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Zero power reactors in support of current and future nuclear power systems

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ABSTRACT

Zero-power reactors stand as indispensable tools for shaping the future of the nuclear industry. Addressing safety concerns, advancing reactor technology, mitigating proliferation risks, fostering education, and promoting economic viability, these reactors hold the key to unlocking the full potential of nuclear energy in a sustainable and responsible manner. As the world seeks cleaner and more efficient energy solutions, the importance of zero-power reactors cannot be overstated in charting the course for the nuclear industry's future. The paper presents a short history of the various zero-power/zero-energy experimental facilities constructed and used worldwide. Many of the names seemed to be lost to history and archives, which means that all the experimental data carried in the those facilities is lost as well. However, re-introducing the various names can spark an interest in "digging up" and revisit experiments of the past, which can help in the design of experiments and new systems in the future. It is clear that a new experimental facility should be built. The next frontier in zero-power reactor design envisions a design for versatility, this future concept addresses diverse energy needs while contributing to a sustainable and responsible nuclear energy landscape. This was demonstrated in the framework of the Zero-power Experimental PHYsics Reactor design proposed by French Atomic Energy Commission.

1. Introduction

The 2015 Accord de Paris sur le climat (a.k.a the Paris Agreement) (United Nations Framework Convention on Climate Change, 2015) for the first time in history acknowledged global warming as a problem recognized by almost 200 countries. The Paris Agreement's ultimate goal is to keep the increase in global average temperature well below 2 °C above the pre-industrial levels, this would significantly limit the risk and effects of climate change. In the environmental community there is almost a full consensus that the continuous increase in global temperature is caused by anthropogenic Greenhouse Gas (GHG) emissions (Parry et al., 2007). In the United States, according to the United States Environmental Protection Agency (US EPA), 50% of total GHG emissions originate from the electricity and industrial sectors, according to the US EPA, large portion of emitted GHG is because of the continuous burn of fossil fuels (oil and natural gas) in the two sectors. Therefore, the current tendency is to reduce these emissions by utilizing renewable sources of energy, such as sun, wind and hydro. However, each one of these solutions have drawbacks in terms of continuous availability (sun and wind), maximal potential capacity, and geographical limitations (hydro).

Unlike fossil fuel-fired power plants, nuclear reactors do not produce large quantities of GHG in any stage of the nuclear fuel cycle (Sovacool, 2008). Although, fossil fuel-fired energy sources are used to support of the nuclear fuel cycle front-end (i.e., uranium mining, fuel manufacturing). Thus, de-carbonization should not only include the electricity generation sector, but the industry sector as well. However, nuclear power suffers from a controversial image, and typically addressed negatively in the public eye. The two main factors being, the safe operation of the plants and the radioactive waste at the back-end of a Nuclear Power Plant (NPP). Nevertheless, including nuclear power in the energy mix is the only viable solution to meet the goals set by the Paris Agreement.

The nuclear power sector is probably the most regulated industrial sector, as it is understood that safe operation of a NPP during its lifecycle is the most important factor. To ensure safe operation, the nuclear research, operators and licensing communities are wholly dedicated to studying the safety of current and future NPPs designs. Generally, there are two main ways to support research related to NPP safety, an analytical approached, which is based on the utilization of best-estimate computer codes. The other is experimental, where it is possible to utilize specially designed experimental systems to imitate NPP relevant condition, such as, thermal-hydraulic loops for flow characteristic studies (O'Brien et al., 2014), or dedicated research reactors.

Research reactors are constructed to provide support to a wide range of civil and commercial needs (e.g., generation of radioactive isotopes for medical purposes, neutron sources, reactor physics, etc.). Typically, these reactors operate at low thermal power levels (about 100 MWth), compared with the commercial NPPs (about 3000 MWth).

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Abbreviations

7uPCX	Seven Percent Critical Experiment
AGR	Advanced Gas-cooled Reactor
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical
BFS	Bolshoy Fizicheskiy Stand (Big Physical Facility)
BORAX	Boiling Reactor Experiment
BR	Belgian Reactor
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
CDA	Core Disruptive Accident
CEA	Commissariat à l’énergie atomique et aux énergies alternatives
CLEAR	China LEAd-based Reactor
DIMPLE	Deuterium Moderated Pile of Low Energy
EFR	European Fast Reactor
ELSY	European Lead-cooled SYstem
FBR	Fast Breeder Reactor
FCA	Fast Critical Assembly
FFTF	Fast Flux Test Facility
GHG	Greenhouse Gas
GLEEP	Graphite Low Energy Experimental Pile
GUINEVERE	Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor
HAZEL	Homogeneous Assembly Zero-Energy Laboratory
HECTOR	Heated Experimental Carbon Thermal Oscillator Reactor
HFIR	High Flux Isotope Reactor
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
IPPE	Institute of Physics and Power Engineering
KATHER	High Temperature Pebble Bed Reactor
KUCA	Kyoto University Critical Assembly
LIDO	A lido in the UK is an outdoor public swimming pool
LMFBR	Liquid Metal-cooled Fast Breeder Reactor
LWR	Light Water Reactor
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
NCA	Nuclear Critical Assembly
NERO	New Experimental Reactor
NESTOR	Neutron Source Thermal Reactor
NPP	Nuclear Power Plant
NRAD	Neutron Radiography
PCF	P Critical Facility
PFR	Prototype Fast Reactor
PWR	Pressurized Water Reactor
RBMK	Reaktor Bolshoy Moshchnosti Kanalnyy (High Power Channel-type Reactor)
RCF	Reactor Critical Facility
SCA	Severe Core Accident
SFR	Sodium-cooled Fast Reactor
SGHWR	Steam Generating Heavy Water Reactor
SMR	Small Modular Reactor
SNEAK	Schnellen NullEnergie-Anlage Karlsruhe
SPERT	Special Power Excursion Reactor Test
SS	Stainless Steel
STACY	STAtic experiment Critical facilitY
SUR	Siemens-Unterrichts-Reaktor 100
TAPIRO	TAratura PIla Rapida Potenza ZerO (Fast Pile Calibration at Zero Power)
TCA	Tank-type Critical Assembly
TMI	Three Mile Island
TREAT	Transient Reactor Test Facility
US EPA	United States Environmental Protection Agency
VENUS	Vulcan Experimental Nuclear Study
VHTRC	Very High Temperature Reactor Critical Assembly
VVER	Vodo-Vodyanoi Energetichesky Reaktor (Water-Water Power Reactor)
ZEBRA	Zero Energy Breeder Reactor Assembly
ZED	Zero Energy Deuterium
ZEEP	Zero Energy Experimental Pile
ZENITH	Zero-Energy High-Temperature Reactor
ZEPHYR	Zero-power Experimental PHYsics Reactor
ZLFR	Zittau Training and Research Reactor
ZOÉ	Zéro de puissance; Oxyde d’uranium; Eau lourde
ZPPR	Zero Power Physics Reactor
ZPR	Zero Power Reactor

The spectrum of research reactors is wide and their utilization vary from facility to facility. Best way to classify those facilities would be in the power rate. The tens of MW reactors, similar to the Jules Horowitz Reactor or the PIK high-flux reactor, are used as advanced neutron beams for materials analysis and source for high neutron fluxes. Facilities with power rating of around 10 MW, similar to the Heinz Maier-Leibnitz FRM II, are utilized for material testing, neutron beams for research, and radioisotope production. The 100 kW to a few MW facilities, similar to the Training-Research-Isotopes-General Atomics (TRIGA), are used worldwide for teaching, research, material irradiation and some radioisotope production. Most of the research reactors in the world belong to this class. Finally, the very low power reactor, also known as ’critical facilities’, are primarily used for fundamental reactor physics and neutron characteristic studies. In some case those facilities would be designed to serve as mock-ups of a power reactor core. The final definition that must be addressed is that of a ’critical assembly’, which according to the International Atomic Energy Agency, is referred to neutron multiplying systems, which is flexible in character, assembled form fissile and other materials. In the US and other Western countries this terminology was utilized to describe criticality experiments like Flattop or Godiva. However, in Eastern countries this terminology was expanded to low power reactor such as the Fast Criticality Assembly in Japan and V-1000 in the Soviet Union. Therefore, in this paper the term zero-power reactor and critical assembly will be used based on how the specific facility was regarded in the country of origin. In this paper, the focus is made on research reactors that operate at zero-power level (several watts). Zero-Power Reactors (ZPRs) have been used since the dawn of nuclear engineering. The first ever reactor to reach self-sustaining chain reaction, designed by Enrico Fermi, the Chicago Pile No. 1 operated at thermal power level of just several watts (Sehgal, 2012). Between 1942 and 1956, most of the constructed reactors can considered as zero-power, reactors such as the French “Zoe”, the Canadian “ZEEP” (Zero-Energy Experimental

Pile), the British "GLEEP" (Graphite Low Energy Experimental Pile), or the Russian F-1 reactor. However, these reactors operated with one end goal, the atomic bomb, and only in 1953 did president Eisenhower announce the Atoms for Peace program, which established the ground for a civil nuclear program for power generation.

With the development of the civil nuclear program, there was a growing need for experimental facilities to study physics problems related to large reactors. Programs dedicated to studying Light Water Reactors (LWRs) under destructive conditions were performed in the Boiling Reactor Experiment (BORAX) (Haroldsen, 2008) and the Special Power Excursion Reactor Test (SPERT) (Heffner and Wilson, 1961; Crocker and Stephan, 1964) in the middle of the 50's. Nuclear data measurements and other reactor physics programs were studied in the French zero-power facilities Eole and Minerve (named after the Greek god of wind and Roman goddess of wisdom respectively) (Cathalau et al., 2014). A more detailed review of past and current ZPRs, and experimental programs carried out in them, is presented in the following section of this paper.

During the nuclear power golden era (60's to 80's), before the Chernobyl accident (Sehgal, 2012), which demonstrated to the world for the first time the shear power of a run away reactor (Three-Mile Island was contained accident, with no failure of the containment), the regulation was much more flexible to the kind of experiments performed in research facilities (destructive test BORAX and SPERT, molten slat reactor experiment etc.). The two accidents (Chernobyl and Three Mile Island) led to drastic changes in the regulation of nuclear reactors, which affected research reactors as well. However, the need to continue and support NPPs has not disappeared, and probably became much more important, the research reactor community had to adapt. A smarter and safer way to model different phenomena in research reactors had to be developed to provide support to the commercial NPPs with the highest requirements for safety. Furthermore, in 2009, the OECD report on "Research and Test Facilities required in Nuclear Science and Technology" highlighted the shortage of research facilities to nuclear and neutron physics measurements for exiting and new reactor upgrade and development. The nature of ZPRs of low operation power means the material balance within the core does not change, as a result spatial flux distribution can be reconstructed with great accuracy. Those make ZPRs ideal for validation and verification activities of simulation tools and instrumentation. Therefore, a complete loss of those facilities will not only mean loss of experimental skills (there is a clear lack of nuclear data evaluators worldwide), but also the future capacity at acquiring new data to support verification, validation and uncertainty quantification of our simulation tools and to experimentally investigate new phenomena, materials, develop instruments, and new reactor concepts to foster innovation.

The need for a new reactor physics testing facility became obvious when the Eole and Minerve closed in 2017, and Masurca, the world last ZPR for fast application, closed in 2018 (was in a refurbishment state to support the ASTRID project). The fist two played key role in the LWR development world, providing data for tool validation, nuclear data, and for the design of standard and advanced LWRs. Following the closure of those facilities the NEA established a working party on scientific issues to identify whether there is a need for future ZPR facility. The message that came out of the working party was clear — there is a need for a new facility to support research in the reactor physics field. The need for additional experimental data is great, as relaying on historical experimental data almost always suffers from poor documentation.

The objective of the paper is to provide an insight to the past and present of the ZPRs, and look into the potential future needs and innovation that can be found in the field of design of new ZPRs.

2. ZPRs - The past and the present

ZPRs present several advantages in nuclear reactor safety studies, the low power reduces the modeling complexity of these facilities, as heat is not generated. The fuel loaded into a ZPR maintains the same material composition through the reactor life-time, due to the low power there is no depletion of fuel. The core is usually designed in such way that instrumentation could be placed in different locations inside and around the core. These reactor characteristics make ZPRs ultra-safe reactors (pictures taken during operation of some research reactor is shown in Fig. 1) that allow complicated modeling of different reactor physics phenomena, which is sometimes impossible in full commercial power plants.

Since the dawn of nuclear power generation ZPRs where used to demonstrate capabilities of different innovative technologies, related to thermal and fast reactors, nuclear data measurements, irradiation of samples, and the generation of radioisotopes. The history of nuclear power generation is full of examples of "firsts", and this section will try to provide an overview for key ZPRs (according to the authors) operated around the world. It should be noted, in the early days (post world war II) the goal of the nuclear countries was to obtain an atomic weapon. Therefore, information regarding some of the first ZPRs is not accessible as they were military facilities. Thus, the focus will be on facilities that are used for civilian programs, towards the peaceful utilization of nuclear power.

2.1. United Kingdom

The first country with a rich history of ZPRs is the United Kingdom, although not at current state, but post world war II, when the United Kingdom was a heaven for nuclear development. Looking back, in 1968 the nuclear installed capacity was about 4200 MW producing 99,139 millions kWh (Hill, 2013), for comparison the United States at the same time had nuclear capacity of 2900 MW producing 35,182 millions kWh. The nuclear community in the UK can credit to herself a number of firsts, such as, the first nuclear reactor in western Europe (Graphite Low Energy Experimental Pile - GLEEP), the first commercial nuclear power station (Calder Hall at Windscale), and others. Research reactors had a large role in the development of the nuclear industry in the United Kingdom, Fig. 2.

In the United Kingdom, the roots of nuclear power generation take place in the military program, in the chase of nuclear weapons. The plutonium generating and military reactors are ignored, and the focus is made on the civilian program. Thus, the first nuclear reactor in Britain was a ZPR - GLEEP. The reactor went critical in August of 1947, with a main purpose — the measurement of thermal neutron absorption cross-sections in different elements. The realization of the cross-section measurement was done by oscillation of different material samples in the core. The rapid movement of the sample from the outer part of the core, to the center and back produced a periodic change in the power level of the pile. From this change, the degree of absorption by the sample could be obtained.

