evaluation parameters and other mass flows or works on the scenario study results. Owing to the short period of the project, there had to be some limitations or simplifications to show the indicators or evaluation parameters. These are as follows:

— The amount of spent fuel in nuclear power plant storage does not include the contribution from historical reactors for several initial years because the template limits the input period to that from 2008.
— The location and the inventory of plutonium and minor actinides are caluculated in a simple model, in which the compositions of spent fuel in nuclear power plant storage and long term storage are constant as 30 year cooled for LWRs, 5 year cooled for ALWRs and HWRs, and 2 year cooled for fast reactors. The composition of the reactor cores are simplified as the average between fresh fuel and discharged fuel, assuming equilibrium core states.
— The inventories of plutonium and minor actinides in a reprocessing facility and a fabrication facility are assumed such that one half of the annual throughput exists in each facility taking account of the process time of one year for the two processes. (Reprocessing: 6 months; fabrication: 6 months.)
— The template for the NG model was made as a workbook which consists of three NG groups and the global sum of the three groups. As the mass flows and the work are calculated to correspond to the nuclear power plant operations in each group, it is difficult to indicate the transfer of the fuel material or work between the groups in the synergistic model. Some scenario study results on synergistic cases are shown in two steps of the templates in order to grasp the necessary additional reprocessing of spent fuel from NG2 and NG3. (The characters of '-org' are attached in the last position of the file name of the first step.)

TABLE III–1. SET OF ANALYSIS CONDITIONS RELATED TO THE FUEL CYCLE (FC)

	FC1 : initial BAU	**FC2 : BAU and FR, other**	**FC3 : BAU+ and FR, other**
Load factors (%):			
LWR (L1)	80	85	80
ALWR (L2)	—	—	80
HWR (H1)	80	85	80
FR (F1, F2, F3)	—	85	85
Plant lifetimes (a):			
LWR (L1)	40	60	40
ALWR (L2)	—	—	60
HWR (H1)	40	60	40
FR (F1, F2, F3)	—	60	60
Other (MSR, ADS)	—	—	60
Historical LWR and HWR deployment scenario profile	Specified	Same as FC1	Same as FC1
Future HWR deployment	6% of total power supplied by thermal reactors	6% of total global nuclear power for homogeneous cases and as initial condition for heterogeneous cases. For heterogeneous cases, power supplied by HWRs is a constant fraction of total nuclear power generated in a group.	6% of total power supplied by thermal reactors
Fuel separation deployment	No recycle	All discharged fuel from LWR(L1) including the historical period from 1970 is available for separation. Separation of accumulated historical inventories of discharged LWR fuel is constrained to take place over a 40 year period with a constant annual separations rate.	All discharged fuel from LWR(L1) and ALWR(L2) including the historical period from 1970 is available for separation. No separation capacity limit.
Initial core enrichment Modelling: different enrichments used for initial core and equilibrium core	No	No	Ratios of initial core enrichment to equilibrium core: LWR (L1): 0.600 ALWR(L2): 0.689
Enrichment tails assay (%)	0.3	0.3	0.3 until 2014; 0.2 from 2015

TABLE III–1. SET OF ANALYSIS CONDITIONS RELATED TO THE FUEL CYCLE (FC) (cont.)

	FC1 : initial BAU	**FC2 : BAU and FR, other**	**FC3 : BAU+ and FR, other**
Fuel separation deployment	No recycle	All discharged fuel from LWR(L1) including the historical period from 1970 is available for separation. Separation of accumulated historical inventories of discharged LWR fuel is constrained to take place over a 40 year period with a constant annual separations rate.	All discharged fuel from LWR(L1) and ALWR(L2) including the historical period from 1970 is available for separation. No separation capacity limit.
	No	No	Ratios of initial core enrichment to equilibrium core: LWR (L1): 0.600 ALWR(L2): 0.689
Enrichment tails assay (%)	0.3	0.3	0.3 until 2014; 0.2 from 2015
Lead time for mining, conversion and fabrication (a)	0	2	0
Fuel recycle assumptions (as applicable):	No recycle	Out of reactor time (cooling + separations + fabrication): LWR(L1): 6 years FR(F1): 3 years Pu losses = 1%	Out of reactor time (cooling + separations + fabrication): LWR(L1), ALWR(L2): 6 years FR (F1, F2, F3): 3 years Pu losses = 1%

TABLE III–2. GAINS TEMPLATE FILE NAME LISTS STORED ON THE ATTACHED CD-ROM

	Scenario file name	Data provider	Note
1	HG0-High-L1H1-FC1-D.xls	D. Jaluvka, SCK-CEN, Belgium	
2	HG0-High-L1H1-FC1-F.xls	A. Ohtaki, JAEA, Japan	
3	HG0-High-L1H1-FC1-N.xls	H. Hayashi, IAEA/INPRO	
4	HG0-High-L1H1-FC2-V.xls	B. Dixon, INL, USA	
5	HG0-Mod-L1H1-FC1-N.xls	H. Hayashi, IAEA/INPRO	
6	HG0-High-L1L2H1-FC3-D.xls	D. Jaluvka, SCK-CEN, Belgium	
7	HG0-High-L1L2H1-FC3-F.xls	A. Ohtaki, JAEA, Japan	
8	HG0-High-L1L2H1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
9	HG0-Mod-L1L2H1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
10	HG0-High-L1H1F1-FC2-V.xls	B. Dixon, INL, USA	
11	HG0-High-L1L2H1F1-FC3-D.xls	D. Jaluvka, SCK-CEN, Belgium	
12	HG0-High-L1L2H1F1-FC3-F.xls	A. Ohtaki, JAEA, Japan	
13	HG0-High-L1L2H1F1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
14	HG0-High-L1L2H1F2-FC3-N.xls	H. Hayashi, IAEA/INPRO	
15	HG0-High-L1L2H1F3-FC3-N.xls	H. Hayashi, IAEA/INPRO	
16	HG0-Mod-L1H1F1-FC2-V.xls	B. Dixon, INL, USA	
17	HG0-Mod-L1L2H1F1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
18	HG0-Mod-L1L2H1F2-FC3-N.xls	H. Hayashi, IAEA/INPRO	
19	HG0-Mod-L1L2H1F3-FC3-N.xls	H. Hayashi, IAEA/INPRO	
20	NG0Sep-High-L1L2H1F1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
21	NG0Sep-High-L1L2H1F2-FC3-N.xls	H. Hayashi, IAEA/INPRO	
22	NG0Sep-High-L1L2H1F3-FC3-N.xls	H. Hayashi, IAEA/INPRO	
23	NG0Sep-Mod-L1L2H1F1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
24	NG0Sep-Mod-L1L2H1F2-FC3-N.xls	H. Hayashi, IAEA/INPRO	
25	NG0Sep-Mod-L1L2H1F3-FC3-N.xls	H. Hayashi, IAEA/INPRO	
26	NG0Sep-High-L1H1F1-FC2-V.xls	B. Dixon, INL, USA	
27	NG0Sep-Mod-L1H1F1-FC2-V.xls	B. Dixon, INL, USA	
28	NG0Syn-High-L1L2H1F1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
29	NG0Syn-High-L1L2H1F1-FC3-N-org.xls	H. Hayashi, IAEA/INPRO	Reprocess in NG2, 3
30	NG0Syn-High-L1L2H1F2-FC3-N.xls	H. Hayashi, IAEA/INPRO	
31	NG0Syn-High-L1L2H1F2-FC3-N-org.xls	H. Hayashi, IAEA/INPRO	Reprocess in NG2
32	NG0Syn-High-L1L2H1F3-FC3-N.xls	H. Hayashi, IAEA/INPRO	
33	NG0Syn-High-L1L2H1F3-FC3-N-org.xls	H. Hayashi, IAEA/INPRO	Reprocess in NG2
34	NG0Syn-Mod-L1L2H1F1-FC3-N.xls	H. Hayashi, IAEA/INPRO	
35	NG0Syn-Mod-L1L2H1F1-FC3-org.xls	H. Hayashi, IAEA/INPRO	Reprocess in NG2, 3
36	NG0Syn-Mod-L1L2H1F2-FC3-N.xls	H. Hayashi, IAEA/INPRO	
37	NG0Syn-Mod-L1L2H1F2-FC3-N-org.xls	H. Hayashi, IAEA/INPRO	Reprocess in NG2
38	NG0Syn-Mod-L1L2H1F3-FC3-N.xls	H. Hayashi, IAEA/INPRO	
39	NG0Syn-Mod-L1L2H1F3-FC3-N-org.xls	H. Hayashi, IAEA/INPRO	Reprocess in NG2
40	NG0Syn-High-L1H1F1-FC2-V.xls	B. Dixon, INL, USA	

