International role of nuclear fission energy generation, status and perspectives

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Abstract

The Fukushima incident in March 2011 caused worldwide a change in the perception of nuclear energy generation. Independent from the decision made by individual nations regarding the future use of nuclear energy for electricity generation, the number of nuclear power plants (NPP) operated worldwide has hardly changed. Essential reasons are mainly rising feedstock prices, increased energy demands and the simultaneous aspiration to reduce substantially the CO_2-emission by fossil fuels. Especially emerging Asian economies are forced to an aggressive exploitation of all electricity generating technologies including nuclear to match their societal and economic demands. Nevertheless, the Fukushima accident initiated worldwide a new quality in the safety assessment and safety culture by considering additional man made or natural disasters. This process is reflected in enhanced bilateral or international co-operations. One of the most striking consequences is that a safe NPP operation demands a continuous retrofitting and evaluation of the plant behavior based on the current state of science and technology, which is part of the German safety practice since the Three-Mile-Island (TMI) incident.

Within this article different new nuclear plant developments with enhanced safety features are presented. Although these concepts as well as their deployment options diverge considerably in design and operational strategy the major nuclear protection goals in terms of confinement, coolability and reactivity control, which have to be met by any plant design, remain the same. Regarding the operational safety increased computational capabilities allow by means of coupled multi-physics and multi-scale method to identify design weaknesses down to the pin scale of a fuel assembly both for steady state and also for plant transients. To master severe accidents the different plant concepts, however, yield to a considerably larger diversity of technical solutions, nearly all of which are based on passive systems that exploit the physical natural laws. A sustainable use of nuclear fuel avoiding large scale deep underground repositories inherently implies a closed fuel cycle and the deployment of fast spectrum reactors, so-called Generation –IV reactors, for which similar nuclear postulations in terms of safety on all levels have to be demonstrated. Within the article for both operational safety and severe accident measures examples are presented to illustrate the main functionality and operational principle.

1 Present status of nuclear electricity generation – observations worldwide and in Europe

At present 435 commercial nuclear reactors (NPP) are operating and almost 2/3rd´s of the 72 plants under construction are erected in Asia [1]. More than 75% of the existing reactor fleet is light water reactors and about 85% of the new built belong to the class of pressurized water reactors (PWR). All commercially operated NPP´s produced in 2013 nearly 11.5% of the global electricity production, which is only slightly less than in the previous years. These commercial plants are complemented by approximately 240 research reactors operated in 56 countries and currently nearly 180 civil nuclear powered ships.

Remarkable is that the countries engaged in new built or strongly envisaging the use of nuclear power as a “nuclear newcomer” belongs either to Eastern Europe or to Asia and the motivation to use nuclear power is mainly triggered by their societal decision to rely to a large quantity on industrial production as one major pillar of economic development- or simply as source of future wealth. The specific reasons of those societies range from vast economic development and rapidly rising electricity consumption, grid independence, fuel independence

(reduced currency export?), cost arguments or access to large scale renewable resources and many other more.
In contrast to the new built, the classic nuclear countries focus either on replacements of their in average more than 25 year old reactor fleet [2], power-uprating and life-time extensions up to 60 years.

Even in Europe the new-built and retrofitting of operating plants has hardly changed the nuclear share in electricity generation, which amounts to about 30%. Although Europe is committed to match the CO_2-conformity goals formulated by the EU commission by 2020 and 2050, the member states can select on their own the means to attain the decarbonisation goals [3] according to the ministry council agreement. Especially in the EU-11 (comprised of Poland, Czech Rep., Slovakia, Hungary, Latvia, Lithuania, Estonia, Romania, Bulgaria, Slovenia and Croatia, the economic growth will cause a considerable increase of electricity consumption and at the same time ever stringent environmental requirements must be fulfilled. Both, the lack of abundance of fossil resources and the trend to a certain energy autarchy lead the EU-11 states to conceive nuclear as a one electricity production option [4].

Independent of the world region considered and the quite diversely motivated basis of the individual countries mainly large scaled NPP´s in the power class of 1GWe and more are currently deployed.

Taking a glance at Germany´s current electricity profile, which decided a phase-out of nuclear by 2022, at present the total installed electricity capacity of renewable energy sources (RES) amounts in average in 2013 to installed 35,886GW peak photo-voltaic capacity and 33,818GW nominal installed wind power, which would fit the entire mean German demand. Only approximately 16% is provided by NPP´s. Nonetheless, the intermittent production lead for the RES to a remarkable share of 24.9% for electricity [5], see fig. 1. Although having a grid priority access, solar photo-voltaics delivered 30TWh and wind 53TWh corresponding to load factors of solar and wind compared to their installed capacities to about 9.5% (solar) and 18% (wind), respectively. This market regulation caused low wholesale electric energy prices in Germany, which in turn made combined cycle gas turbines (CCGT), which are most thermally efficient, noncompetitive. In the long term especially targeting at a RES share exceeding 50% significant storage volumes for electricity and the maturity of storage concept technologies are required, unless the decarbonisation will not be matched.

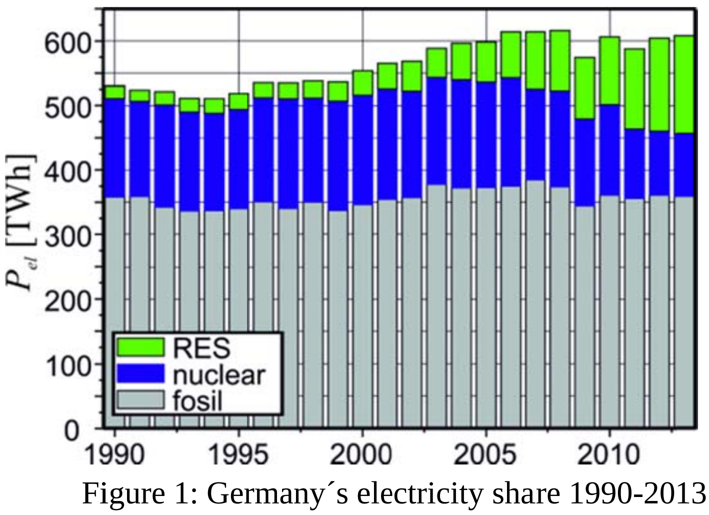


Figure 1: Germany´s electricity share 1990-2013 [5].

Doubtless the quite ambitious climate protection policy goals and the limitations of RES based energy production in the near term will demand compromises from all main actors. Otherwise the economic competitiveness of regions not adapting these goals will be undermined. From the scientific point of view, each technology and fuel option must be considered including nuclear energy and heat generation in the energy mix either as a midterm or long term bridging option to prevent energy poverty and to assure reasonable energy prices.

2 Boundary conditions for NPP deployment-Large reactors/ vs. small medium sized reactors

Considering nuclear as an option either as bridging technology or as one major pillar of the energy mix of an individual country, the question of the appropriate reactor size for the base load configuration arises immediately: large reactors (LR) or small scaled modular reactors (SMR) ? . The arguments for the choice of LR or SMR may be grouped in social (acceptance, risk perception), political (independence, CO_2 limitations), economic (resources, price, risk) or technological (technical maturity, safety performance) criteria. Mainly the decision matrix is composed of mixtures of all these arguments and the ranking is strongly dependent on the national boundary conditions.

2.1 Economic considerations

Large NPP's require a considerable capital investment per MWh/unit including all costs for projection, deployment, operation & maintenance (O&M) and finally decommissioning. Additionally nuclear power utilization demands a long term strategy of the energy policy and its development. Due to these high financial exposures, the long pay back times (envisaged duration of the investment 80-100 years from planning to decommissioning) present a high investment risk if entirely financed by private shareholders. Compared to coal, RES or gas fired plants the capital costs amount to about 55% [6] and hence the capital intensive investment represents a strong exposure to market risks aside from other critical aspects as political frame (licensing, inspection, regulations,…) and social factors as e.g. public acceptance. As a consequence, private operators in a liberalized market often based on competition and sometimes with priority access of other energy sources require a stable energy politics environment. In contrast to purchase a LR there are numerous arguments for deploying SMRs as identified by [6, 7, 8] such as

- the need for flexible power generation for wider range of users and applications;
- the replacement of aging fossil fired units;
- the potential for enhanced safety margin through inherent and/or passive safety features;
- the economic consideration-better affordability freedom in upgrading;
- potential for integration innovative energy systems: cogeneration & non electric applications (desalination, process heat) and
- hybrid energy systems composed of nuclear with RES.

