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A COMPARISON OF ADVANCED NUCLEAR TECHNOLOGIES

Andrew C. Kadak, Ph.D

MARCH 2017

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*__Andrew C. Kadak__ is the former president of Yankee Atomic Electric Company and professor of the practice at the Massachusetts Institute of Technology. He continues to consult on nuclear operations, advanced nuclear power plants, and policy and regulatory matters in the United States. He also serves on senior nuclear safety oversight boards in China. He is a graduate of MIT from the Nuclear Science and Engineering Department. His career of over forty years in the nuclear industry has provided many of the insights found in this report.

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This policy paper represents the research and views of the author. It does not necessarily represent the views of the Center on Global Energy Policy.

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PREFACE

This paper is one of a series of three being released by the Center on Global Energy Policy (CGEP) at the School of International and Public Affairs (SIPA) of Columbia University that focuses on the future of nuclear energy. These papers were made possible, in part, by a grant from the Sasakawa Peace Foundation (SPF) of Japan. SPF played no role, however, in the drafting or review of this paper series.

The series consists of the following three papers:

- "A Comparison of Advanced Nuclear Technologies," by Dr. Andrew Kadak
- "The Role of Policy in Reviving and Expanding the US Global Nuclear Leadership," by Tim Frazier
- "The Geopolitics of Nuclear Power and Technology," by Dr. Nicola de Blasio and Richard Nephew

CGEP chose three different sets of authors to prepare these papers to ensure a wide, diverse range of experiences and perspectives. CGEP also chose to work on these papers more or less in concert, with primary research and drafting of the paper on advanced nuclear reactor design taking place slightly earlier than the two policy papers. As such, though each of these papers reflects some understanding of the research, ideas, and concepts articulated in the other two, there are organic differences in emphasis, concentration, and interest.

There are also areas of clear convergence and stark divergence between and among the three papers. For example, all three papers operate from a baseline that views nuclear power as a useful – if not a necessary – part of the global energy mix. The broader, and important, debate of whether there is a role for nuclear power in a low-carbon society is outside the scope of these papers.

Even with this basic agreement, each of the three papers diverges on key aspects of nuclear power (such as the treatment of and concern with the threat of nuclear proliferation from widespread use of nuclear power). There are other areas in the papers in which differences of opinion exist, and most important, differing conclusions are reached—even when looking at the same historical episodes and present circumstances.

CGEP strongly believes in the importance of bringing together unique perspectives to address the most pressing energy issues. In the competition and comparison of ideas, and in debate and disagreement, the institution sees the acme of academic purpose. We hope this series of papers prompt a discussion about nuclear power and the trade-offs that exist in its pursuit.

EXECUTIVE SUMMARY

The global nuclear industry is presently exploring a wide range of potential technologies for advanced nuclear reactor design. These designs carry with them the potential to revolutionize nuclear power, improving the performance of nuclear reactors across a range of important characteristics. However, the very scale of the potential options for nuclear reactor design can be daunting.

This paper examines the range of advanced nuclear power reactor designs being developed at the present time. The study was completed on behalf of the Center on Global Energy Policy at Columbia University, and it used a grant provided by the Sasakawa Peace Foundation. The interest is in new, emerging nuclear technologies that are more cost competitive (relative to renewable and fossil fuel generation sources) and that lower the risk of construction delay and help manage the issues of nuclear nonproliferation and nuclear waste.

This study compares the various advanced nuclear technologies being developed by the United States and other nations to provide decision-makers a better understanding of the options available to accomplish the sought-after goals, using five critical criteria:

1. Safety risks.
2. Cost.
3. Waste issues.
4. Regulation.
5. Risk of contributing to nuclear weapons development.

The study presents detailed information on these new reactor concepts, which was derived from the publicly available details provided by the companies and research and development facilities associated with the respective reactor designs. It then assesses each reactor according to the specific criteria using a consistent, though somewhat subjective, evaluation methodology.

Key findings for policy makers include the following points:

1. For nuclear energy to play a significant role in dealing with climate change, government and private sector support is needed for innovative reactor design development and to realize the improved safety and efficiency of new plants.
2. A new regulatory system based on risk-informed requirements will greatly improve the ability to reduce costs and to bring these new designs to market without compromising safety.
3. Nonproliferation goals can be best achieved by working on political solutions versus technical limitations. This enables the use of reactors to consume nuclear waste and to provide for essentially a long-term sustainable nuclear energy enterprise using fast reactors.
4. A level playing field is needed for nuclear energy, similar to that provided for other clean-air sources, such as solar and wind.

As expected, the study did not identify any one reactor that uniquely or perfectly addresses concerns across all five criteria. However, as described in a summary table located at the end of this report, there are some reactor designs that more efficiently and effectively satisfy the interest of enhanced safety, reduced cost, mitigated waste issues, managed regulatory questions, and reduced risk of contributing to nuclear weapons proliferation. Consideration should be given by policy makers to identify mechanisms for prioritizing further research and development of these types of reactor designs.

INTRODUCTION

While it could be argued that the nuclear power industry is in the midst of another hiatus in some parts of the world, countries such as China and Russia are rapidly expanding their nuclear fleets to deal with the challenges of global climate change and resource limitations. The United Arab Emirates, Argentina, and Vietnam are building new nuclear plants. Many other nations, such as Indonesia, Turkey, Belarus, and Poland, are interested in new nuclear technologies in order to avoid continued dependence on fossil fuels. More developed nations, such as India, Finland, and Sweden, are either building new nuclear plants or considering adding additional nuclear capacity. In total, over forty-five nations are actively considering nuclear plants as part of their energy mix.

A review of commercial nuclear options shows there are a wide variety of technologies currently being offered to address the global climate change problem and the energy needs of specific countries. Many of these are new, innovative designs that address enhanced passive safety and have missions beyond simple electricity generation, such as reducing the quantity of nuclear waste, providing process heat for industrial applications, and desalinizing water. At last count, fifty reactors under development around the world were waiting for markets to materialize and for the funding to support continued development. Many of the current plants on the market are 1,200 megawatt electrical (MWe) or larger light-water reactors that have been designed with improved safety systems and increased size to capture the economies of scale. Other developments focus on more small modular reactors, in the 50 to 300 MWe range. These can address regional electricity grid capabilities at a lower investment cost than larger plants. The small modular reactors are seeking to gain the economies of mass production to counter the economies of scale. The high cost of large plants, which can run into many billions of dollars, is a deterrent in both developed and developing nations.

There is also a group of new innovative reactor designs that have yet to make it beyond the conceptual design stage, but they offer great potential. These technologies are unconventional in that they utilize helium gas, molten salts, or liquid metals as the primary coolant instead of water. These nontraditional technologies will require many years of testing and demonstration prior to widespread application. Importantly, in order to be successful in the market, they must be shown to be more economic than light-water reactors or other competing electricity-generating sources.

The common trend in all these new reactor plants is the movement to more modularity in design and construction in order to reduce the time and cost of construction. In addition, some designs focus on simplicity, utilizing more natural and passive cooling systems instead of relying on active systems that require electric pumps to provide the needed cooling water.

This paper explores the current major options available today as well as new, innovative technologies still on the drawing board along five criteria:

1. Safety risks.
2. Cost.
3. Waste issues.
4. Regulation.
5. Risk of contributing to nuclear weapons development.

This review will highlight various technologies by classifying them into four major groups:

1. Generation III light-water reactors, which can typically be classified into two types: traditional and innovative design.
2. Small modular reactors, which are also water based.
3. New designs that utilize nonconventional coolants, such as the high-temperature helium-cooled gas reactors and various liquid metal and molten salt reactors.
4. Nontraditional reactors, such as the battery-type reactors and small MWe size plants (for local application).

Grouping the different reactors in this way enables the paper to discuss the critical questions of cost, proliferation, and waste, as the groups, by and large, face similar risks and challenges. In cases where there are differences in specific reactors in the group, they will be addressed accordingly. Cost information, especially for new reactors that have not been built, is highly speculative, if available at all. In those cases, cost will not generally be included. Additionally, since new reactor construction has only recently been restarted with improved designs, first-of-a-kind cost numbers can also be misleading and are generally very high compared to estimates. What data are available will be presented, but they should not be used to judge future costs of similar plants. In general, the economic analysis used to make decisions when purchasing a nuclear plant is technology specific and country specific, based on local laws, regulations, and labor rates.

The analysis shows there is currently no one reactor that uniquely addresses all five criteria to an optimal degree. Potential options show better results in some areas and poorer results in others. These trade-offs are critical for the evaluation of nuclear reactor design, especially given that the investment in nuclear power requires a long-term view to justify the high initial cost.

Before plunging into the details of our wide range of reactor options, the following section provides a review of the basic principles of nuclear technology for readers less familiar with some of the more complex aspects of the sector.

NUCLEAR ENERGY PRIMER

1.1 Basics of Nuclear Technology

1.1.1 The Basics of Nuclear Energy

There are two basic ways to produce nuclear energy: nuclear fission and nuclear fusion. Both are processes by which atoms are altered to release energy. In extreme synthesis, fission is the division of one heavy atom into its two smaller atoms, while fusion is the combination of two light atoms into a larger one. Nuclear power utilizes the resulting energy to heat water and, ultimately, produce electricity.

In nuclear fission, a heavy element, such as uranium or plutonium, is bombarded by neutrons from an atomic nucleus, which causes the atoms to split. In the process, high-energy neutrons are ejected, becoming projectiles that can then initiate other fission reactions. In nuclear fusion, the nuclei of more than one atom are fused under extreme pressure and temperature. Fusion is a process that occurs within stars, and while much promise has been seen in using nuclear fusion as a way of producing reliable, clean energy, it has yet to be developed to a point where more energy is produced than is needed to create the fusion reaction. Nuclear fission has, therefore, formed the basis of nuclear energy production worldwide.

For nuclear fission to work, however, it is not merely enough to split an atom. Rather, a system must be designed so the heat produced by the nuclear fission reaction can be safely extracted to produce electricity.

Solving how to most efficiently and effectively produce energy in a safe, reliable, and sustainable way has been the main focus of nuclear reactor design since the 1940s. Over the intervening seventy years, scientists and operators have learned many lessons, both about nuclear science and the limitations of human design and engineering. Reactor designs have adjusted to include more safety features, particularly in response to highly public accidents, such as Three Mile Island, Chernobyl, and Fukushima.

For the most part, uranium has been the fuel of choice for these reactors. Uranium occurring naturally in the earth's crust is a mixture largely of two isotopes: uranium-235 (U-235) and uranium-238 (U-238). U-238 makes up about 99.3% of uranium in the earth's crust, while U-235 is far rarer, accounting for the remaining 0.7%. However, of the two isotopes, U-235 is fissionable, and thus needed for self-sustaining nuclear reactions. Reactor designs have, therefore, had to take the relative scarcity of U-235 into consideration by either developing specific designs that can utilize this low value of natural uranium or by increasing the overall amount of U-235 in the fuel. The latter process is known as fuel enrichment. Most reactors in use today use U-235 enriched to less than 5%.

There are some reactors that utilize plutonium or thorium instead of or complementary to the use of uranium. Plutonium[1] can be made in a nuclear reactor when uranium-238 absorbs a neutron. Like uranium, it also has various isotopes that are useful in creating and sustaining nuclear reactions. Thorium is a naturally-occurring element that when struck by a neutron in a reactor is converted to U-233, which can also fission, which then sustains the chain reaction.

1.1.2 Principal Elements of a Reactor

The most fundamental elements of a nuclear reactor are the fuel and the system used to control the resulting chain reaction and remove heat. Nuclear reactors predominantly use control rods. These are neutron absorbers that control the rate of the nuclear reaction to achieve a desired level of heat production. In addition, a moderator is used to slow down the neutrons emitted from the fissioning process. This allows for more efficient use of neutrons in splitting uranium atoms.

Most nuclear reactors produce electricity by generating steam that turns turbines that then spin electric generators to produce electricity, as is done in conventional electric generating stations. The most common reactor design uses normal water (H2O) to cool and moderate the reactor and to channel its heat energy to the turbines.[2] These reactors are called light-water reactors. They are generally large and use low (less than 5%) enriched uranium as fuel. At these enrichment levels, nuclear plants have no chance of exploding like a nuclear bomb.

Pressurized water reactors (PWRs) have both a primary cooling circuit, which flows through the core of the reactor at high pressure, and a secondary circuit, which generated steam at lower pressure. Water in the reactor core reaches 325°C, and it is kept under pressure to prevent it from boiling. Pressure is maintained by steam in a pressurizer.

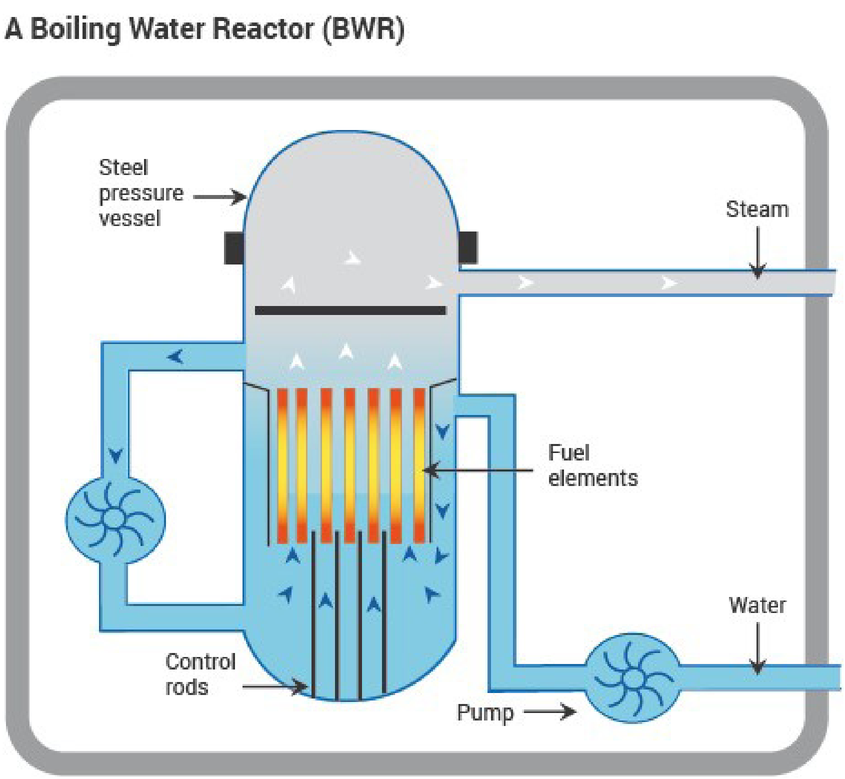
Boiling water reactors (BWRs), on the other hand, only have a single water circuit. The water is at lower pressure, and it boils directly in the core at 285°C. These reactors operate with 12%–15% of the water in the top part of the core as steam. The steam then passes directly from the core to the turbines, which are, therefore, part of the reactor circuit and need to be shielded.

Steam turbines have spinning blades that turn when the steam flows past them. In the process, the steam expands, cools down, and condenses, and the resulting liquid water is recycled.

Other reactor designs use different types of moderators, such as heavy water. This is water containing deuterium, an isotope of hydrogen. These reactors can use natural uranium as a fuel.