TABLE III–2. GAINS TEMPLATE FILE NAME LISTS STORED ON THE ATTACHED CD-ROM (cont.)

	Scenario file name	Data provider	Note
41	NG0Syn-Mod-L1H1F1-FC2-V.xls	B. Dixon, INL, USA	
42	HG0-High-L1L2H1F1A1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
43	HG0-High-L1L2H1F2A1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
44	HG0-High-L1L2H1F3A1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
45	HG0-Mod-L1L2H1F1A1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
46	HG0-Mod-L1L2H1F2A1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
47	HG0-Mod-L1L2H1F3A1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
48	HG0-High-L1L2H1F1M1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
49	HG0-High-L1L2H1F2M1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
50	HG0-High-L1L2H1F3M1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
51	HG0-Mod-L1L2H1F1M1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
52	HG0-Mod-L1L2H1F2M1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
53	HG0-Mod-L1L2H1F3M1-FC3-N.xls	D. Ulanov, H. Hayashi, IAEA	
54	HG0-Mod-L1H1H2-FC2-D.xls	G. Edwards, AECL, Canada	
55	HG0-Mod-L1H1H2H3-FC2-D.xls	G. Edwards, AECL, Canada	

Annex IV

GROWTH RATE TABLES FOR GAINS FRAMEWORK BASE CASES

This annex contains the growth rate tables discussed in Section 7 for the framework base cases. All of the tables use a similar format with data listed by calendar year. The format includes separate groups of columns for the moderate and high growth scenarios. Tables IV–1 and IV–2 cover historical and projected growth, respectively, for the homogeneous world framework base cases, while Table IV–3 provides this information for the heterogeneous world.

For the homogeneous world cases, within each group there are subgroups for total power demand and total power capacity. Power demand reflects the amount of electricity generated, which is equivalent to power capacity times the capacity factor for the reactor type. Within each subgroup, there are separate columns for the contribution by each reactor type.

For the heterogeneous world cases, only the power demand is listed, with separate columns for the power demand in each of the three nuclear strategy groups NG1, NG2 and NG3.

Table IV–1 is historical growth from 1970 to 2010 for use in all homogeneous framework cases. The table provides data for heavy water reactors (HWRs) and light water reactors (LWRs) only. The data are identical for both the moderate and high cases for the historical period from 1970 to 2008.

TABLE IV–1. HISTORICAL GROWTH FOR USE IN HOMOGENEOUS FRAMEWORK CASES

	Moderate scenario						**High scenario**					
	Power demand (GW·a)			**Power capacity (GW(e))**			**Power demand (GW·a)**			**Power capacity (GW(e))**		
Year	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**
1970	9.0	0.0	9.0	10.6	0.0	10.6	9.0	0.0	9.0	10.6	0.0	10.6
1971	15.5	0.5	15.1	18.3	0.5	17.7	15.5	0.5	15.1	18.3	0.5	17.7
1972	23.0	0.6	22.4	27.0	0.7	26.4	23.0	0.6	22.4	27.0	0.7	26.4
1973	33.3	1.1	32.2	39.2	1.3	37.9	33.3	1.1	32.2	39.2	1.3	37.9
1974	47.1	1.4	45.6	55.4	1.7	53.7	47.1	1.4	45.6	55.4	1.7	53.7
1975	55.8	1.4	54.4	65.7	1.7	64.0	55.8	1.4	54.4	65.7	1.7	64.0
1976	67.9	1.4	66.5	79.9	1.7	78.2	67.9	1.4	66.5	79.9	1.7	78.2
1977	78.2	1.4	76.8	92.0	1.7	90.3	78.2	1.4	76.8	92.0	1.7	90.3
1978	91.1	2.1	88.9	107.1	2.5	104.6	91.1	2.1	88.9	107.1	2.5	104.6
1979	95.7	2.8	92.9	112.5	3.3	109.2	95.7	2.8	92.9	112.5	3.3	109.2
1980	108.7	2.8	105.9	127.9	3.3	124.6	108.7	2.8	105.9	127.9	3.3	124.6
1981	125.5	3.0	122.5	147.7	3.5	144.2	125.5	3.0	122.5	147.7	3.5	144.2
1982	138.9	3.0	135.9	163.4	3.5	159.9	138.9	3.0	135.9	163.4	3.5	159.9

TABLE IV–1. HISTORICAL GROWTH FOR USE IN HOMOGENEOUS FRAMEWORK CASES (cont.)