The following ZPR constructed was named DIMPLE, a rather forced abbreviation of Deuterium Moderated Pile of Low Energy (Hill, 2013). The reactor went critical in July of 1954, and consisted of a heavy water moderated design, which was enclosed in a tank and radially shielded by graphite. The reactor was utilized for nuclear data studies, with a similar technique as in GLEEP. Later the reactor was utilized as a benchmark facility for calculational methods validation developed for water cooled reactors. In particular, the reactor was utilized for the design of a prototype Steam Generating Heavy Water Reactor (SGHWR) (Middleton, 1975). The final role of the reactor was related to the criticality field, and pursuing an experimental programme relevant to the manufacture, transport, storage and re-processing of reactor fuel (Ingram, 1984). A short complementary program to the DIMPLE, established four years later, enabled commercial firms to get

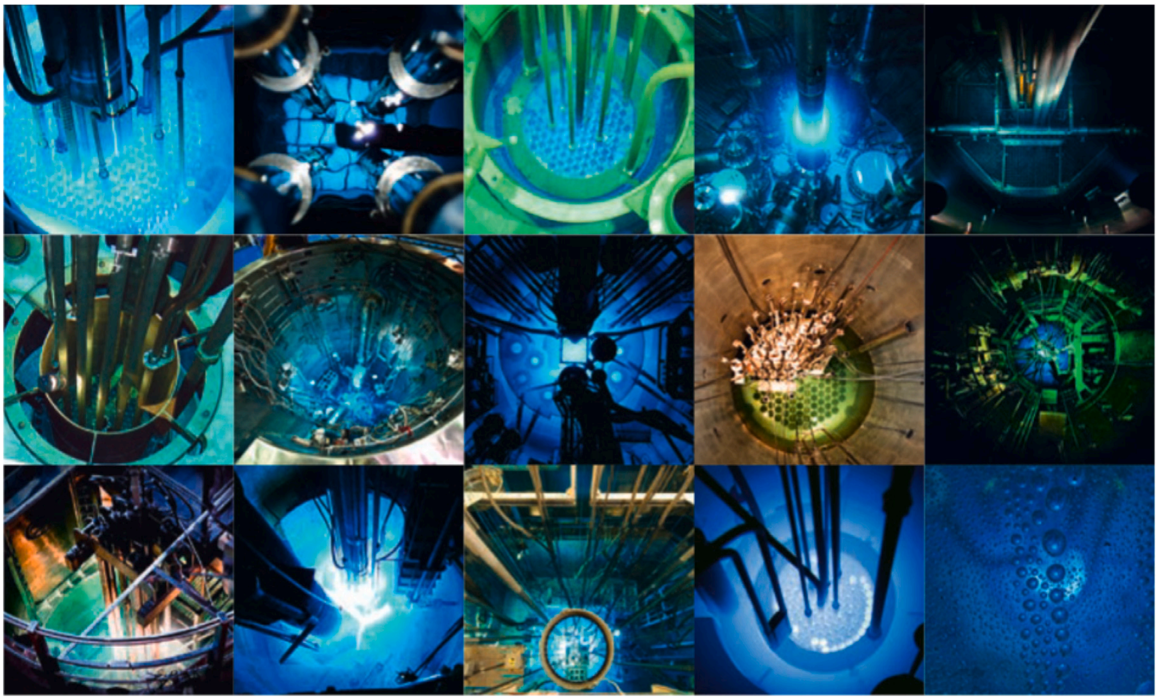


Fig. 1. Core view of different research reactors - Sandia National Laboratories, TU Delft, Centre National de l'Énergie des Sciences et Techniques Nucléaires, Technische Universität München, Czech Technical University in Prague, Dalat Nuclear Research Institute, Australian Nuclear Science and Technology Organisation, Research Centre Rez, Korea Atomic Energy Research Institute, CEA - Commissariat à l'énergie atomique et aux energies alternatives, Oregon State University, Jožef Stefan Institute (IAEA, 2016).

involved in the nuclear business. The HAZEL (Homogeneous Assembly Zero-Energy Laboratory) was constructed in 1958 to study basic nuclear characteristics of heavy water moderation in homogeneous systems (Sabel et al., 1961). The program was stopped after only six month of operation (Hill, 2013).

Being a naval empire, the nuclear needs of the United Kingdom did not stop at weapons and power. The need for nuclear naval propulsion attracted focus as well. The LIDO reactor was a thermal pool-type reactor, which operated at power level of about 100 kW (slightly high for a ZPR). The reactor started it operation 1958 under the responsibility of the Admiralty, and was used to test materials to be used for shielding reactors (an important consideration for a nuclear submarine). Latter LIDO was replaced by the Rolls-Royce designed zero-energy reactor NEPTUNE.

The ZPR history in the United Kingdom contains a large number of zero-energy facilities, which were build to support research of specific reactor designs. Such was the New Experimental Reactor (NERO), which was constructed to support the design of the Advanced Gas-cooled Reactors (AGRs). Later in life the reactor provided information on sodium-cooled systems. NERO achieved first criticality in February of 1957. Later NERO, was replaced by a water-moderated reactor — JUNO. JUNO was utilized in a similar manner as NERO, but focusing on light- or heavy-water reactors, providing supplementary results to the research conducted in DIMPLE (Fry, 2015). To extend the studies in NERO, the Heated Experimental Carbon Thermal Oscillator Reactor (HECTOR) was constructed. Commissioning of HECTOR took place in March 1963, where the focus was made on studies related to fuels, moderators and structural materials in power reactors in a wide range of experimental conditions. Particularly, the reactor was used to study plutonium-239 and −240 build-up in the fuel of the Magnox and AGR designs.

In parallel, to the operation of DIMPLE, a second reactor, known as NESTOR (Neutron Source Thermal Reactor), was constructed and reached criticality in December of 1960. The purpose of NESTOR was to provide the neutrons required for the experimental assemblies of nuclear fuels and moderators which were used to obtain design data of future systems. The NESTOR was designed to validate calculation methods employed to predict the activation of nuclear reactor grade steels. As well as, utilized for the development and calibration of instrumentation (Fry, 2015). The two facilities (NESTOR and DIMPLE) were among the world's longest running and most contributing research facilities in the world. They provided a large amount of data on safety and performance of nuclear reactors to the industry.

Up until now, only thermal systems were covered. However, the ZPR history in the United Kingdom includes several experimental facilities that were dedicated to the study of fast reactor technologies. The first two zero-power fast critical facilities in Britain were — the Zephyr (not an acronym as in the French reactor) and Zeus, constructed in 1954 and 1955 respectively. The difference between these two reactors was the utilized fuel, the Zephyr was loaded with plutonium while Zeus was loaded with enriched uranium. The reactors were used to study different physics parameters related to fast reactors, such as — neutron energy spectrum, breeding ratio (Shepherd, 1956), structural materials studies for fast reactors, and fast spectrum kinetics (Fenning, 1956).

Following Zephyr and Zeus, a bigger experimental facility was constructed in 1962 to support the fast reactor research programs, this reactor was known as the Zero Energy Breeder Reactor Assembly or ZEBRA (Smith, 1962). The reactor was designed as a mock-up assembly representing such facilities as the Prototype Fast Reactor(PFR) (Jensen and Olgaard, 1996), the Japanese fast reactor MONJU (Aoki, 2004), and the proposed European Fast Reactor (EFR) (Lefevre et al., 1996). Furthermore, the facility provided support for nuclear data validation, instrument development and calibration. Some experiments were designed to test methods used to treat heterogeneity by imposing void in different locations in the core, material compaction, and materials stratification. The fuel in the reactor was similar to other facilities in the world (in particularly the MASURCA reactor in France). ZEBRA provided a large quantity of high quality data for the design of fast systems.

The final facility that should be mentioned in the UK context is the zero-energy high-temperature reactor (ZENITH), which was the experimental facility used to validate the Dragon design. The conditions

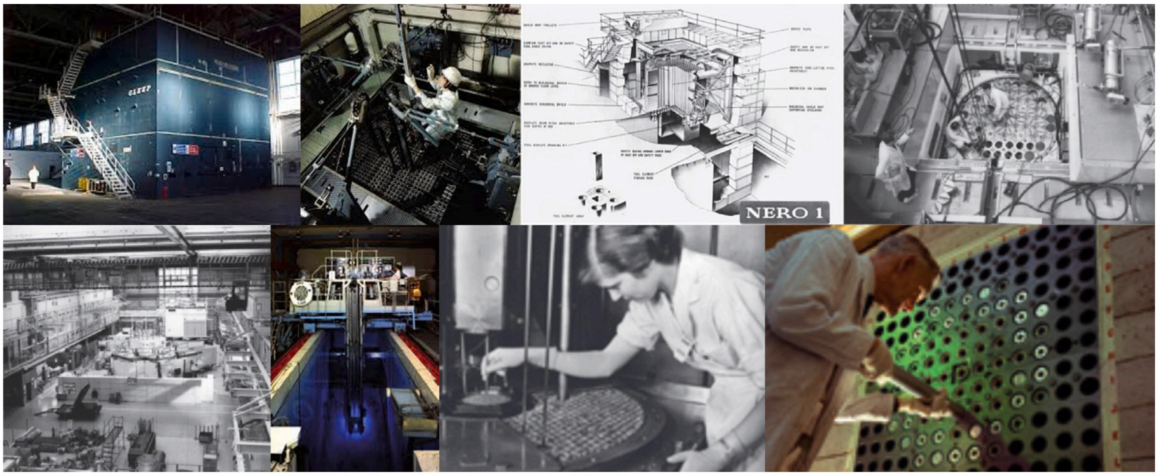


Fig. 2. United Kingdom zero-energy facilities - GLEEP,ZEBRA, NERO, DIMPLE, LIDO, ZEPHYR, NESTOR (Hill, 2013; Fry, 2015).

Fig. 3. United Kingdom decommissioning of GLEEP, ZEBRA, NESTOR and DIMPLE (Magnox).

at ZENITH were similar to those in Dragon, containing uranium, thorium and graphite. Some of the experiments conducted in the reactor reached temperatures up to 1000 °C. Thus, the activities in the two facilities (Dragon and ZENITH) were used to deliver a design of a commercial high-temperature reactor to replace the first generation of AGRs. However, at the time the advantage over the AGR was not clear or economical. In the current UK nuclear landscape, it is seen that there is a clear preference for high-temperature reactors to be constructed (World Nuclear News, 2022). But, the technology seems to be imported, with the HTGR Japanese design preferred (NikkeiAsia, 2023), which makes it unclear why existing designs based on experimental facilities developed in the UK is not used to help advance the technology forward.

The story of the nuclear industry in the United Kingdom is grim, from an atomic empire (Hill, 2013), during the previous decade, to an almost complete standstill in the 21st century. Currently, in the United Kingdom there is no operational zero-power research facility in civilian hands. The only research reactor in operation is the Neptune reactor, which is a Rolls-Royce facility supporting the naval propulsion programme . Most of the facilities mentioned thus far, are demolished and some are even returned to their previous states as green fields, Fig. 3.

2.2. France

France in the eyes of many is considered as nuclear heaven, where in 2017 more than 70% of the electricity generated was from NPPs (Schnider et al., 2017) (Fig. 4). The dawn of nuclear power in France started not with a reactor, but with a "pile" (as fuel and graphite were piled together to form a critical mass). The first French pile reached criticality on December 15th, 1948 at Fontenay-aux-Roses (outskirts of Paris), and was named ZOÉ - Zéro de puissance; Oxyde d'uranium Eau lourde (zero-power; uranium oxide; heavy water) (cea, 2012). The reactor operated at about 150 kWth, and was utilized for the first nuclear reactor physics studies in France. The reactor was designed to study materials behavior under irradiation (graphite, control absorbers, structural material, etc.). However, it became clear that more sophisticated reactors were needed, critical mockups, irradiation facilities, safety studies, and prototype reactors for industrial demonstrators. Critical mockups for neutron studies, characterized by high operating flexibility, easy access for measurements, geometry flexibility, and a power nearly equal to zero.

The focus in the first years of the nuclear program in France was on improving nuclear data knowledge of natural uranium fueled reactors (i.e., moderated by heavy-water or graphite). That was the main task of

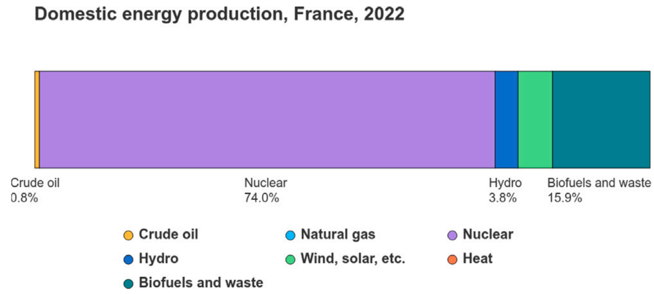


Fig. 4. French electricity generation by sector (Intenational Energy Agency, 2022).

the ZOÉ and later AQUILON reactor in Saclay, France (reached criticality in 1956). In parallel, France looked into propulsion, for that goal the ALIZÉE reactor was constructed in 1959, this reactor was light-water moderated with enriched uranium. It joined it's predecessor, AQUILON, in this task. In parallel, France start to build larger test reactor in the mega-watt scale. However, as more discoveries were made in the field, a new reactor is constructed in the Sacley facility of the Commissariat à l'énergie atomique (the French atomic energy commission), the MINERVE reactor. The task of this was to measure neutron parameters, such as neutron spectra, resonance integrals, and reactivity effects, utilizing techniques developed by scientists at Sacley as J. Yvon, J. Horowitz, G. Vendryes and J. Bourgeois (all world-renowned pioneers in nuclear reactor physics). The development included miniature fission chambers, activation detectors and oscillation techniques.

In the sixties, additional zero-power facilities were built, as a demand for additional studies was rising. The MARIUS reactor was constructed at the Marcoule site of CEA in 1960, and in 1965 was transferred to Cadarache. The reactor was built for basic neutronic studies of graphite moderated cores, and for parametric studies of the French gas-cooled reactors via what is known a "substitution" method, i.e., progressively modifying the fuel element lattice without changing the reactor core. Furthermore, the reactor was utilized for qualification of components for future power reactors. In addition to MARIUS, the CÉSAR reactor, that was dedicated as well to the gas-cooled reactor program (with a slight modification) was designed to work in the hot conditions of the gas-cooled reactor. The reactor reached first criticality 1964, at the Cadarache site. The reactor was designed to allow lattice studies, measurement of temperature coefficients by the means of oscillation of irradiated fuels. Later, in 1971, the CÉSAR reactor was converted to support studies related to High-Temperature Reactors (HTR), including investigating different fuels, such as — pebble cores and prismatic lattices.