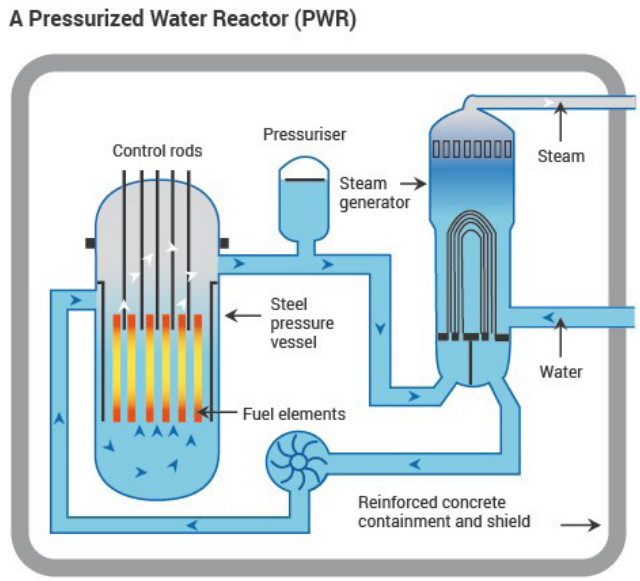
Figures 1.1-1.3 below, courtesy of the World Nuclear Association, help to articulate the general principles of reactor design, comparing PWRs and BWRs in particular:

Figure 1.1 Pressurized Water Schematic



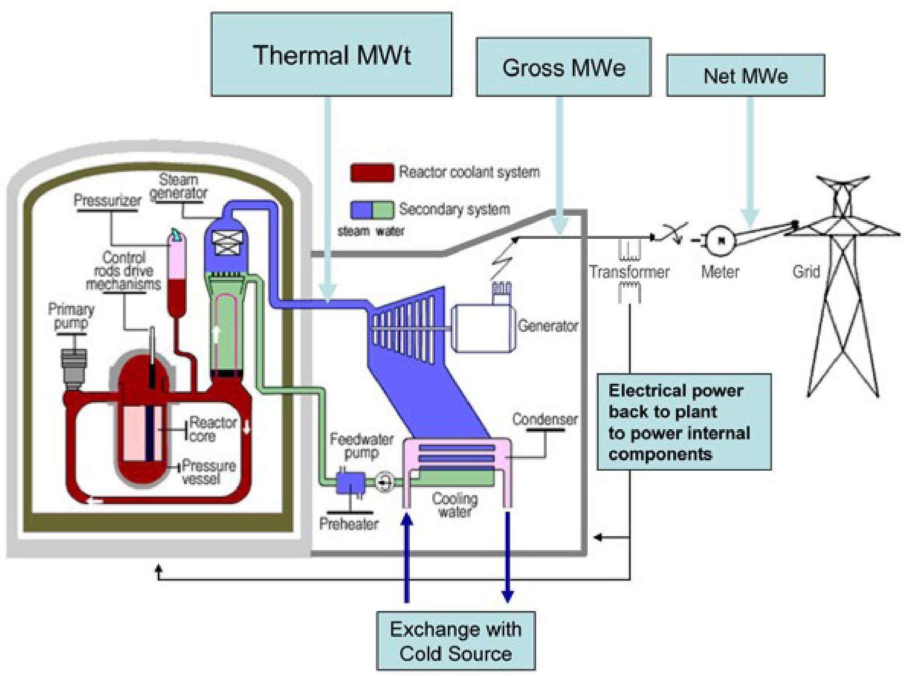
Source: World Nuclear Association

Figure 1.2 Boiling Water Reactor Schematic



Source: World Nuclear Association

Figure 1.3 Nuclear Plant Power Cycle



Source: World Nuclear Association

Nuclear reactor power outputs are rated in the following ways:

- Thermal megawatt (MWth), a function of the design, refers to the quantity and quality of the produced steam.
- Gross megawatt electric (MWe) refers to the power produced by the attached steam turbine and generator.
- Net MWe is the power that can be fed into the grid after deducting the power needed to run the plant itself.

1.1.3 Challenges in Nuclear Reactor Design

In assessing nuclear power, it is also important to evaluate the side effects and unintended consequences of development. There are five critical considerations:

1. Safety risks.
2. Cost.
3. Waste issues.
4. Regulation.
5. Risk of contributing to nuclear weapons development.

With respect to safety, concerns center on safety systems needed to deal with potential accidents. Much emphasis has been placed on the creation of passively safe designs that do not require active cooling systems but use instead passive natural cooling and less human intervention. This reduces the risk and possibility of human error or indecision. Advanced reactor designs have sought to do more to reduce further the risk levels that come along with the use of this technology.

The issue of cost remains one of the most difficult for nuclear energy to overcome. In the 1950s, there was widespread discussion of limitless, cheap nuclear energy. Over the decades that have followed, though, nuclear energy remains a comparatively expensive option, despite the climate benefits and reliability. Some of these costs are due to the safety systems required of nuclear reactors. Reactor designers have sought to find ways to make nuclear energy cheaper while still providing all the safeguards that nuclear power demands. However, other costs or economic pressures have created a different economic environment. This is due to the relative abundance of carbon-intensive energy sources, such as natural gas and coal.

Waste issues also persist, as the generation of nuclear reactions results in the creation of highly radioactive materials. Fortunately, they are small in volume. Managing nuclear waste has emerged as a particularly difficult political problem, even when technical solutions have been found. This is in part because of the reluctance of people to allow for the disposal of nuclear waste near their communities. This is particularly true in the United States. After years of technical study and the submission of a construction permit for the Yucca Mountain deep geological repository for high-level waste in Nevada, the US government decided to stop the review process by the Nuclear Regulatory Commission—essentially canceling the project for political reasons.

Reactor designers have sought to engineer ways of reducing the produced waste or immobilizing it, but these solutions can be expensive. Other designers have chosen to use the nuclear waste as a fuel in their designs, essentially removing it and reducing the volume of waste that needs to be disposed. These types of reactors require some reprocessing of the spent fuel from reactors in order to recycle it in their reactor designs. This raises the issue of potential proliferation concerns should plutonium, which can be used as the fuel for nuclear weapons, be separated from the used fuel.

Regulation has always affected the future of nuclear power, both in the United States and across the globe. As discussed, the safe and secure use of nuclear energy could play an important role in achieving international climate goals while providing economically competitive power. The role of the regulator has been to ensure a level of safety commensurate with acceptable public risk. In the United States, regulators are slowly moving toward risk-informed and performance-based regulation. If implemented completely, this will allow designers to focus on safety-critical systems.[3] For new and innovative reactor systems, a fundamental change in the water-based regulations that are now in existence is also needed. Otherwise, the key advantages of new designs cannot be realized. The old prescriptive regulations are not appropriate for passively or inherently safe designs, which could offer significant advantages when compared to current light-water reactors. Forcing these new technologies to comply with outdated deterministic requirements based on older designs would be detrimental to making advances. A more risk-informed, technology-neutral regulatory system would be better suited for these new technologies. Overall, governments need to establish efficient, stable, and reliable regulatory frameworks, while putting in place policy incentives to support research and development efforts with respect to fuel cycles.

The issue of proliferation is not normally a concern in the United States or Japan, but it is in other nations that are seeking or might seek to develop nuclear weapons. There are two ways in which nuclear reactor development potentially contributes to nuclear weapons development. First, because reactors contain mostly U-238, they produce plutonium. During the course of the operation of the reactor, fissions from plutonium made in the reactor produce heat, allowing the reactor to run for close to two years without refueling. Once the fuel is removed, the many isotopes of plutonium produced are contained in the fuel. In the aggregate, the plutonium coming out of nuclear reactors is not very well suited for nuclear weapons since almost pure Pu-239 is needed for a weapon.[4] In any case, the plutonium must be chemically separated (reprocessed) from the fuel to be available for weapons production. In order to set an example to other countries, the US policy has been to discourage reprocessing in its fuel cycle. At this point, it is simply cheaper to buy new uranium than to pay the cost of reprocessing and recycling. These economics have limited the spread of reprocessing technology in addition to proliferation concerns. France, Russia, and the United Kingdom have reprocessed spent nuclear fuel as part of their waste minimization strategy. Japan has built a reprocessing plant, but it is idle due to the Fukushima accident and their difficulties with the demonstration fast reactor, which would use the fuel from the reprocessing plant. Reprocessed plutonium can also be used in light water reactors as a fuel in a mixed oxide (MOX) form. This fuel contains a mixture of plutonium isotopes and uranium. MOX has been used in light water reactors in France.

The second way is that nuclear fuel created in the enrichment process of uranium for use in reactors can itself be used to create materials for nuclear weapons. Most enrichment plants are designed to enrich to about 5% U-235, which is all most reactors need. Future nuclear reactor options call for enrichments up to 20%. For nuclear weapons, enrichments on the order of 93% would be required. Although enrichment plants can be modified to produce higher levels of enrichment, strict monitoring and controls by the International Atomic Energy Agency (IAEA) are required for enrichment plants and reprocessing plants operating in non-nuclear-weapon states under the Nuclear Nonproliferation Treaty (NPT). The risk of uranium diversion or enrichment to weapons-grade levels is, therefore, real but limited in most countries around the world.

GENERATION III AND LARGE LIGHT-WATER REACTORS

As noted in the previous section, most operating commercial nuclear power stations in the world are light-water reactors.

Early in the development of this growing industry in the United States, Combustion Engineering and Babcock & Wilcox also developed their versions of a PWR, joining the pioneers at GE and Westinghouse. At the time, the United States led the nuclear industry, and their fundamental technologies were licensed to other companies in Europe (France and Germany) and Asia (Japan and Korea). These nations have since modified the fundamental technology and adapted it to each nation's ability to manufacture components for internal fabrication and export. Russia independently developed their own version of a pressurized water reactor (VVER), which Russia sold to the former Soviet Socialist Republics. Russia is currently exporting their latest version of the VVER to developing nations, such as Vietnam and China. China initially started their nuclear program with French-designed pressurized water reactors, which they adapted and modified to their needs by increasing size and making other improvements. Many new Chinese reactors are of the indigenous design, and most of the components are made in China. China has also purchased both the Westinghouse AP-1000 and EPR reactors. Both types are under construction in China.

The industry has evolved the design of light-water reactors since the 1960s, going through three generations of design. Each design was larger than the last and had enhanced safety features and improved performance. In the United States today, Westinghouse and General Electric continue to advance the technology of light-water reactors by attempting to improve safety and economics, making the reactors less vulnerable to core-melt accidents. Eight of the Westinghouse AP-1000s are being built in the United States and China (four in each country). France and Germany have developed the European Pressurized Reactor (EPR), which is a more advanced but less innovative light-water reactor. It is currently under construction in Finland, China, and France. Korea has adapted the Combustion Engineering design and is building four APR-1400 plants in Korea and four in Abu Dhabi. In Japan, Mitsubishi has also developed an advanced PWR (APWR), which is rated at 1530 MWe.

General Electric continues to develop the boiling water reactor with the advanced boiling water reactor (ABWR) that has been built in Japan and proposed for the United States. Two Japanese companies (Toshiba and Hitachi) have developed BWRs in Japan. Hitachi has partnered with General Electric to design the Economic Simplified Boiling Water Reactor (ESBWR), which recently received design certification from the US Nuclear Regulatory Commission.

There are many designs for light-water reactors internationally, but all build on the same pressurized or boiling water reactor principles. Shown in Table 2.1, Table 2.2, and Table 2.3 is the current status of light-water reactor offerings and proposals. As can be seen, strong international competition for new reactor technology exists. Each vendor offers variants regarding how safety functions are met and size in order to meet the expectations of market needs by country. Emphasis is also on reducing capital costs to allow the vendors to compete in the international market. Not all these reactors will be described, but they are shown to indicate the number of nuclear power stations available for purchase.

Table 2.1 Operational Advanced Power Reactors

Developer	Reactor	Size (MWe Gross)	Design Progress, Notes
GE Hitachi, Toshiba	ABWR	1380	Commercial operation in Japan since 1996–7. US design certification, 1997. UK design certification application, 2013. Active safety systems.
KHNP	APR-1400	1450	Shin Kori 4 in South Korea, operating since January 2016. Korean design certification, 2003. US design certification application underway.

Table 2.2 Advanced Power Reactors Under Construction

Developer	Reactor	Size (MWe Gross)	Design Progress, Notes
Westinghouse	AP-1000 (PWR)	1250	Under construction in China and the United States; many units planned in China. US design certification, 2005. Canadian design certification in progress.
Areva (and EDF)	EPR (PWR)	1750	Was to be future French standard, with French design approval. Being built in Finland, France, and China
KHNP	APR-1400 (PWR)	1450	Under construction: Shin Kori 4, Shin Hanul 1, and Shin Hanul 2 in South Korea and Barakah in United Arab Emirates. Korean design certification, 2003. US design certification application underway.
CNNC and CGN (China)	Hualong One (PWR)	1150	Main Chinese export design. Under construction at Ningde.
Gidropress	VVER-1200 (PWR)	1200	Under construction at Leningrad and Novovoronezh plants as AES-2006.

Table 2.3 Advanced Power Reactors Ready for Deployment

Developer	Reactor	Size (MWe Gross)	Design Progress, Notes
GE Hitachi	ESBWR	1600	Planned for Fermi and North Anna in the United States. Developed from ABWR but contain passive safety systems. Design certification in the United States, September 2014.
Mitsubishi	APWR	1530	Planned for Tsuruga in Japan. US design certification application for US-APWR. EU design approval for EU-APWR, October 2014.
Areva, Mitsubishi	Atmea1 (PWR)	1150	Planned for Sinop in Turkey. French design approval, February 2012. Canadian design certification in progress.
Candu Energy	EC6 (PHWR)	750	Improved CANDU-6 model. Canadian design certification, June 2013.
Gidropress	VVER-TOI (PWR)	1300	Planned for Nizhny Novgorod in Russia and Akkuyu in Turkey. Russian design certification in progress for EUR.

While the basics of the technology are the same for each type, the differentiating features are largely how each approaches the safety function and overall simplicity and cost of the plants. The European, Korean, and Japanese versions of the PWRs are larger than standard plants (to capture economies of scale), while the AP-1000 focus is on reducing the amount of piping and cabling and number of safety systems needed to achieve more passive safety. The AP-1000, rather than relying on active pumps to provide emergency cooling water should mishaps occur that interrupt that supply of cooling water, rely on gravity to supply cooling water. New designs also include additional safety features to capture the lessons learned from the Chernobyl and Fukushima accidents. All new designs for reactors target a sixty-year operating life, up from forty years in previous reactor designs.[5]

2.1 Safety

It can be reliably stated that Generation III reactors currently on the market have higher safety margins than earlier designs by a factor of ten. This is based on probabilistic safety analyses. These improvements are plant specific and design specific and were based on years of learning lessons from the past, using probabilistic risk (or safety) analysis in design, and simplifying plants. These simplifications took advantage of more passive, rather than active, safety systems that did not require electric power for their operation. Additionally, some plants have added “core catchers,” or devices and systems to retain the core in the reactor vessel. This differs from containment, which surrounds the reactor vessel. These enhancements have greatly improved the safety performance of Generation III light-water reactors.

2.2 Nonproliferation

From a nonproliferation point of view, light-water reactors are quite similar. They all utilize low-enriched uranium-235 (less than 5%). Enrichment plants do pose some proliferation risk. With some modification, they can produce highly enriched uranium-235, which is needed for simple atomic bombs. The International Atomic Energy Agency (IAEA) monitors existing enrichment plants to ensure that such modifications are not made and that uranium is not diverted in those nations that allow such inspections.

Since most uranium in the core of light-water reactors is uranium-238, plutonium is produced as a by-product. The spent or used fuel requires reprocessing for use in fast neutron reactors, or it's recycled into fuel that can be used in light-water reactors. These reprocessing plants, if not monitored by the IAEA, can produce plutonium that can be diverted for use in crude nuclear weapons. However, the plutonium produced is reactor grade. Other plutonium isotopes contaminate it, which makes it difficult to make a nuclear weapon. France, the United Kingdom, and Russia have reprocessed their used fuel, reducing quantities of high-level waste to be disposed of while converting the still useful uranium and plutonium into mixed oxide (MOX) fuel for use in some of their reactors. In the United States, the US Navy has reprocessed naval reactor spent fuel, but a commercial reprocessing plant in Barnwell, South Carolina, was never started due to a presidential decree at the time. Consideration was being given to a new reprocessing plant to recycle spent fuel in the United States, but the economics of such a proposal did not justify the cost due to the low price of uranium.

2.3 Nuclear Waste

In terms of nuclear waste, all light-water reactors currently in operation and proposed worldwide have the same issue associated with finding suitable locations for long-term waste disposal. (The waste is mostly used fuel that will not be reprocessed.) Waste from reprocessing plants is solidified in borosilicate glass, which reduces the volume by over 90% from the spent fuel form. France and other reprocessing countries are currently storing this high-level glass waste in air-cooled vaults.

While the United States has the first operating geologic repository for nuclear waste in New Mexico, at the Waste Isolation Pilot Plant for plutonium contaminated wastes, Finland and Sweden lead the world by siting the first geological repository for high-level spent fuel waste. The United States and other nations are still struggling with the political question of where to site a geological repository for high-level nuclear waste. In the United States, the proposed Yucca Mountain geological repository was canceled[6] in 2009 for political reasons. This was after the license application was submitted to the Nuclear Regulatory Commission for final review. Should Yucca Mountain be restarted under the new administration or the voluntary siting process continue to identify a new site, the additional future challenges for disposal include addressing the transportation and infrastructure that need to be established to allow shipment of spent nuclear fuel to a repository. In any case, the cost of high-level waste disposal is included in the cost of nuclear-generated electricity. The government has been collecting $0.001 per kilowatt-hour generated by nuclear plants. To date, over $30 billion has been collected to finance the repository construction and operation.