	Moderate scenario						High scenario					
	Power demand (GW·a)			Power capacity (GW(e))			Power demand (GW·a)			Power capacity (GW(e))		
Year	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR
1983	154.3	4.5	149.8	181.6	5.3	176.3	154.3	4.5	149.8	181.6	5.3	176.3
1984	181.0	6.5	174.5	212.9	7.6	205.3	181.0	6.5	174.5	212.9	7.6	205.3
1985	207.3	7.7	199.7	243.9	9.0	234.9	207.3	7.7	199.7	243.9	9.0	234.9
1986	228.9	9.0	219.9	269.3	10.6	258.7	228.9	9.0	219.9	269.3	10.6	258.7
1987	247.7	9.7	237.9	291.4	11.4	279.9	247.7	9.7	237.9	291.4	11.4	279.9
1988	257.9	9.7	248.1	303.4	11.4	291.9	257.9	9.7	248.1	303.4	11.4	291.9
1989	264.9	9.7	255.2	311.7	11.4	300.3	264.9	9.7	255.2	311.7	11.4	300.3
1990	272.0	10.5	261.5	320.0	12.4	307.6	272.0	10.5	261.5	320.0	12.4	307.6
1991	273.6	10.7	262.9	321.9	12.6	309.3	273.6	10.7	262.9	321.9	12.6	309.3
1992	276.1	11.7	264.4	324.9	13.8	311.1	276.1	11.7	264.4	324.9	13.8	311.1
1993	283.8	13.5	270.3	333.9	15.8	318.0	283.8	13.5	270.3	333.9	15.8	318.0
1994	286.4	13.5	272.9	336.9	15.8	321.0	286.4	13.5	272.9	336.9	15.8	321.0
1995	288.5	13.7	274.8	339.4	16.1	323.4	288.5	13.7	274.8	339.4	16.1	323.4
1996	293.4	14.3	279.1	345.2	16.8	328.4	293.4	14.3	279.1	345.2	16.8	328.4
1997	294.0	14.9	279.1	345.9	17.5	328.4	294.0	14.9	279.1	345.9	17.5	328.4
1998	294.6	15.5	279.1	346.6	18.2	328.4	294.6	15.5	279.1	346.6	18.2	328.4
1999	295.3	16.1	279.1	347.4	19.0	328.4	295.3	16.1	279.1	347.4	19.0	328.4
2000	297.2	16.9	280.3	349.6	19.8	329.8	297.2	16.9	280.3	349.6	19.8	329.8
2001	298.8	16.9	282.0	351.6	19.8	331.7	298.8	16.9	282.0	351.6	19.8	331.7
2002	303.4	17.5	285.9	356.9	20.5	336.4	303.4	17.5	285.9	356.9	20.5	336.4
2003	306.1	18.0	288.1	360.2	21.2	338.9	306.1	18.0	288.1	360.2	21.2	338.9
2004	310.9	18.0	292.8	365.8	21.2	344.5	310.9	18.0	292.8	365.8	21.2	344.5
2005	312.7	18.5	294.1	367.8	21.8	346.1	312.7	18.5	294.1	367.8	21.8	346.1
2006	313.1	19.0	294.1	368.4	22.3	346.1	313.1	19.0	294.1	368.4	22.3	346.1
2007	315.3	19.0	296.3	370.9	22.3	348.6	315.3	19.0	296.3	370.9	22.3	348.6
2008	316.2	19.0	297.2	372.0	22.3	349.7	316.2	19.0	297.2	372.0	22.3	349.7

TABLE IV–1. HISTORICAL GROWTH FOR USE IN HOMOGENEOUS FRAMEWORK CASES (cont.)

	Moderate scenario						**High scenario**					
	Power demand (GW·a)			**Power capacity (GW(e))**			**Power demand (GW·a)**			**Power capacity (GW(e))**		
Year	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**
2009	329.1	19.7	309.4	387.2	23.2	363.9	333.6	20.0	313.6	392.5	23.5	369.0
2010	342.0	20.5	321.5	402.3	24.1	378.2	351.1	21.1	330.0	413.0	24.8	388.3

Table IV–2 is the moderate and high growth cases for the business-as-usual (BAU) homogeneous world cases. These cases only include HWRs and LWRs. Data are provided starting in 2011, with earlier data provided in Table IV–1. Data extend to 2130 to allow analysts to ensure stability of their models beyond the end of the scenario in 2100. Specific features of these cases include a fixed electricity generation ratio of LWRs:HWRs of 94:6 during four separate growth periods. Each growth period is modelled as linear growth (not exponential) to reach a specific level of generation by the end of the period:

(a) 2009–2030: reaching 600 GW·a for the moderate case and 700 GW·a for the high case.
(b) 2031–2050: reaching 1000 GW·a for the moderate case and 1500 GW·a for the high case.
(c) 2051–2100: reaching 2500 GW·a for the moderate case and 5000 GW·a for the high case.
(d) 2101 and beyond: no additional growth, only replacement of retiring reactors.

TABLE IV–2. MODERATE AND HIGH GROWTH PROJECTIONS FOR THE BAU FRAMEWORK CASES

	Moderate scenario						**High scenario**					
	Power demand (GW·a)			**Power capacity, (GW(e))**			**Power demand (GW·a)**			**Power capacity (GW(e))**		
Year	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**
2011	354.9	21.3	333.6	417.5	25.0	392.5	368.5	22.1	346.4	433.6	26.0	407.6
2012	367.8	22.1	345.7	432.7	26.0	406.7	386.0	23.2	362.8	454.1	27.2	426.9
2013	380.7	22.8	357.9	447.9	26.9	421.0	403.4	24.2	379.2	474.6	28.5	446.1
2014	393.6	23.6	370.0	463.1	27.8	435.3	420.9	25.2	395.6	495.1	29.7	465.4
2015	406.5	24.4	382.1	478.2	28.7	449.5	438.3	26.3	412.0	515.7	30.9	484.7
2016	419.4	25.2	394.2	493.4	29.6	463.8	455.8	27.3	428.4	536.2	32.2	504.0
2017	432.3	25.9	406.4	508.6	30.5	478.1	473.2	28.4	444.8	556.7	33.4	523.3
2018	445.2	26.7	418.5	523.8	31.4	492.3	490.7	29.4	461.2	577.2	34.6	542.6
2019	458.1	27.5	430.6	538.9	32.3	506.6	508.1	30.5	477.6	597.8	35.9	561.9
2020	471.0	28.3	442.7	554.1	33.2	520.9	525.5	31.5	494.0	618.3	37.1	581.2
2021	483.9	29.0	454.9	569.3	34.2	535.1	543.0	32.6	510.4	638.8	38.3	600.5

TABLE IV–2. MODERATE AND HIGH GROWTH PROJECTIONS FOR THE BAU FRAMEWORK CASES (cont.)