During this period another ZPR joined the water moderated reactor family, the ÉOLE reactor at Cadarache, which reached criticality in December of 1965. The reactor, was at the beginning, dedicated for studies related to heavy-water reactors. In later years it was changed to support the light-water reactors constructed in France. The reactor was design as a mock-up of a water cooled reactor, and was utilized for lattice studies and qualification of fuels.

Finally, in order to pursue the design on fast reactors, the HARMONIE source reactor and the MASURCA critical mock-up started up almost simultaneously at Cadarache. HARMONIE started its operation in 1965. The reactor was loaded with highly enriched uranium (above 90% of ^{235}U) mobile kernels that can be removed from the shields. Experimental canals allow for abroad variety of neutron spectra, and the possibility to achieve pulsed-mode experiments. HARMONIE was considered as a high value experimental tool, especially because of the first neutronics qualifications of shielding materials for fast reactors.

All the contribution of MASURCA to the design of fast reactors can easily fill a paper. However, even today, the results obtained in the facility are limited only for the internal use of the CEA, and not available to the reactor physics community. MASURCA was put into the reactor mix of CEA in 1966. Over the years the reactor produced a large number of criticality benchmarks to investigate fast reactor behavior. The size of the reactor allowed achieving large representative cores of fast reactors, as it could contain up to 2 tons of plutonium. The studies performed in MASURCA allowed the construction of the two fast reactor demonstrators PHÉNIX and SUPERPHÉNIX (Schneider, 2009). MASURCA was the last ZPR to be built in France (cea, 2012).

What the future holds — from the list of mentioned reactors, only the AZUR reactor continues its operation for naval purposes. The EOLE and MINERVE were the last to shut down in December of 2017 (Blaise et al., 2016), as a result of post-Fukushima safety studies. The MASURCA reactor was in a long status of refurbishment for the support of the CEA Sodium-cooled Fast Reactor (SFR) project ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) (Chenaud et al., 2013; Bertrand et al., 2018). However, the ASTRID project suffered a fatal blow on the government level (Reuters Team, 2018). Thus, although there is not official support, the MASURCA reactor refurbishment process was stopped. This means that currently France is without any operational ZPRs. Therefore, CEA is investing in the construction of a new ZPR at the Cadarache site — the ZEPHYR (Zero-power Experimental PHYsics Reactor). The description of the ZEPHYR and it possible capabilities will be presented in a separate section of this paper. Some of the past facilities are shown in Fig. 5

2.3. United States

The history of the nuclear energy, and in particularly the research reactors history, in the United States is probably one of the most interesting stories to be told. However, not all the information about all the facilities that operated in the United States is available to the public. It is possible to estimate, that in the United States alone, before the Three Mile Island accident (Sehgal, 2012) there were around 300 research reactors constructed at Department of Energy and Department of Defence laboratories.

In 1990's a change in the research path happened in the United States, shifting from experimental work (construction of research reactors for different tasks) to a more "numerical simulations" oriented path. Apart from several important facilities, such as the ATR (Advanced Test Reactor) and its critical version ATRC (Advanced Test Reactor Critical) at Idaho National Laboratory, and the HFIR (High Flux Isotope Reactor) at Oak Ridge National Laboratory all the critical mock-ups of national laboratories have been closed or put in long preservation (IAEA, 2012). Thus, de facto, the experimental research activities shifted to the American university (about 30 reactors in operation across the country) (Rogers, 2002). The non-perpetuation of critical models (and research reactors in general) in the United States was justified by an almost non-existent innovation in LWR design, as well as the abandonment of the industry (Schnider et al., 2017).

Nevertheless, not all the needs of the industry can be supplied by university facilities. Therefore, the time came when life-extension of existing NPPs became an issue, there was an urgent need to construct a ZPR facility. The Seven Percent Critical Experiment (7uPCX) was constructed in 2007 at Sandia National Laboratory (Harms et al., 2015), the experiment is dedicated to uranium-oxide (enriched up to 7%) core physics and associated criticality-safety issues. The challenges in terms of the extension of the current LWR fleet cycle have made 7uPCX the "billion dollar reactor", with respect to the expected gains for nuclear operators, both over the cycle and over the maintenance periods.

Following the rising needs for multi-physics studies, the Department of Energy decided to restart the safety test reactor TREAT (Transient Reactor Test Facility) (Jensen et al., 2017), with the aim of conducting analytical experiments to understand the physical phenomena in the couplings, and the qualification of accident tolerant fuels. In support of TREAT, the NRAD (Neutron Radiography) facility was brought online as-well (Bess et al., 2014). The reactor can be utilized for improvement of nuclear data knowledge.

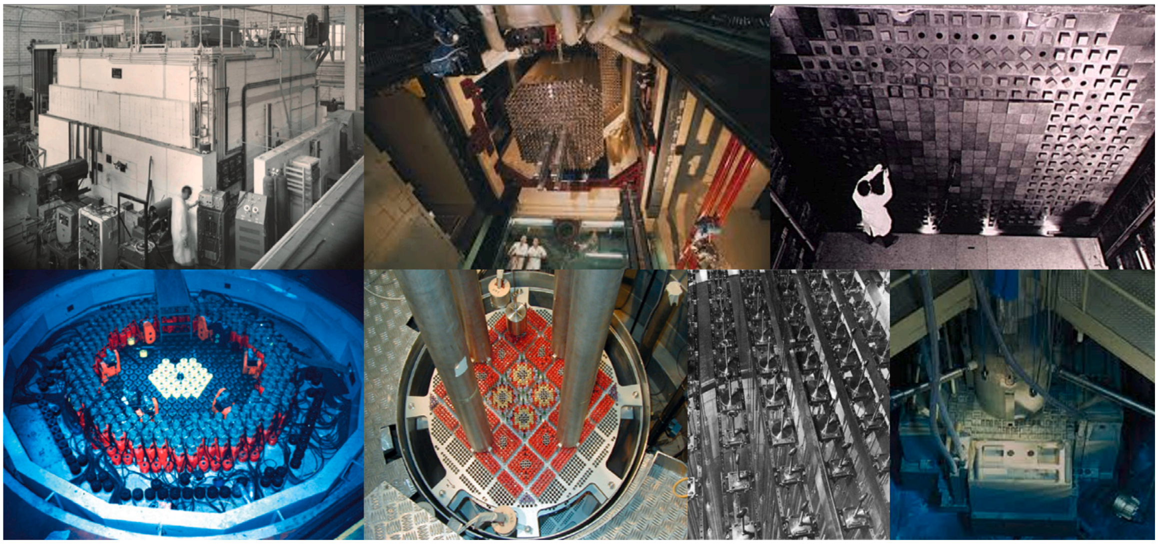


Fig. 5. France zero-energy facilities - ZOÉ, MASURCA, MARIUS, CESAR, EOLE, AQUILON, MINERVE.

An additional zero-power facility, that is utilized for nuclear reactor physics studies is the ZPR located at Rensselaer Polytechnic Institute. The RCF (Reactor Critical Facility). The reactor was constructed in 1964, and was designed to provide support in reactor physics studies, tool validation benchmarks, and material test support (Steiner et al., 2006).

Several facilities in the past were named ZPR at different national laboratories, sometimes they were completely independent facilities, sometimes different experiments in the same core (Annon). The first facility ZPR-1, was utilized for naval reactors studies. The next ZPR-2 (1953) experiment was designed to study the construction of the Savannah River production reactor. The following ZPR-3, was a fast reactor experiment. The reactor design consisted of two rectangular boxes, which were separated in order to ensure reactor shut-down, and brought closer to achieve criticality (Fig. 6). The third ZPR was used to study properties related to fast reactors, such as critical mass, geometry, power distributions. In 1953, the core of the ZPR-1 was modified to provide fast neutrons to exponential assemblies, this experimental facility received the name ZPR-4. ZPR-6, was of a similar design, but larger, to ZPR-3. The reactor provided valuable information to validate computational schemes. The experiment in a series involved slightly different reactor cores (different materials, geometry) with a wide variety of measurements, such as critical configurations, reaction rate distributions, reactivity worth distributions for various materials, control rod worth and kinetics properties.

Following ZPR-6 was the ZPR-9, which was also focusing on fast reactors. However, the seventh ZPR, was a heavy water thorium–uranium fueled designs. This experiment was designed to study the properties of thorium fuels, which attracted interest in the 1960 due to thorium being higher abundance in nature than uranium. The neutron absorption in thorium generates a fissile isotope of uranium - ^{233}U. ZPR-7 was a mock-up for a possible test boiling water reactor, for the purpose of ^{233}U breeding, it provided information on vital core parameters (power distribution, reactivity, kinetic parameters etc.). The final ZPR-9 experiment, which was used to obtain data on possible fast reactor designs, Several candidate nuclear rocket designs were tested, and it served as a detailed engineering mock-up of the Fast Test Reactor. Furthermore, there were also several assemblies investigating gas-cooled fast reactors.

The last reactor in this overview is the final step in the ZPR program, the Zero Power Physics Reactor (ZPPR). The experiments in the ZPPR were used to simulate as closely as possible the behavior of a full size fast reactors and then use the experiment's results to validate and refine the data and computational methods utilized in their designs. During its critical operations from 1969 to 1990, ZPPR generated large quantities of high-quality experimental data that was used to demonstrate and improve confidence in analytical nuclear design codes, methods, and in the nuclear data. In 1990, it was placed in a standby shutdown state by Argonne National Laboratory and was ultimately transferred to the newly established Idaho National Laboratory in 2005.

The future of ZPRs is not clear, there are no planned constructions of new research reactors. However, the research community is highly invested in studies for innovative core designs (Lyons and Kotek, 2018). These research efforts can greatly benefit from utilization of a ZPR facility. The ZEPHYR is an example of a facility that can provide the experimental needs for the future development.

2.4. Russia

Russia, probably, maintained the largest number of operational research reactors — about 60, with about 30 being critical assemblies (ZPRs). Most of the facilities are grouped in — Kurchatov institute centre in Moscow, research institute of atomic reactors in Dimitrovgrad, and IPPE (Institute of Physics and Power Engineering) Obninsk. Most of the research facilities are engaged in support of the naval propulsion program, the older generation of reactors (mainly RBMK-type, from the Russian abbreviation of High Power Channel-type Reactor). Although, most of the ZPRs in Russia were constructed between 1980 and 2002, there are several interesting facilities that can support the needed research for current and advanced systems (Aborina et al., 2002).

The PCF (P Critical Facility) at Kurchatov institute, started operation in 1987. This is a light water moderated facility, with maximal power level of 200 W. It is designed to study UO_2 fuels with enrichment level of up to 6.5%, support advanced LWR designs, nuclear data measurements, and reactor safety. The core structure is flexible and can support a wide range of light-water moderated reactors. Furthermore, the reactor is equipped with a heating system that can vary the moderator temperature from 15 to 90 °C. An additional facility in the Kurchatov institute that has similar operational functionalities as the PCF, is a reactor known as DELTA. The reactor was constructed in 1985, and it is loaded with highly enriched uranium fuel. The reactor is designed to accommodate research related to VVER (in Russian stands for Water-Water Power Reactor) type reactors (I.B.P. Inc, 2009).

VVER reactors are the main export of the Russian power reactor industry. Therefore, most of the research facilities are engaged in studies related to this type of reactors. There are two additional facilities that

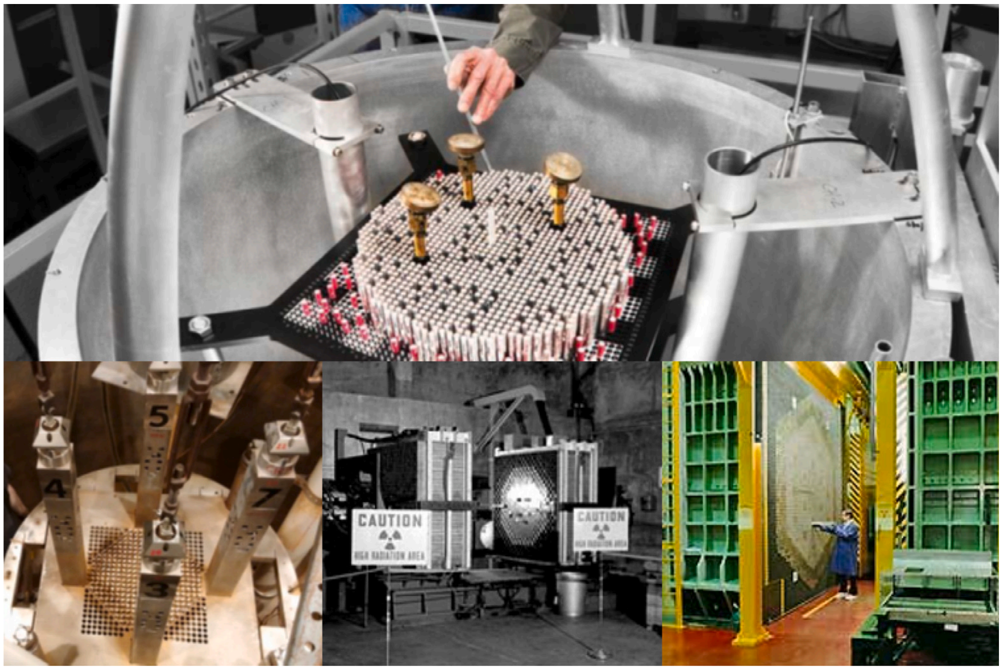


Fig. 6. United States zero-energy facilities - 7uPCX, RCF, ZPR-3, ZPPR.

are located in Kurchatov, the V-1000 and the SK-FIZ. The V-1000 went critical in 1986 (construction started in 1977), with the main purpose to qualify core designs for VVER-400 and -1000. The reactor tank volume is about 70 cubic-meters, which allows almost a 1:1 modeling of reference reactor cores (Aborina et al., 2002). On the other hand, SK-FIZ, which was constructed in 1997, is a small experimental mock-up (core volume of 8.5 cubic-meters) intended for the support of the VVER systems. The facility is utilized for the development of rhodium detectors (Aborina et al., 2002).