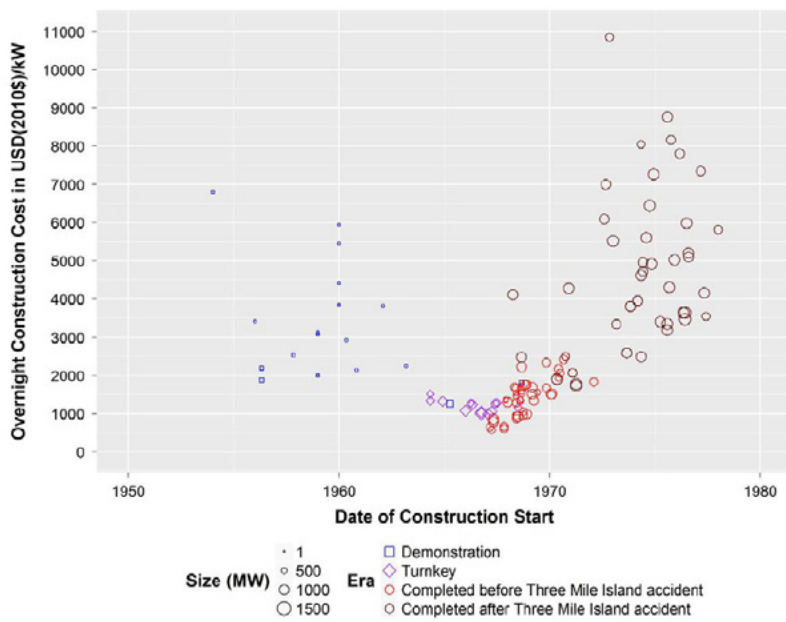
Since most new reactor technologies for light-water reactors are fundamentally the same, the following section will only summarize the major offerings currently on the market and outline the differences and unique natures of the plants relative to others in this group. The decision to purchase one over another largely depends on economics and the national perception of safety that each might offer. For example, in Europe, filtered vented containments appear to be necessary additions, which some plants offer in the design.

2.4 Economics

On the economic front, designers are generally seeking to capitalize on the economies of scale to reduce the cost of power on a per kilowatt-hour basis. This comes at a price in terms of the initial capital investment, which can be quite high—on the order of approximately $10 billion per plant, depending on the country. While the life of the investment has been increased to sixty years as a design goal, the initial high investment is a deterrent to new nuclear construction for many utilities and countries.

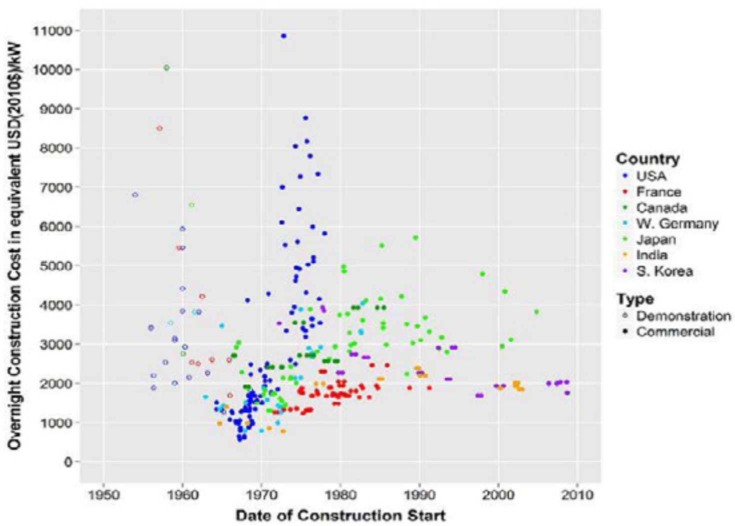
Historically, the cost of nuclear plant construction has increased due to increased regulatory requirements and escalated construction costs. The Breakthrough Institute issued a report[7] on the historical costs of global nuclear construction. Shown in Figure 2.1, there is great variability in the overnight cost of nuclear power, but this depends on local circumstances. It also greatly increased after the Three Mile Island accident, which imposed many additional regulatory requirements. When one compares worldwide construction costs by country, the results shown in Figure 2.2, the United States is an outlier in overnight costs, while Korea and France continue to maintain a relatively stable $2,000–$2,500/kWe. This is largely due to design standardization and a more rational regulatory system.

Figure 2.1 Historical Overnight Construction Costs in the United States



Source: Breakthrough Institute

Figure 2.2 Global Overnight Nuclear Construction Costs (Lovering)



Source: Breakthrough Institute

Since new nuclear construction in the United States has been only recently restarted, the costs of present projects, such as the V. C. Summer and Vogtle AP-1000 plants, have experienced schedule and cost overruns due to restarting an industry that has been dormant for over twenty-five years. With standardization and subsequent plant construction, it is expected that the next plants built in the United States will be considerably less expensive. An interesting study completed in 2008[8] tracked the increasing cost estimates of new nuclear construction. These were only estimates based on studies, not real construction costs in the United States. Table 2.4 below summarizes these estimates over time. There are some differences in each study as to what is included in the capital cost, but the trends are clear. The Georgia power estimate appears to be the closest to what is currently being experienced for those plants under construction in the United States. The reasons for the increases are cost of goods, cost of labor, escalated regulatory requirements over the years, and lack of suppliers who are able to provide nuclear-grade components on a worldwide basis.

Table 2.4 Nuclear Plant Construction Cost Estimates

Forecast	Overnight Cost	Total Plant Cost	Total Plant Cost - 2 Units
	($/kW)	($/kW)	(billions$)
DOE (2002)	$1,200		
	$1,500		
MIT (2003)	$2,000		
Keystone Center (2007)	$2,950	$3,600	
	$2,950	$4,000	
Moody's Investor Services (2007)		$4,000	
		$6,000	
Florida Power & Light (2007)	$3,108	$5,492	$12.1
	$4,540	$8,081	$17.8
Progress Energy (2008)			$14.0
Georgia Power (2008)			$6.4 for 45% of 2 plants

Source: SYNAPS Energy Economics

The overnight cost is only one of the key variables of the cost of electricity from nuclear plants. Other very important factors include financing cost, fuel cost, and operating and maintenance costs, which include staffing and regulatory fees. The most historically stable and least impactful on the cost of electricity is the cost of fuel, which is about 10% of the cost of power. This is why the price of electricity for operating nuclear plants is relatively stable and not a function of volatile fuel costs, such as the cost of natural gas, coal, or oil.

2.5 More Innovative Light-Water Reactor Designs

2.5.1 AP-1000

The AP-1000 nuclear plant is a pressurized water reactor currently under construction in China and in the United States. The maximum thermal power capacity is 3,415 MWth, with a net electrical generation of 1,115 MWe. Designed by Westinghouse, the primary goal of the design is to reduce the number of costly components, piping, and cabling and increase overall safety of the plant.[9] Westinghouse accomplishes this goal be relying more on passive safety features and avoiding active cooling pumps for safety functions.

The AP-1000 has 50% fewer safety-related valves, 35% fewer pumps, 80% less safety-related piping, 85% less control cable, and 45% less seismic building volume. These developments should clearly improve the economics of the Westinghouse plant. The construction of the plant also employs significant modularity in the fabrication of large structures at a central factory to be shipped to the site for erection.

Eight such plants are under construction in the United States and China. Figure 2.3 shows the Sanmen plant in China. Table 2.5 shows the key plant design features.

Figure 2.3 Sanmen AP-1000 Under Construction in China

Source: Westinghouse

Table 2.5 Key Plant Design Features of AP-1000

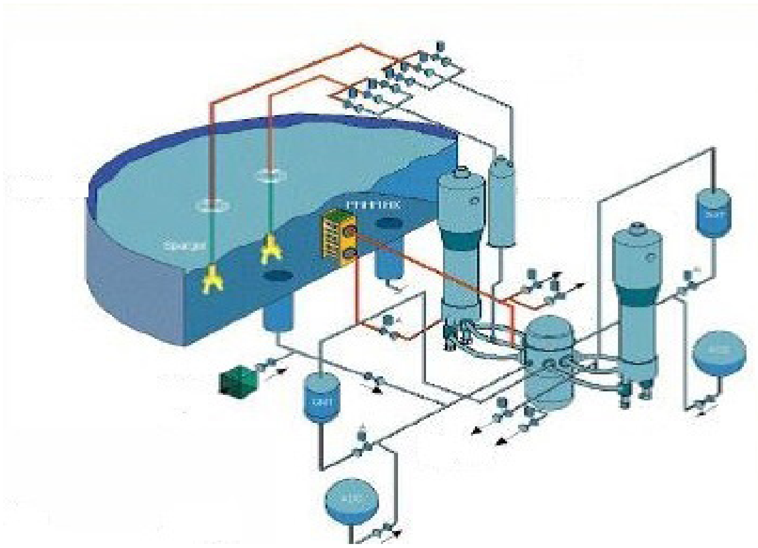
Reactor Thermal Power	3,415 MWth
Reactor Electrical Power	1,115 MWe
Containment	Single
Core Inlet/Outlet Temperature	280.7°C /321.1°C
Number of Fuel Assemblies	157
Fuel Assembly Length	14 ft.
Core Damage Frequency	2.4×10^{-7}
Emergency Safeguards	Passive In-Vessel Retention System
Number of Steam Generators	2
Main Coolant Pumps	4 Canned Rotor
Refueling Interval	18 Months
Construction Period[10]	3 Years

Source: Westinghouse

The design employs two large steam generators and canned rotor main coolant pumps to avoid problems associated with seal leakage. The containment is self-cooled in the event of a loss of coolant accident. Emergency core cooling water high in the containment allows for gravity flow into the reactor vessel and the reactor cavity housing the reactor vessel to ensure complete core coverage should a major pipe break. In the case of severe accidents, the primary reactor system is depressurized, allowing for gravity cooling water flow. The plant also boasts an in-reactor vessel retention system in the event of a core-melt accident. This limits the consequences of severe accidents. The core damage frequency is also very low, estimated at 5×10^{-7}, which is about one hundred times lower than current designs.

The plant employs two passive safety systems. One is used to cool the core in the event of a major pipe break, and the other is used to cool the containment. The core cooling system is shown in Figure 2.4 below. The plant has an in-containment refueling water storage tank (IRWST), which drains by gravity into the core upon reactor depressurization. The amount of water in the tank is sufficient to cover the reactor fuel and the reactor cavity above the fuel assembly height.[11]

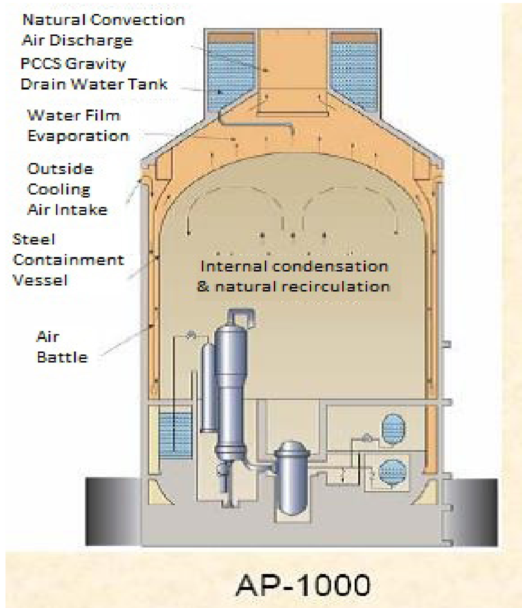
Figure 2.4 Schematic of AP-1000 Safety Systems



Source: Westinghouse

Similarly, to handle the containment of excessive pressurization upon a loss of coolant accident or fuel damage, the containment is naturally cooled by a water tank above the containment structure. This drains on the containment steel shell and provides needed cooling to avoid excessive pressurization and failure of the containment.

Figure 2.5 AP-1000 Containment Cooling System



Source: Westinghouse

The AP-1000's innovation can also be found in increased modularization of construction, where large structures are prefabricated in a factory to be shipped to the site for installation. This shortens the time to construct the plants as compared to "stick-build" techniques of the past.

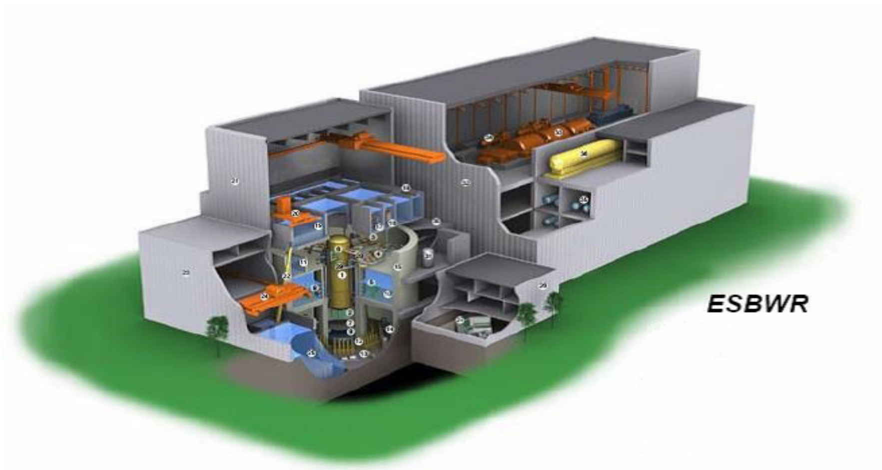
The cost of AP-1000s being built in the United States is still high. The four units under construction in Georgia and South Carolina are estimated to cost about $8 billion each. The cost of power from these plants is estimated to be about $0.08 to $0.10/kWh. Since this is the first time in over twenty-five years that new plants have been built in the United States, there have been delays incurred for both projects, raising the costs above estimates.

2.5.2 Economic Simplified Boiling Water Reactor (ESBWR)

This boiling water reactor was developed by General Electric with Hitachi from Japan. The design objective of this plant was to take advantage of increasing size and simplicity of design to reduce the cost of power. The thermal output of this reactor plant is 4,500 MWth, and it has a net electrical output of 1,600 MWe.[12, 13] The innovative feature of this size design is that the reactor core is cooled by natural circulation with no active recirculation pumps. Additionally, the safety features to address loss of coolant accidents and severe accidents are passive systems that rely on gravity to supply water to the reactor core. Risk analysis shows that the ESBWR probability of a core-melt accident is as low as 10^{-8} (an improvement of a factor of fifty over previous BWR designs). As an additional safety feature for the unlikely core-melt accident, a core catcher is included. This mitigates the damage caused by such severe accidents. At present, no orders for the ESBWR have been made, but there is interest from many countries, including the United States and the United Kingdom.

A graphic of the plant layout is shown in Figure 2.6 below.

Figure 2.6 ESBWR Plant Layout



Source: GE/Hitachi

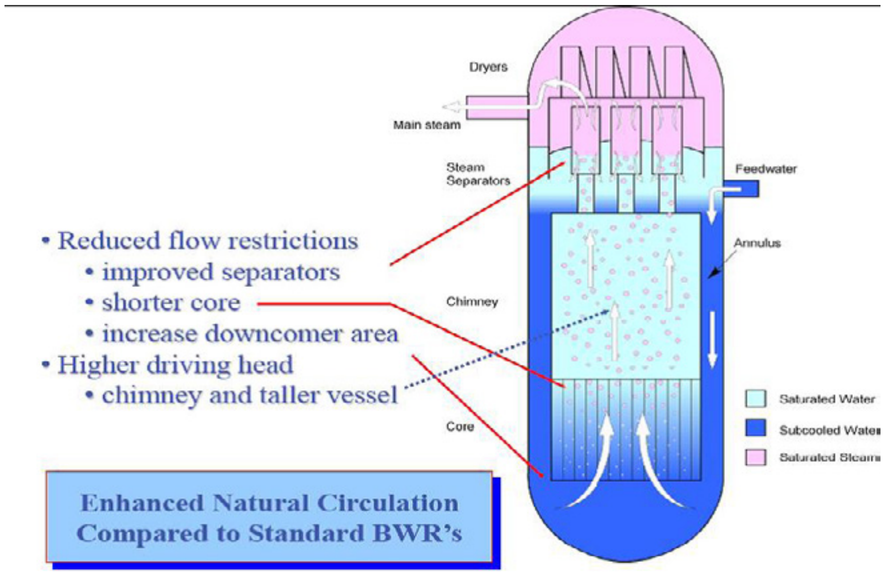
Table 2.6 Key Plant Design Features of ESBWR

Reactor Thermal Power	4,500 MWth
Reactor Electrical Power	1,600 MWe
Containment	Single
Core Inlet/Outlet Temperature	215.6°C /287.7°C
Number of Fuel Assemblies	1,132
Fuel Assembly Length	3.79 m
Core Damage Frequency	$< 10^{-6}$
Emergency Safeguards	Passive (Core Catcher)
Number of Steam Generators	Direct Cycle (Steam from Reactor to Turbine)
Main Coolant Pumps	0 (Natural Circulation)
Refueling Interval	18–24 Months
Construction Period[14]	4 Years

International Atomic Energy Agency

The ability to cool the reactor core using only natural circulation is accomplished by reducing the length of the 1,132 fuel assemblies and increasing the height of the reactor vessel. This allows for the chimney effect to generate a sufficient flow of water (Figure 2.7).