But, according to numerous studies [6, 8] SMR are not significantly cheaper than LR´s and moreover, the capital return time is even larger than for larger reactors although they may offer a higher decision flexibility to expand their unit size. Additionally, SMR´s cannot be conceived as a simple scale reduction of a LR. Also the power output of several SMR to the grid cannot be simply considered as the sum of the modules; the SMR technology presents an entirely different product with respect to fuelling, operation but mainly with respect to the safety features and the applied technology. Among these technology issues their safety behavior is due to the smaller dimensions considerably different. SMR have usually a smaller specific power density than LR allowing the use of a set of passive measures to master essential safety functions or even a full encapsulation of the reactor. Nevertheless, SMRs fulfill in some markets already an essential role as to act as base load source in remote regions or as grid stabilization in regions with moderate energy consumption like in China or India.

2.2 Current situation of NPP deployment

More than 95% of the currently operating reactor fleet belongs to the class to generation II plants, which in principle have been designed in the sixties and seventies. Also the presently installed generation III reactors are mainly evolutionary designs of Gen-II systems. The major reason for this development may be conceived as a risk minimization strategy of the

shareholders. The comparability of Gen-II and operating Gen-III plant enables to a considerable extend the use of the accumulated experience of the currently operating fleet and therefore it facilitates the licensing aspects. The designs rely on well proven physics principles and no technological leaps are required. All aspects together yield for the operating Gen-III reactors a similar performance and sustainability as for Gen-II units. Another class of Gen-III reactors currently under construction in USA and China are very innovative; they rely mainly on novel passive safety features to assure core coolability e.g. in case of Loss-of-Coolant (LOCA) accidents and to remove the residual heat.
What are the peculiarities of the operating Gen-III reactors?

There are essentially two drivers for the new Generation III plants, which arise from both hardened design objectives and economic design objectives. The design objectives may be subdivided in two classes- the nuclear safety and the public acceptability. Regarding nuclear safety in Generation III units severe accident measures have been already integrated in the design to attain considerably lower core damage frequencies and a significant reduction of potential radiological consequences. Another essential feature is that external events and hazards are considered in design and emergency management measures, which end up in a more robust safety architecture. In order to attain public acceptability, the design is devoted to minimize the environmental impact for all operational stages and to prevent situations, in which off-plant areas are submerged to any emergency planning.

Especially the competition with other sources hardened the economic objectives. In the front row here is the profitability of the project, which in turn demands plant availabilities of more than 90% along the whole life-time, short re-fuelling and outage durations resulting in long cycle length and reduced investments caused by design simplifications and short erection times. According to this list, LR´s are preferred to SMR units. Another economic aspect is the investment protection, which translates into anticipated operation times of at least 60 up to 80 years and a low difficult-to-repair failure rate, which in turn demands to credit mainly for proven technologies. The latter argumentation chain holds mainly for liberalized markets, where temporal economic ups or downs even at low interest rates shall allow for profits for the shareholders within a reasonable time. The frame for NPP development today is conducted in contrast to former times by a set of regulations, standardizations and requirements elaborated in the international context of utilities [9], technical survey organizations (TSO) [10], worldwide co-operations and collaborations as well as international institutions like the nuclear energy agency (NEA) and the IAEA [11]. All these regulations are publicly available and continuously updated.

3 Safety concept of an NPP

3.1 General safety approach

The major protection goals for NPP´s have not been changed since the early days and scope only three aspects:

- confinement of the radionuclide inventory;
- coolability at any time irrespective of origin and source and
- control of reactivity.

This protection goals led to the implementation of a defence in depth (DiD) strategy, for different levels are assigned to specific reactor states from 1 to 5, see . The challenge is to provide enough margins between the different levels of safety to prevent cliff edge effects. The subsequent safety demonstration is characterized by a risk informed safety strategy, in which at first the protection goals are transferred into fundamental safety functions to be provided by the individual plant system design. The individual demonstration is conducted by both probabilistic and deterministic methods, in which for the latter a set of initiating events (PIE) are postulated and their progression is analyzed.

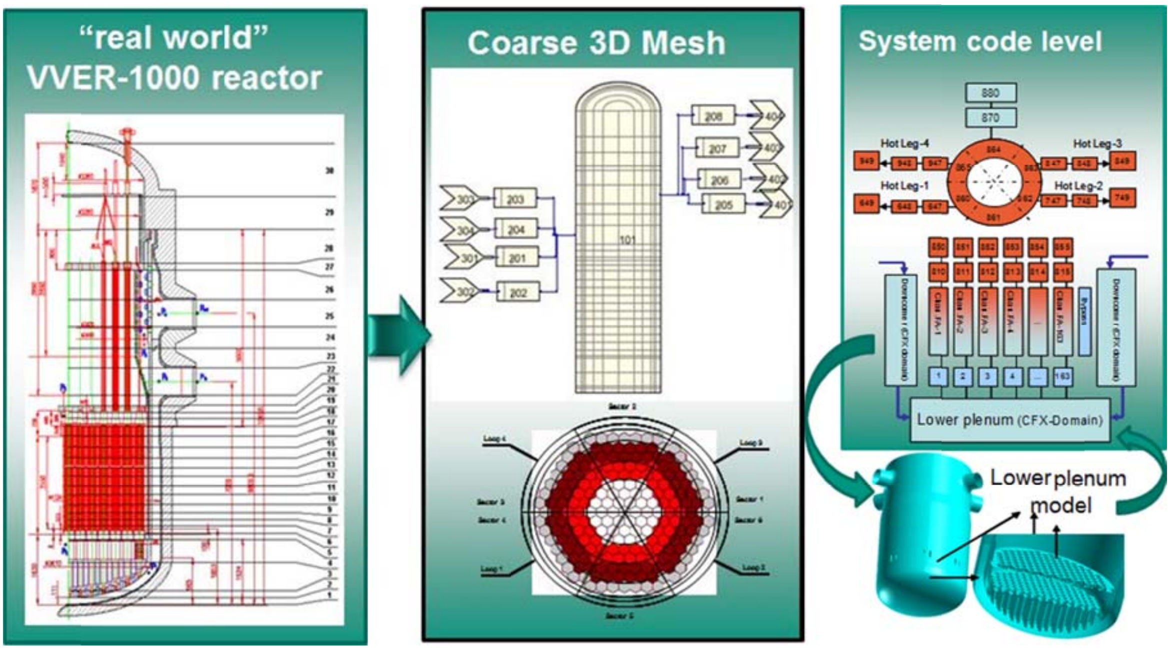
Table 1: Safety level categorization; corresponding aim, measures & consequences of a NPP.

	condition	aim	measures	consequences
1	normal	prevention of anormal operation or failures	conservative design, high quality construction, qual. personnel	no measures
2	operational failure	condition control, detection/ identification of reason	control, limitation/ protection measures and survey functions	after short time restart
3	design basis accident (DBA)	control of DBA within design (e.g. multiple failures of safety functions)	engineering safety charact. and implementation of controlled accident measures	planned restart anticipated (after inspection, repair,qualification)
4	severe accident (BDBA)	control of critical plant states incl. prevention of propagation	complementing measures and accident management	**re-start not required**
5	post severe accident	mitigation of radiologic consequences	off- plant emergency measures	**no plant restart assumed**

3.2 Design basis safety (operational safety)

A nuclear power plant is a complex system, in which different physical phenomena such as neutronics, thermal-hydraulics and thermo-mechanics are interconnected. Moreover, all of them are occurring on different scales from micro- via meso- to macro-scale and are additionally characterized by non-linear feedbacks. This mutual interaction impacts considerably the safety performance and poses significant challenges to develop qualified computational multi-physics and multi-scale tools to describe the temporal behavior of the plant.

Especially today and very likely in the near future, the enlarged and still growing com-



(a) (b) (c)

Figure 2: Transfer of reactor core (a) into nodal model (b) and then connection to thermal-hydraulic channels.

putational capabilities and the memory resources allow a more refined representation of the phenomena taking place in NPP than in the past. These efforts not only allow for a more detailed on-line plant monitoring by reduced models to support operational safety in real-time but much more to elaborate and analyze with a high local discretization safety margins down to a fuel pin scale not only steady state but also in transients. Subsequently, some state of the art examples are highlighted and discussed.

The thermal-hydraulic transport of mass and energy is an essential feature in daily reactor operation, and hence necessitates a reliable fast running predictive codes to flexibly adapt the power and to assess the feedbacks between neutronic and thermal hydraulics. Fast running methods are provided by 3D coarse mesh approaches to describe the reactor pressure vessel and the core as depicted in figure 2 and the piping is modelled via 1D nodes.. Here, a profound a-priori knowledge is required to describe the individual phenomena within the reactor pressure vessel, core and primary/secondary system of a NPP as denoted in figure 2c. The numerical codes contain large set of models and correlations to describe the heat transport within fuel pellet, over the gap and on the fuel rod surface as well as in the fluid domains. All important correlations were derived from representative experiments and the codes are extensively validated against experimental data before they are used for safety assessment of NPPs.