	Moderate scenario						High scenario					
	Power demand (GW·a)			Power capacity, (GW(e))			Power demand (GW·a)			Power capacity (GW(e))		
Year	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR
2022	496.8	29.8	467.0	584.5	35.1	549.4	560.4	33.6	526.8	659.3	39.6	619.8
2023	509.7	30.6	479.1	599.6	36.0	563.7	577.9	34.7	543.2	679.9	40.8	639.1
2024	522.6	31.4	491.2	614.8	36.9	577.9	595.3	35.7	559.6	700.4	42.0	658.4
2025	535.5	32.1	503.4	630.0	37.8	592.2	612.8	36.8	576.0	720.9	43.3	677.7
2026	548.4	32.9	515.5	645.2	38.7	606.5	630.2	37.8	592.4	741.4	44.5	696.9
2027	561.3	33.7	527.6	660.4	39.6	620.7	647.7	38.9	608.8	762.0	45.7	716.2
2028	574.2	34.5	539.7	675.5	40.5	635.0	665.1	39.9	625.2	782.5	46.9	735.5
2029	587.1	35.2	551.9	690.7	41.4	649.3	682.6	41.0	641.6	803.0	48.2	754.8
2030	600.0	36.0	564.0	705.9	42.4	663.5	700.0	42.0	658.0	823.5	49.4	774.1
2031	620.0	37.2	582.8	729.4	43.8	685.6	740.0	44.4	695.6	870.6	52.2	818.4
2032	640.0	38.4	601.6	752.9	45.2	707.8	780.0	46.8	733.2	917.6	55.1	862.6
2033	660.0	39.6	620.4	776.5	46.6	729.9	820.0	49.2	770.8	964.7	57.9	906.8
2034	680.0	40.8	639.2	800.0	48.0	752.0	860.0	51.6	808.4	1011.8	60.7	951.1
2035	700.0	42.0	658.0	823.5	49.4	774.1	900.0	54.0	846.0	1058.8	63.5	995.3
2036	720.0	43.2	676.8	847.1	50.8	796.2	940.0	56.4	883.6	1105.9	66.4	1039.5
2037	740.0	44.4	695.6	870.6	52.2	818.4	980.0	58.8	921.2	1152.9	69.2	1083.8
2038	760.0	45.6	714.4	894.1	53.6	840.5	1020.0	61.2	958.8	1200.0	72.0	1128.0
2039	780.0	46.8	733.2	917.6	55.1	862.6	1060.0	63.6	996.4	1247.1	74.8	1172.2
2040	800.0	48.0	752.0	941.2	56.5	884.7	1100.0	66.0	1034.0	1294.1	77.6	1216.5
2041	820.0	49.2	770.8	964.7	57.9	906.8	1140.0	68.4	1071.6	1341.2	80.5	1260.7
2042	840.0	50.4	789.6	988.2	59.3	928.9	1180.0	70.8	1109.2	1388.2	83.3	1304.9
2043	860.0	51.6	808.4	1011.8	60.7	951.1	1220.0	73.2	1146.8	1435.3	86.1	1349.2
2044	880.0	52.8	827.2	1035.3	62.1	973.2	1260.0	75.6	1184.4	1482.4	88.9	1393.4
2045	900.0	54.0	846.0	1058.8	63.5	995.3	1300.0	78.0	1222.0	1529.4	91.8	1437.6
2046	920.0	55.2	864.8	1082.4	64.9	1017.4	1340.0	80.4	1259.6	1576.5	94.6	1481.9
2047	940.0	56.4	883.6	1105.9	66.4	1039.5	1380.0	82.8	1297.2	1623.5	97.4	1526.1

TABLE IV–2. MODERATE AND HIGH GROWTH PROJECTIONS FOR THE BAU FRAMEWORK CASES (cont.)

	Moderate scenario						High scenario					
	Power demand (GW·a)			Power capacity, (GW(e))			Power demand (GW·a)			Power capacity (GW(e))		
Year	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**	**Total**	**HWR**	**LWR**
2048	960.0	57.6	902.4	1129.4	67.8	1061.6	1420.0	85.2	1334.8	1670.6	100.2	1570.4
2049	980.0	58.8	921.2	1152.9	69.2	1083.8	1460.0	87.6	1372.4	1717.6	103.1	1614.6
2050	1000.0	60.0	940.0	1176.5	70.6	1105.9	1500.0	90.0	1410.0	1764.7	105.9	1658.8
2051	1030.0	61.8	968.2	1211.8	72.7	1139.1	1570.0	94.2	1475.8	1847.1	110.8	1736.2
2052	1060.0	63.6	996.4	1247.1	74.8	1172.2	1640.0	98.4	1541.6	1929.4	115.8	1813.6
2053	1090.0	65.4	1024.6	1282.4	76.9	1205.4	1710.0	102.6	1607.4	2011.8	120.7	1891.1
2054	1120.0	67.2	1052.8	1317.6	79.1	1238.6	1780.0	106.8	1673.2	2094.1	125.6	1968.5
2055	1150.0	69.0	1081.0	1352.9	81.2	1271.8	1850.0	111.0	1739.0	2176.5	130.6	2045.9
2056	1180.0	70.8	1109.2	1388.2	83.3	1304.9	1920.0	115.2	1804.8	2258.8	135.5	2123.3
2057	1210.0	72.6	1137.4	1423.5	85.4	1338.1	1990.0	119.4	1870.6	2341.2	140.5	2200.7
2058	1240.0	74.4	1165.6	1458.8	87.5	1371.3	2060.0	123.6	1936.4	2423.5	145.4	2278.1
2059	1270.0	76.2	1193.8	1494.1	89.6	1404.5	2130.0	127.8	2002.2	2505.9	150.4	2355.5
2060	1300.0	78.0	1222.0	1529.4	91.8	1437.6	2200.0	132.0	2068.0	2588.2	155.3	2432.9
2061	1330.0	79.8	1250.2	1564.7	93.9	1470.8	2270.0	136.2	2133.8	2670.6	160.2	2510.4
2062	1360.0	81.6	1278.4	1600.0	96.0	1504.0	2340.0	140.4	2199.6	2752.9	165.2	2587.8
2063	1390.0	83.4	1306.6	1635.3	98.1	1537.2	2410.0	144.6	2265.4	2835.3	170.1	2665.2
2064	1420.0	85.2	1334.8	1670.6	100.2	1570.4	2480.0	148.8	2331.2	2917.6	175.1	2742.6
2065	1450.0	87.0	1363.0	1705.9	102.4	1603.5	2550.0	153.0	2397.0	3000.0	180.0	2820.0
2066	1480.0	88.8	1391.2	1741.2	104.5	1636.7	2620.0	157.2	2462.8	3082.4	184.9	2897.4
2067	1510.0	90.6	1419.4	1776.5	106.6	1669.9	2690.0	161.4	2528.6	3164.7	189.9	2974.8
2068	1540.0	92.4	1447.6	1811.8	108.7	1703.1	2760.0	165.6	2594.4	3247.1	194.8	3052.2
2069	1570.0	94.2	1475.8	1847.1	110.8	1736.2	2830.0	169.8	2660.2	3329.4	199.8	3129.6
2070	1600.0	96.0	1504.0	1882.4	112.9	1769.4	2900.0	174.0	2726.0	3411.8	204.7	3207.1
2071	1630.0	97.8	1532.2	1917.6	115.1	1802.6	2970.0	178.2	2791.8	3494.1	209.6	3284.5
2072	1660.0	99.6	1560.4	1952.9	117.2	1835.8	3040.0	182.4	2857.6	3576.5	214.6	3361.9
2073	1690.0	101.4	1588.6	1988.2	119.3	1868.9	3110.0	186.6	2923.4	3658.8	219.5	3439.3

TABLE IV–2. MODERATE AND HIGH GROWTH PROJECTIONS FOR THE BAU FRAMEWORK CASES (cont.)