For the support of the fast reactor programs, Russia operate two reactors the BFS-1 and −2 (from Russian - Big Physical Facility) (Dulin et al., 2014). The reactors begun operation in 1961 and 1969 respectively. The reactors are utilized for the study of large fast reactors, such as the sodium-cooled fast reactors. The facilities are similar to the MASURCA reactor in France. The reactor is used to study the physics of the Russian fast reactor BN series, and for nuclear data validation for fast neutron spectrum.

Most of the research reactors mentioned here are going to continue their operation in the future. The Kurchatov reactors are to operate until 2029 (Gagariskiy, 2018). The situation in Russia, is such, that the nuclear industry has complete support from the research community to ensure safe operation of current and future reactors. The mentioned ZPR cores are shown in Fig. 7.

2.5. Japan

Japan, like the previously mentioned nations, has many research teams, which operate a wide range of critical facilities (both thermal and fast). However, the Japanese nuclear industry suffered a blow after the Fukushima Daiichi accident, which followed the great tsunami, which hit the coast of Japan in March 2011 (Sehgal, 2012). Currently, seven years after the tsunami only part of the 54 reactor in Japan have returned to normal operation. The accident at Fukushima affected the research reactor community as well, with almost all the research reactors moved into a hibernation state.

There are two main critical facilities utilized in Japan, the Fast Critical Assembly (FCA), and the Tank-type Critical Assembly (TCA). The FCA was constructed in 1967 and it is Japan's only facility for the study of neutronic characteristics of fast reactors. The design of the FCA is similar to that of the American ZPPR, i.e., a split-table configuration (Fig. 8). FCA experimental core is constructed by inserting plate-type or block-type fuels (U and Pu) and core materials (Na, Stainless Steel, etc.) into each of the half-assembly fuel drawers. The facility has very high flexibility of material that can be inserted, similar to the capabilities at the MASURCA reactor in France. However, the reactor is currently not in operation, and its highly enriched uranium fuel and plutonium fuel were sent to the United States as part of an agreement between the countries on reduction of fuel enrichment in civilian facilities (ENERGY.GOV, 2016). The reactor produced, during its operation, a large number of high-quality experimental data for the research community around the world (JAEA, 2018a).

The TCA (Fig. 8) was constructed in 1962, the main purpose of the reactor was to provide benchmark data for validation of computational methods and tools developed for LWR design in Japan. The reactor is similar to the French facility EOLE, with power level of upto 200 W. Currently, the reactor is used for training and in support of Japan's Static Experiment Critical Facility (STACY) (JAEA, 2018b).

The two additional facilities that complete the list of the critical facilities fleet in Japan, are the Kyoto University Critical Assembly (KUCA, Fig. 8) and the Toshiba NCA (Nuclear Critical Assembly). KUCA is a multi-core type critical facility, which was constructed in 1974, which was the main site for reactor physics studies for all universities in Japan. It has three independent cores, namely, two solid moderated cores (A, B cores) and one light water-moderated core (C core). The reactor was utilized to study — thorium fuels, nuclear transmutation, critical experiments of high-enriched uranium fuels with different spectrum, subcriticality measurements, nuclear characteristics of coupled core systems, and more (Sano et al., 2018). The Toshiba NCA is a tank-type, light-water moderated nuclear facility, it reached its first criticality in December of 1963. The reactor is designed for studies related to LWR physics and training of personal (Umano et al., 2014). Unfortunately, the future of these facilities it yet to be determined.

The final facility to mention in the Japanese context is the Very High Temperature Reactor Critical-assembly (VHTRC) (Yasuda et al., 1987). The reactor was constructed by the Japanese Atomic Energy Authority to study phenomena related to high temperature gas cooled reactor. The core was a prismatic graphite-moderated loaded with 2%–4% UO_2 (coated fuel particle, BISO type). The reactor used to provide

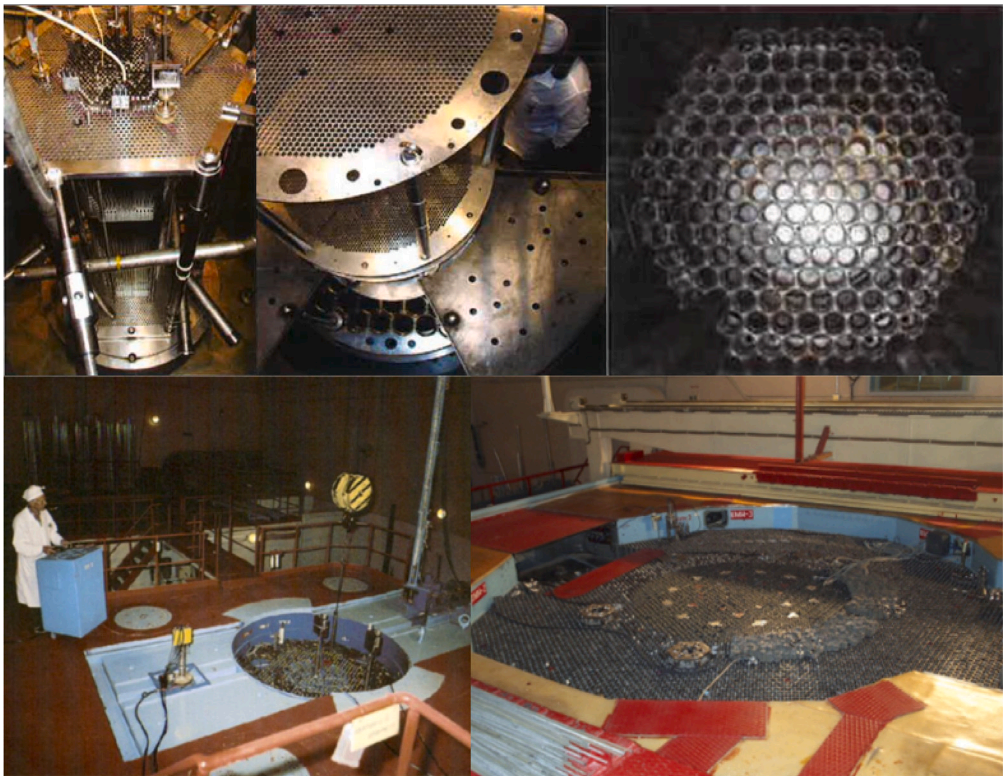


Fig. 7. Russia zero-energy facilities - SK-FIZ, PCF, V-1000, BFS-1 and -2.

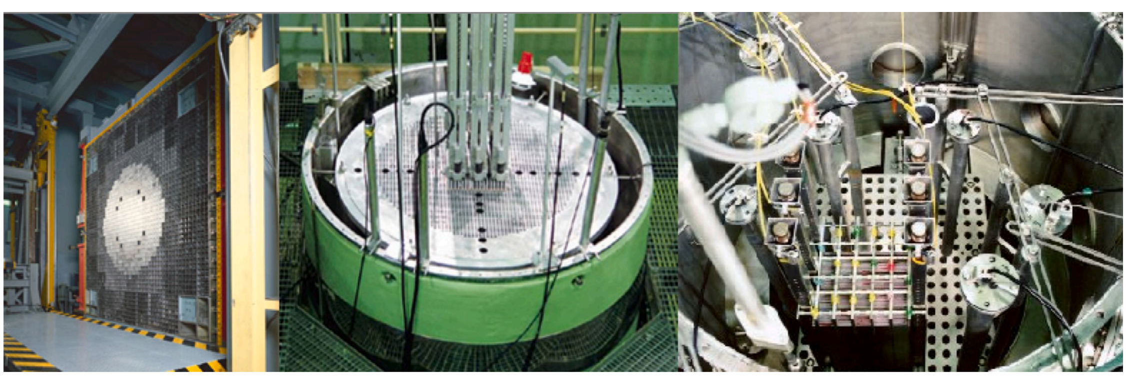


Fig. 8. Japan zero-energy facilities - FCA, TCA, KUCA.

vital information on gas-cooled reactor behavior, such as critical mass, reactivity worth, temperature coefficients, neutron flux distribution and kinetic parameters.

2.6. China

The nuclear energy policy in China is focused on "of-the-shelf" technology purchase, both for power and experimental reactors (mainly from Russia), due to its considerable financial resources. The research reactor fleet in China includes 20 operational facilities (IAEA, 2012), from them four ZPRs attract interest.

Venus-1 (Fig. 9), is China's accelerator-driven sub-critical system, which was constructed in 2005. The core of Venus-1 is a coupled one of a fast neutron zone and a thermal neutron zone. This reactor is mainly utilized for nuclear data validation and criticality safety experiments (Shi et al., 2007). The reactor provided information on highly subcritical experiments (reactor multiplication factor of 0.98), which led to the design of Venus-2, to support the need of additional criticality studies and training. The Venus-2 (Zhu et al., 2018) (Fig. 9) was constructed in 2012 to complete the studies start in Venus-1. The main purpose of Venus-2 is to provide a proof of concept for a thermal accelerator-driven system and training.

DF-IV (Fig. 9) reactor was constructed in 1970 for the purpose of fast neutron technology studies. In was stopped in 2007 for refurbishment phase to support the Chinese experiential fast reactor concept, changing the high-enriched uranium fuel to mixed-oxide fuel (plutonium based) (Nuclear Engineering International, 2017). There is no information available about this reactor at it current status.

Finally, China is strongly invested in studies related to Generation-IV technology (Gen-I.V. International Forum, 2017). The China Lead-based Reactor-0 (CLEAR, Fig. 9) (Wu, 2016), is a small reactor located at the Institute of Nuclear Energy Safety Technology in Heifei. CLEAR-0 is designed for validation of computational tool and nuclear data dedicated to lead-cooled systems, development of instrumentation, and support the following experimental facilities in the CLEAR program. The construction of was finished in 2015, under its conceptual design, the core sits in a pit (Fig. 9), which is covered by a biological shield during operation. Given the flexibility of the materials utilized in the

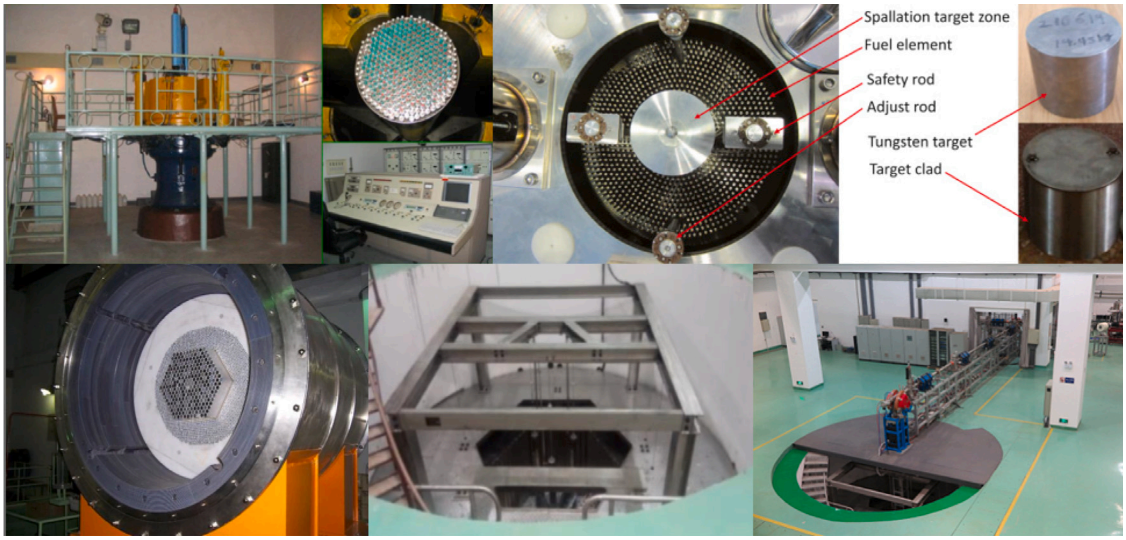


Fig. 9. China zero-energy facilities - DF-IV, VENUS-2, VENUS-1, CLEAR-0.

core, it is possible study different reactor configurations. The reactor can operate in two modes — a critical mode for fast reactor validation and a subcritical mode driven by an accelerator for accelerator-driven systems validation.

Thus the future of experimental facilities in China is possibly bright, as new technologies arrive into the country. This leads to an increase in the need for research facilities for safety studies and concept validation.

2.7. Other notable facilities

Until now, past and present facilities of countries with major nuclear programs were listed. However, outside of those countries there are additional facilities that are of interest to the current overview. The following sections will cover some of those facilities inside and outside of Europe.

2.7.1. Europe

For research targeting LWR technologies there are several facilities currently operating around the continent. The first one is the Vulcan Experimental Nuclear Study (VENUS, Fig. 10), which is located in the SCK·CEN institute in Mol, Belgium (Kochetkov et al., 2012). The reactor was constructed in 1964, and was designed to validate reactor physics calculation codes developed for light-water installation. Currently, the reactor is converted to support research towards Gen-IV type reactors, the reactor is loaded in its fast configuration, known as VENUS-F. The facility aims to study the physics of lead-cooled fast reactors under the project known as GUINEVERE. The execution of the GUINEVERE project consisted of coupling a subcritical fast lead core with a particle accelerator that acts as external neutron source.

Another Belgian facility of interest is the Belgian Reactor 1 (BR-1), it was critical for the first time in May of 1956 (Ruan, 2003) (Fig. 10). BR-1 is an air-cooled reactor with graphite as the moderator. It is a flexible instrument for fundamental research and training. After the start-up period, BR-1 was mainly used for research in reactor and neutron physics. Until after the start-up of BR-2 in 1964, BR-1 was also used for the production of radioisotopes for medical applications. The reactor is utilized for instrumentation calibration, which occurs in the central channel, where the neutron spectrum is Maxwellian (perfectly thermalized). This makes the reactor a perfect reference facility for detector calibration.

Two additional zero-power facilities are located across the boarder in Switzerland. The first is the PROTEUS facility (Fig. 10) at the Paul Scherrer Institute. The reactor started its operation in 1968 and was shutdown in 2011. During its operation the reactor was utilized to study different reactor concepts, such as gas-cooled fast reactors, the tight-pitch, high conversion light water reactor, and the modular high temperature reactor. Before ending its operation, PROTEUS was utilized to study LWR technologies for the Swiss nuclear utilities (Leibundgut, 2017). The reactor used to allow high level of flexibility and mock-up almost any reference system. However, the institute decided to shutdown the facility due to strategic reasons.