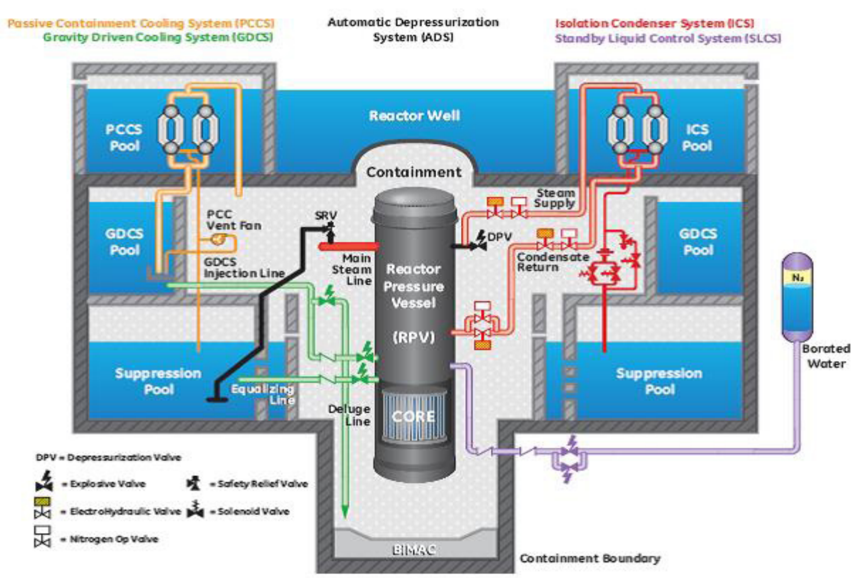
Figure 2.7 ESBWR Natural Circulation



Source: GE/Hitachi

The safety features incorporated in the design include passive features that are actuated upon receipt of abnormal signals that depressurize the reactor vessel. This allows for water to naturally flow into the core and provide needed core cooling. Figure 2.8 summarizes the key safety features incorporated in the design. For the ESBWR design, General Electric-Hitachi reintroduced the use of isolation condensers to remove decay heat. This is another passive system that enhances the safety of design.

Figure 2.8 ESBWR Safety Features



Source: GE/Hitachi

GE Hitachi claims this plant can be constructed in thirty-six months. This is based on the experience of the Japanese who have built several ABWRs on similar schedules. The cost of power from ESBWRs is not publically available, except to say the plants are competitive to other nuclear power stations.

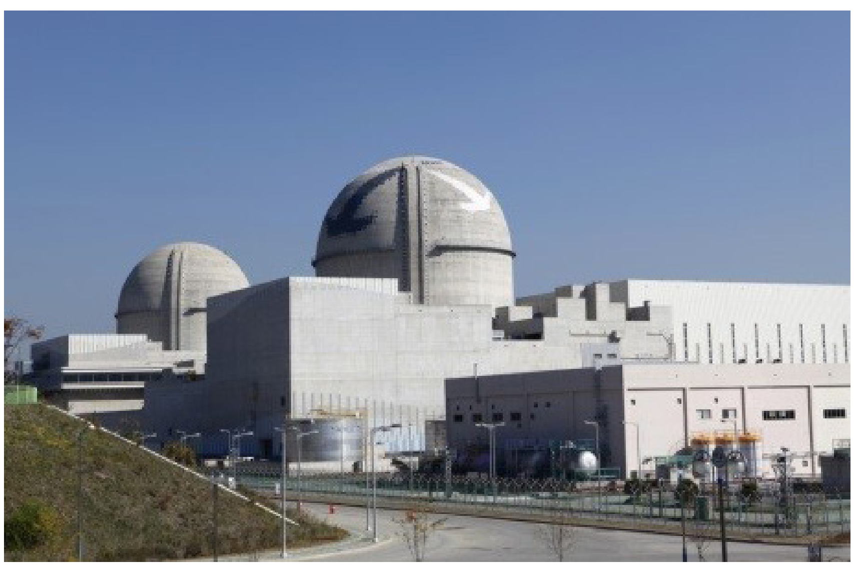
2.6 Standard Design Large Pressurized Water Reactors

This class of PWRs is represented by conventional design reactor plants. The major innovations are increased power output, additional trains of emergency equipment to provide water to the core, more advanced digital controls, filtered vented containments for severe accident mitigation, and more efficient power conversion systems. These reactors include the French European Pressurized Reactor (EPR); the Chinese CAP-1400, based on Areva's 1,000 MWe plant; the Korean APR-1400; the Mitsubishi, Advanced China PWR, ACP-1000; and the Russian VVER-1200. These evolutionary standard designs take advantage of the experience gained from operating the past generation of plants.

2.6.1 APR-1400 Korea

Korea Hydro & Nuclear Power (KHNP) has established a strong foothold in the nuclear market by first completing the construction of the evolutionary 1,400 MWe pressurized reactor plant in Korea. The Shin Kori plant was recently connected to the grid. Three other such plants are under construction in Korea and four in Abu Dhabi. These plants are derivatives of the original Combustion Engineering design of the 1980s, System 80. They incorporate the enhancements described earlier. See Figure 2.9.

Figure 2.9 Shin Kori Nuclear Power Station



Source: KHNP

Key design features of the APR-1400 are shown in Table 2.7 below.

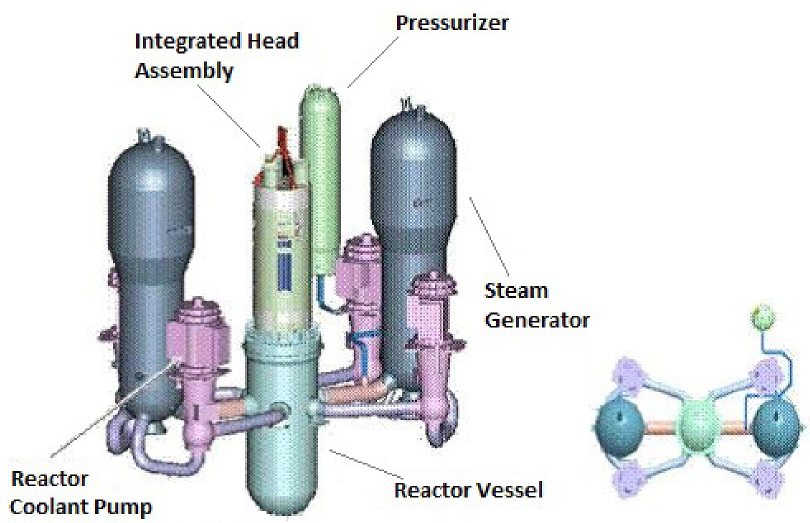
Table 2.7 Key Design Features of APR-1400

Reactor Thermal Power	3,983 MWth
Reactor Electrical Power	1,400 MWe
System Pressure	2,250 PSIA
Core Inlet/Outlet Temperature	555°F/615°F
Number of Fuel Assemblies	241
Fuel Assembly Length	12.5 ft.
Core Damage Frequency	< 10E6
Emergency Safeguards	Active (4 Independent Trains)
Steam Generators	2
Main Coolant Pumps	4
Containment	Single
Refueling Interval	18 Months
Construction Period	45–51 Months

Source: International Atomic Energy Agency

The basis configuration of the APR-1400 is a traditional PWR, as shown in Figure 2.10.

Figure 2.10 APR-1400 Primary System Arrangement



Source: KHNP

The APR-1400 includes additional safety features to manage severe accidents, including direct water injection lines into the reactor vessel as well as depressurization systems to assist after accident cooling.

The cost of the APR-1400 is best represented by the price paid by Abu Dhabi in the purchase of four such units. Quoted estimates are in the range of $20 billion for four reactor plants that generate 5,600 MWe[15] in aggregate. The plant is considered a stick build, and it uses traditional nuclear construction methods.

2.6.2 The European Pressurized Reactor (EPR)

The EPR is a 4,500 MWth pressurized water reactor that generates 1,660 MWe.[16] Four of these reactors are under construction: one in Finland, one in France, and two in China. Shown in Figure 2.11 below is the first EPR. It started construction in Olkiluoto, Finland, and is expected to be completed in 2018.

Figure 2.11 Olkiluoto EPR Under Construction

Source: Areva/EDF

The EPR builds on the combined expertise of Areva and Siemens to design a pressurized water reactor that increased safety in a more traditional but robust way. The size of the plant was increased over previous French and German designs to capture economies of scale and to make the plants more competitive. The design boasts more independent and redundant safety systems and a core catcher, should the plant's safety systems fail and deal with potential fuel melt accidents. By adding more systems, this more standard design increases cost and complexity. At present, the Olkiluoto and French Flamanville are seriously behind schedule. High cost overruns are due to some of first of a kind contruction problems.

The basic design details are shown in table 2.8 below.

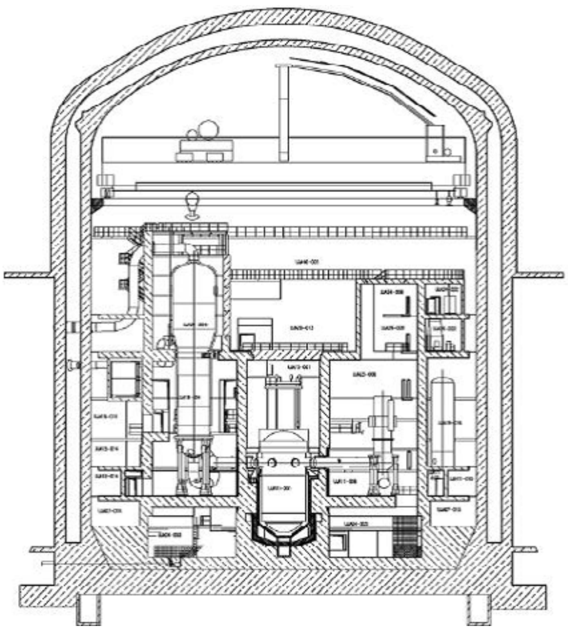
Table 2.8 Key Plant Design Features of EPR

Reactor Thermal Power	4,500 MWth
Reactor Electrical Power	1,660 MWe
System Pressure	2,250 PSIA
Core Inlet/Outlet Temperature	295.6°C/329.8°C
Number of Fuel Assemblies	241
Fuel Assembly Length	480 cm
Core Damage Frequency	5×10^{-7}
Emergency Safeguards	Active (4 Independent Trains)
Steam Generators	4
Main Coolant Pumps	4
Containment	Double
Refueling Interval	18 Months
Construction Period	5 Years[17]

Source: International Atomic Energy Agency

A unique feature of the EPR design is the double containment structure and the addition of a core-melt retention system to mitigate the damage caused by severe accidents. Shown in Figure 2.12 below is a cutaway drawing of the plant's reactor containment system.

Figure 2.12 Cross Section of EPR Containment



Source: Areva/EDF

Overall, the plant is a standard evolutionary PWR design that has been scaled up from previous French versions. Due to the size of the plant, the economies of scale in terms of power production have yet to be demonstrated relative to their actual cost of construction and operation. It is likely the Chinese Taishan plants under construction might be the first EPRs to produce power in late 2017. This is due to the construction delays experienced in Finland and France. The cost of the two plants is estimated to be $8.7 billion.[18, 19]

What is interesting to note is that the cost of these plants varies tremendously by country. Overall cost depends on cost of construction in the country, regulatory systems, and delays incurred. For example, the two Flamanville plants were estimated to cost $11.8 billion in 2015, and the Olkiluoto plants are estimated to cost $9.5 billion in aggregate.[20] The expectation is that lessons will be learned from past construction experiences that reduce these capital costs in the future.

2.6.3 VVER-1200 Russia

The Water-Water Energetic Reactor (VVER) is an evolutionary PWR based on the VVER-1000 reactor designed by the Organization of General Designer of the reactor plant and OKB Gidropress. The Kurchatov Institute in Moscow offered scientific oversight. The plant generates 3,212 MWth and an electrical output of 1,198 MWe. It meets international norms established by the International Atomic Energy Agency and 2000 international quality assurance standards. There have been over five hundred operating years of experience with the VVER-1000. Shown in Figure 0.13 is the Novovoronezh Nuclear Power Plant-2 in Russia. This was the first VVER-1200, and it went into operation in 2016. There are four other VVER-1200s under construction in Russia, Kaliningrad, and Belarus.[21]

Figure 2.13 Novovoronezh VVER Nuclear Plant

Source: Rosatom

The unique feature of Russian reactors is the use of horizontal rather than vertical steam generators which are found in most PWRs in the world. The plant has newer features, such as active and passive safety systems to deal with accidents and transients. The core damage frequency is estimated to be 10^{-6} per year.

The key plant parameters are shown in Table 2.9.[22]

Table 2.9 Key Plant Design Features of VVER-1200

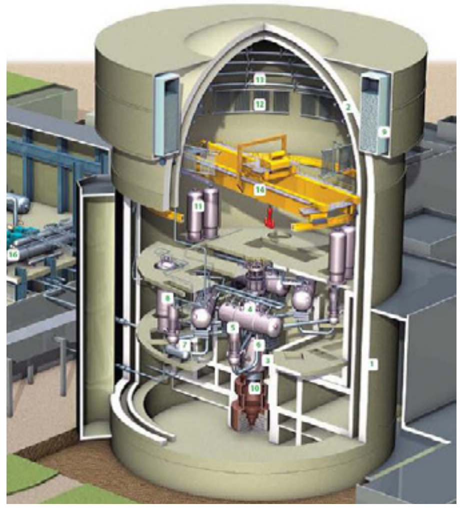
Reactor Thermal Power	3,212 MWth
Reactor Electrical Power	1,198 MWe
Core Inlet/Outlet Temperature	298.2°C/328.9°C
Number of Fuel Assemblies	163
Fuel Assembly Length	457 cm
Core Damage Frequency	7.37×10^{-7}
Emergency Safeguards	Active (4 Independent Trains)
Containment	Double
Steam Generators	4 Horizontal
Main Coolant Pumps	4
Refueling Interval	18–20 Months
Construction Period	54 Months

Source: International Atomic Energy Agency

The plant's design is conventional and has added safety features, including four independent safety trains. These provide redundancy and diversity to deal with plant transients and accidents. The plant also has a double containment system and a core catcher for retention of fuel should a severe accident occur. See Figure 2.14.

Russia is aggressively marketing the VVER-1200 with financial proposals to Turkey, Egypt, Finland, Hungary, and Bangladesh, and it proposes to finance the cost to build, operate, and sell power to local power companies and countries.