	Moderate scenario						High scenario					
	Power demand (GW·a)			Power capacity, (GW(e))			Power demand (GW·a)			Power capacity (GW(e))		
Year	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR
2074	1720.0	103.2	1616.8	2023.5	121.4	1902.1	3180.0	190.8	2989.2	3741.2	224.5	3516.7
2075	1750.0	105.0	1645.0	2058.8	123.5	1935.3	3250.0	195.0	3055.0	3823.5	229.4	3594.1
2076	1780.0	106.8	1673.2	2094.1	125.6	1968.5	3320.0	199.2	3120.8	3905.9	234.4	3671.5
2077	1810.0	108.6	1701.4	2129.4	127.8	2001.6	3390.0	203.4	3186.6	3988.2	239.3	3748.9
2078	1840.0	110.4	1729.6	2164.7	129.9	2034.8	3460.0	207.6	3252.4	4070.6	244.2	3826.4
2079	1870.0	112.2	1757.8	2200.0	132.0	2068.0	3530.0	211.8	3318.2	4152.9	249.2	3903.8
2080	1900.0	114.0	1786.0	2235.3	134.1	2101.2	3600.0	216.0	3384.0	4235.3	254.1	3981.2
2081	1930.0	115.8	1814.2	2270.6	136.2	2134.4	3670.0	220.2	3449.8	4317.6	259.1	4058.6
2082	1960.0	117.6	1842.4	2305.9	138.4	2167.5	3740.0	224.4	3515.6	4400.0	264.0	4136.0
2083	1990.0	119.4	1870.6	2341.2	140.5	2200.7	3810.0	228.6	3581.4	4482.4	268.9	4213.4
2084	2020.0	121.2	1898.8	2376.5	142.6	2233.9	3880.0	232.8	3647.2	4564.7	273.9	4290.8
2085	2050.0	123.0	1927.0	2411.8	144.7	2267.1	3950.0	237.0	3713.0	4647.1	278.8	4368.2
2086	2080.0	124.8	1955.2	2447.1	146.8	2300.2	4020.0	241.2	3778.8	4729.4	283.8	4445.6
2087	2110.0	126.6	1983.4	2482.4	148.9	2333.4	4090.0	245.4	3844.6	4811.8	288.7	4523.1
2088	2140.0	128.4	2011.6	2517.6	151.1	2366.6	4160.0	249.6	3910.4	4894.1	293.6	4600.5
2089	2170.0	130.2	2039.8	2552.9	153.2	2399.8	4230.0	253.8	3976.2	4976.5	298.6	4677.9
2090	2200.0	132.0	2068.0	2588.2	155.3	2432.9	4300.0	258.0	4042.0	5058.8	303.5	4755.3
2091	2230.0	133.8	2096.2	2623.5	157.4	2466.1	4370.0	262.2	4107.8	5141.2	308.5	4832.7
2092	2260.0	135.6	2124.4	2658.8	159.5	2499.3	4440.0	266.4	4173.6	5223.5	313.4	4910.1
2093	2290.0	137.4	2152.6	2694.1	161.6	2532.5	4510.0	270.6	4239.4	5305.9	318.4	4987.5
2094	2320.0	139.2	2180.8	2729.4	163.8	2565.6	4580.0	274.8	4305.2	5388.2	323.3	5064.9
2095	2350.0	141.0	2209.0	2764.7	165.9	2598.8	4650.0	279.0	4371.0	5470.6	328.2	5142.4
2096	2380.0	142.8	2237.2	2800.0	168.0	2632.0	4720.0	283.2	4436.8	5552.9	333.2	5219.8
2097	2410.0	144.6	2265.4	2835.3	170.1	2665.2	4790.0	287.4	4502.6	5635.3	338.1	5297.2
2098	2440.0	146.4	2293.6	2870.6	172.2	2698.4	4860.0	291.6	4568.4	5717.6	343.1	5374.6
2099	2470.0	148.2	2321.8	2905.9	174.4	2731.5	4930.0	295.8	4634.2	5800.0	348.0	5452.0

TABLE IV–2. MODERATE AND HIGH GROWTH PROJECTIONS FOR THE BAU FRAMEWORK CASES (cont.)

	Moderate scenario						High scenario					
	Power demand (GW·a)			Power capacity, (GW(e))			Power demand (GW·a)			Power capacity (GW(e))		
Year	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR
2100	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2101	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2102	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2103	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2104	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2105	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2106	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2107	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2108	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2109	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2110	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2111	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2112	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2113	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2114	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2115	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2116	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2117	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2118	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2119	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2120	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2121	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2122	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2123	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2124	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2125	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4

TABLE IV–2. MODERATE AND HIGH GROWTH PROJECTIONS FOR THE BAU FRAMEWORK CASES (cont.)

	Moderate scenario						High scenario					
	Power demand (GW·a)			Power capacity, (GW(e))			Power demand (GW·a)			Power capacity (GW(e))		
Year	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR	Total	HWR	LWR
2126	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2127	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2128	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2129	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4
2130	2500.0	150.0	2350.0	2941.2	176.5	2764.7	5000.0	300.0	4700.0	5882.4	352.9	5529.4

Table IV–3 is the combined historical and projected growth values by group for the heterogeneous framework base cases. For NG1, the total combination of LWRs and fast reactors is listed. The values for HWRs and LWRs in NG2 are listed in separate columns. The transition points are the same as described for Table IV–2, but divided by NG.

TABLE IV–3. GROWTH VALUES BY REACTOR TYPE BY GROUP BY YEAR FOR THE MODERATE AND HIGH HETEROGENEOUS FRAMEWORK BASE CASES

Moderate growth						High growth				
	NG1	NG2		NG3			NG1	NG2		NG3
Year	LWR/FR	LWR	HWR	LWR		Year	LWR/FR	LWR	HWR	LWR
1970	4.5	4.5	0.0	0		1970	4.5	4.5	0.0	0
1971	7.8	7.3	0.5	0		1971	7.8	7.3	0.5	0
1972	11.5	10.9	0.6	0		1972	11.5	10.9	0.6	0
1973	16.6	15.5	1.1	0		1973	16.6	15.5	1.1	0
1974	23.5	22.1	1.4	0		1974	23.5	22.1	1.4	0
1975	27.9	26.5	1.4	0		1975	27.9	26.5	1.4	0
1976	34.0	32.5	1.4	0		1976	34.0	32.5	1.4	0
1977	39.1	37.7	1.4	0		1977	39.1	37.7	1.4	0
1978	45.5	43.4	2.1	0		1978	45.5	43.4	2.1	0
1979	47.8	45.0	2.8	0		1979	47.8	45.0	2.8	0
1980	54.4	51.6	2.8	0		1980	54.4	51.6	2.8	0
1981	62.8	59.8	3.0	0		1981	62.8	59.8	3.0	0

TABLE IV–3. GROWTH VALUES BY REACTOR TYPE BY GROUP BY YEAR FOR THE MODERATE AND HIGH HETEROGENEOUS FRAMEWORK BASE CASES (cont.)