Thus the only nuclear operating facility in Switzerland is the CROCUS reactor (Fig. 10), at the École Polytechnique Fédérale de Lausanne, which was constructed in 1983 (Laminard et al., 2016). The zero-power facility is moderated by water, mainly dedicated to teaching radiation and reactor physics. The number of experiments conducted in the reactor, are numerous — investigation of mechanical noise induced by fuel rod vibrations, void fraction determination using neutron noise measurements, qualification of reflector materials for current generation of reactors, and many more. The reactor is widely known in the research community.

In Germany, the Zittau Training and Research Reactor (ZLFR) was build to simulate the concept of a Soviet/Russian VVER. Built by the Zittau Technical University with the assistance of the then Central Institute for Nuclear Research at Rossendorf and commissioned in 1979. The main objective was the need for student training for the growing nuclear industry. In addition to the ZLFR, many German institutes operated the Siemens training reactors (SUR) 100 (Siemens-Unterrichts-Reaktor 100). The reactors were 100 mW in power, consisting of a homogeneous core enclosed in a graphite reflector. This reactor type had a strong emphasis on training of university students. In the context of advanced thermal reactors, it worth noting the high temperature pebble bed reactor (KATHER), the facility provided a substantial amount of insight into pebble bed reactors. The data was used to develop models in codes DIFF-2D and CITATION for analysis of pebble bed geometries (Pohlen, 1982; Bredberg et al., 2020).

Finally, the last European light-water facility that will be reviewed here is the LR-0 reactor (Fig. 10) at the Nuclear Research Institute in Řež, Czech Republic, commissioned in 1972. The reactor is designed for VVER core studies. It is very close in principle to the installation of the Kurchatov Institute - V-1000, mentioned in Section 2.4. The main programs concern the experimental validation of computer codes and nuclear data. The reactor is constructed in such a way, that it allows flexible zone reconfiguration, flexible reactor operation, standard and special support plates for mock-up experiments, and a wide range of measurement technology (Košťál et al., 2013).

In support of fast reactor research, excepting of VENUS-F, there is only a single facility currently operating in Europe. The TAPIRO

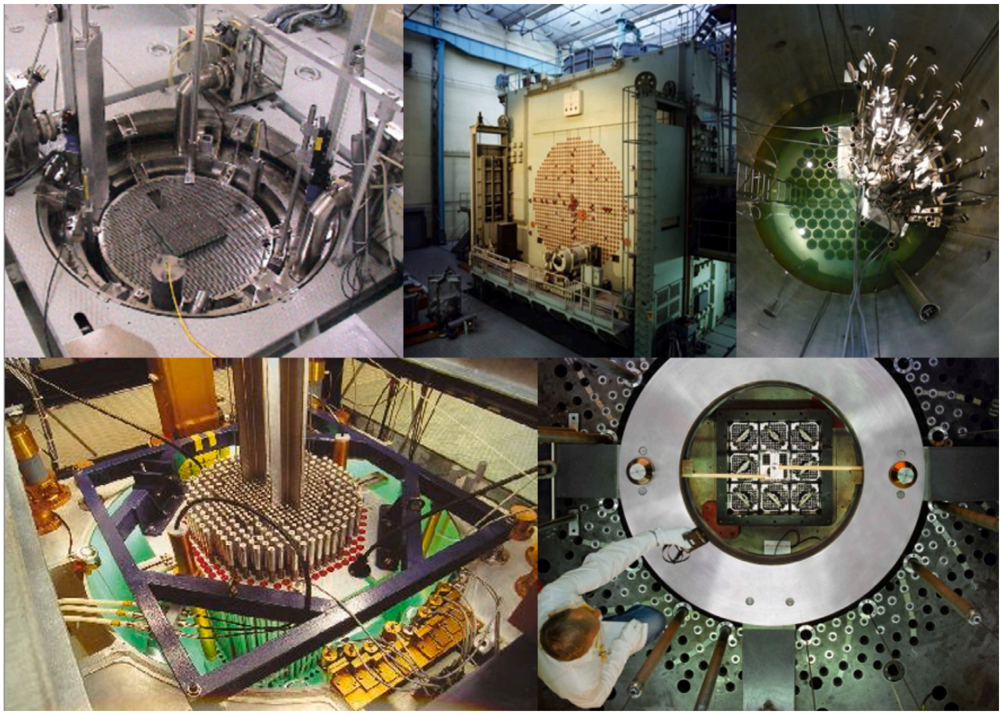


Fig. 10. European zero-energy facilities - VENUS, BR-1, LR-0, CROCUS, PROTEUS.

- TAratura PIla Rapida potenza zerO (Fast Pile Calibration at Zero Power, Fig. 11) (Esposito et al., 2007) at the Italian National Agency for New Technologies, Energy and Sustainable Economic Development facility near Rome, Italy. The reactor was built in 1971 to support an experimental program on fast reactors. The neutron spectrum in the reactor's core center is its unique feature, as this is almost a pure fission spectrum. The reactor enables validation of calculation codes for Gen-IV reactors design, fast neutrons damage studies, nuclear data testing benchmarks, and qualification of chains of innovative detectors.

However, although VENUS and TAPIRO are able to support research related to fast reactor, there is no facility outside of Russia's BFS reactor (operational), that can provide support for large core studies. This was not always the case when the Schnellen NullEnergie-Anlage Karlsruhe (SNEAK, Fig. 11) (Helm et al., 1984) facility was operational. The reactor constructed in 1966, and was shutdown in 1985, the reactor was almost a twin to the French MASURCA. It allowed for a large flexibility of materials (fuel, coolant and structural) and core configurations. In recent years, experimental programs related to CDA (Core Disruptive Accident), performed in the SNEAK, were under investigation in the framework of the ZEPHYR project (Margulis et al., 2017a,b, 2018b, 2019a). Other experiments for determination of critical size, power distribution, reactivity worth of materials, were performed. Currently, the fuel from the decommissioned facility is located at the MASURCA stock-pile in France.

2.7.2. Outside of Europe

The review of Russia, China, Japan, and the United States, was presented in separate sections. This section will deal with the rest of the world. Although, there was a large number of facilities that operated around the world that did not enter this review. Here only facilities that are still in exploitation will be mentioned.

Canada, a country that relies on nuclear energy as one of the main sources of electricity, runs a fleet of CANada Deuterium Uranium (CANDU) reactors. This fleet requires continuous support by research facilities for safety and modernization studies. The ZED (Zero Energy Deuterium) experiment (Horner, 2017), which was built in 1960 and replaced Canada's first nuclear pile known as ZEEP, provides this support. The ZEEP was utilized for the generation of plutonium in Canada's nuclear weapons program. The ZED-2 is a flexible reactor when it comes to coolant, and it is possible to change between light- and heavy-water, and CO_2. The facility conducted many experiments for qualification of different fuel types (oxide, metallic, silicide and carbide) and core designs. Today ZED-2 supports the development of reactors and advanced fuel cycles. As well as, the development, characterization, and calibration of in-core and ex-core flux detectors for use in power reactors.

Although Brazil is not a nuclear "super power" (in comparison to Russia, USA, UK, France or China), its two power station (Angra 1 and 2) generate about 3% of its electricity demand World Nuclear Association (2018). There are additional planned reactors to be constructed in Brazil. However, no clear construction dates are set yet. The nuclear industry in Brazil is being supported by a single research reactor operated by the Instituto de Pesquisas Energéticas e Nucleares (Institute of Energy and Nuclear Research), which is located at the University of Sao Paulo. The MB-01 is a small light water moderated zero-power facility designed as part of naval propulsion program of the Brazilian navy. The reactors core is a rectangular grid of 27 × 29 with enriched UO_2 to 4.5%, similar to some configurations tested in the French EOLE reactor. The facility is utilized for the development of instrumentation, nuclear data and kinetic parameters estimation (Bitelli et al., 2003). Furthermore, the reactor is currently utilized for as a reference core for the design of a new Brazilian nuclear multi-purpose reactor (Mai and Siqueira, 2011).

2.8. Summary

The presented overview of the past and present zero-power facilities around the world is intended to demonstrate their importance to nuclear research. As the overview presented the zero-power facilities that were utilized in various ways to support the nuclear industry's development, providing vital information for safe operation, new concept and instrumentation development, nuclear data needs and much more. The overview showed that the current state of the ZPRs around the world is in decline, with most of the iconic facilities being shut due to

Fig. 11. European zero-energy facilities - SNEAK, TAPIRO.

different reasons. However, if the world community wishes to reach the global warming targets set by the United Nations in the Paris agreement of 2015 (United Nations Framework Convention on Climate Change, 2015), nuclear power must be considered seriously.

The ZPRs future is well dependent on the future of the nuclear power growth, and vice versa (Nuclear Energy Agency, 2023). History showed that each time a new concept was proposed, a zero-power experiment was the first step towards the final reactor design. Such need is not existent any more, there is no need for a ZPR that will be dedicated to a certain reactor concept as the computational tools available today are more advanced than in previous decades. However, high level of experimental representativity performed in the future ZPRs would be a key for supporting the future of nuclear energy expansion.

3. Design consideration for the future facility

When it comes for the development of innovative technology in the nuclear sector, the first step was the design of a zero-power experimental facility. The success of the zero-power experiment led to rapid expansion of the programmes it targeted (as can be very well be seen with the light water reactors and to extent with the sodium-cooled fast reactors). The cost benefit in the past was such that ZPRs were constructed to test even the smallest ideas, as was demonstrated in the previous section. However, those facilities were constructed under the portraying eyes of governmental agencies, with in many cases open checks to do so. This is not the case any more, in many cases the first facilities to pay the price of budget cuts are the physics reactors and focus is made on constructing materials test reactors (e.g., Jules Horowitz Reactor and Versatile Test Reactor). Reactor physics and reactor design is taking a back seat in the nuclear world with clear shift into the material field.

The question of a physics oriented facility rose in many forums and in 2023 reached that centre stage of the Nuclear Energy Agency, which hosted a topical meeting under the title — The demise of zero power reactors: From concern to action (Nuclear Energy Agency, 2023). The goal was understand whether a new ZPR is needed. The gathered evidence showed that there is a clear support for a new facility. However, there was no clear consensus on the design, with most of the votes going towards a light water-based design. The presentation of one such facility is the objective of the next section, but in this section, the underlying consideration how to design such a facility is made.

Although it is not clear how the future facility will look like, one thing is clear it has to provide the best experimental platform. In mechanical engineering dimensionless analysis is used to provide a way to plan and carry out experiments, and enables one to scale up results from model to prototype. In reactor physics the equivalence can be seen in the representativity approach (Orlov, 1980). Considering this method one can still design an experiment that would be useful for reducing the uncertainties related to computational models without the need of a ZPR demonstrator (similar to the Russin V-1000), as was done since the dawn of the nuclear industry.

The basis for the representativity approach commences with the definition of the sensitivity coefficient. Generally, the sensitivity coefficient S of a response parameter R to the perturbed parameter α is defined as Williams (1986) -

$$S = \frac{\Delta R}{R} \Big/ \frac{\Delta \alpha}{\alpha} \tag{1}$$

In this case, the response parameter is the core effective multiplication factor (k_{eff}), and the perturbed parameters are the macroscopic cross-sections (Σ). The estimation of sensitivity coefficients of k_{eff} is performed according to a formula derived from the classical standard perturbation theory (Williams, 1986) -

$$S_{k,\Sigma_{r,n}} = -\frac{\Sigma_{r,n}}{k_{\mathrm{eff}}} \frac{\left\langle \phi^{+}, \left(\frac{\partial M}{\partial \Sigma_{r,n}} - \frac{1}{k_{\mathrm{eff}}} \frac{\partial F}{\partial \Sigma_{r,n}} \right) \phi \right\rangle}{\left\langle \phi^{+}, \frac{F}{k_{\mathrm{eff}}^{2}} \phi \right\rangle} \tag{2}$$

Where M and F are the loss and production operators respectively, ϕ being the angular neutron flux with ϕ^{+} is the adjoint operator, $\Sigma_{r,n}$ is the incidental macroscopic cross-section of isotope n and response r and $S_{k,\Sigma_{r,n}}$ is the energy dependent sensitivity vector. In a sense, the $\partial M/\partial \Sigma_{n,r}$ and $\partial F/\partial \Sigma_{n,r}$ expressions in Eq. (2) represent functions of scattering plus capture and fission cross-section data respectively. The evaluation of Eq. (2) is essentially the integration of the forward and adjoint fluxes and the cross-sections over the entire space and angle.

After obtaining the sensitivity vectors (as a function of energy), the propagated uncertainties are calculated from the cross-section data using covariance matrices available with most of the major nuclear data evaluation. The variance for the k_{eff} is determined as Ronen (1988)-

$$\sigma^{2}_{k_{n,r}} = S_{k,\Sigma_{n,r}} C_{\Sigma_{n,r},\Sigma_{n',r'}} S^{t}_{k,\Sigma_{n',r'}} \tag{3}$$

where $C_{\Sigma_{n,r},\Sigma_{n',r'}}$ is the covariance matrix of size $S{\times}S$ corresponding to isotope n and reaction r, the two different subscripts (for isotopes n and n' and for the reactions r and r') are made to ensure that both diagonal and off-diagonal covariance data is included. The typical treatment of uncertainties contains mainly energy-correlated responses for different isotopes (e.g., for ^{238}U capture-capture cross-sections). However, typically in the covariance data, there are "cross-correlated" (or " non block-diagonal") responses linked to the data assimilation process used. Utilization of these matrices allows for more realistic final results (i.e., the off-diagonal correlations represent a physical constraint in the process of differential measurement analysis, such as a total cross section is the sum of its partials, hence correlating the capture to the fission).