Figure 2.14 Cross Section of VVER Containment



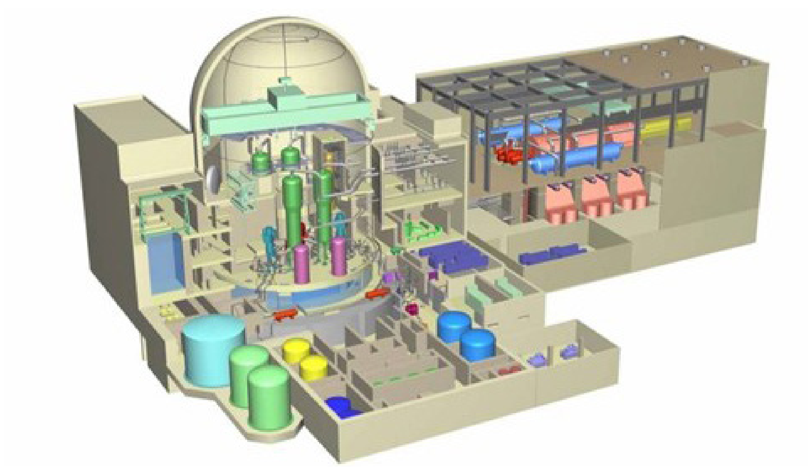
Source: Rosatom

Length of construction for a power unit (beginning with the initial concrete placement and ending with connection to the grid) should not exceed fifty-four months, with a twelve-month interval for commissioning. In 2012, two VVER-1200s (AES-2006) were contracted with Belarus at a cost of $10 billion, or $4,200/kWe.[23]

2.6.4 APWR

Mitsubishi, in collaboration with a number of Japanese utilities and the Japanese government, has designed an evolutionary PWR rated from 1,600 to 1,700 MWe, depending on the market. The APWR is a standard type of pressurized water reactor with improvements in safety capabilities to deal with potential transients and accidents. This includes four independent safety trains to provide emergency core cooling water in the event of accidents involving loss of coolant. The goal of the APWR is to increase the plant's power-generating capability and to reduce the cost of power. An improvement in this area is to increase the efficiency of the power cycle to 39%, above the normal 33% for such plants. Two APWRs have been ordered for the Tsuruga plant site, but it was recently discovered there is an existing seismic fault at the site. This makes construction unlikely. Several APWRs are under licensing review in several countries.[24]

Figure 2.15 APWR Plant Schematic



Source: Mitsubishi

Table 2.10 APWR Key Plant Design Features

Reactor Thermal Power	4,466 MWth
Reactor Electrical Power	1,538–1,700 MWe
Core Inlet/Outlet Temperature	280°C/325°C
Number of Fuel Assemblies	257
Fuel Assembly Length	370 cm
Core Damage Frequency	5×10^{-7}
Emergency Safeguards	Active (4 Independent Trains)
Steam Generators	4
Containment	Single
Main Coolant Pumps	4
Core Damage Frequency	1×10^{-7}
Refueling Interval	24 Months
Construction Period	48 Months

Source: International Atomic Energy Agency

Once the Japanese political situation responds to the Fukushima accident and is resolved, it is expected that the APWR will be used in Japan as the main pressurized water reactor along with the ABWRs.

2.7 Chinese Pressurized Water Reactors

China has aggressively developed nuclear plants, starting with the original 900 MWe French (Framatome) designs in the 1980s and spanning to the present Generation III AP-1000 and the EPR. China's strategy is to purchase the reactor plant from a vendor and modify the design to make it more "Chinese," in the sense that it allows them to own the technology. The two major companies that provided China with reactors were Framatome (now Areva) and Westinghouse. China's goal is to self-manufacture all needed components. Westinghouse has agreed to the transfer of the AP-1000 technology, while the French have not. Early in its development program, China also developed indigenous designs in the 300 to 600 MWe range. Shown in Table 0.11 are reactors that are being deployed and developed in China. Depending on the class, they are essentially copies of other pressurized water reactors described in this paper—from the very large 1,500 MWe to the small modular integral units.[25]

Table 2.11 Chinese PWRs Deployed and Under Development

Reactor Type	Description
CNP-300, -600, -1000	Indigenous Chinese Designs (2 or 3 Loop)
ACP-100	Small Modular Integral Reactor Modeled from CNP Plants
CAP-200, -150, -50	Small Modular Integral Reactor Modeled from CAP Plants
CPR-1000	Upgraded 900 MWe French Original 3-Loop Plants
ACPR-100	Small Modular Integral Reactor
HPR-1000	1200 MWe 3-Loop PWR
CAP-1400	AP-1000 Stretched to 1,400 MWe

Source: International Atomic Energy Agency

The advanced PWRS can be summarized briefly as follows:[26]

2.7.1 CPR-1000

This reactor is the Chinese version of the original Areva three-loop plants, which were first built in the 1980s. They form the bulk of the new nuclear plants under construction in China, but they are restricted from export due to licensing restrictions by the French.

The plant is rated at 1,080 MWe with a design life of sixty years. They use less than 5% enriched uranium fuel. The new plants have digital control rooms and operate on an eighteen-month refueling cycle. Each plant has 157 fuel assemblies that are 4.3 meters long. The calculated core-melt frequency is estimated to be less than 1×10^{-5}. Essentially all the components for this design are made in China. The plant thermal efficiency is aimed at 32.9%. Shown in Table 2.12 are the key technical parameters of the CPR-1000.

Table 2.12 Key Technical Parameters of CPR-1000

Reactor Thermal Power	2,905 MWth
Reactor Electrical Power	1,080 MWe
Core Inlet/Outlet Temperature	292°C/327°C
Number of Fuel Assemblies	157
Fuel Assembly Length	12 ft.
Emergency Safeguards	Active
Steam Generators	3
Main Coolant Pumps	3
Containment	Single
Core Damage Frequency	$< 10^{-5}$
Refueling Interval	18 Months
Construction Period	48 Months

Source: International Atomic Energy Agency

Figure 2.16 shows the generational development of the French-inspired CPR plants, from the original Daya Bay Nuclear Power Plants in the foreground to the Ling Ao twin reactors in the background. The site houses six nuclear power plants in Guangdong Province near Hong Kong.

Figure 2.16 Daya Bay Nuclear Power Plants

Source: China General Nuclear Power Corporation

2.7.2 ACPR-1000

This advanced Chinese PWR is being designed and built by the China Guangdong Nuclear Power Corporation. It is based on the CPR-1000. The Chinese have full intellectual property rights for the ACPR. This design has double containment, a core catcher, and other severe accident mitigation systems to meet post-Fukushima safety requirements. The plant's thermal efficiency is aimed at 33%[27, 28]. Shown in Table 2.13 are the key plant characteristics of the ACPR-1000.

Table 2.13 Key Plant Parameters of ACPR-1000

Reactor Thermal Power	3,500 MWth
Reactor Electrical Power	1,150 MWe
Core Inlet/Outlet Temperature	292°C/311°C
Number of Fuel Assemblies	177
Fuel Assembly Length	3.66 m
Emergency Safeguards	Active
Steam Generators	3
Main Coolant Pumps	3
Containment	Double with Core Catcher
Core Damage Frequency	< 10−5
Refueling Interval	18 Months
Construction Period	48 Months

Source: International Atomic Energy Agency

Figure 2.17 Yangjiang ACPR-1000

Source: China General Nuclear Power Corporation

2.7.3 Hualong-1, HPR-1000

The HPR-1000 series of plants is a combination of the China National Nuclear Corporation's ACP-1000 and the China General Nuclear Power Corporation's ACPR-1000.[29] Both reactors are conventional three-loop pressurized water reactors that use the ACP-1000 core design but have slightly different safety systems. This is based on the company's design. These reactors incorporate the latest international safety standards, including backup passive safety systems, severe accident mitigation systems, and upgraded seismic protection capabilities. The power rating is 1,150 MWe with a fuel cycle of eighteen to twenty-four months and an operating life of sixty years. The plant's thermal efficiency is aimed at 36.6%. Table 2.14 below summarizes key technical parameters.

Table 2.14 HPR-1000's Key Technical Parameters

Reactor Thermal Power	3,050 MWth
Reactor Electrical Power	1,150 MWe
Core Inlet/Outlet Temperature	/301°C
Number of Fuel Assemblies	177
Fuel Assembly Length	3.66 m
Emergency Safeguards	Active with Some Passive Elements
Steam Generators	3
Main Coolant Pumps	3
Containment	Double with Core Catcher
Core Damage Frequency	$< 10^{-5}$
Refueling Interval	18 Months
Construction Period	48 Months

Source: International Atomic Energy Agency

The estimated capital cost in China is expected to be $3,000/kWe, which translates into about $3.5 billion. These reactors are available for export, and the first is to be built in Pakistan.

Figure 2.18 Hualong-1, HPR-1000, Fangchenggang 3 and 4

Source: China General Nuclear Power Corporation

2.7.4 CAP-1400

The CAP-1400 design is based on an upgraded AP-1000 with the cooperation of Westinghouse.[30] The China State Nuclear Power Technology Corporation (SNPTC) supports it. By increasing the plant size, the expectation is that the cost per kilowatt-hour will be reduced while using the same basic passive and modularity principles. The electrical capacity is 1,500 MWe with an eighteen-month fuel cycle at 4.95% enrichment and a design life of sixty years. The plant's thermal efficiency is aimed at 34.5%.

Shown in Table 2.15 are the key technical parameters for the CAP-1400.

Table 2.15 Key Technical Parameters for CAP-1400

Reactor Thermal Power	4,058 MWth
Reactor Electrical Power	1,500 MWe
Core Inlet/Outlet Temperature	/304°C
Number of Fuel Assemblies	177
Fuel Assembly Length	3.66 m
Emergency Safeguards	Active with Some Passive Elements
Steam Generators	3
Main Coolant Pumps	3
Containment	Double with Core Catcher
Core Damage Frequency	$< 10^{-5}$
Refueling Interval	18 Months
Construction Period	50–56 Months

Source: International Atomic Energy Agency

The first CAP-1400 plant, Shidaowan, is under construction in the Shandong Province, next to the high-temperature reactor pebble bed modular (HTR-PM) demonstration. Shown in Figure 2.19 is an artist rendering of the CAP-1400 plants.

Figure 2.19 CAP-1400 Plant Under Construction in the Shandong Province

Source: China General Nuclear Power Corporation

The government estimates the cost to build two CAP-1400s will be in the range of $6.5 billion, with 80% of the components made in China.

2.8 Standard Boiling Water Reactors

2.8.1 Advanced Boiling Water Reactor (ABWR)

General Electric and its partners, Hitachi and Toshiba, have developed an advanced version of the boiling water reactor.[31] Thus far, six are operating in Japan, and two are under construction in Taiwan. The ABWR builds on six generations of boiling water reactors designed by General Electric. The major improvements found in the ABWR include elimination of external recirculation pumps and associated piping, replacing them with bottom-mounted internal reactor pumps; improved depressurization systems to handle accident situations; the addition of electrically inserted control rods that complement the existing hydraulic control units; and an improved pressure suppression system. The ABWR has also included the equivalent of a core catcher to contain melted fuel under severe accident conditions in which all traditional safety systems fail. Additionally, the plant has been upgraded to address the issues identified as a result of the damage caused by the tsunami in Fukushima, Japan.

The plant also has a fully digital reactor protection system and is able to be started up automatically without human action, if desired. The ABWR was the first evolutionary Generation III reactor to be operated. Shown below is the Shika Nuclear Power Plant in Japan.

Figure 2.20 Shika ABWR, Japan

Source: GE/Hitachi/Toshiba

Table 2.16 Key Plant Design Features of ABWR

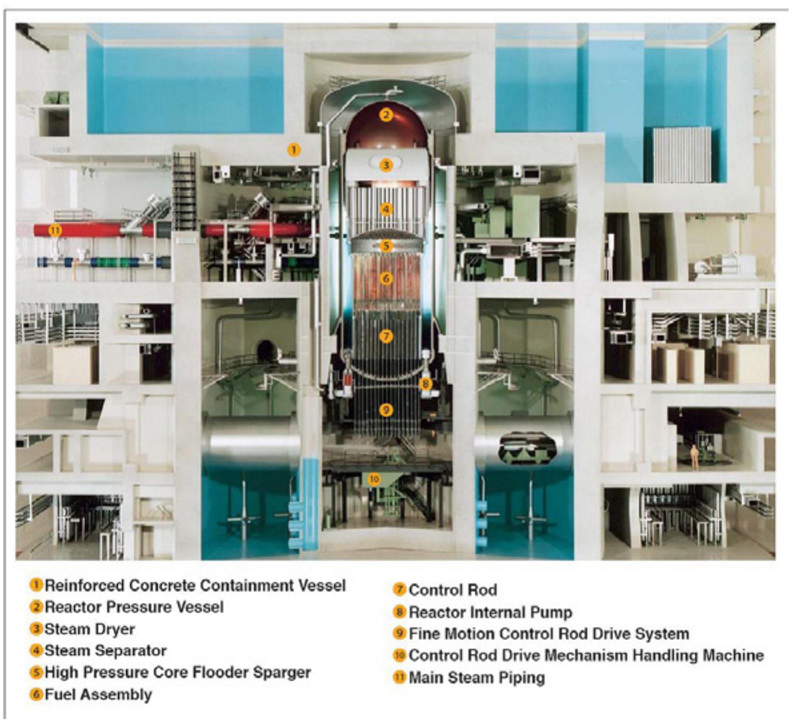
Reactor Thermal Power	3,926 MWth
Reactor Electrical Power	1,371 MWe
Recirculation Pumps	10 (Direct Boiling in Core)
Steam Pressure (PSIA)	7.17 MPa
Core Inlet Temperature	215°C (14.5% Exit Quality)
Number of Fuel Assemblies	872
Fuel Assembly Length	4.47 m
Containment	Single (Wet and Dry Well)
Core Damage Frequency	1.6×10^{-7}
Emergency Safeguards	Active (3 Trains)
Refueling Interval	18–24 months
Construction Period	48 months

Source: International Atomic Energy Agency

The ABWR has the advantage of operational experience, in comparison to the new evolutionary designs. The plants in Japan were built on time and on budget within thirty-nine to forty-one months. Two ABWRs originally planned for the South Texas project were estimated to cost $14 billion.

A cross-sectional layout of the ABWR is shown in Figure 2.21 below. It shows the key features of the ABWR.

Figure 2.21 Cross Section of the ABWR Plant



Source: GE/Hitachi/Toshiba

There are several different versions of the ABWR. General Electric and Hitachi have developed one, and Toshiba has adapted the design by increasing the power level to 1,600 MWe and including a filtered containment vent to meet the regulatory requirements of Europe.

2.9 Small Modular Reactors (SMR)

This class of reactors is largely in response to the high capital cost of light-water reactors, which can be as high as $8 to $10 billion per 1,200 MWe plant. Most small modular reactors are water based since they are the easiest to downsize and are most understood by regulators. This should make the licensing easier, but that's an assumption that has yet to be validated. The new designs are meant to take advantage of modularization in terms of facility size, and to reduce the number of stand-alone components, such as steam generators. Many of these designs incorporate the key components of a pressurized reactor into a single vessel. Typically included are the pressurizer and steam generator, thus simplifying the piping and physical arrangement. Such designs are referred to as integral pressurized reactors since all the major components are in one reactor vessel. These plants range from 35 MWe to 300 MWe. Korea, the United States, and several other countries have developed detailed designs of SMRs. Russia is deploying 35 MWe icebreaker reactors on barges to supply power to remote locations.

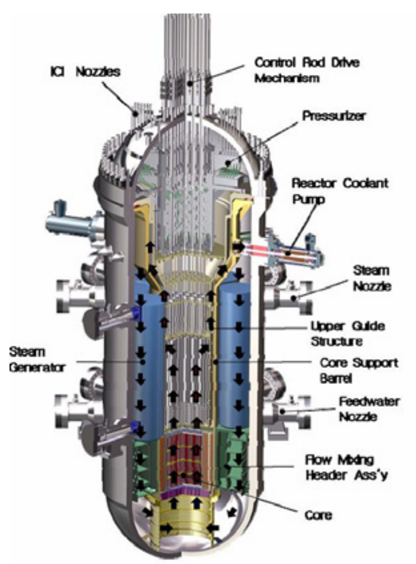
What follows is a brief description of current small modular reactors under development. It lists first those that are more advanced in terms of design and licensing.

2.9.1 SMART

SMART is the first licensed small modular reactor in the world. The Korean Nuclear Safety and Security Commission licensed it in 2012.[32] It is rated at 330 MWth and has a rated electrical power of 100 MWe. This light-water reactor is designed for flexibility. Depending on the configuration chosen, it can produce electricity and heat for desalinization of water for one hundred thousand people. The SMART reactor plant is an integral pressurized water reactor, which means key components are in one pressure vessel. This reduces the amount of external piping systems required and reduces the risk of a loss of coolant accident. Additionally, the plant advertises a construction period of thirty-six months, which is considerably less than the sixty-plus months for larger reactor plants.