Of course, fast computing code systems allow for parameterized computations or to conduct exploring potentially domains with a reduced or even problematic safety margin. Moreover, by means of them the impact of design measures on the reactor performance can be screened. However, one of the drawbacks of these codes is their limited spatial resolution, which demands a more sophisticated description of the thermal-hydraulics. Naturally, this is matched by CFD. But, even a relatively small domain like a reactor pressure vessel (RPV) would require such an enormously large number of cells to depict all important phenomena with an adequate resolution that it can be hardly solved within the next one or two decades. One option for a correct transfer from micro to macro scale provides the coarse grid CFD, which is depicted exemplarily in figure 3. There, on a micro-scale all phenomena are computed exactly and then transferred as volumetric forces or interfacial tension to a meso-scale. A similar approach is used from meso- to macro scale. By this approach the prediction quality within the qualified parameter range of the CFD models is by far better than for the nodal methods; but this is on cost of the considerably larger computational times, so that this approach is mainly used for design verification. Such a CFD model for the RPV can then be connected to a loop model based on a 1D coarse mesh. An essential corner-stone of the computational modeling is the validation. Especially in the context of the IAEA, the OECD-NEA

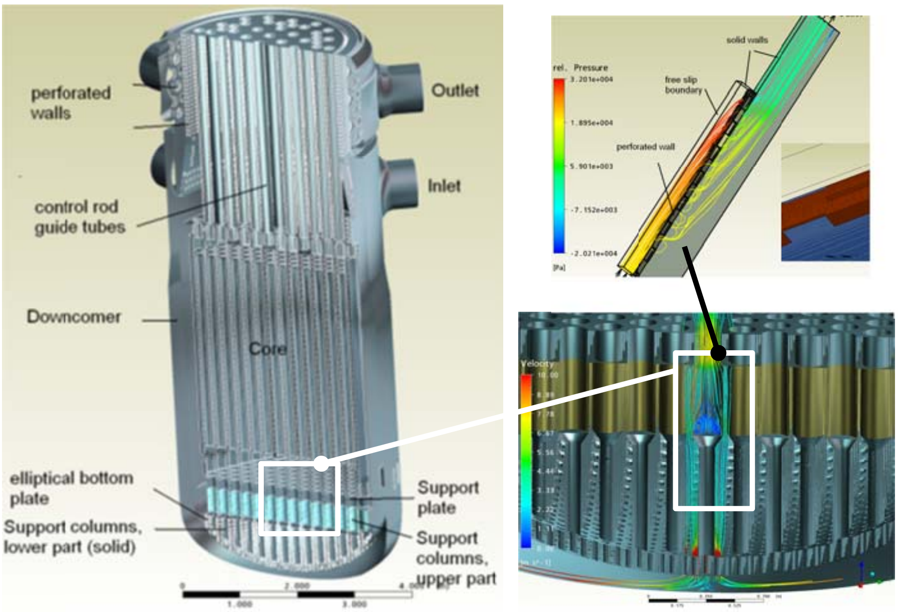


Figure 3: Transfer from micro-scale CFD (top right) to meso scale (bottom right) and then to reactor scale (left.).

or international collaborations benchmarks have been formulated or set-up to validate numerical code packages. These benchmarks scope thermal-hydraulics problems, reactor-physics topics or coupled neutron-kinetic thermal-hydraulic interactions as well as real-world plant data and transients. Figure 4 illustrates the comparison of the temporal evolution of computed void fraction distributions at three different axial heights in a fuel assembly for a pump trip, which has been conducted in the framework of an OECD/NEA Benchmark [12]. As the figure shows, the different validated current code packages are capable to depict both void fraction as well as temporal evolution with a high degree confidence.

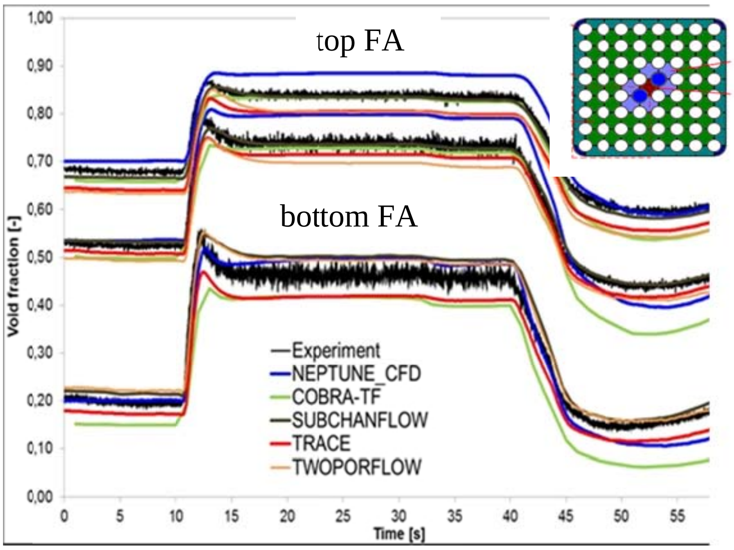


Figure 4: Computed void fraction distributions at three different axial heights by different codes during a pump trip in comparison with the experiment [12].

Another similarly challenging problem is posed by neutronic-thermal-hydraulic (N-TH) interaction. A classical problem in this context is a boron dilution transient. In case of an unintended reactor flooding with deborated water, the reactivity and hence the power production increases. The Doppler feedback leads in turn to a power reduction and the reactivity declines. The figure 5 (top) depicts the temporal evolution of the overall reactivity while the lower figure illustrates the distribution of the borated water within the core computed by nodal diffusion codes coupled with sub-channel codes at fuel assembly level.

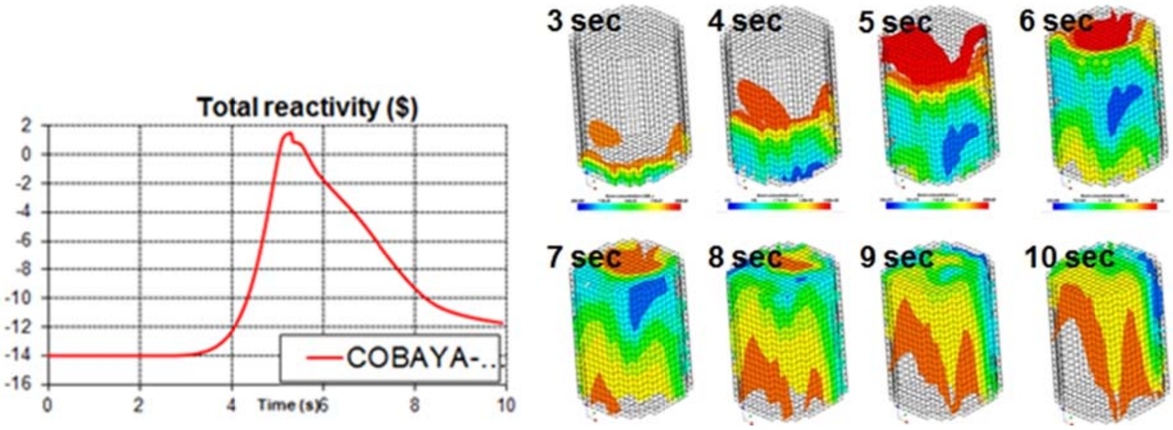


Figure 5: Thermal-hydraulics – neutronic coupling for a boron dilution transient in which deborated water (blue) enters the core leading to first to reactivity increase and by feedbacks afterwards to a reduction [13] and corresponding reactivity change (left).