Finally, the reactivity variation sensitivity, which is obtained directly from the definition of reactivity $\Delta\rho = \rho_1 - \rho_2 = 1/k_1 - 1/k_2$ (Tommasi et al., 2010), is given as -

$$S(\Delta\rho, \mu) = \frac{1}{\Delta\rho}\left(\frac{S(k_2, \mu)}{k_2} - \frac{S(k_1, \mu)}{k_1}\right), \quad (4)$$

where ρ stands for reactivity, k is the effective multiplication factor of the two different core configurations: 1 (the reference one) and 2 (the perturbed one), and μ means the restriction of the operator to the terms involving μ. Note that when the reactivity variation is too small, i.e., $\Delta\rho \to 0$, Eq. (4) is no longer valid since the sensitivity of the reactivity variation to a perturbation in cross-section i diverges, i.e., $\lim_{\Delta\rho\to 0} \Delta S(\Delta\rho, \Sigma_i) = \infty$. This result has no physical meaning and hence Eq. (4) is valid as long as the reactivity variation is larger than some lower threshold.

The representativity method is used to extract a quantitative relationship between a particular integral response of an experimental mock-up and the same response in a power reactor that want to be designed. It is based on the similarity of the sensitivity profiles of both integral responses. The representativity is linked to the definition of a correlation coefficient (called by analogy representativity coefficient r_{RE}), defined, as far as could be determined, for the first time by Orlov (1980), as -

$$r_{\mathrm{RE}} = \frac{\boldsymbol{S}_{\boldsymbol{R}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{E}}}{\sqrt{\boldsymbol{S}_{\boldsymbol{R}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{R}}} \cdot \sqrt{\boldsymbol{S}_{\boldsymbol{E}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{E}}}}, \quad (5)$$

where:

- The subscripts $\boldsymbol{E}$ and $\boldsymbol{R}$ correspond to the experimental mock-up and reference power systems, respectively.
- $\boldsymbol{S}$ is the sensitivity vector of the integral quantity to nuclear data in the two systems. In the case of reactivity variation representativity, the vector is defined by Eq. (4) or Eq. (2) for multiplication factor.
- $\boldsymbol{V}$ is the variance–covariance matrix between nuclear data.
- $\boldsymbol{S}_{\boldsymbol{E}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{E}}$ and $\boldsymbol{S}_{\boldsymbol{R}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{R}}$ represent respectively the priori variance $\boldsymbol{E}$ and $\boldsymbol{R}$ due to nuclear data uncertainties, propagated by the classical "sandwich" rule (Ronen, 1988).

The numerator in Eq. (5) represents formally the covariance between the experiment and the reactor response, while the denominator is simply the product of the square-root of the variance of $\boldsymbol{E}$ and the square-root of the variance $\boldsymbol{R}$. The larger the magnitude of r_{RE}, the higher the information transferred from the mock-up test to the target systems designs. When the similarity of $\boldsymbol{S}_{\boldsymbol{E}}$ and $\boldsymbol{S}_{\boldsymbol{R}}$ increases, the value of the representativity factor r_{RE} reaches an optimum value of unity, which indicates fully correlated neutron systems, with respect to the variance–covariance matrix used.

Finally, the representativity approach also enables the prediction of a *posteriori* reduction in the reactor response uncertainty, ϵ_R^*, after having injected the experimental information into a complete Bayesian assimilation process ("adjustment") of multigroup cross-sections. The reduction factor is given by the expression in Eq. (6) (Orlov, 1980).

$$\begin{aligned}(\epsilon_R^*)^2 &= (\epsilon_R)^2 \cdot \left(1 - \frac{r_{RE}^2}{1 + \delta E^2/(\boldsymbol{S}_{\boldsymbol{E}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{E}})}\right) \\ &= (\epsilon_R)^2 \cdot \left(1 - \frac{r_{RE}^2}{1 + \delta E^2/\epsilon_E^2}\right) = (\epsilon_R)^2 \cdot (1 - \omega r_{RE}^2),\end{aligned} \quad (6)$$

where δE^2 is the experimental uncertainty of the response E, and $\omega = (1 + \delta E^2/\epsilon_E^2)^{-1}$ is what can be called the "experimental weighting" factor or the "experimental importance", which represents the amount of transferable precision (i.e., experimental uncertainty) of the integral parameter versus the propagated uncertainty from nuclear data. In the limit of $r_{RE} = 1$ and the ratio $\delta E^2/\epsilon_E^2 \to 0$, the reduction factor, ϵ_R^*/ϵ_R, can vanish.

The C/E bias from the experimental parameter can be transposed to the target parameter calculation bias $\hat{\boldsymbol{R}}$-$\boldsymbol{R}_0$ (a posteriori calculation – prior calculated value) can be written as follows (Orlov, 1980) –

$$\frac{\hat{\mathbf{R}} - \mathbf{R}_0}{\mathbf{R}_0} = \alpha \frac{E - C}{C} \quad (7)$$

where the transposition factor α is expressed as:

$$\alpha = \frac{cov(\mathbf{R}_0, E)}{\delta E^2 + \boldsymbol{S}_{\boldsymbol{E}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{E}}} = \frac{r_{\mathrm{RE}}}{1 + \delta E^2/\boldsymbol{S}_{\boldsymbol{E}}^{t} \cdot \boldsymbol{V} \cdot \boldsymbol{S}_{\boldsymbol{E}}} \cdot \frac{\epsilon_r}{\epsilon_E} = \omega \cdot r_{\mathrm{RE}} \cdot \frac{\epsilon_r}{\epsilon_E} \quad (8)$$

Utilizing the representativity approach in the experiment design can ensure that the data generated in the experiment will be relevant to the particular system under investigation, and the obtained measurements can be used to reduce uncertainties in the fed nuclear data. Reducing uncertainness in nuclear data is vital to produce best estimates of reactor neutronic related quantities, such as criticality, reactivity coefficients, delayed neutron fraction and flux distribution. This in tern will increase the confidence of regulators when upgrades or new concept placed on their tables. This approach is widely utilized by the experimental physics groups within CEA. The EOLE reactor was the best example of a facility where the experiment were designed in such a way that the C/E deviation is directly the calculation error that would have been obtained in the reference application (in the case of EOLE a PWR or BWR), i.e. $r = 1$.

4. The future facility

The experimental programs performed in the various ZPRs were very fruitful for understanding LWR and Fast Reactor physical phenomena, but today these facilities are facing closure or are becoming dedicated for a single purpose. During the last seven decades ZPR facilities helped the nuclear industry to improve the knowledge of reactor physics parameters (such as k_{eff} and β_{eff}), measurements for nuclear data evaluation, reduction of their underlying uncertainties, and generating data for tool validation (distribution of reaction rates, detailed power maps, etc.). Those activities contributed to the development and improvement of nuclear power plants (high burnup, fuel design and qualification, new core design), life extension (dosimetry on vessel and internal materials examination), safety (absorbing material characterization, reactivity coefficients, importance function estimations). Lastly, these facilities play key roles in the training of engineers, scientists and operators (Pavel et al., 2023). Thus, there is a strong drive from the various communities (academic, scientific, and industrial) to see new ZPRs to be constructed in the world.

However, there is a debate between the communities on what kind of a facility should be constructed. As identified by the NEA there is a slight disarrangement between the industry and the scientific/academic communities. The first would like to see construction of dedicated ZPR to support the design of their technologies. On the other hand, the scientific/academic would like to see a flexible generic facility, which can contribute to the future data needs for many decades with core adaptability to any system. Nevertheless, all agree on a list of capabilities that need to be implemented in the new facility, such as multi-physics to supports MSRs, HTGRs and some of the SMR prototypes, high temperature capabilities to heat various core parts for integral Doppler measurements, qualification of fuels, steady-state and transients measurements, spectral flexibility and others.

Multi-physics will play a pivotal role in the case for a ZPR construction to be approved. Due to the nature of the reactor the operational conditions are standard pressure and temperature. As was shown in Section 2, some past experimental facilities incorporated heating, but in many it was no part of the design. However, the impact of temperature on the fuel, coolant and moderator neutronic behavior is profound, through their corresponding nuclear data. Therefore, incorporating

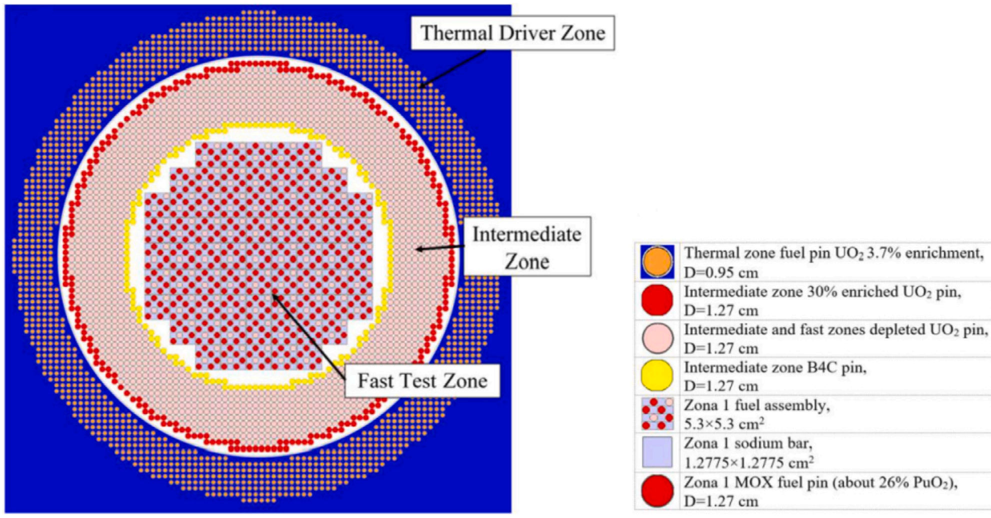


Fig. 12. ZEPHYR coupled fast/thermal reference core layout.

thermal effects into experiment design will improve the quality of the data obtained from the experimental facility and would be relevant for the reference systems at their full power conditions. Including temperature effects will help to increase the representativity factor (Margulis et al., 2018a, 2019b), an example will be shown later in this section.

An ambitions project that could potentially have most of the answers to the needs of the various groups and the capabilities sought for, is the Zero-power Experimental PHYsics Reactor (ZEPHYR) proposed by the CEA (Blaise et al., 2016). In parallel with R& D studies on SFR, (mainly the ASTRID technological demonstrator), a renewed interest emerged in fast-thermal coupled cores, a topic which was already investigated in the 70's (Ros et al., 2016). These configurations consist of getting fast-spectrum neutronic characteristics in a reduced central zone, also called the "measurement zone" or "experimental zone", while criticality is achieved using a thermal zone". As this latter contains most of the fission, such configurations allow a substantial reduction of fissile materials in the core, as less material is needed to achieve criticality in thermal/reflected system, and higher flexibility due to the thermal spectrum kinetics parameters.

The main issue is then to provide a proper fast spectrum in the center by using an adapted spectral conversion zone, surrounded by the thermal spectrum zone. This work has been extensively revisited and major coupled cores design characteristics and conclusions can be found in Ros et al. (2017b). Hence, within the new awaited innovative feature of the ZEPHYR project, coupled core physics is one of the most promising outcomes for performing neutron physics and for improve both codes and nuclear data.

Fig. 12 presents the proposed coupled configuration for the ZEPHYR core. The thermal part of a fast/thermal coupled core can work either in a booster or coupling region and can then amplify local effects in the fast zone, suitable for nuclear data improvement. Another attribute of a fast/thermal coupled core is the improvement of the representativity of the spectrum and reactivity effects in the fast zone with respect to infinite fast lattices, thus reducing the number of fuel materials while guaranteeing the correct neutron spectrum on a large area (Blaise et al., 2016). But, this is not limited only for fast reactor applications, experiments considering representative experimental zones to support molten-salt and gas-cooled reactors can also benefit. The last can be achieved by loading the center of the core with the appropriate representative materials.

The spectral flexibility of the ZEPHYR core was demonstrated by Ros et al. (2017a). The goal of the study was to maximize the reaction rates and reactivity effects in the epi-thermal energy range, which achieved by maximizing the following ratios -

$$\frac{\int_{10keV}^{10eV}\phi^{+}(E)\Sigma(E)\phi(E)dE}{\int_{20MeV}^{0eV}\phi^{+}(E)\Sigma(E)\phi(E)dE} \qquad (9)$$

$$\frac{\Delta\rho_{10eV-10keV}}{\Delta\rho_{total}} \qquad (10)$$

where ϕ^{+} and ϕ are adjoint and direct flux, respectively, Σ is the macroscopic cross-section, and $\Delta\rho$ is the reactivity variation. In order to provide a systematic adoption of the ZEPHYR core to the target spectrum range several reactions were considered. For the entire energy domain fission of ^{239}Pu and ^{235}U was considered, absorption of ^{103}Rh was chosen to represent energy range 0-1eV, capture of ^{238}U was used to evaluate the sensitivity in the range of 1–10 eV, and finally, absorption of ^{56}Fe is used to estimate the behavior in the epi-thermal range.

The principle of the coupled core deign follows three main steps. First the conversion zone is adapted, this zone contains high enriched uranium in order to ensure transformation of thermal neutrons to fast neutrons. Then a transition area is modified, to achieve an epi-thermal flux in the central zone a material with good elastic and inelastic scattering properties is loaded (such as light and heavy water, graphite, steel etc.). Finally, an absorption layer is introduced that will ensure total removal of neutrons under 10eV, which have the strongest influence on reaction rates. An R-Z representation of the described optimization problem is shown in Fig. 13.

Combining the spectral adaptation method and the representativity method shown in Section 3, one can adjust the spectrum in the centre of the core to target specific needs for specific reactor types. An example for an enhanced epithermal zone is shown in Fig. 14, emphesising the 100 keV enegrgy peak As well as Doppler measurements at relevant spectra with samples at various temperatures will allow experiments to tackle uncertainties related to data which is traditionally extrapolated. Similar, criticality/safety assessment relate to burn-up credit and dynamical measurements (unlike the traditional quasi-static approach in ZPRs) can be performed at the proper relevant conditions.