The plant utilizes eight helical tube steam generators that produce 30°C superheated steam housed in the reactor pressure vessel. This is shown in Figure 2.22 below.

Figure 2.22 SMART Reactor Vessel



Source: KEPCO

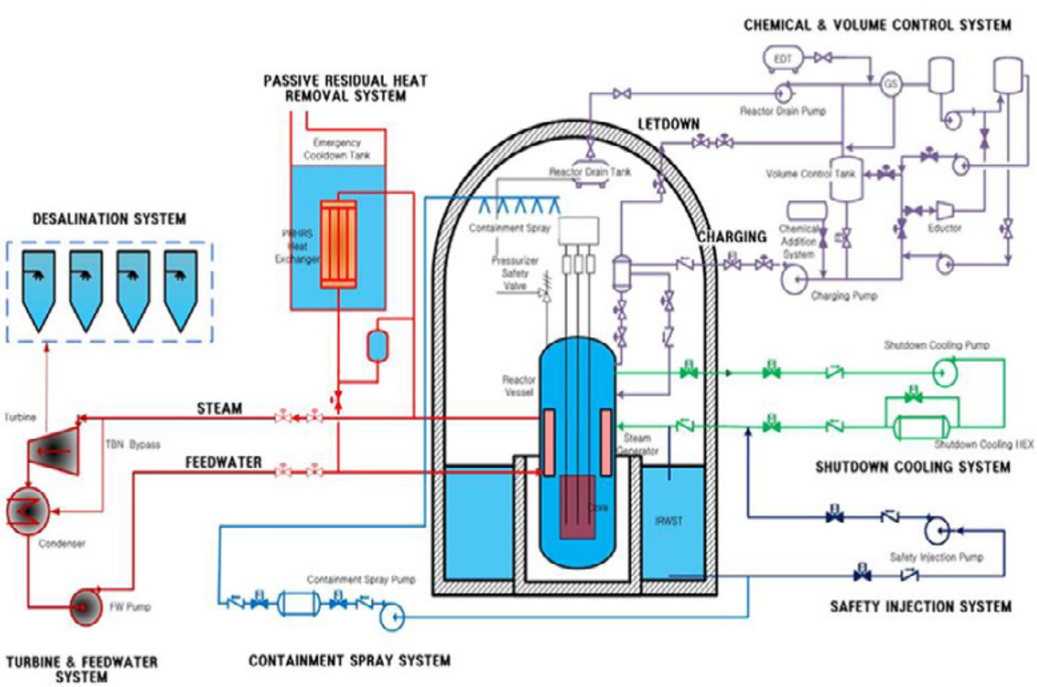
Table 2.17 Key Plant Design Features of SMART

Reactor Thermal Power	330 MWth
Reactor Electrical Power	100 MWe
System Pressure (PSIA)	15 MPa
Core Inlet/Outlet Temperature	296°C/323°C
Number of Fuel Assemblies	57
Fuel Assembly Length	2 m (Active Core Height)
Core Damage Frequency	1×10^{-7}
Emergency Safeguards	Active (4 Trains)
Steam Generators	8 Helical Tubes (Internal)
Main Coolant Pumps	4 Canned Rotor
Containment	Single
Core Damage Frequency	< 10−6 per Year
Refueling Interval	18 Months
Construction Period	36 Months

Source: International Atomic Energy Agency

A schematic of the plant hydraulic systems is shown in Figure 2.23 below. It is essentially a smaller version of a standard PWR, with the exception of the integral pressure vessel housing the core, pressurizer, and steam generators.

Figure 2.23 SMART Hydraulic System Schematic



Source: KEPCO

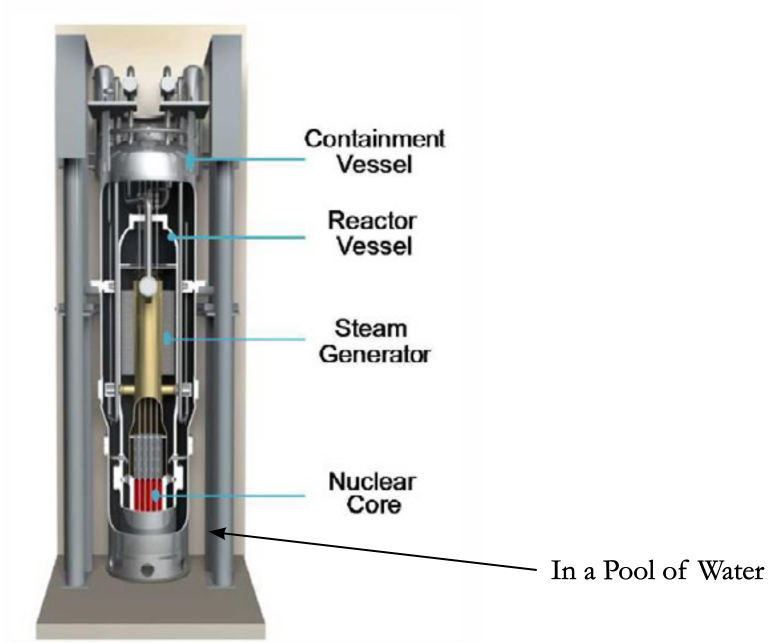
Although the SMART system has been available in Korea for four years, to date no orders have been placed. Korea still relies on large PWR plants for their baseload generation needs. Exports, to date, have not been successful. No cost information is available.

2.9.2 NuScale

The NuScale plant was originally developed at Oregon State University with the Idaho National Laboratory. It is now undergoing licensing by the Nuclear Regulatory Commission.[33] This plant is unique in many ways. The thermal capacity of each module is only 160 MWth with a gross electrical output of 50 MWe. The plant can contain up to twelve such modules that generate 600 MWe. The unique feature of this plant is that all reactor modules are self-contained integral reactors placed in one large pool of water. Once the reactor needs to be refueled, the entire vessel is removed to a refueling area in the large pool. Another already refueled vessel replaces it. Natural circulation cools the reactor, and external pumps and pipes are not needed. Additionally, the entire reactor assembly (power module) can be shipped to the site from the factory by truck, train, or barge. The vessel is sixty-eight feet long and nine feet in diameter, and it weighs seven hundred tons. This type of mass manufacturing should greatly reduce capital costs.

The other unique feature of this design is that the steel containment is evacuated and also underwater. This enhances heat removal capability in an accident. Since the reactor and containment are immersed in water, natural heat removal processes prevent the release of radioactive materials in the unlikely event of a core melt. See Figure 2.24.

Figure 2.24 NuScale Reactor Module



Source: NuScale Power

Key design characteristics of the NuScale reactor are summarized in Table 2.18 below.

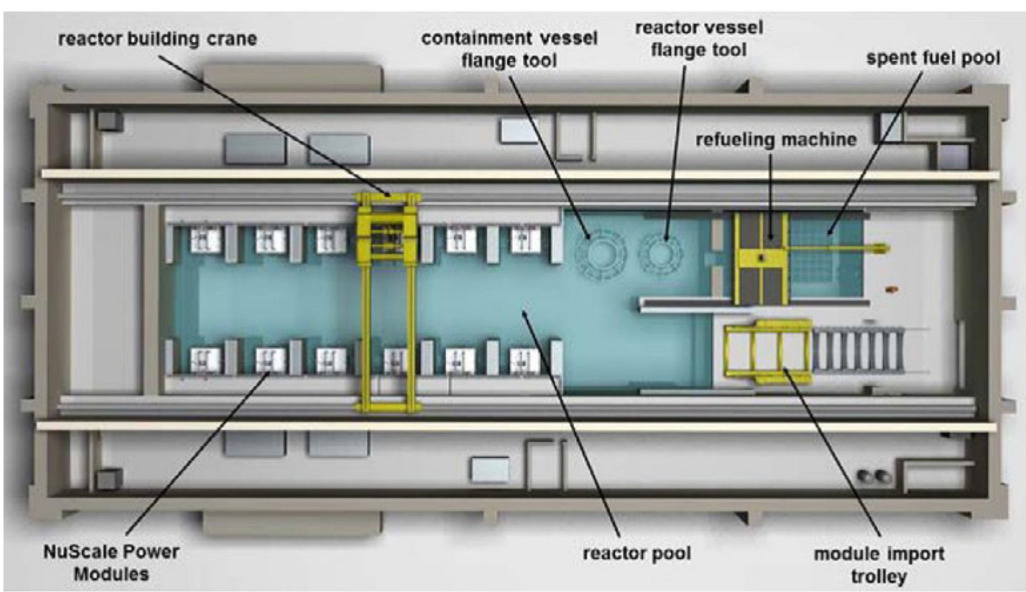
Table 2.18 NuScale Key Plant Design Features per Module

Reactor Thermal Power	160 MWth
Reactor Electrical Power	50 MWe
System Pressure (PSIA)	12.76 MPa
Core Inlet/Outlet Temperature	149°C/302°C
Number of Fuel Assemblies	37
Fuel Assembly Length	2 m (Active Core Height)
Core Damage Frequency	< 10−8 per Year
Emergency Safeguards	Passive Decay Heat Removal and Immersion in Water
Steam Generators	2 Internal Helical Coils
Main Coolant Pumps	0 (Natural Circulation)
Core Damage Frequency	< 10^{-6} per Year
Containment	Single
Refueling Interval	24 Months
Construction Period	36 Months

Source: International Atomic Energy Agency

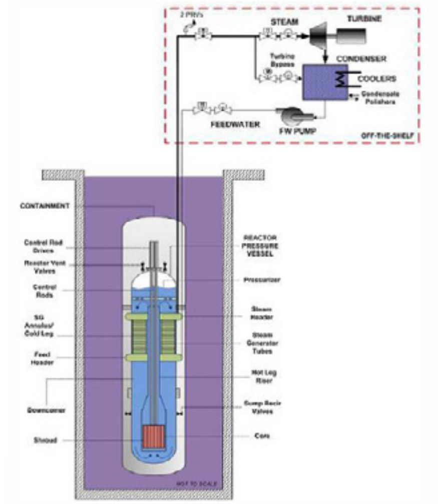
The advertised capital cost is $5,100/kWe for a twelve-module plant, or about $3.1 billion. Each module contains its own power conversion system. This avoids the problems with single-turbine power trains in that other modules can operate while one is being repaired or refueled. The twelve modules are operated from a single control room. Figure 2.25 shows a typical pool configuration for twelve reactor vessels. Figure 2.26 shows the power conversion system for each reactor vessel module. Twelve such turbine plants would be installed for the complete plant.

Figure 2.25 Plant Layout for NuScale



Source: NuScale Power

Figure 2.26 NuScale Power Conversion System



Source: NuScale Power

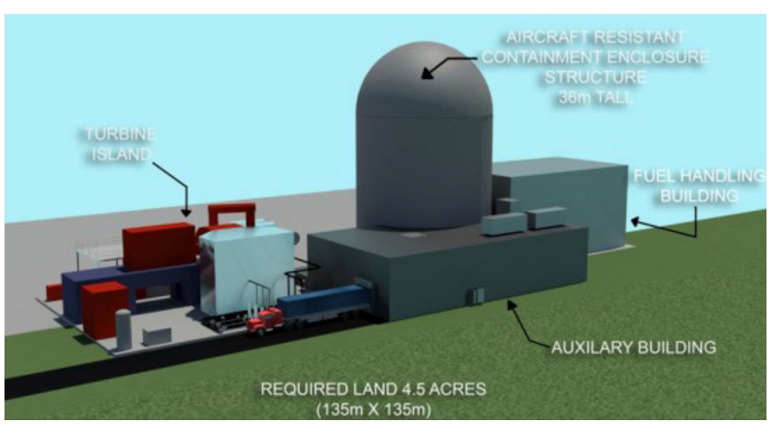
At present, there is a consortium of utilities interested in this technology in the United States.

2.9.3 SMR-160

Holtec International is developing a passively cooled natural circulation reactor generating 160 MWe. This reactor plant is early in its development but claims to be "walk-away safe" since there are no active systems needed for emergency core cooling. The plant electrical rating is 160 MWe, and it is built largely underground. The reactor containment is a tall, slender vessel containing the reactor vessel and an attached steam generator and pressurizer. The reason for the tall vessel is that the core is naturally cooled using temperature density differences of the coolant. The lack of pipes and connection to the reactor vessel are viewed as positive attributes that help avoid accidents involving the loss of coolant. This plant is not an integral PWR since the steam generator is outside the reactor vessel. The reactor core is relatively small and has about sixty fuel assemblies in a pressurized water reactor configuration.[34] The plant operates on a two-batch fuel cycle of eighteen to twenty-four months. The spent fuel pool is housed in the containment. The plant is presently in conceptual design, and many features are subject to change.

Shown in Figure 2.27 below is a conceptual graphic of the plant arrangement.

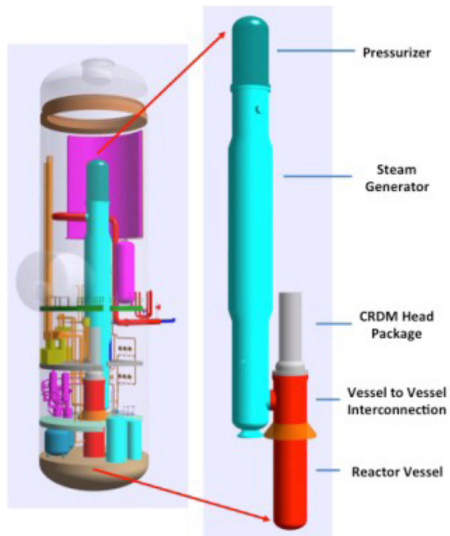
Figure 2.27 Holtec SMR-160 Artist Rendering



Source: Holtec International

The primary system arrangement is shown in Figure 2.28 below. It indicates the two major vessels in the steel containment structure.

Figure 2.28 SMR-160 Primary System Arrangement



Source: Holtec International

Shown in Table 2.19 are the key technical parameters for the SMR-160.

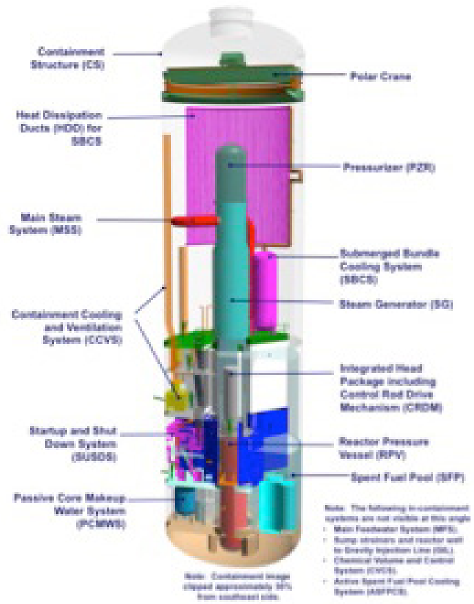
Table 2.19 Key Technical Parameters of SMR-160

Reactor Thermal Power	160 MWth
Reactor Electrical Power	50 MWe
System Pressure (PSIA)	Not Available
Core Inlet/Outlet Temperature (°F)	Not Available
Number of Fuel Assemblies	Not Available
Fuel Assembly Length	6 ft.
Core Damage Frequency	< 10^{-8} per Year
Emergency Safeguards	Passive Decay Heat Removal and Immersion in Water
Containment	Single
Steam Generators	1
Main Coolant Pumps	0 (Natural Circulation)
Core Damage Frequency	< 10^{-6} per Year
Refueling Interval	18–24 Months
Construction Period	36 Months

Source: International Atomic Energy Agency

The steam generator and pressurizer are attached to the reactor vessel. This eliminates primary coolant pipes and pumps. A more detailed look inside the containment vessel is shown in Figure 2.29 below.

Figure 2.29 SMR-160 Containment Internal Arrangement



Source: Holtec International

A unique operating feature of this reactor is that it operates without the use of boron. This is similar to a boiling water reactor, which only uses control rods to control reactor power over the life cycle of the core. This saves on a costly chemical and volume control system, waste processing, and corrosion issues. The steam generator is designed to provide superheated steam to prolong the steam turbine life and offer higher thermal efficiency.

As noted in the above figure, the containment structure has passive heat removal elements in the event of an accident. This is shown on the figure as heat dissipation ducts.