Nodal methods as presented before represent still “state-of-the-art” methods to predict safety parameters of a plant and are well validated through thousands of computations. In combination with thermal-hydraulic sub-channel codes, they represent the class in literature often called best estimate tools (BE). The interplay of neutronics and thermal-hydraulics poses the strongest challenge. The neutronics provides the nuclear power generated by fission and the power map is transferred to the individual pin, where the heat is removed by the

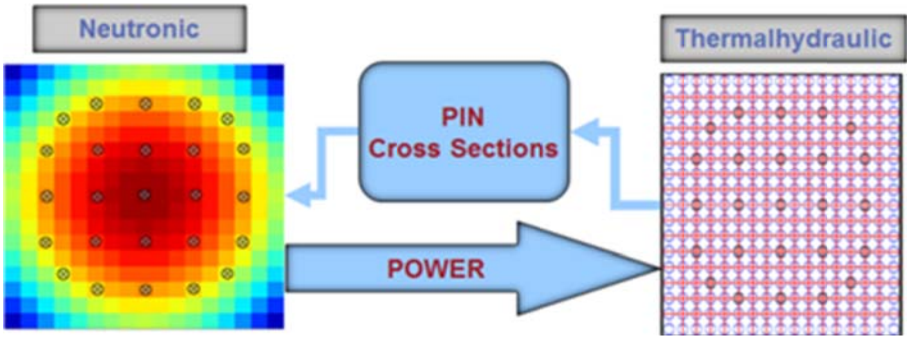


Figure 6: Interplay of neutronics (at core level, left) and thermal-hydraulics (on fuel pin level in a fuel assembly-right) to be considered for pin-based multi-physics high fidelity computations.

coolant. By the coolant the fuel temperatures change, which in turn leads to a change in the nuclear cross-sections altering in turn the fission power generation, see figure 6. Unfortunately, both occur on different time scales. A direct prediction of local safety parameters demands therefore computations on a pin-level rather than on fuel assembly (FA) or cell level to reduce the conservatism.

The trend from FA to pin based high fidelity modelling solutions is rather heterogeneous. One route focuses first on a homogenization of the heterogeneous FA configuration to a node and a subsequent integration as a node into a 3D core model (route 1 in figure 7). The individual pin-power distribution and the corresponding safety parameters are then obtained by means of pin-power-reconstruction (PPR) methods. A full pin-based solution considers throughout the entire computation the individual pin configuration. The major advantage is that local safety parameters are directly computed without any simplifications of PPR methods.

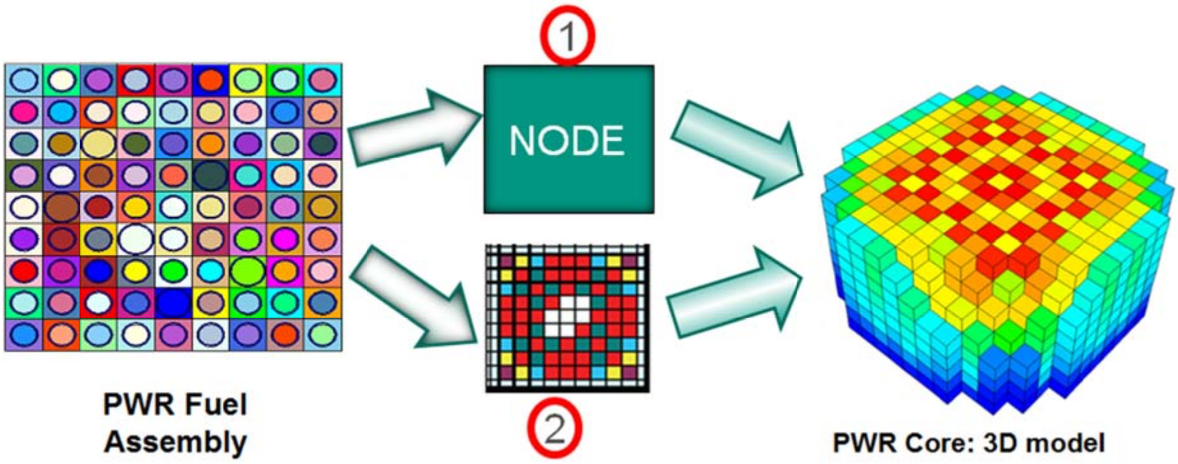


Figure 7: Different options for high fidelity coupled N-TH computations. Path (1) via homogenization and subsequent PPR methods. Route (2) full pin-based solutions.

A major drawback of these so called pin-based solutions is the high computational cost and the extremely demanding modelling. But, on the other hand, they provide an exact solution. The figure 8 shows a pin-based solution with respect to power evolution of a rod-ejection accident transient in a PWR. The corresponding temporal evolution of the fuel temperatures is depicted in figure 8 (right) and compared to a nodal solution. In most cases, the nodal solution provides a conservative solution, however in this specific case the full pin-by-pin solution exhibits that the nodal solution underestimates by 70 K the fuel temperature. This example illustrates that the potential of entirely pin based solutions offers a more exact prediction of the safety parameters.

An emerging new trend observed in coupled N-TH reactor simulations is devoted to quasi-exact neutronic calculations by means of Monte-Carlo methods, in which the thermal-hydraulics is directly embedded. Although it poses additionally even higher computational demands, the precision is considerably better. In figure 9 such a computational result is shown for the axial distribution of the fuel temperature of a typical PWR FA [15]
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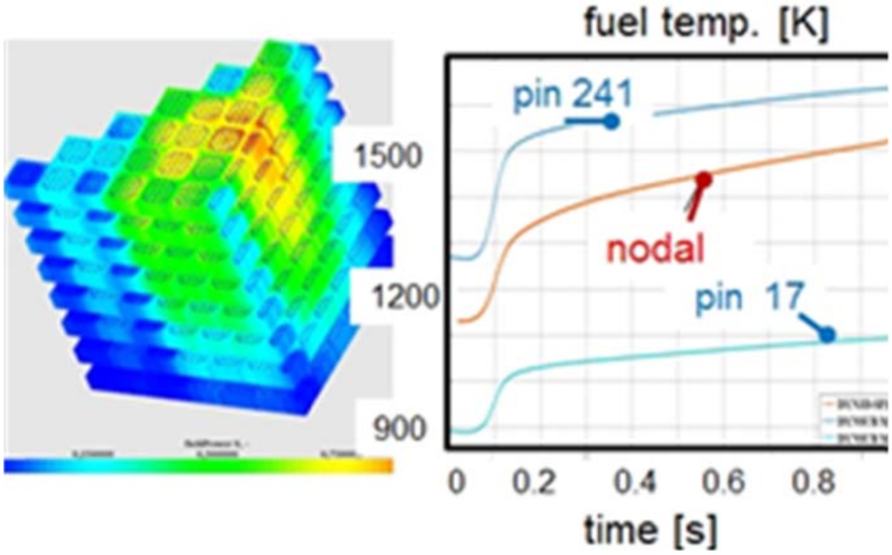

Figure 8: Pin based solution of the power distribution (left) during a rod ejection accident (REA). Comparison of the temporal evolution of fuel pin temperatures for a pin-based solution compared to a nodal solution [14].

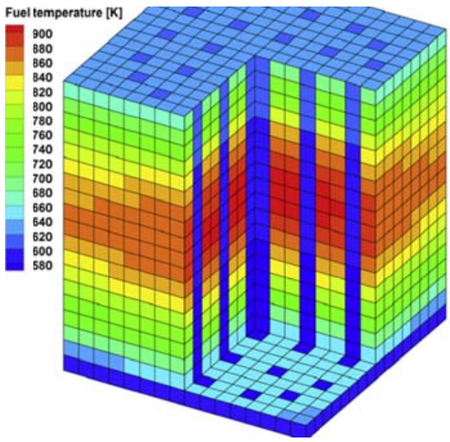

Figure 9: Inline coupled Monte-Carlo-neutronics – thermal-hydraulics computation of the fuel temperature of a PWR FA [15]- dark blue=guide tube channels.

3.3 Beyond design basis safety (BDBA)

In contrast to Gen-II reactors measures to cope with beyond design accidents are integral part of the reactor design of Generation-III plants. What does this mean?
Gen-III reactor designs implicitly assume the occurrence of reactor states exceeding the design basis and adequate design provisions to carry out preventive and mitigative accident measures. These 'a-priori' setting of safety related design pre-requisites doesn't abandon standard reactor control measures such as control rods and borated water but additionally allocates specific provisions e.g. for an improved emergency core cooling capability in BDBA-situations, see figure 10. Remarkably, the focus of emergency core cooling systems (ECCS) development has been directed mainly to systems acting by passive means and not necessarily requiring signals of intelligence to accomplish a safe state of the plant in case of a BDBA. A detailed classification of passive safety systems to render a plant within a safe condition can be found in [16]. The expression passive means mainly gravity or density difference driven systems caused mainly by temperature gradients. In addition, the following safety relevant phenomena appeared in connection with the innovative Gen-III reactor concepts:

- behavior of large pools of liquid,
- effects of non-condensable gases on condensation heat transfer,
- condensation on containment structures,
- behavior of containment heat removal systems,
- thermo- and fluid dynamics as well as pressure drop in different geometric configurations,
- steam-liquid interaction, etc. All these effects emerged into integrated engineering solutions of passive safety systems in various advanced water cooled nuclear power plants. An

example of the set of different ECCS systems employed within the AP1000 design consisting of