Moderate growth						**High growth**				
	NG1	**NG2**		**NG3**			**NG1**	**NG2**		**NG3**
Year	**LWR/FR**	**LWR**	**HWR**	**LWR**		**Year**	**LWR/FR**	**LWR**	**HWR**	**LWR**
1982	69.4	66.5	3.0	0		1982	69.4	66.5	3.0	0
1983	77.2	72.6	4.5	0		1983	77.2	72.6	4.5	0
1984	90.5	84.0	6.5	0		1984	90.5	84.0	6.5	0
1985	103.7	96.0	7.7	0		1985	103.7	96.0	7.7	0
1986	114.4	105.4	9.0	0		1986	114.4	105.4	9.0	0
1987	123.8	114.1	9.7	0		1987	123.8	114.1	9.7	0
1988	128.9	119.2	9.7	0		1988	128.9	119.2	9.7	0
1989	132.5	122.7	9.7	0		1989	132.5	122.7	9.7	0
1990	136.0	125.5	10.5	0		1990	136.0	125.5	10.5	0
1991	136.8	126.1	10.7	0		1991	136.8	126.1	10.7	0
1992	138.1	126.4	11.7	0		1992	138.1	126.4	11.7	0
1993	141.9	128.4	13.5	0		1993	141.9	128.4	13.5	0
1994	143.2	129.7	13.5	0		1994	143.2	129.7	13.5	0
1995	144.2	130.6	13.7	0		1995	144.2	130.6	13.7	0
1996	146.7	132.4	14.3	0		1996	146.7	132.4	14.3	0
1997	147.0	132.1	14.9	0		1997	147.0	132.1	14.9	0
1998	147.3	131.8	15.5	0		1998	147.3	131.8	15.5	0
1999	147.6	131.5	16.1	0		1999	147.6	131.5	16.1	0
2000	148.6	131.7	16.9	0		2000	148.6	131.7	16.9	0
2001	149.4	132.6	16.9	0		2001	149.4	132.6	16.9	0
2002	151.7	134.2	17.5	0		2002	151.7	134.2	17.5	0
2003	153.1	135.0	18.0	0		2003	153.1	135.0	18.0	0
2004	155.4	137.4	18.0	0		2004	155.4	137.4	18.0	0
2005	156.3	137.8	18.5	0		2005	156.3	137.8	18.5	0
2006	156.6	137.6	19.0	0		2006	156.6	137.6	19.0	0
2007	157.6	138.7	19.0	0		2007	157.6	138.7	19.0	0

TABLE IV–3. GROWTH VALUES BY REACTOR TYPE BY GROUP BY YEAR FOR THE MODERATE AND HIGH HETEROGENEOUS FRAMEWORK BASE CASES (cont.)

Moderate growth						High growth				
	NG1	NG2		NG3			NG1	NG2		NG3
Year	**LWR/FR**	**LWR**	**HWR**	**LWR**		**Year**	**LWR/FR**	**LWR**	**HWR**	**LWR**
2008	158.1	139.1	19.0	0.0		2008	158.1	139.1	19.0	0.0
2009	163.9	144.2	19.7	1.3		2009	166.1	146.0	20.0	1.5
2010	169.7	149.2	20.5	2.6		2010	174.0	153.0	21.1	3.0
2011	175.5	154.2	21.3	3.9		2011	182.0	159.9	22.1	4.5
2012	181.3	159.2	22.1	5.2		2012	190.0	166.8	23.2	6.1
2013	187.1	164.3	22.8	6.5		2013	197.9	173.7	24.2	7.6
2014	192.9	169.3	23.6	7.8		2014	205.9	180.6	25.2	9.1
2015	198.7	174.3	24.4	9.1		2015	213.9	187.6	26.3	10.6
2016	204.5	179.3	25.2	10.4		2016	221.8	194.5	27.3	12.1
2017	210.3	184.4	25.9	11.7		2017	229.8	201.4	28.4	13.6
2018	216.1	189.4	26.7	13.0		2018	237.8	208.3	29.4	15.2
2019	221.9	194.4	27.5	14.3		2019	245.7	215.2	30.5	16.7
2020	227.7	199.4	28.3	15.6		2020	253.7	222.2	31.5	18.2
2021	233.5	204.5	29.0	16.9		2021	261.6	229.1	32.6	19.7
2022	239.3	209.5	29.8	18.2		2022	269.6	236.0	33.6	21.2
2023	245.1	214.5	30.6	19.5		2023	277.6	242.9	34.7	22.7
2024	250.9	219.6	31.4	20.8		2024	285.5	249.8	35.7	24.2
2025	256.7	224.6	32.1	22.1		2025	293.5	256.7	36.8	25.8
2026	262.5	229.6	32.9	23.4		2026	301.5	263.7	37.8	27.3
2027	268.3	234.6	33.7	24.7		2027	309.4	270.6	38.9	28.8
2028	274.1	239.7	34.5	26.0		2028	317.4	277.5	39.9	30.3
2029	279.9	244.7	35.2	27.3		2029	325.4	284.4	41.0	31.8
2030	285.7	249.7	36.0	28.6		2030	333.3	291.3	42.0	33.3
2031	294.2	257.0	37.2	31.7		2031	350.8	306.4	44.4	38.5
2032	302.6	264.2	38.4	34.8		2032	368.2	321.4	46.8	43.6
2033	311.0	271.4	39.6	37.9		2033	385.6	336.4	49.2	48.8

TABLE IV–3. GROWTH VALUES BY REACTOR TYPE BY GROUP BY YEAR FOR THE MODERATE AND HIGH HETEROGENEOUS FRAMEWORK BASE CASES (cont.)

Moderate growth						High growth				
	NG1	NG2		NG3			NG1	NG2		NG3
Year	LWR/FR	LWR	HWR	LWR		Year	LWR/FR	LWR	HWR	LWR
2034	319.5	278.7	40.8	41.0		2034	403.0	351.4	51.6	53.9
2035	327.9	285.9	42.0	44.2		2035	420.5	366.5	54.0	59.1
2036	336.4	293.2	43.2	47.3		2036	437.9	381.5	56.4	64.2
2037	344.8	300.4	44.4	50.4		2037	455.3	396.5	58.8	69.4
2038	353.2	307.6	45.6	53.5		2038	472.7	411.5	61.2	74.5
2039	361.7	314.9	46.8	56.6		2039	490.2	426.6	63.6	79.7
2040	370.1	322.1	48.0	59.7		2040	507.6	441.6	66.0	84.8
2041	378.6	329.4	49.2	62.9		2041	525.0	456.6	68.4	90.0
2042	387.0	336.6	50.4	66.0		2042	542.4	471.6	70.8	95.2
2043	395.5	343.9	51.6	69.1		2043	559.8	486.6	73.2	100.3
2044	403.9	351.1	52.8	72.2		2044	577.3	501.7	75.6	105.5
2045	412.3	358.3	54.0	75.3		2045	594.7	516.7	78.0	110.6
2046	420.8	365.6	55.2	78.4		2046	612.1	531.7	80.4	115.8
2047	429.2	372.8	56.4	81.6		2047	629.5	546.7	82.8	120.9
2048	437.7	380.1	57.6	84.7		2048	647.0	561.8	85.2	126.1
2049	446.1	387.3	58.8	87.8		2049	664.4	576.8	87.6	131.2
2050	454.5	394.5	60.0	90.9		2050	681.8	591.8	90.0	136.4
2051	465.5	403.7	61.8	99.1		2051	708.2	614.0	94.2	153.6
2052	476.4	412.8	63.6	107.3		2052	734.5	636.1	98.4	170.9
2053	487.3	421.9	65.4	115.5		2053	760.9	658.3	102.6	188.2
2054	498.2	431.0	67.2	123.6		2054	787.3	680.5	106.8	205.5
2055	509.1	440.1	69.0	131.8		2055	813.6	702.6	111.0	222.7
2056	520.0	449.2	70.8	140.0		2056	840.0	724.8	115.2	240.0
2057	530.9	458.3	72.6	148.2		2057	866.4	747.0	119.4	257.3
2058	541.8	467.4	74.4	156.4		2058	892.7	769.1	123.6	274.5
2059	552.7	476.5	76.2	164.5		2059	919.1	791.3	127.8	291.8