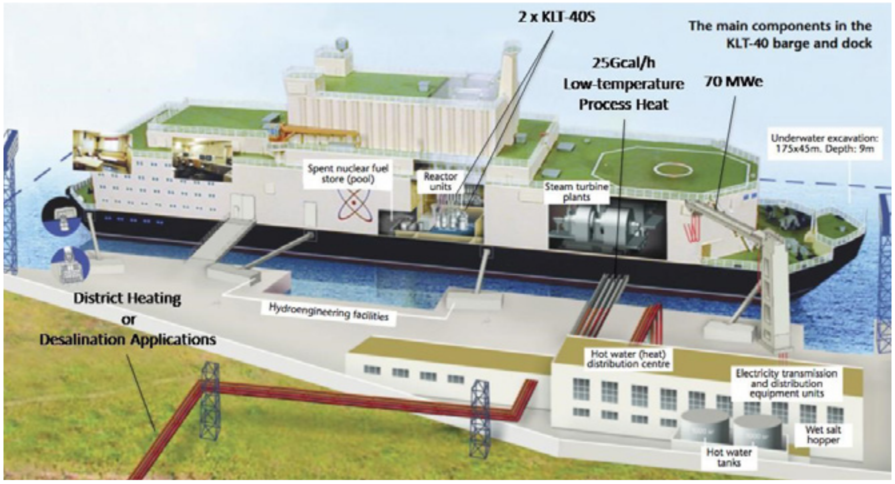
The projected capital cost of this plant is estimated to be $650 million (2015). This plant is presently being privately developed without government support. Holtec International believes it can construct this plant with all component manufactured in the United States in about half the time of a large light-water reactor.

It remains to be seen whether this plant advances beyond the conceptual design stage and can meet the aggressive cost and schedule targets set above.

2.9.4 KLT-40

One of the most advanced small modular reactors is the Russian KLT-40, which is based on Russian icebreakers that use small reactors for power and propulsion.[35] It is Russia's intent to use these types of reactors for barge-mounted power stations for remote locations. Shown below is an artist rendering of a floating nuclear power station. This barge plant has two KLT-40 reactors, each producing 38 MWe from 150 MWth. The plant uses low-enriched uranium as fuel and has an onboard refueling capability and spent fuel storage pool. Every twelve years, the twenty-thousand-tonne barge is sent for a complete overhaul and refueling. The Russian designers estimate a combined 70% capacity factor, which is low for light-water reactors.

Figure 2.30 KLT-40 Barge



Source: Wallstreet Daily

The plant is a four-loop pressurized water reactor. It uses forced circulation with passive emergency core cooling. The details of the design are not available. The fuel is unique and uses less than 20% enriched uranium-235 in a uranium silicide form. This allows for a four-year refueling interval. Another version of this plant, at half the rated power, is a two-loop plant. It has the same enrichment and could have a ten-year refueling interval. Shown in Table 2.20 are the key technical parameters of this plant.

Table 2.20 Key Technical Parameters of KLT-40

Reactor Thermal Power	150 MWth
Reactor Electrical Power	40 MWe
System Pressure (PSIA)	Not Available
Core Inlet/Outlet Temperature (°F)	Not Available
Number of Fuel Assemblies	121
Fuel Assembly Length	1.67 m
Core Damage Frequency	Not Available
Emergency Safeguards	Active and Passive
Steam Generators	4
Containment	Single
Main Coolant Pumps	4
Core Damage Frequency	Not Available
Refueling Interval	28 Months (Batch)
Construction Period	Under Construction

Source: International Atomic Energy Agency

The first floating nuclear plant, Akademik Lomonosov, is under construction. Construction completion is scheduled in 2016.

2.9.5 mPower

Work on this small integrated pressurized water reactor has been halted due to the inability to attract customers or additional investors. Once the leader in US small modular reactor development, Babcock & Wilcox was developing the mPower reactor.

2.9.6 Westinghouse 225

Westinghouse has been developing a small modular reactor intermittently over the last ten years. It is essentially an integral pressurized water reactor similar to others but with a capability of 800 MWth and 225 MWe.[36] They are attempting to use as much of the AP-1000 technology in terms of fuel size and reactor internals as possible with fewer fuel assemblies (eighty-nine). The core is actively cooled using eight main coolant pumps mounted horizontally on the side of the reactor vessel.

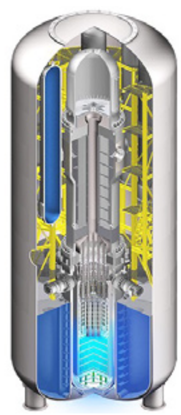
An artist rendering of the Westinghouse 225 (W 225) is shown below.

Figure 2.31 Westinghouse 225 SMR

Source: Westinghouse

As is typical of integral pressurized water reactors, the Westinghouse SMR contains the reactor core, the steam generator, and pressurizer in a 25-meter high and 3.5-meter diameter reactor vessel, which can be shipped by truck or train from the factory. The containment and reactor vessel are shown in Figure 2.32 below. As can be seen inFigure 0.32, the plant has water supplies surrounding the core to enhance the passive emergency cooling in the event of an accident. The containment is designed to be underground.

Figure 2.32 Containment Vessel of Westinghouse 225 SMR



Source: Westinghouse

Shown in Table 2.21 are the technical parameters of the Westinghouse 225 integral small modular reactor.

Table 2.21 Key Technical Parameters of W 225

Reactor Thermal Power	800 MWth
Reactor Electrical Power	225 MWe
System Pressure (PSIA)	Not Available
Core Inlet/Outlet Temperature (°F)	Not Available
Number of Fuel Assemblies	89
Fuel Assembly Length	8 ft.
Core Damage Frequency	< 10^{-8} per Year
Emergency Safeguards	Not Available
Steam Generators	1
Main Coolant Pumps	8 External
Core Damage Frequency	< 10^{-6} per Year
Refueling Interval	24
Construction Period	Not Available

Source: Westinghouse

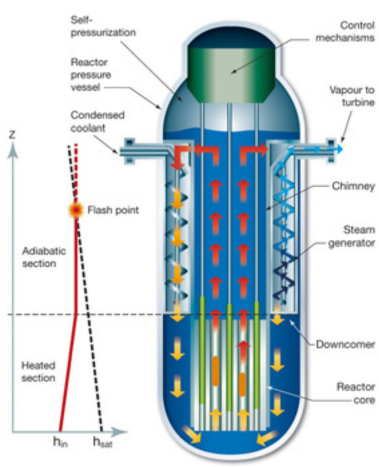
The design life of the plant is sixty years with a twenty-four-month refueling interval.

At present, Westinghouse is looking for teaming partners and customers to continue the design effort.

2.9.7 Central Argentina de Elementos Modulares (CAREM)

Argentina has been working on the design of the CAREM reactor since 1980, and it is now under construction.[37] This is a small modular pressurized water reactor generating 25 MWe. It is expected to open in 2017. The plant is being built at the site of the Atucha I Nuclear Power Plant in Lima, about 110 km northwest of Buenos Aires. The plant is an integral pressurized water reactor similar to others. Shown in Figure 2.33 below is an artist cutaway of the reactor vessel showing coolant flow paths.

Figure 2.33 CAREM Coolant Flow Path



Source: CAREM

Table 2.22 Key Technical Parameters of CAREM

Reactor Thermal Power	100 MWth
Reactor Electrical Power	27 MWe
System Pressure (PSIA)	Not Available
Core Inlet/Outlet Temperature (°F)	Not Available
Number of Fuel Assemblies	61
Fuel Assembly Length	1.4 m
Core Damage Frequency	Not Available
Emergency Safeguards	Passive
Steam Generators	12
Main Coolant Pumps	Natural Circulation
Core Damage Frequency	< 10^{-6} per Year
Refueling Interval	Not Available
Construction Period	Not Available

Source: International Atomic Energy Agency

The plant is a natural circulation plant. It does not require pumps for recirculation of coolant but relies on dynamic pressure and the temperature head of the coolant. An interesting feature of this reactor is that it does not have a pressurizer to control pressure but relies on dynamic feedback pressure control through the thermal hydraulics of the operating cycle. The stability of the thermal hydraulics is complex and will be researched during the demonstration phase of the reactor development. Should the design be proven, a 100–200 MWe reactor plant is planned. The cost of this demonstration plant is about $446 million.

2.9.8 Other SMR Designs in Development

As mentioned in the introduction, there are many reactor designs under development in the small modular reactor category. Table 2.23 provides an indication of the development worldwide. Some of these will be discussed later in the paper. It is quite obvious there is international interest in small modular reactors. Not all these will be developed into real reactors, though.[38]

Table 2.23 Small (25 and More MWe) Reactor Designs at Earlier Stages (or Shelved)

Name	Capacity	Type	Developer
EM2	240 MWe	HTR, FNR	General Atomics, United States
VK-300	300 MWe	BWR	RDIPE, Russia
AHWR-300 LEU	300 MWe	PHWR	BARC, India
CAP-150	150 MWe	Integral PWR	SNERDI, China
ACPR-100	140 MWe	Integral PWR	CGN, China
IMR	350 MWe	Integral PWR	Mitsubishi Heavy Industries, Japan
PBMR	165 MWe	HTR	PBMR, South Africa
SC-HTGR (Antares)	250 MWe	HTR	Areva, France
Xe-100	48 MWe	HTR	X-energy, United States
Gen4 Module	25 MWe	FNR	Gen4 (Hyperion), United States

Name	Capacity	Type	Developer
MCFR	Unknown	MSR/FNR	Southern Co., United States
TMSR-SF	100 MWth	MSR	SINAP, China
PB-FHR	100 MWe	MSR	UC Berkeley, United States
Integral MSR	192 MWe	MSR	Terrestrial Energy, Canada
Moltex SSR	60 MWe	MSR	Moltex, United Kingdom
ThorCon MSR	250 MWe	MSR	Martingale, United States
Leadir-PS100	36 MWe	Lead-Cooled	Northern Nuclear, Canada

3. High-Temperature Gas Reactors

There are two dominant types of high-temperature gas reactors: the pebble bed, formerly introduced by Germany and currently being developed in China, and the prismatic block type, originally developed by General Atomics in the United States and currently being demonstrated in Japan. The operational goal of high-temperature gas reactors is to take advantage of the higher temperatures for increased thermal efficiency and, hence, power production and for process heat applications, which need higher temperatures. The coolant in the reactor is helium gas, which is passed to heat exchangers. This transfers the heat from the helium gas to either steam, for conventional steam electric plants, or directly to high-temperature gas turbines. The most advanced high-temperature gas reactor is a pebble-bed reactor (HTR-PM) currently under construction Weihai, China. The current schedule calls for a start up datein late 2017.

3.1. Nonproliferation

As a class of reactors, high-temperature helium-cooled reactors require higher uranium-235 enrichment. Some reactors have enrichments as high as 19%. This is still considered low-enriched uranium, but the concern exists that the enrichment facility could be modified to produce higher levels. (Up to 95% is needed for weapons.) In terms of reprocessing, the fuel for pebble bed or prismatic reactors is made from tiny uranium particles coated with silicon carbide. This makes extraction of the uranium exceedingly difficult. Generally uranium-fueled high-temperature gas reactors are considered a once-through fuel cycle, and the spent fuel is directly disposed. For pebble-bed reactors, since they are online refueling plants with pebbles discharged during operation, one of the criticisms is that pebbles can be discharged prematurely and diverted. However, the amount of fuel in one pebble is very small. It would require the diversion of tens of thousands of pebbles to make sufficient plutonium for a weapon after reprocessing. Refueling a prismatic reactor is a complex process that requires shielded casks and fuel-handling systems, which are readily detected by monitoring systems of the IAEA.

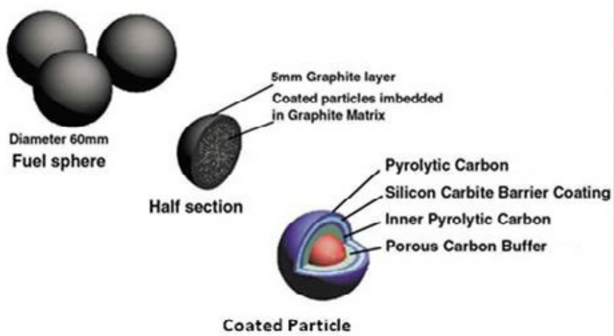
3.2. Nuclear Waste

One of the safety features of high-temperature gas reactors is that the power density of the core is very low. This means that, for the same size power plant output, the size of the core and the amount of fuel needed are much larger. High-temperature gas reactors, therefore, typically have a higher volume of nuclear waste that has to be disposed of in a repository. A study done at MIT[39] addressed this question for the Yucca Mountain repository. It showed that, while the number of waste shipments to the repository would be higher, the space occupied in the repository would be lower for the same amount of power produced by the reactor. This is largely due to the heat generated by the waste being significantly lower when compared to spent fuel from a light-water reactor (lower power density), allowing for closer placement of the waste. Additionally, it was identified that the pebble itself, being a hardened graphite ball containing fuel particles coated with silicon carbide, is a superior waste form. It is insoluble in water and has a very low leach rate, providing an additional safety margin in case of any water intrusion into the repository.

3.3. Pebble-Bed Reactors

The unique feature of pebble-bed reactors is that the fuel is contained in graphite pebbles about the size of tennis balls. Each pebble contains about nine grams of uranium fuel contained in ten thousand silicon-coated particles about the size of poppy seeds. See Figure 3.1.

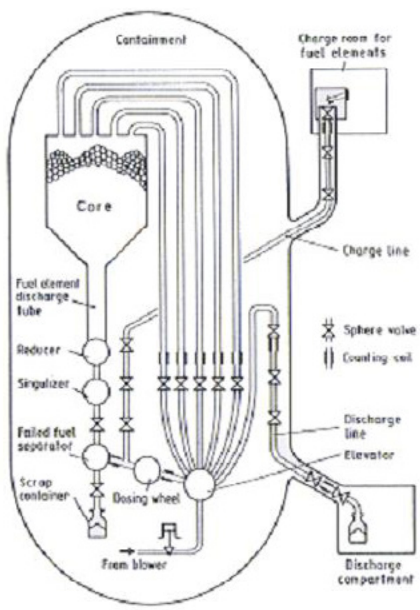
Figure 3.1 Pebble-Bed Reactor Fuel



Source: PBMR South Africa

The fuel is continuously circulated in the core with fresh fuel pebble additions. This maintains constant power operation without shutdowns needed for refueling of other nuclear reactors every twelve to twenty-four months. See Figure 3.2.

Figure 3.2 Pebble Flow Path Schematic



Source: Juelich Research Institute

An important differentiating feature between light-water reactors and helium is that the use of inert helium gas has the advantage of very low activity during operation. The power density is also very low (about 1/30 of a light-water reactor). This is a safety advantage in that the fuel cannot melt down during a loss of coolant accident. To maintain the safety advantage and simplicity in design, pebble-bed reactors are typically limited to about 250 MWth. Depending on the power conversion cycle chosen, they produce about 100 MWe.

Due to the core neutronic design, the pebble-bed reactor naturally shuts itself down without the use of control rods when the temperature gets too high. This makes it an extremely safe reactor. This inherent safety feature provides an additional safety margin for the design.

3.3.1. HTR-PM Pebble-Bed Reactor

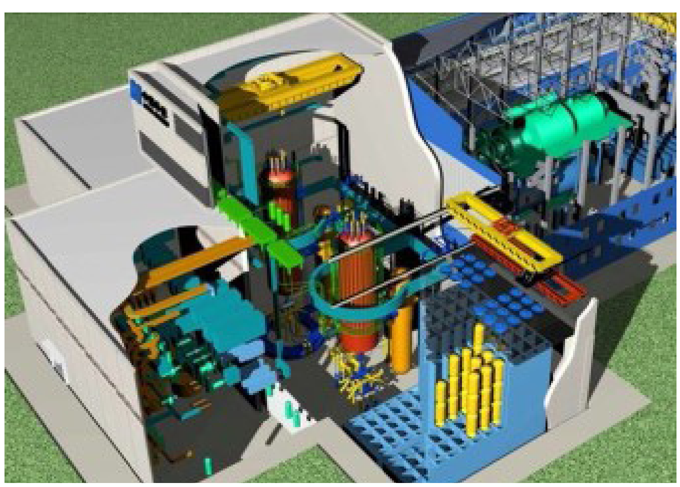
The Chinese have been developing a pebble-bed reactor since the 1990s. This has been largely at Tsinghua University in Beijing. At one point in time, the pebble bed was being developed in South Africa and was considered for the Next Generation Nuclear Plant for the United States, and then both projects were canceled. At present, the only active construction project is in China. The Chinese expect to be able to export this technology once the demonstration project is complete and the reactor has operated for several years.