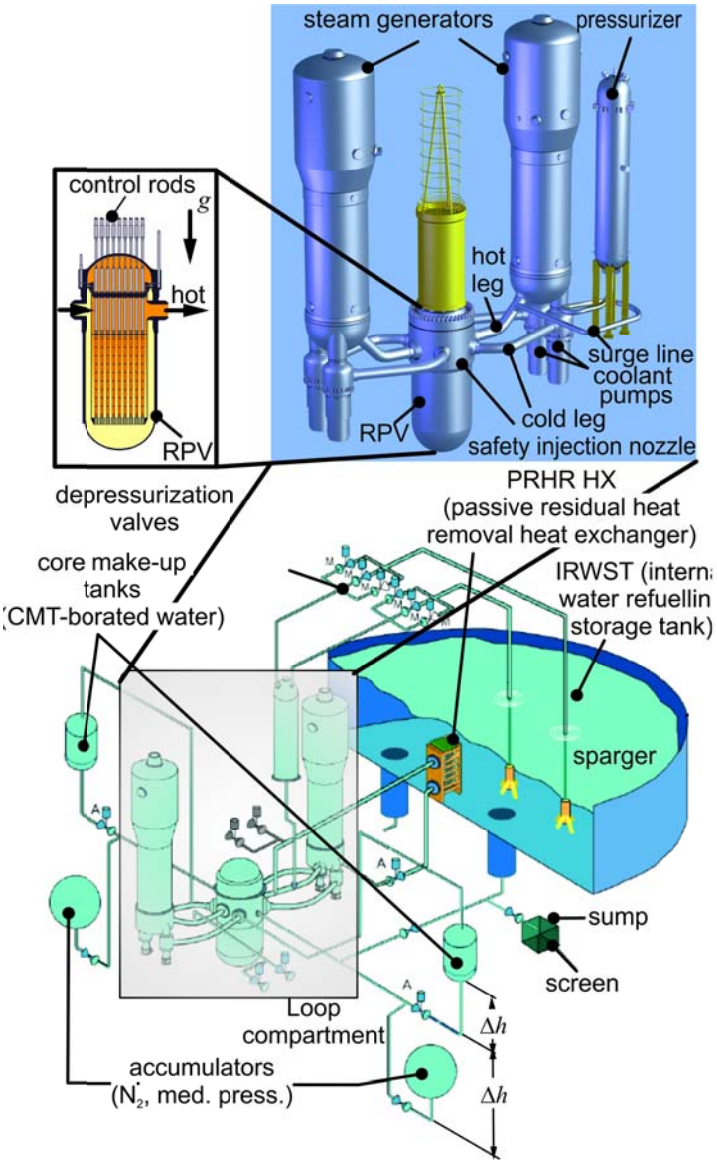

Figure 10: Safety system (AP 1000) [17].

- core make-up-tanks (borated water),
- accumulators (for water replacement),
- coolant make-up from in-containment refueling water storage tank (IRWST) driven by gravity and
- passive residual heat removal (PRHR) system based on gravitational forces

to name the most important, see figure 10.

In a web-based video in [17] the functionality of the passive safety systems is illustrated for a main steam line break (MSLB). In case of the occurrence of a severe accident with core melt different strategies are applied in the evolutionary Gen-III systems. One branch of the plant designs follows an in-vessel retention strategy, such as foreseen in the AP1000 (see fig.11,left), while others as the EPR is focused on spreading of the corium in a dedicated domain underneath the RPV, which is called an external core-catcher , figure 11, right .

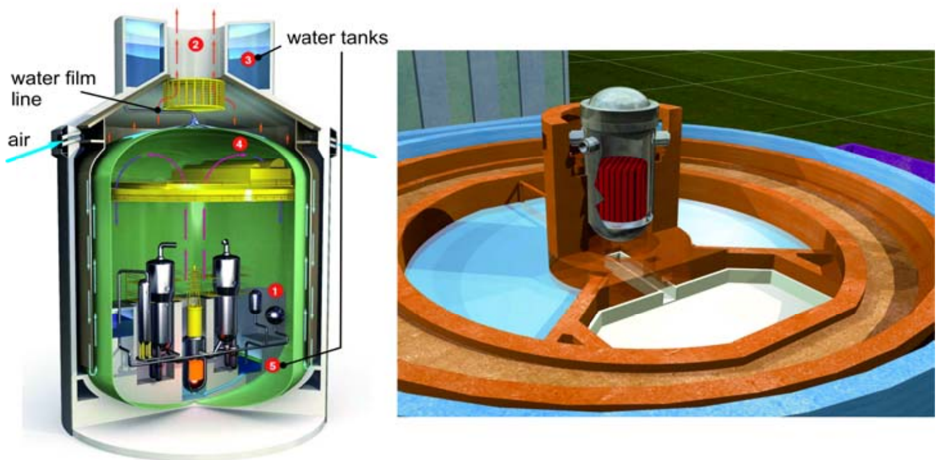

Figure 11: Different design based safety provisions to control core meltdown situations: in-vessel retention strategy [17, left] and core-catcher of the EPR [18, right].

However, the safe confinement of the corium represents only one aspect in case of a severe accident. In case of a failure of the corium in-vessel retention, dedicated containment measures have to be implemented to ensure containment integrity that may be affected by uncontrolled hydrogen explosions or molten core concrete interaction (MCCI), etc.. Within the Helmholtz-program Nuclear Waste Disposal and Safety (NUSAFE) and international collaborations some of the key severe accident phenomena are investigated at the Karlsruhe Institute of Technology (KIT) as depicted in figure 12.

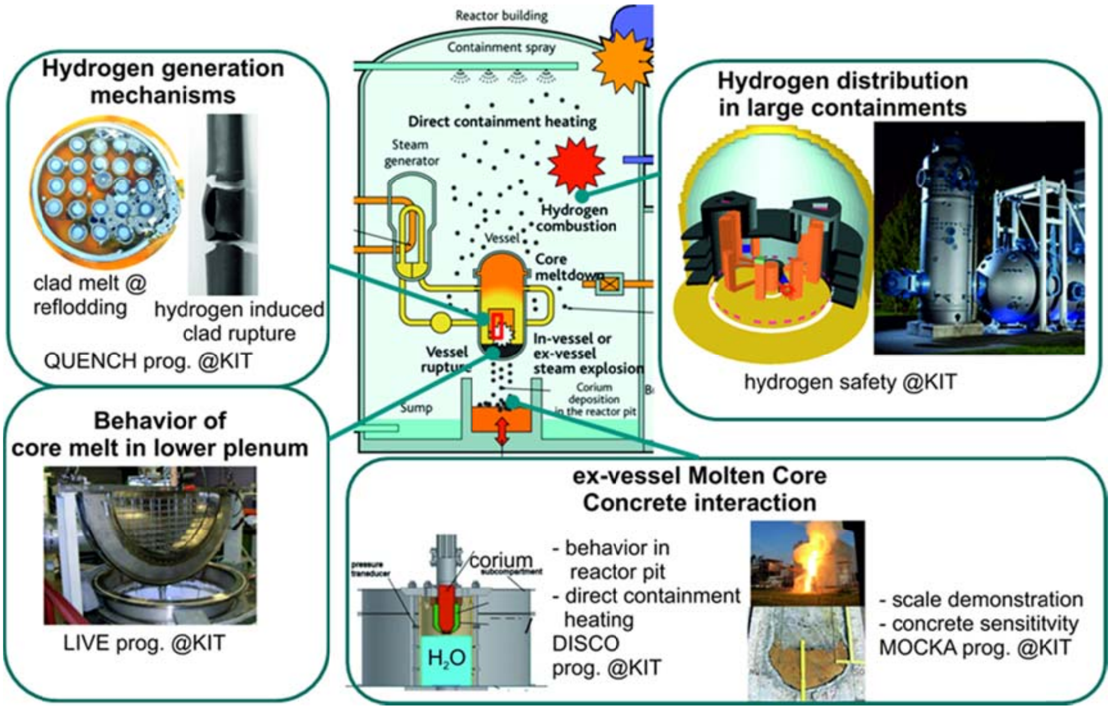

Figure 12: Potential containment phenomena occurring in a case of a severe accident [19] and associated R&D experimental programs at KIT.