TABLE IV–3. GROWTH VALUES BY REACTOR TYPE BY GROUP BY YEAR FOR THE MODERATE AND HIGH HETEROGENEOUS FRAMEWORK BASE CASES (cont.)

Moderate growth						High growth				
	NG1	NG2		NG3			NG1	NG2		NG3
Year	LWR/FR	LWR	HWR	LWR		Year	LWR/FR	LWR	HWR	LWR
2060	563.6	485.6	78.0	172.7		2060	945.5	813.5	132.0	309.1
2061	574.5	494.7	79.8	180.9		2061	971.8	835.6	136.2	326.4
2062	585.5	503.9	81.6	189.1		2062	998.2	857.8	140.4	343.6
2063	596.4	513.0	83.4	197.3		2063	1024.5	879.9	144.6	360.9
2064	607.3	522.1	85.2	205.5		2064	1050.9	902.1	148.8	378.2
2065	618.2	531.2	87.0	213.6		2065	1077.3	924.3	153.0	395.5
2066	629.1	540.3	88.8	221.8		2066	1103.6	946.4	157.2	412.7
2067	640.0	549.4	90.6	230.0		2067	1130.0	968.6	161.4	430.0
2068	650.9	558.5	92.4	238.2		2068	1156.4	990.8	165.6	447.3
2069	661.8	567.6	94.2	246.4		2069	1182.7	1012.9	169.8	464.5
2070	672.7	576.7	96.0	254.5		2070	1209.1	1035.1	174.0	481.8
2071	683.6	585.8	97.8	262.7		2071	1235.5	1057.3	178.2	499.1
2072	694.5	594.9	99.6	270.9		2072	1261.8	1079.4	182.4	516.4
2073	705.5	604.1	101.4	279.1		2073	1288.2	1101.6	186.6	533.6
2074	716.4	613.2	103.2	287.3		2074	1314.5	1123.7	190.8	550.9
2075	727.3	622.3	105.0	295.5		2075	1340.9	1145.9	195.0	568.2
2076	738.2	631.4	106.8	303.6		2076	1367.3	1168.1	199.2	585.5
2077	749.1	640.5	108.6	311.8		2077	1393.6	1190.2	203.4	602.7
2078	760.0	649.6	110.4	320.0		2078	1420.0	1212.4	207.6	620.0
2079	770.9	658.7	112.2	328.2		2079	1446.4	1234.6	211.8	637.3
2080	781.8	667.8	114.0	336.4		2080	1472.7	1256.7	216.0	654.5
2081	792.7	676.9	115.8	344.5		2081	1499.1	1278.9	220.2	671.8
2082	803.6	686.0	117.6	352.7		2082	1525.5	1301.1	224.4	689.1
2083	814.5	695.1	119.4	360.9		2083	1551.8	1323.2	228.6	706.4
2084	825.5	704.3	121.2	369.1		2084	1578.2	1345.4	232.8	723.6
2085	836.4	713.4	123.0	377.3		2085	1604.5	1367.5	237.0	740.9

TABLE IV–3. GROWTH VALUES BY REACTOR TYPE BY GROUP BY YEAR FOR THE MODERATE AND HIGH HETEROGENEOUS FRAMEWORK BASE CASES (cont.)

Moderate growth						High growth				
	NG1	NG2		NG3			NG1	NG2		NG3
Year	LWR/FR	LWR	HWR	LWR		Year	LWR/FR	LWR	HWR	LWR
2086	847.3	722.5	124.8	385.5		2086	1630.9	1389.7	241.2	758.2
2087	858.2	731.6	126.6	393.6		2087	1657.3	1411.9	245.4	775.5
2088	869.1	740.7	128.4	401.8		2088	1683.6	1434.0	249.6	792.7
2089	880.0	749.8	130.2	410.0		2089	1710.0	1456.2	253.8	810.0
2090	890.9	758.9	132.0	418.2		2090	1736.4	1478.4	258.0	827.3
2091	901.8	768.0	133.8	426.4		2091	1762.7	1500.5	262.2	844.5
2092	912.7	777.1	135.6	434.5		2092	1789.1	1522.7	266.4	861.8
2093	923.6	786.2	137.4	442.7		2093	1815.5	1544.9	270.6	879.1
2094	934.5	795.3	139.2	450.9		2094	1841.8	1567.0	274.8	896.4
2095	945.5	804.5	141.0	459.1		2095	1868.2	1589.2	279.0	913.6
2096	956.4	813.6	142.8	467.3		2096	1894.5	1611.3	283.2	930.9
2097	967.3	822.7	144.6	475.5		2097	1920.9	1633.5	287.4	948.2
2098	978.2	831.8	146.4	483.6		2098	1947.3	1655.7	291.6	965.5
2099	989.1	840.9	148.2	491.8		2099	1973.6	1677.8	295.8	982.7
2100	1000.0	850.0	150.0	500.0		2100	2000.0	1700.0	300.0	1000.0
2101	1000.0	850.0	150.0	500.0		2101	2000.0	1700.0	300.0	1000.0
2102	1000.0	850.0	150.0	500.0		2102	2000.0	1700.0	300.0	1000.0
2103	1000.0	850.0	150.0	500.0		2103	2000.0	1700.0	300.0	1000.0
2104	1000.0	850.0	150.0	500.0		2104	2000.0	1700.0	300.0	1000.0
2105	1000.0	850.0	150.0	500.0		2105	2000.0	1700.0	300.0	1000.0
2106	1000.0	850.0	150.0	500.0		2106	2000.0	1700.0	300.0	1000.0
2107	1000.0	850.0	150.0	500.0		2107	2000.0	1700.0	300.0	1000.0
2108	1000.0	850.0	150.0	500.0		2108	2000.0	1700.0	300.0	1000.0
2109	1000.0	850.0	150.0	500.0		2109	2000.0	1700.0	300.0	1000.0
2110	1000.0	850.0	150.0	500.0		2110	2000.0	1700.0	300.0	1000.0