An example of an innovative experimental programme that was under investigation to be implemented in the ZEPHYR facility was related to the neutronics of severe accident situations in fast reactors (sodium and lead). As severe core accidents, with fuel meltdown, occur at high temperatures (about 2800 °C), and the experiments are conducted at normal conditions (i.e. 20 °C), an innovative methodology must be developed to ensure proper representativity between hypothetical case and the critical facility. In order to achieve this target an optimization methodology was developed to translate temperature into material

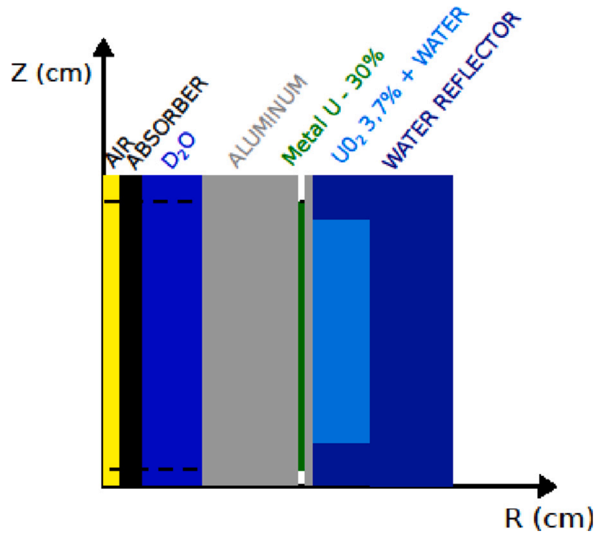


Fig. 13. ZEPHYR coupled core R-Z layout for epi-thermal enhanced flux.

Table 1
Design parameters for the CFV-V0 core.

Core design parameter	CFV-V0
Nominal thermal power [MWth]	1500
Primary coolant	Sodium
Inner core geometrical dimensions	
Lower Blanket height [cm]	30
Lower Fissile zone height [cm]	25
Intermediate Blanket height [cm]	25
Upper Fissile zone height [cm]	35
Inner Core radius [cm]	133.5
Assembly pitch [cm]	17.5
Outer fissile zone geometrical dimensions	
Lower Blanket height [cm]	30
Fissile zone height [cm]	100
Outer fissile zone radius [cm]	162.6
Assembly pitch [cm]	17.5
Fissile zones PuO_2 enrichment (inner/outer) 22.8/22.8	
Effective delayed neutron fraction (β_{eff}) [pcm]	364
Void effective reactivity effect ($) core at equilibrium	−1.2

Table 2
Design parameters for the ELSY core.

Core design parameter	ELSY
Nominal thermal power [MWth]	1500
Primary coolant	Lead
Inner core geometrical dimensions	
Lower Fuel Expansion zone [cm]	96
Fissile zone [cm]	120
Upper Fuel Expansion zone [cm]	24
Core Radius [cm]	290
Assembly pitch [cm]	17.5
Fissile zones PuO_2 enrichment (inner/intermediate/outer)	14.5/15.5/18.5
Effective delayed neutron fraction (β_{eff}) [pcm]	346

Table 3
Comparison of representativity factor for the new core layout.

Reference system	ASTRID[a]	ELSY[a]
ZEPHYR coupled	0.6051	0.5905
ZEPHYR-S	0.9609	–
ZEPHYR-L	–	0.9556

[a] Reference power systems at 900 °C.

density effects. As the sensitivity vectors obtained from calculations in Eq. (5) relate to the macroscopic cross-section, it is possible to manipulate those through alteration in temperature (which affects the microscopic cross-section) or the material density.

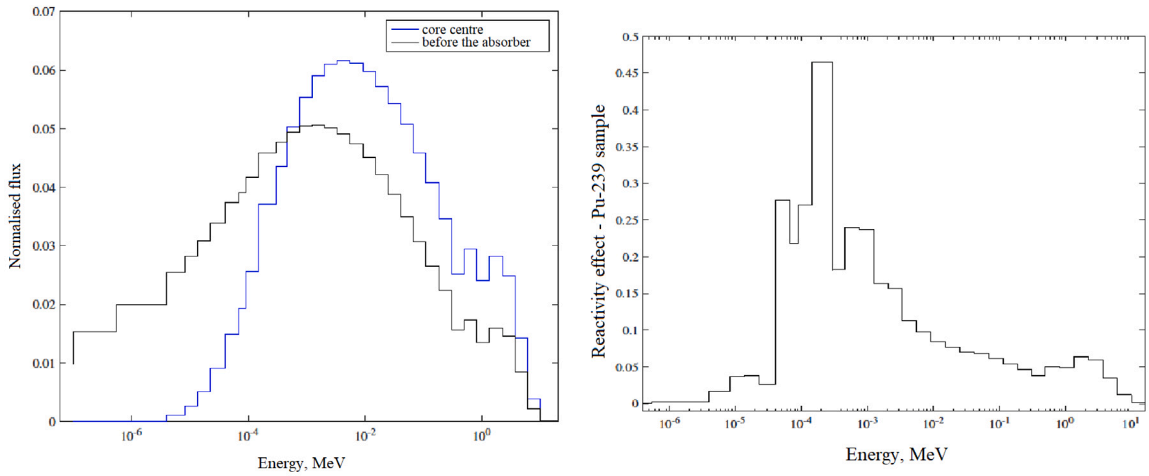
The study was conducted on two cores the sodium-cooled ASTRID (Fig. 15) design and the prototypic European Lead-cooled SYstem (ELSY) (Fig. 16), the main characteristics of the two cores is summarized in Tables 1 and 2, respectively. For each reactor two degraded configurations were considered, as shown in Fig. 17 for the ASTRID core and Fig. 18 for the ELSY core. The affected zone includes the central fuel assembly and an adjacent ring or two, depending on the degraded configuration. It is found that when a single fuel assembly in the core's center undergoes degradation, the impact on the core's multiplication factor is negligible and cannot be detected with ex-core instrumentation. This is an important issue for this type of core in general, and for designing in-core and ex-core monitoring systems in particular. Therefore, a larger affected area is considered. However, in case of the ASTRID degradation sequences (Fig. 17), when two molten zones are formed in the upper and lower fissile zones, the impact on the sensitivity profiles is not profound as well. Therefore, an even larger zone is considered, with degraded fuel assemblies in the two adjacent rings around the center.

In the context of the coupled ZEPHYR thermal-fast core, as shown in Fig. 12, it was impossible to reach a high level of representativity. This was due to the dominance of the thermal zone on the calculations. Although, for small samples it is still possible to utilize the coupled core concept, as the reactivity perturbation is small, which leads to the cancellation of the sensitivity coefficients of the thermal region. However, this cannot be said for large perturbations on the assembly level size. Therefore, two alternative full fast core designs are proposed for the new fast ZEPHYR configuration, i.e., ZEPHYR-S(odium) and ZEPHYR-L(ead), which will serve as the reference configuration for studies related to reactivity variations on the full core level. The two alternative configurations are shown in Fig. 19. It should be noted that these two alternative core designs were a preliminary pre-conceptual design and further research is required for full evaluation of these concepts. A comparison between the main parameters of the alternative designs is presented in Table 3. The S core is slightly larger with respect to the L core and both cores are larger than the entire coupled core configuration.

Two sets of degraded configurations were examined in the ASTRID and ELSY cores. The severe accidents scenarios were examined in the core centre affecting one or two rings of fuel assembles, as stated previously. The outlined configurations are shown in Figs. 20 and 21. The optimization process was set to identify the degraded configuration that would be loaded into the ZEPHYR-S or -L core to achieve a representativity factor of 0.85, where the search for a plutonium content in the degraded zone of the ZEPHYR assembly is the optimization parameter. Several temperature variations are considered, from 900 °C to 1000 °C, 2000 °C and 3000 °C.

Considering the reactivity change between the first two ASTRID configurations (shown in Figs. 20(a) and 19(b), and summarized in Table 4). The results of the optimization are based on the utilization of the CEA covariance matrix V01. The optimization results of "Perfect" fuel (this term refer to optimization result fuel not the available actual fuel in the CEA stockpile) show that it is possible to achieve the required minimal value of $r_{RE} = 0.85$ by adjusting the PuO_2 content in the degrade zone. The impact of the temperature on the results is clear, where the representativity slightly drops as the temperature of the degraded zone increases. Nevertheless, the representativity for all the temperature variations satisfies the required minimal value of $r_{RE} = 0.85$. Furthermore, the "Perfect" fuel was then replaced by the available fuel in the MASURCA stock-pile, and in this case as well the results showed that it is possible to reach the minimal level of $r_{RE} = 0.85$.

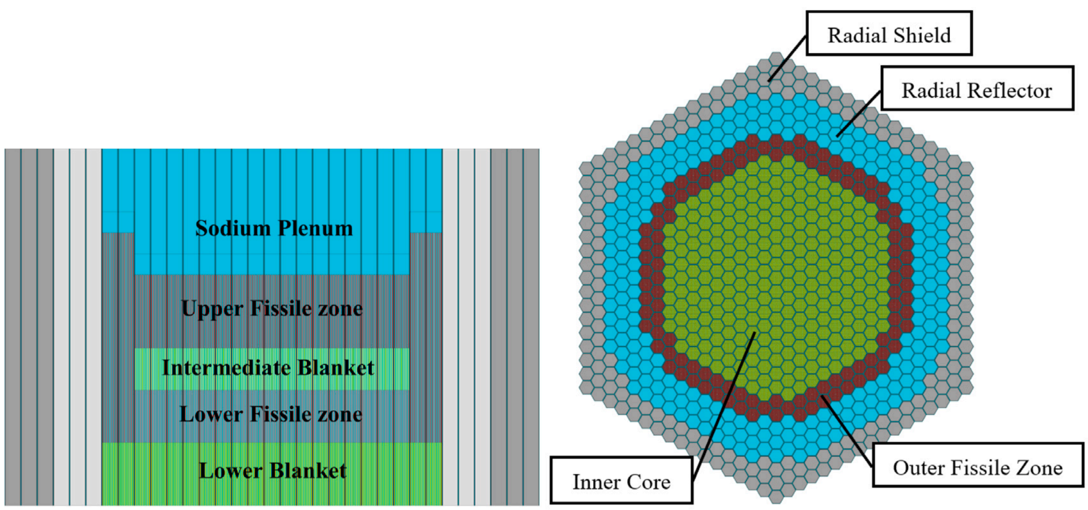
The second degraded configuration under investigation considers degradation of each one of the fissile zones of the ASTRID core on its



(a) Normalized flux spectrum at the center of the core and before the absorber layer

(b) $\Delta\rho$ energy breakdown of a ^{239}Pu sample

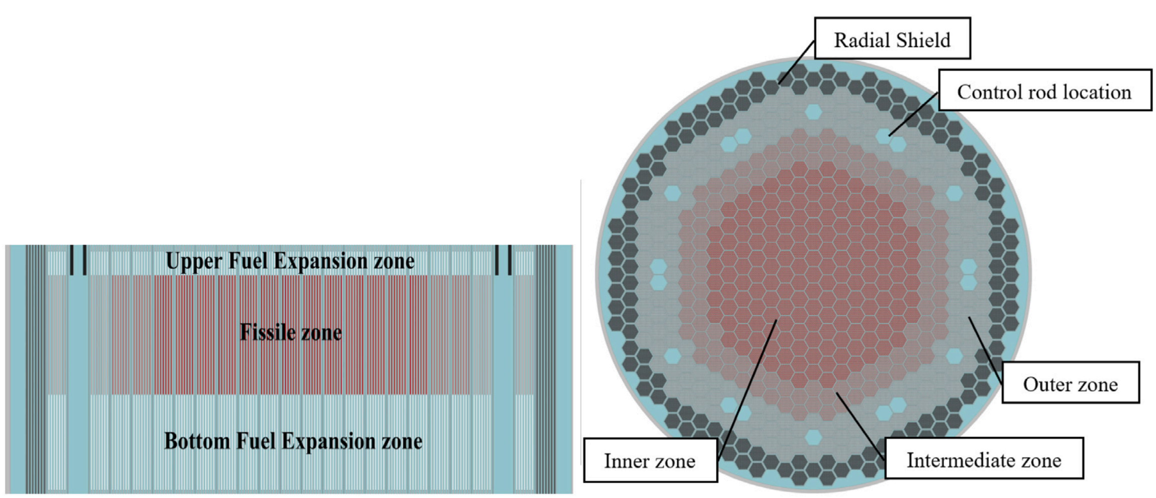
Fig. 14. Epi-thermal core characteristics (Ros et al., 2017a).



(a) Axial cut.

(b) Radial cut.

Fig. 15. ASTRID CFV-V0 core layout.



(a) Axial cut.

(b) Radial cut.

Fig. 16. ELSY core layout.

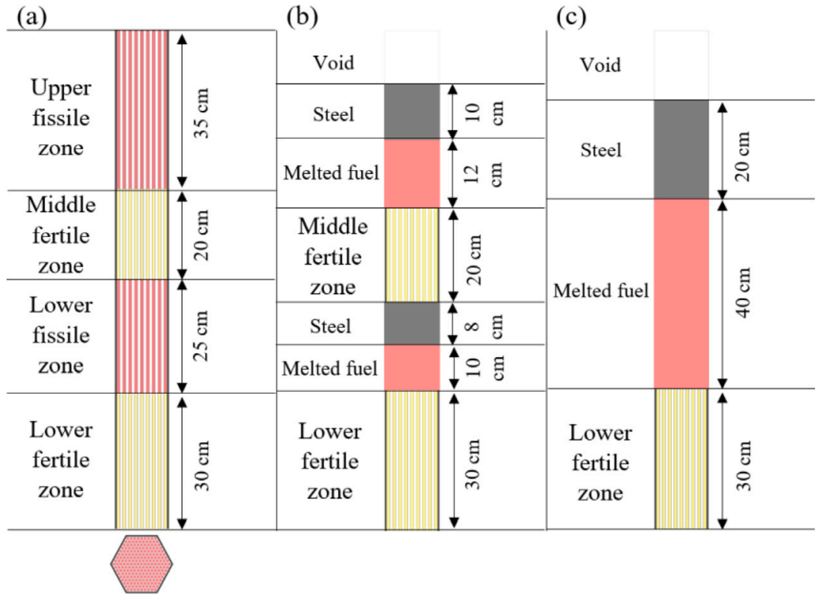


Fig. 17. ASTRID SCA sequences under investigation, (**a**) voided reference configuration (**b**) two zone degraded configuration. (**c**) single degraded configuration.