3.4 LR under development

After the financial crisis in 2009 and the Fukushima incident 2011, the majority of the NPP deployed belongs to the class of LR exceeding 1GWe gross output and out of those most are PWR´s. The deployment intention of all countries is based on a long-term utilization of nuclear electricity meaning that the intended lifetime of the reactors is of the order of 60years. In their context, nuclear is seen to act as an almost CO_2-neutral energy source providing the electricity generation grid backbone.

Based on the progress in the scientific know-how to describe and validate DBA and BDBA safety methods, the power plant designer translated the hardened economic and safety related requirements into different plant designs, which vary significantly from each other at nearly all level as e.g. fundamental core design, number of hydrodynamic loops, design based operational safety and last but not least the philosophy with respect to provisions and management of severe accidents. This previous itemization is by far not exhaustive, but since all measures deduced from initial design necessitate aside from validation aspects the approval by national and international safety authorities, the integral LR plant design posed a significant challenge to the individual companies, which resulted both in the reduction of NPP producers and additionally in the formation of strategic alliances.

Aside from the economic aspects, the industrial partners share fundamental know-how regarding nuclear safety since the Three Mile Island accidents. In the last decades, common efforts from manufactures and utilities were undertaken in the frame of international co-operations and organizations such as the WANO (World Association of Nuclear Operators) aiming to get consensus on common safety design criteria for NPP, to share operation experience collected in each country and in the IAEA e.g. the event notification reports, event analysis reports and to foster mutual exchange of professionals and technical support for safety-relevant issues.

On the opposite side governments and technical survey organizations (TSO) established an international cooperation on a worldwide basis as e.g. through the IAEA (International Atomic Energy Agency). The regulatory authorities of the Western European countries have create the WENRA association devoted to intensify the cooperation and work out standardized regulations and safety requirements for the licensing of nuclear power plants.

In addition, a vast bandwidth of worldwide collaborations on dedicated topics exit within the nuclear community such as the Global Nuclear Energy Partnership (GNEP), International Framework for Nuclear Energy Cooperation (IFNEC), Multinational Design Evaluation Program (MDEP), the Contact Expert Group **(**CEG) and in Europe for example the European Atomic Energy Community **(**EURATOM**),** which themselves are complemented often by bilateral agreements on safety standards and best practice guidelines.

As result of these international activities and the beneficial interaction of manufactures and regulators, new reactors of Gen-III have been developed taking profit of the extensive operational experience of hundreds of NPP of Gen-II, of the advances in nuclear technology, material sciences, computer codes, etc. and considering the overall safety requirements continuously updated and published by the IAEA that reflects the state-of-the –art of science and technology. Hereafter, selected reactor designs without claiming for completeness are briefly described.

The European Pressurized water Reactor (EPR), depicted in Figure 13a, is based on a 4 loop evolutionary PWR design evolving from both the N4 (France) and the Konvoi (Germany) design; its rated power is about 1600 MW_{el} and it consists of 4 train active safety systems, a strong double containment design (primary containment designed for low pressure core melt, Corium spraying area, shield building), protection of the plant against commercial airplane crash by protected buildings (containment, fuel building, part of the safeguard buildings) and by physical separation (part of the safeguard buildings, diesels, …). The large core (241 FA) allows for a power upgrade, an economical fuel management allowing for 50% mixed oxide (MOX) core loading and long cycles up to 24 months [20].

Another concept currently deployed successfully by KEPCO Korea is the APR1400, figure 13b. It is also PWR with a rated power around 1400 MW_{el} using a compact core. This reactor design originated from the CE80+ developed by Combustion Engineering in the 80's (certified in USA in 1996). It is a 2-loop design with 2 steam-generators and 4 pumps having 2-train active safety systems and 4 independent mechanical trains for safety injection systems. The containment consists of a single concrete containment with steel liner with a high resistance against earthquakes. In contrast to the EPR, the severe accident management strategy focuses on an in-vessel Corium retention through external reactor vessel cooling by means of water provided from the IRWST and additionally a boric acid make-up pump [21]. Among these two PWR designs shown here, several other PWR are currently erected as the AES family (Russia), AP1000 (Westinghouse-Toshiba) and others are certified as the ATMEA (MHI-AREVA) and the APWR 1000 (MHI).

The interest in light water boiling water reactors (BWR) is considerable smaller than in PWR´s and aside the already licensed plant types such as the AB1600 (Toshiba), ESBWR (General Electric) and KERENA (AREVA), the Advanced Boiling Water Reactor (ABWR from Hitachi-General Electric) is currently erected. The safety philosophy of such reactors regarding the control of severe accidents is similar to the one of a PWR plant, while marginal differences naturally arise with respect to the operational safety due to the diverging principle.

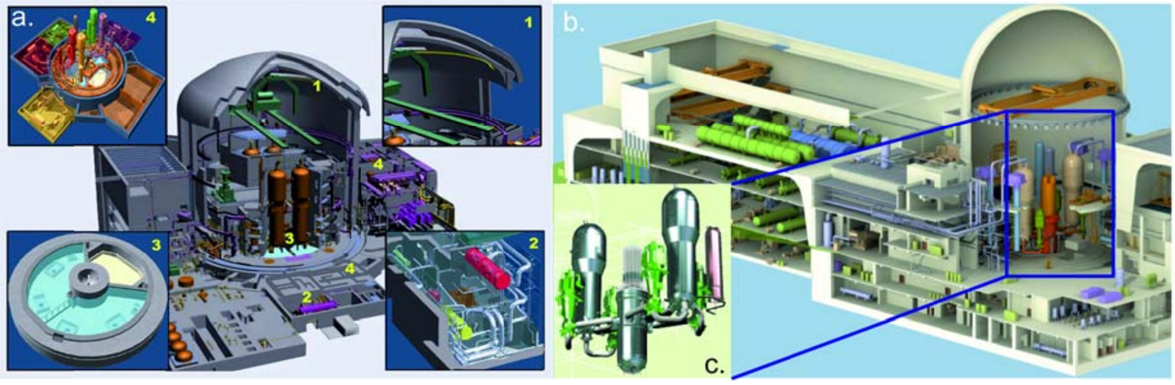

Figure 13: (a) Cut through the nuclear island of the EPR and safety systems installed. 1=double walled containment, 2= containment heat removal system, 3= corium spreading area and IRWST, 4=4 train safe guard system. b.) plant cut-away of the APR1400 by KEPCO and (c) primary circuit [21].

3.5 SMR technologies under development

The target for the SMR deployment does not only focus on electricity production but also on process heat and water desalination. Even though the capital cost per installed MWh is in principle higher than the one LR´s, the MSR's represent an attractive alternative to the LR's for some countries due to the following peculiarities:

- System simplification,
- compactness,
- modularity allowing for uprate the plant on one site by deploying multiple units,
- use of cheaper and more standardized construction techniques,
- reduction of the amount of used parts, and
- high availability and short outage times.

There is a vast variety on literature available on SMR's, [22-28]. One superior option voting for a SMR deployment is their potentially more robust safety performance by using simplified active and passive safety systems due to their lower volumetric power density.

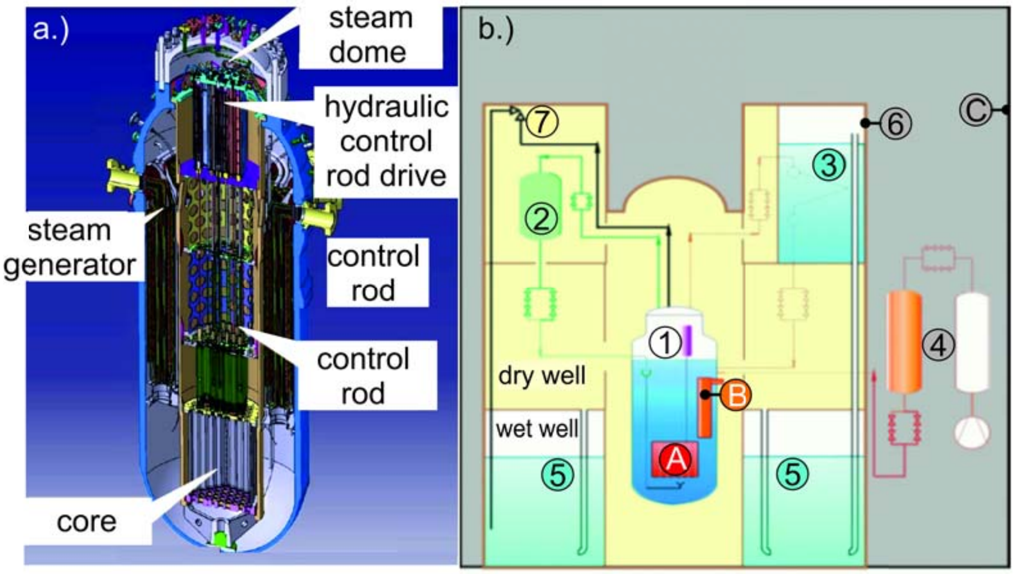

Figure 14: (a) integral core design of the CAREM 25 reactor. CAREM 25 safety systems: 1=first shutdown system, 2=second shutdown system,3=residual heat removal system primary system, 4=emergency injection system, 5=pressure suppression pool, 6=containment, 7=safety valves, A=core, B=steam generators, C=reactor building.