ABBREVIATIONS AND ACRONYMS

ADS	accelerator driven system
AECL	Atomic Energy of Canada Limited
ALWR	advanced light water reactor
ANL	Argonne National Laboratory
BARC	Bhabha Atomic Research Centre
BAU	business-as-usual
BR	breeding ratio
BWR	boiling water reactor
CEA	French Alternative Energies and Atomic Energy Commission
CIAE	China Institute of Atomic Energy
CIEMAT	Research Centre for Energy, Environment and Technology
CNEA	National Atomic Energy Commission
CNFC	closed nuclear fuel cycle
CNFC–FR	closed nuclear fuel cycle with fast reactors
CP	collaborative project
CR	conversion ratio
DESAE	dynamics of energy systems — atomic energy
DOE	United States Department of Energy
EFIT	European Facility for Industrial Transmutation
EFPD	effective full power day
ELSY	European lead cooled system
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
ENERGOATOM	National Nuclear Energy Generating Company ENERGOATOM
EP	evaluation parameter
EURATOM	European Atomic Energy Community
FBR	fast breeder reactor
FP	fission product
FR	fast reactor
GAINS	Global Architecture of Innovative Nuclear Energy Systems Based on Thermal and Fast Reactors Including a Closed Fuel Cycle
GDW	globally driven world
HLW	high level waste
HM	heavy metal
HTR	high temperature reactor
HWR	heavy water reactor
IGCAR	Indira Gandhi Centre for Atomic Research
INFCE	International Nuclear Fuel Cycle Evaluation
INL	Idaho National Laboratory
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
INS	innovative nuclear energy system
IPCC	Intergovernmental Panel on Climate Change
IPPE	Institute of Physics and Power Engineering
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre
KAERI	Korea Atomic Energy Research Institute
KI	key indicator
LWR	light water reactor
MA	minor actinide
MEPhI	Moscow Engineering Physics Institute
MESSAGE	Model for Energy Supply Strategy Alternatives and their General Environmental Impacts

MNA	multilateral nuclear approach
MOX	mixed oxide
MSR	molten salt reactor
NDC	Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle
NDW	nationally driven world
NEA	Nuclear Energy Agency
NES	nuclear energy system
NFC	nuclear fuel cycle
NFCSS	nuclear fuel cycle simulation system
NG	nuclear strategy group
NU-HWR	natural uranium fuelled heavy water reactor
OECD	Organisation for Economic Co-operation and Development
P&T	partitioning and transmutation
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor
R&D	research and development
RBMK	graphite moderated fuel channel reactor
RD&D	research, development and demonstration
Rosatom	State Atomic Energy Corporation “Rosatom”
SCK•CEN	Belgian Nuclear Research Centre
SF	spent fuel
SFR	sodium cooled fast reactor
SMR	small and medium sized reactor
SNF	spent nuclear fuel
SRES	Special Report on Emissions Scenarios
SWU	separative work unit
TEPS	tool for energy planning studies
ToR	terms of reference
TRU	transuranic
UOX	uranium oxide
VISION	verifiable fuel cycle simulation
VÚJE	Nuclear Power Plant Research Institute Trnava
WNA	World Nuclear Association
WWER	water cooled water moderated power reactor

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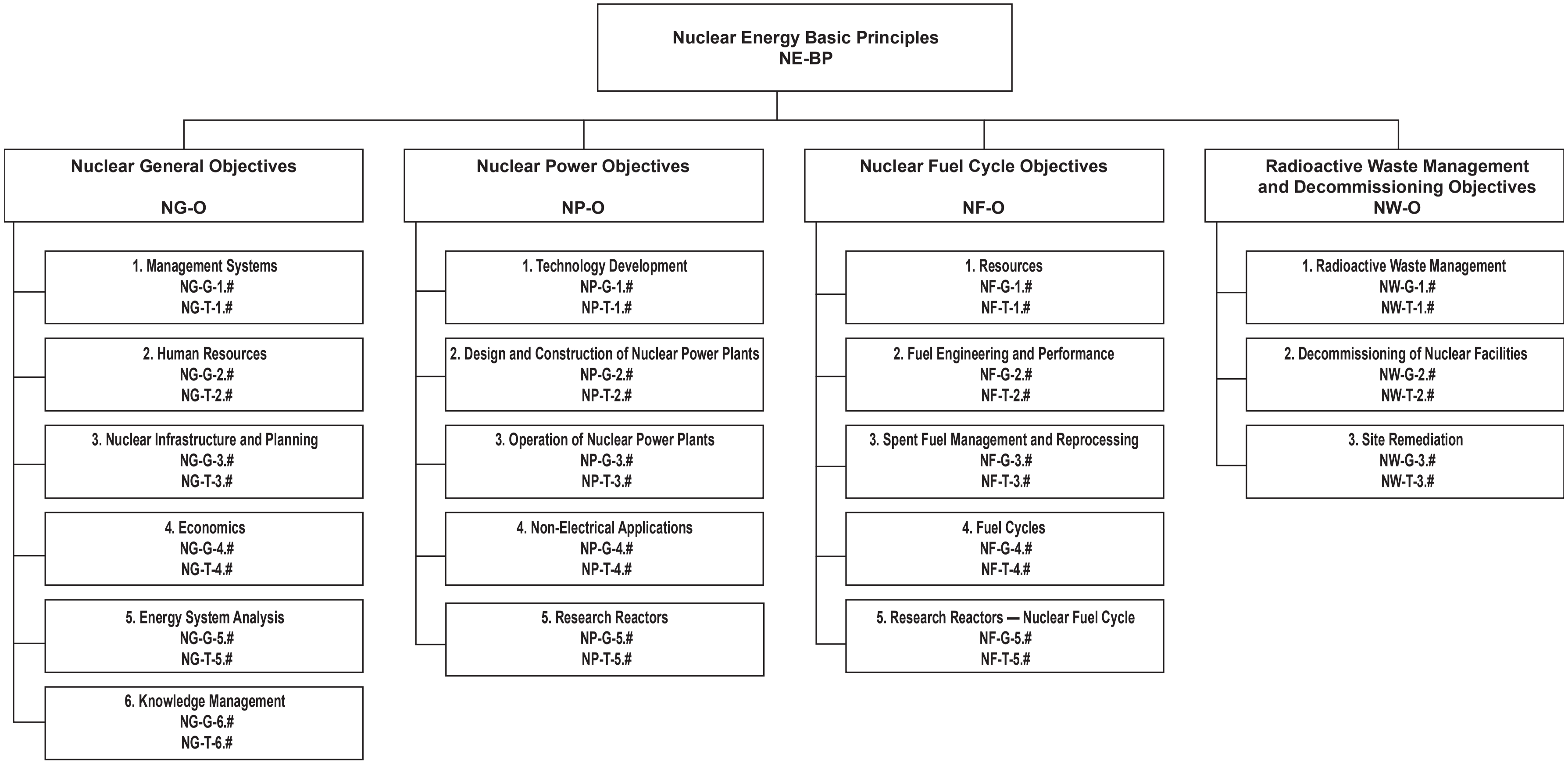

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