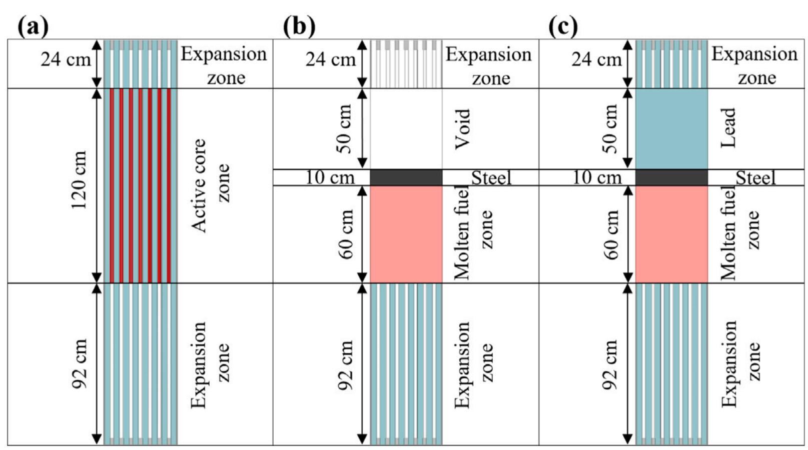
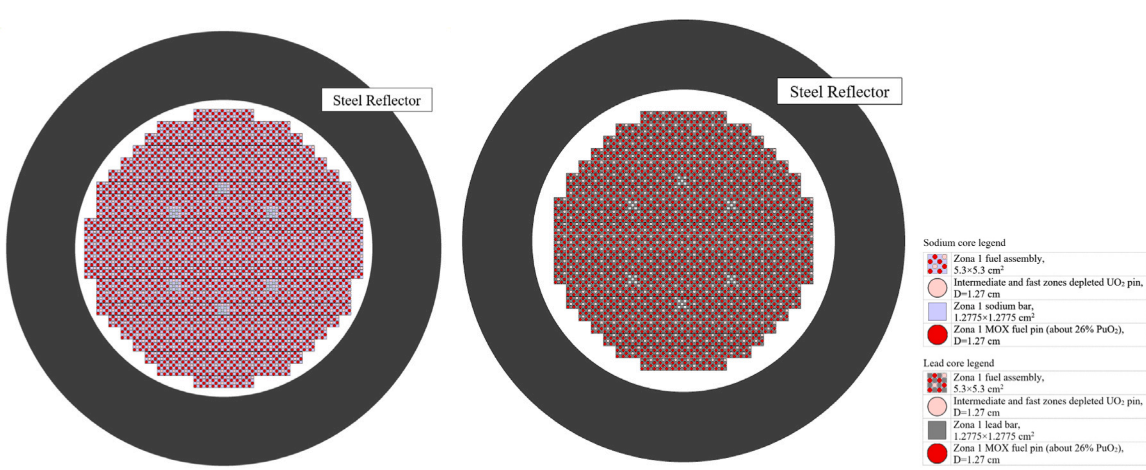
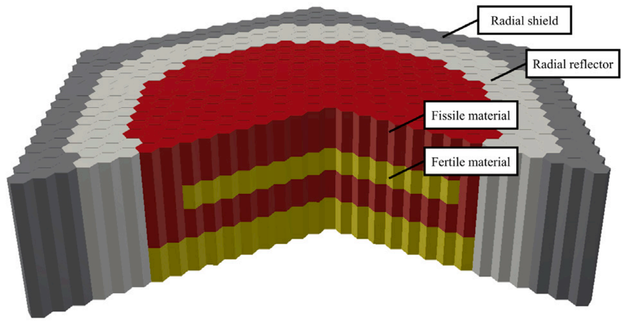


Fig. 18. Degraded configuration considered in ELSY SCA studies. (a) reference intact assembly. (b) molten fuel zone with a void on top. (c) reflooded molten pool.

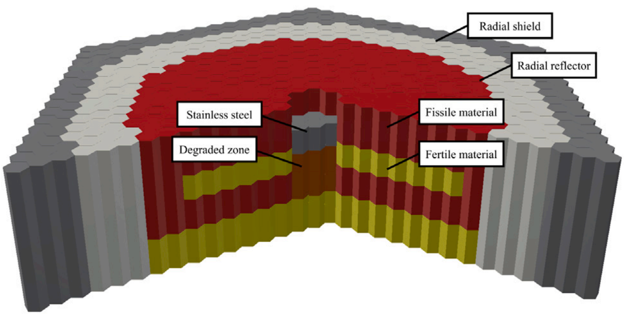


(**a**) ZEPHYR-S(odium). (**b**) ZEPHYR-L(ead). (**c**) Legend.

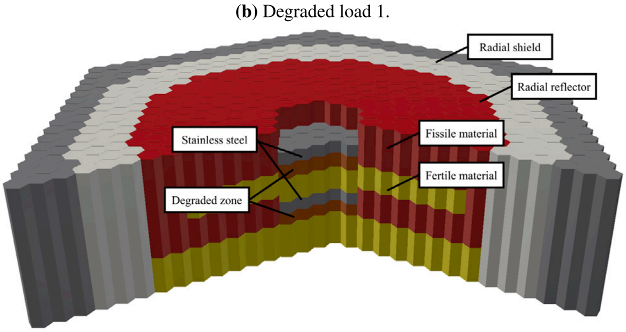
Fig. 19. Alternative full fast core designs.



(a) Reference.

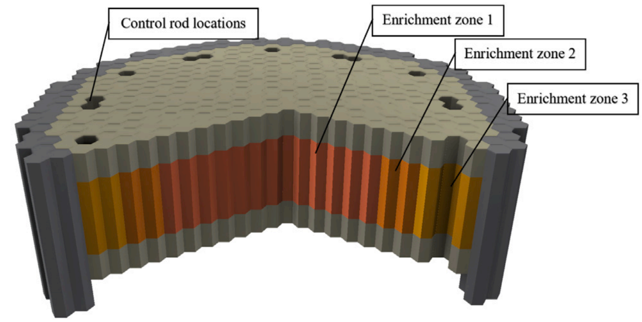


(b) Degraded load 1.

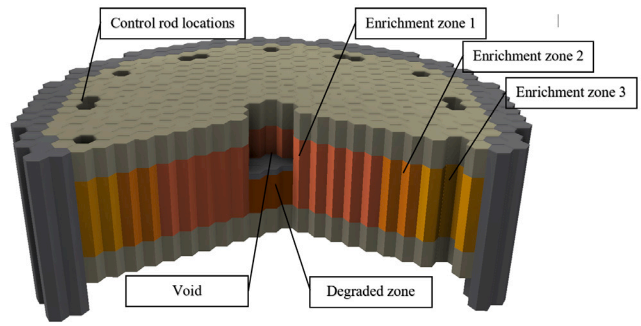


(c) Degraded load 2.

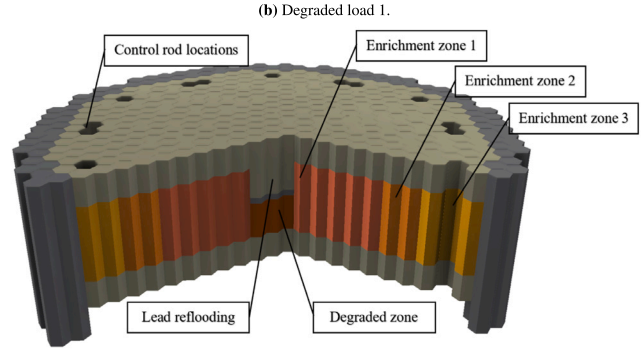
Fig. 20. Configuration for core optimization of ASTRID core degradation.



(a) Reference.



(b) Degraded load 1.



(c) Degraded load 2.

Fig. 21. Configuration for core optimization of ELSY core degradation.

Table 4
Single molten zone optimization results for representativity of full core reactivity variation obtained with COMAC-V01.

Temperature variation	"Perfect" fuel		MASURCA stockpile	
Reference-Degraded 1	PuO_2 content	r_{RE}	# of PuO_2 plates	r_{RE}
900–1000 °C	98.9%	0.90	55	0.85
900–2000 °C	93.6%	0.86	52	0.85
900–3000 °C	92.6%	0.85	51	0.85

own, as represented in Fig. 20(c). The current degraded layout differs from the previous one by the size of the degraded zones. The degraded zone of the current configuration spreads over two surrounding rings around the central fuel assembly, unlike the previous degraded single zone configuration that occupies the central fuel assembly and one surrounding ring. The reason for a larger degrade zone is the fact that there was no profound behavior in the sensitivity profiles for some of the key reactions of several isotopes (such as ^{23}Na and ^{56}Fe). In other words, one-ring perturbation is simply too small.

Similar to the single degraded zone results, the optimization results of "Perfect" fuel show that it is possible to achieve the required minimal value of $r_{RE} = 0.85$ by adjusting the PuO_2 content in the degrade zones. Moreover, it was possible to match the target level of representativity with actual fuel from the MASURCA stock-pile, as shown in Table 5.

The ELSY core consists of only a single axial fissile zone (Fig. 21(a)), but characterized by three radial enrichment zones, as mentioned in Table 2. Thus, the worst severe core accident scenario in the ELSY core is the voiding and compaction of the central core zone (Fig. 21(b)), with lead reflooding of the voided zone above the compacted fuel (Fig. 21(c)). The representativity analysis for the ELSY reactor considers a single temperature variation from 900 °C at nominal state (Fig. 21(a)) up to 3000 °C at the degraded zone.

The results of the ELSY reactivity variations are summarized in Table 6. The single zone optimization shows that it is possible to reach the minimal level of representativity of 0.85 with utilizing a "Perfect" fuel. However, one should bear in mind that ELSY is still a conceptual design, while the ASTRID project was in its final design stages. Thus, the MOX fuel provided in the ELSY core is an Americium-free MOX, which creates challenges when considering MASURCA fuel. The presence of a strong absorber, such as Americium-241 strongly alters the system characteristics. The representativity value decreases sharply when the MASURCA plates are loaded, as presented in Table 6.

Designing experimental programs considering influences of various parameters, which ensures high level representativity at nominal or a-nominal conditions of reference systems is of high interest. Such program can provide experimental data for code validation, insight

Table 5
Two molten zones (Fig. 20(c)) optimization results for reactivity variation representativity obtained with COMAC-V01.

Temperature variation	"Perfect" fuel			MASURCA stockpile		
	PuO_2 content			# of PuO_2 plates		
Reference-Degraded 2	Lower zone	Upper zone	r_{RE}	Lower zone	Upper zone	r_{RE}
900–1000 °C	45.4%	45.4%	0.90	13	20	0.87
900–2000 °C	48.7%	78.3%	0.86	14	35	0.85
900–3000 °C	94.5%	5.2%	0.90	29	4	0.87

Table 6
Single molten zone optimization results for representativity of full core reactivity variation (Figs. 21(b) and 21(c)) obtained with JANIS-4.0 covariance data for ENDF/B-VII.1.

Configuration	"Perfect" fuel		MASURCA stockpile	
	PuO_2 content	r_{RE}	# of PuO_2 plates	r_{RE}
Ref-Deg. 1	83.9%	0.86	128	0.78
Ref-Deg. 2	85.9%	0.87	128	0.78

regarding nuclear data uncertainties targeting, and potentially serve as a ground for instrumentation development (in-core and ex-core). Therefore, when considering construction of a new ZPR, it should be taken into account the manufacturing capability available to execute the most ambitious experiments. CEA's Cadarache site is a great example, it contains unique opportunity in terms of availability of fuel in the MASURCA facility stockpile of various types of materials (MOX, UOx, metallic fuels, different coolants, etc.) for different reactor types (e.g., sodium, lead, molten salts and others). The flexibility of the stockpile enables the construction of different representative systems. This advantage of the unique stockpile is utilized through this research for the design of an innovative experimental program. Therefore, choosing a location of construction of a new ZPR must take into account access to a wide range of materials and manufacturing capability. Besides France, there are not many countries that have a large nuclear sector. The UK is the only country that comes to mind that has a suitable capability within its material stockpile to benefit future facilities that is found in France. Thus, UK potentially can host such an innovative facility in collaboration with other countries.

5. Summary

Throughout the history of the nuclear industry ZPRs were a key factor in the advancement of the reactor design. These facilities were able to deliver the fist generation of LWRs, AGRs, and the few SFRs, which were operated around the world in the last 60–70 years. In some cases, experiments carried out in those facilities were before their time. For example, GLEEP was used to study the physics of the SGHWR, and ZENITH being used to design the next generation of AGR, pushing the temperature to a 1000 °C, which is relevant to all current HTGRs.

This paper tries to bring back the names of the many experimental facilities that were in operation around the world. Most of the names are probably forgotten, as the people operating them are retired or not among us. Data from those facilities sometimes is lost in a dark archive of a laboratory or saved in a personal archive. But, in some cases, this data can be key for a faster technology development. Therefore, it is crucial to save this data and bring it forward to help the sector grow faster. Projects like the NEA International Handbook of Evaluated Reactor Physics Benchmark Experiments is of the best example the desire to save the data of the past. But, it is not an easy task, in most cases the documentation is missing key values and parameters, which can render the found data useless. Therefore, in parallel to bringing back the names of past facilities back into life, the second objective of this paper is to present a look on what a future facility should look like and what kind of experiments could be implemented in it.

The CEA-led ZEPHYR project is the best example that is possible to find in the last 5 years. The proposed reactor design was aimed to keep capabilities of the past facilities and introduce much more. The facility offers the advantages of a high flexibility design and experimental potential. Through the representativity approach and the spectral adaptation method, the core of the ZEPHYR could be adapted to emphasize the neutron spectrum at the energy levels that would contribute to the reduction of uncertainties relevant to any reference design. In addition, an innovative experimental design approach taking into account effects that previously were never considered (i.e. temperature of the reference system) was shown.

The operational conditions associated with ZPR allow those facilities to be ambitious with the experiments conducted in them and their design. Therefore, in many cases constructing an innovative ZPR can also play crucial role in progressing the dormant nuclear industry around the world. To an extent, the nuclear industry that managed to deliver the Phenix, SuperPhenix, MSRE, Dounreay fast reactor, PFR, Dragon, Monju and FFTF is gone. There is a need for developing new skills to build advanced systems, and an ambitious ZPR project could be just the key to opening those missed skill sets. Finally, a ZPR is an excellent place to train the future generation of students in the nuclear engineering sector, which at the moment, sometimes go through the entire their entire education experience without seeing a nuclear facility.

CRediT authorship contribution statement

Marat Margulis: Writing – review & editing, Writing – original draft, Visualization, Validation, Resources, Methodology, Investigation, Data curation, Conceptualization.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

Data will be made available on request.

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