As one example of an SMR currently being erected in Argentina is the CAREM 25 (Central ARgentina de Elementos Modulares), which is a 27MWe PWR based entirely on natural convection and self-pressurized primary system with a nominal pressure of 12.25MPa. The CAREM reactor represents a so-called integral design, in which the steam generators are integrated within the reactor pressure vessel, as shown in figure 14a. The severe accident provisions are based on prevention by ensuring long grace period of 3 days without any intervention by autonomous systems and mitigation measures to enable an in-vessel corium retention and passive autocatalytic recombiners (PAR) for hydrogen control. The CAREM plant layout is illustrated in figure 14b.

3.6 Nuclear waste - Origin –volume and management strategies

In the public perception, the nuclear waste issue and its management are conceived as the major drawback of nuclear energy utilization. The major apprehensions are not only related to the time scales ensuring a safe confinement but also the concerns about radiotoxic consequences to be expected if the confinement is lost or unauthorized access to the disposal site . As mentioned in §2 of the Atomic Law, the nuclear energy utilization is a generation contract not only with respect to capital return time but also regarding the related waste generated during operation and fuel management including long-lived fission products and Minor Actinides and finally their decommissioning and disposal. The latter mentioned aspects require a closed and, moreover publically accepted waste management strategy consisting of a set of consecutive nuclear installations to be operated.

Which amount of masses and volumes are involved ?

Figure 15 shows the change of the fuel composition of 17x17 Fuel assembly of a light water reactor after 3 full power years (fpy) with a mean burn-up of 33GWh/t. The major fraction of the FA still consists of U^{238}(≈94.5%), which is unaffected, while the fissile U^{235} burned down to 0.73% and a minor fraction of U^{236}. About 3.41% is a set of are highly radioactive fission products such as Xenon or Cesium but also valuable constituents like Molybdenum with a short decay time. Driver for the major public concern are the Minor Actinides as e.g. Pu, Np, Am and Curium generated by neutron capture of Uranium.

Each year, nuclear power generation facilities worldwide generate about $2.10^5 m^3$ of low-level (LLW) and intermediate-level radioactive waste (ILW). Additionally, approximately $10^4 m^3$ of high-level waste (HLW) including used fuel designated as waste are produced [29]. About 94% of the waste volume is LLW type and another 5.9% belongs to ILW. Although the HLW is less than 0.1% it contains about 95% of the total inventory of radioactivity [30].

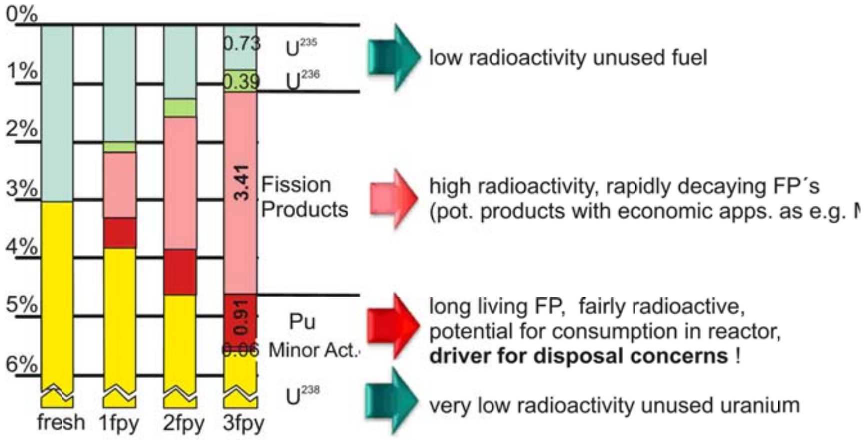

Figure 15: Fuel composition change of a fresh 17x17 LWR FA during 3 full power year operation up to 33GWd/t.

A 1000 MW_{el} light water reactor (LWR) will generate b approximately 200-350 m^3 LLW and ILW waste per year. By fission and its products a NPP produces about 10 m^3 (20-25 tons) of used fuel per year. This requires about 75 m^3 disposal volume following encapsulation if it is entirely treated as nuclear waste. In case of a reprocessing of the spent fuel, the volume reduces dependent on the technology used by a factor of 10 to 30 (theoretically substantially more) meaning that about 3-7 m^3 of vitrified waste in form of glass are the subject of nuclear waste disposal. Due to the heat generation, the glass pellets require about 28 m^3 disposal volume in a dedicated canister [29]. The technological progress e.g. volume reduction techniques, abatement technologies, etc. as well as optimization of work flows, a substantial minimization of waste was achieved in the recent decades. Nonetheless, nuclear utilization poses a societal challenge since it demands a consistent and enduring waste management policy to ensure environmentally sound solutions preventing any hazard to both workers and general public. Even, abandoning the nuclear energy option for electricity generation, there is a need to preserve the knowledge related to ionizing radiation, radiation physics, radioisotopes, etc. due to the large application of nuclear technology in areas not related with electricity generation such as medical diagnostic, automation and control, water treatment, etc. . Moreover, one should relate the numbers of NPP waste production to that of a coal fired power plant of the same size, which produces aside from CO_2 about $4 \cdot 10^5$tons ash a year containing heavy metals such as As, Cd, Hg, Pb or Thallium [31], requiring an adequate storage.

The nuclear waste in Germany is continuously monitored by the Bundesministerium für Strahlenschutz (BfS) [32]. The expected amount of nuclear waste to be conditioned in the future in intermediate and final repository is also well known and any time quantifiable.

Summarizing one can state that irrespective of societal decision taken, nuclear energy utilization requires reprocessing, conditioning and transportation to a safe confinement. All these processes are oversight by the regulatory body according to the national nuclear regulations. Regarding the waste disposal. there are several options feasible either in the temporal and the spatial frame, necessitating societal acceptance and simultaneously matching safety constraints. Regarding the temporal time window, the choice is at first an intermediate storage deciding in request further re-processing options or an ultimate solution by vitrification of the entire inventory. With respect to the spatial solution options, there are on the one hand near soil storage solutions but requiring as drawback permanent access control and confinement integrity and on the other hand deep underground disposals with or without an access option demanding also an analysis for a long term safe confinement.

3.7 Transmutation and Generation-IV

In the view of the nuclear waste generation and their interrelated issues, the utilization solely of light water reactors will lead to an accumulation of the minor actinides (MA) such as Americium, Curium, Neptunium and also Plutonium. The energy released by the fission of Plutonium can be recovered by means of a fast spectrum reactor allowing for a sustainable use of uranium resources. This potential has been identified quite early several decades ago. In May 2001 under the lead of the United States Department of Energy, the Generation IV international forum (GIF) has been founded. The top level requirements postulated by GIF for the Gen-IV reactors are: sustainability (meaning transmutation capability), enhanced economics (lower life cycle costs), improved safety (low probability or even absence of any off-site emergency measures) and non-proliferation. This GIF initiative currently consists of 12 countries. The EU is involved by cooperations within international frameworks such as the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) or within the Europe the Sustainable Nuclear Energy Technology Platform (SNETP).

One of the major aspects for the fast spectrum reactors is their transmutation capability. Transmutation hereby describes the transfer of radionuclides by neutron induced fission or neutron capture into another element as illustrated in figure 16. By dedicated design measures, fast spectrum reactor systems are therefore able either to breed fissile material or to destruct fissile minor actinides. One of the drawbacks of the fast reactor systems is that they require a dedicated fuel cycle incorporating a dedicated fuel fabrication and specific fuel reprocessing plant, which intrinsic implies a nuclear logistics system.

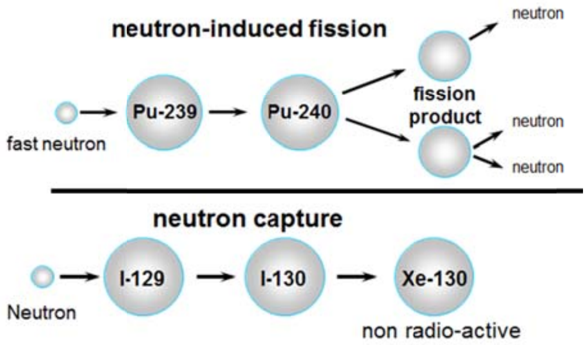

Figure 16: Transmutation of nuclei by fission of fast neutrons (top) or by neutron capture (bottom).

Figure 17 depicts such a potential fuel cycle including partitioning and transmutation. As clearly seen in figure 17, each conversion process causes losses and naturally also the presence of a temporary storage so that a final repository cannot be prevented. However, anticipating such an option would require a substantially reduced volume of temporary storage and final repository and additionally it will considerably reduce the radioactive inventory, the total radiotoxicity and also the time scales for mandatory monitoring [33].

Of course, the implementation of a “new fuel cycle “ especially in countries like Germany targeting for an exit of nuclear causes concerns. Therefore, a detailed study has been conducted in the context of the German Academy of Sciences (ACATECH) investigating not only technological aspects or potential hazards including man made but also societal concerns and the boundary conditions to be set or at least to be prepared. One major aspect is that not only national scenarios but also mixed options considering countries continuing nuclear energy utilization and countries exiting nuclear energy generation have been studied. Independent of the P&T scenario anticipated, a full destruction of the long lived fission products is a one century enterprise. Mandatory for such P& T option is in any case the use of fast spectrum reactors.

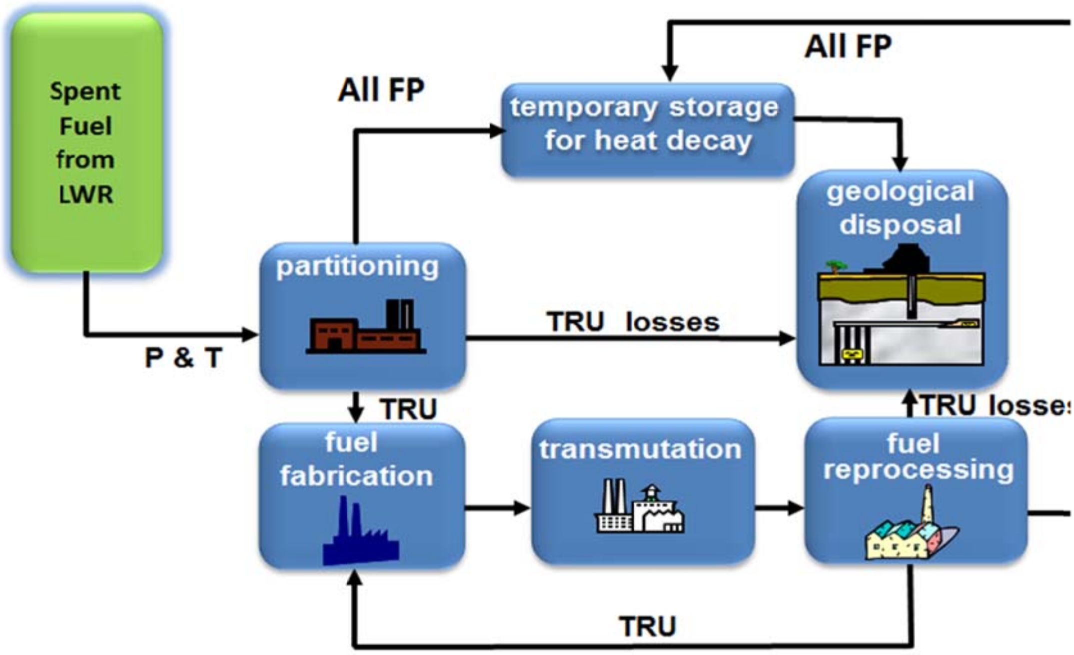

Figure 17: Potential fuel cycle required for transmutation of minor actinides (FP=fission product, TRU=Transuranic elements, P&T=Partitioning and Transmutation).

4 Fast spectrum generation IV reactors

Nearly all reactor types considered in the context of Gen-IV initiative are fast spectrum reactors matching to a large extend the requirements postulated in the Gen-IV roadmap. The most mature in this context are sodium-cooled fast reactors (SFR) or lead cooled fast reactors (LFR) currently considered in the worldwide development as to be the viable reactor types, since at least for the SFR a considerable operational experience has been gained in the last four decades in several countries worldwide. The reputation of the SFR is considered in the public as critical since these reactors have been in the past either designed to as equilibrium systems to produce as much fissile materials as they consume (sustainability) or even in direction to produce ("breed") a Plutonium stockpile during the cold war. But, by dedicated design and fuel composition means, SFR are also able to transmute long-living fission products. A cross-sectional cut as well as photographs of the erection of a SFR in India shows the figure 18a. To allow for a high core safety similar to LWR´s the amount of MA in the fuel shall not exceed 2-5%. In any case SFR-reactors are fast "critical" systems in contrast to so-called accelerator driven systems (ADS), in which MA can be burned. These ADS Systems, which most prominent example is the MYRRHA reactor [33] are so-called sub-critical reactors, which are externally triggered by a proton beam accelerator. The protons generate in the target the neutron source, see figure 18b. By design, the core enveloping the target is subcritical so that the power decays in case of a loss of the beam or a beam shut-down.

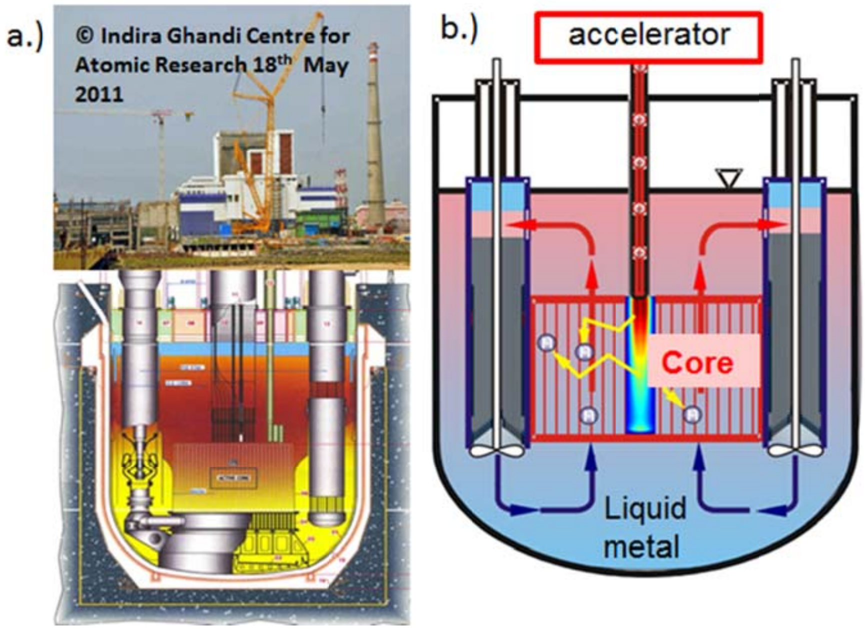

Figure 18: (a) Sodium cooled fast reactor currently erected in India and core cross-section. (b) Schematical cross-sectional cut of an eutectic PbBi cooled sub-critical ADS reactor developed at SCK-CEN, Belgium.

5 Summarizing comments

Nuclear energy utilization represents still a substantial part of the worldwide electricity production, mainly generated by generation II power plants. Nuclear energy production is pursued in numerous countries mainly in Asia as a long term electricity production backbone ensuring fossil fuel independence. As a consequence, most reactors currently deployed are large scaled light water reactors of the generation III rather than SMR´s which would fit better in a dispatched organized grid. The development of these generation III reactors benefitted considerably from the scientific progress especially with respect to their safety performance not only for the design basis but also with respect to the beyond design basis accidental behaviors and measures now already integrated in the design. The fundaments of this safety performance gain are changes in the safety culture by the internationalization through all nuclear interest holders, as science, industry, technical survey organizations and governments in the last two decades. Irrespective of the further energy utilization nuclear is a generation contract.

Hereby, the waste management, the processing and the logistics play an essential role, demanding a continuous monitoring and a sensible long-term oriented technological planning complemented by public acceptance. The amount of nuclear waste and its volume is small compared to the ones of conventional fossil based energies. In this context, partitioning and transmutation in fast spectrum reactors offers a credible option to minimize the burden on future generation either by national efforts or integrated in a regional context. Independent of the societal decision on the future use of nuclear fission for energy production, the development of education in nuclear engineering must persist of vital interest to an industrialized country like Germany to assure not only a credible nuclear safety assessment capability but also further investigations to tackle the technical and scientific challenges related to the final disposal of nuclear waste, which is still far ahead of us.

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