Tsinghua University's Institute of Nuclear and New Energy Technology (INET) is the technology developer and has conducted extensive tests of the fuel and major components to support the demonstration plant. Most major components have been fabricated in China, allowing for control of the supply chain.

The Chinese pebble-bed reactor builds on the successful test demonstration of the HTR-10 reactor, which went operational in 1999. This 10 MWth reactor had a small steam plant that was able to produce 5 MWe electricity and provided the fundamental knowledge for the design of the HTR-PM demonstration plant. Both plants underwent extensive licensing proceedings with the National Nuclear Safety Administration, which required demonstration of fuel and safety performance. In addition, INET built a special testing facility to test full-scale steam generators, gas circulators, fuel-handling systems, and heat transfer capabilities. Today, China has almost completed construction of two 250 MWth helium-cooled reactors. These will provide heat to two steam generators, which will supply a single 200 MWe steam turbine.

A schematic of a pebble-bed reactor plant layout is shown in Figure 3.3 below.

Figure 3.3 Schematic of HTR-PM Pebble-Bed Reactor



Source: INET

The plant-specific parameters for the Chinese HTR-PM design are shown below.[40]

Table 3.1 Key Plant Parameters of HTR-PM

Reactor Thermal Power	250 MWth
Reactor Electrical Power	100 MWe
System Pressure (PSIA)	7 MPa (Helium)
Core Inlet/Outlet Temperature	250°F/750°F
Number of Fuel Assemblies	420,000 Pebbles
Fuel Assembly Length	Not Applicable
Core Damage Frequency	Not Available
Emergency Safeguards	Inherently Safe (No Meltdown Possible)
Steam Generators	1 per Plant
Containment	Confinement (Vented)
Main Coolant Pumps	1 Helium Blower
Refueling Interval	Online (Continuous)
Construction Period	Under Construction

Source: INET

At present, the plant is nearing construction completion. The scheduled opening is in late 2017. Shown in Figure 3.4 is the loading of the reactor vessel into the HTR-PM reactor building.

Figure 3.4 HTR-PM Under Construction

Source: INET

The smaller size of the pebble-bed reactor makes economics a challenge, when compared to larger light-water reactors. Chinese estimates for power production suggest the cost will be within 20% of the larger pressurized water reactors. This is judged as a reasonable differential for faster construction times and increased safety. These reactors are also more suitable for inland locations due to easier siting and lower heat rejection needs. This is due to higher thermal efficiency (about 45%). The total investment cost for this first-of-a-kind reactor, including all research, development, licensing, and construction, is estimated to be about $1 billion.

From the standpoint of proliferation risk, pebble-bed reactor fuel uses low-enriched uranium (higher than light-water reactor fuel, though), but the amount of fuel in each pebble is only seven to nine grams. To accumulate enough uranium or plutonium from used pebbles, thousands of pebbles would need to be stolen and reprocessed for weapons use. For pebble reactor fuel, the reprocessing task is made much more difficult due to the silicon carbide shells containing the fuel. These shells would have to be removed to gain access to the minute quantities of uranium or plutonium in each microsphere. Thus, the intrinsic and extrinsic barriers to proliferation are higher than for light-water reactors.

The biggest advantage of pebble-bed reactors is the inherent safety. The biggest disadvantage is the low power density of the reactor. This is the reason for the safety advantage, but it makes the reactor vessel quite large (about the same size as a 1,000 MWe boiling water reactor vessel). Thus, while the plant is safer, the power output is much lower, making the economics of the reactor and other small modular reactors a challenge. Most people, however, believe the safety versus economic trade-off favors safety.

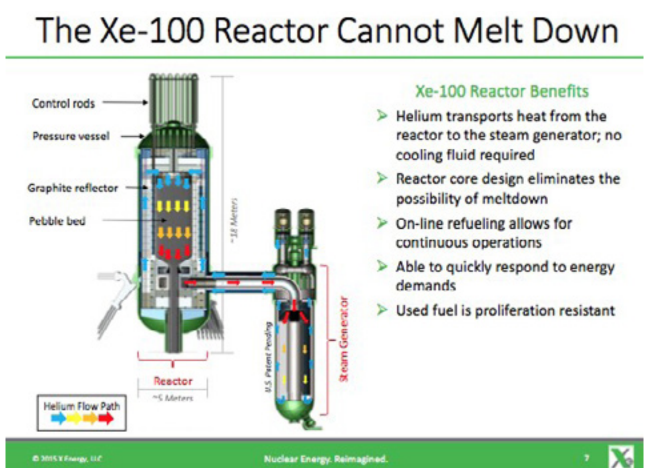
To improve the cost competitiveness of small modular reactors, it will be important to modularize the pebble bed to allow for factory fabrication of plant modules. This would capitalize on economies of production versus economy of scale. MIT performed several studies[41 42] that outline how such modularization can be accomplished for reactors of the 100 MWe size.

Once the demonstration project is complete, the standardized Chinese design should be straightforwardly applied to future projects—especially if modularized. The pebble bed meets the operational safety goal of inherent safety, which includes no possibility of meltdowns and inherent shutdown should the plant overheat.

3.3.2. X-energy

In the United States, a private entrepreneur is developing a pebble-bed reactor very similar to the Chinese version described above, except is it about one-fourth the size.[43, 44, 45] This plant is a single 125 MWth reactor producing steam for a single 45 MWe steam turbine. This is compared to the Chinese twin reactors producing 200 MWe. The designers have relied on technology developed in South Africa and Germany for similar designs. The X-energy plant proposes to use uranium carbide fuel developed by the Idaho National Laboratory with funding from the Department of Energy (DOE). The X-energy fuel contains more silicon carbide fuel particles per pebble (25,000) and fewer total pebbles in the core (170,000). This is commensurate with its smaller size. X-energy recently received a grant from the DOE to continue the development of their reactor concept for the US market. Shown in Figure 3.5 below is a graphic describing its major design goals.

Figure 3.5 X-energy 100 Graphic



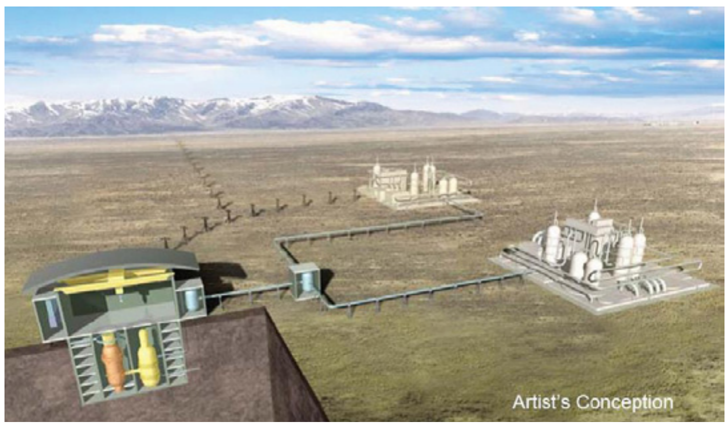
Source: X-energy

3.4. Gas Turbine Modular High-Temperature Reactor (GT-MHR)

General Atomics has been the leader in the development of high-temperature gas reactors for over thirty years. The high-temperature reactor they have developed is a helium-cooled prismatic block–fueled reactor that produces 600 MWth.[46, 47] Due to the use of helium coolant, the reactor is operated at high temperatures. Helium gas is piped directly into a gas turbine, which maximizes the energy value of the coolant. They advertise a 47% thermal efficiency. Their design was the reference design for the US Next Generation Nuclear Plant (NGNP) project which was to be used to demonstrate hydrogen and electricity production. This has been shelved, however, due to lack of commercial interest and conflicts over how future research and development of the prototype should be funded.

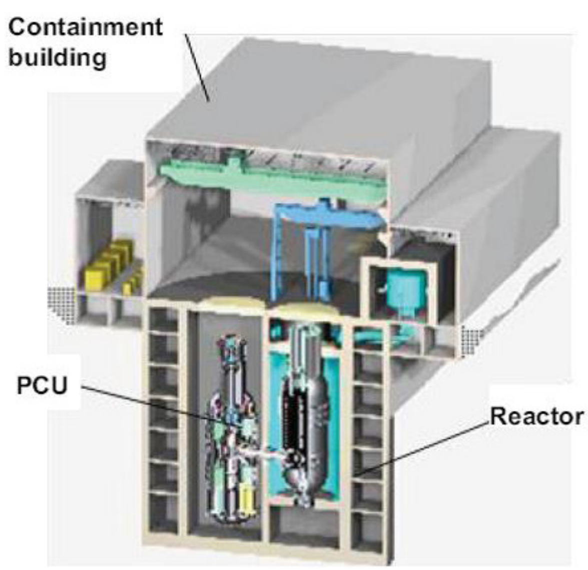
A graphic representation of the plant is shown below.

Figure 3.6 Artist Rendering of NGNP Plant



Source: DOE

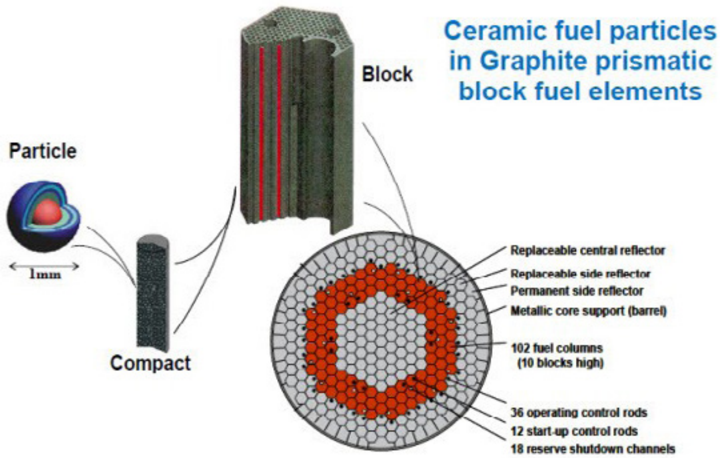
Figure 3.7 Prismatic Reactor Plant Layout



Source: General Atomics

The General Atomics gas reactor design uses the same microsphere silicon-coated fuel particles as the pebble bed, but it configures the fuel into "compacts" (shown below). They are inserted into graphite blocks that form the fuel assembly. These blocks are stacked ten high and make up the reactor core.

Figure 3.8 Prismatic Fuel and Core Design



Source: General Atomics

The GT-MHR is a 600 MWth reactor core that generates about 300 MWe. The key design parameters are shown in Table 3.2

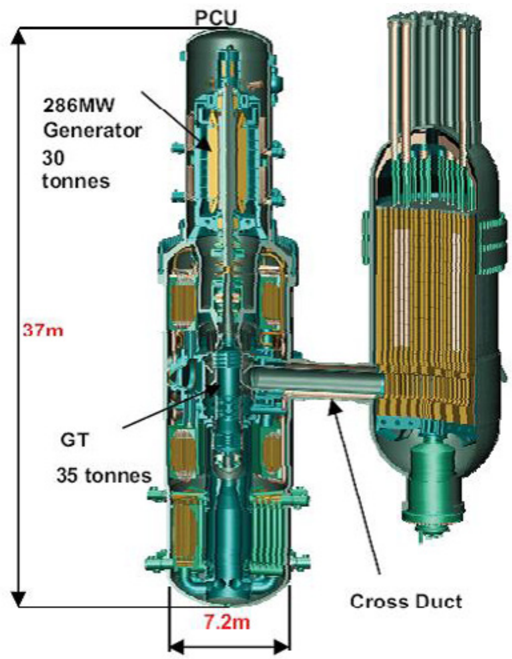
Table 3.2 Design Parameters of Prismatic GT-MHR

Reactor Thermal Power	600 MWth
Reactor Electrical Power	286 MWe
System Pressure (PSIA)	7.07 MPa (Helium)
Core Inlet/Outlet Temperature	491°C/850°C
Number of Fuel Assemblies	102 Fuel Columns × 10 Blocks High = 1,020 Blocks
Fuel Assembly Length	7.9 m (Total Height of 10 Blocks)
Core Damage Frequency	Not Available
Emergency Safeguards	Inherently Safe
Steam Generators	0 (Direct Gas Helium Gas Turbine)
Containment	Confinement Vented
Main Coolant Pumps	1 Compressor
Refueling Interval	18 Months
Construction Period	Not Available

Source: General Atomics

The GT-MHR is a direct-cycle plant that uses a helium gas turbine in a vertical configuration. This is shown in Figure 3.9 below.

Figure 3.9 GT-MHR Vessel Configuration



Source: General Atomics

The technical challenge associated with this design is the direct-cycle gas power conversion system. While the reactor performance has been demonstrated, use of gas turbine technologies of this size has not been done—especially at high temperatures and thrust loads as proposed by General Atomics. Russia has been working with General Atomics to develop gas turbine technology for this type of design.

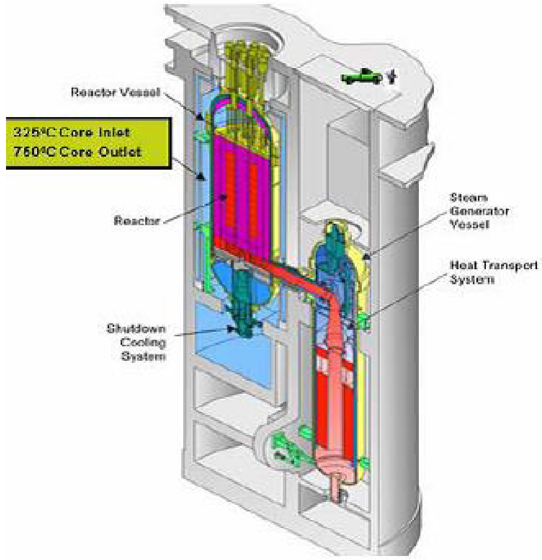
Based on the experience of General Atomics's Fort Saint Vrain reactor, which operated in Colorado, and Peach Bottom in Pennsylvania, the activity level of the helium coolant was very low. This permitted direct-cycle operation (no heat exchangers). The key safety feature of this reactor, as with the pebble-bed reactor, is the particle fuel coated with silicon carbide. Idaho National Laboratory, in conjunction with the Oak Ridge National Laboratory and Babcock & Wilcox, have developed a new uranium carbide fuel. This shows superior performance to the presently used uranium dioxide, which itself is quite good.

At present, General Atomics is not marketing the GT-MHR, but research and development are continuing in Japan on this type of prismatic high-temperature gas reactor. General Atomics has chosen to develop the Energy Multiplier Module (EM2), which is a nuclear waste burner, high temperature fast reactor that will be described later in this paper.

3.5 Antares

Other similar reactors are being developed in Japan and France that use the same fuel form but different power conversion systems. The French ANTARES plant chose to use a more traditional steam cycle instead of the more challenging direct-cycle gas turbine.[48] In choosing this approach, they compromised some thermal plant efficiency for a more proven technology. The plant configuration is pictured below.

Figure 3.10 ANTARES Plant Arrangement



Source: AREVA

The plant uses two steam generators that produce 625 MWth and about 300 MWe if not used for other process heat applications.

Figure 3.11 ANTARES RV and Steam Generator Configuration

Source: AREVA

Table 3.3 Nominal Operating Parameters of ANTARES

Reactor Thermal Power	625 MWth
Reactor Electrical Power	300 MWe
System Pressure (PSIA)	6 MPa (Helium)
Core Inlet/Outlet Temperature	325°C/750°C
Number of Fuel Assemblies	102 Fuel Columns × 10 Blocks High = 1,020 Blocks
Fuel Assembly Length	7.9 m (Total Height of 10 Blocks)
Emergency Safeguards	Inherently Safe
Steam Generators	2
Containment	Confinement (Filtered Vent)
Main Coolant Pumps	2 Helium Gas Circulators
Refueling Interval	18 Months
Core Damage Frequency	Not Available
Construction Period	Not Available

Source: AREVA

The Industry Alliance is using the ANTARES reactor as the reference gas reactor for the Next Generation Nuclear Plant, should it be restarted.

3.6. High-Temperature Engineering Test Reactor (HTTR)

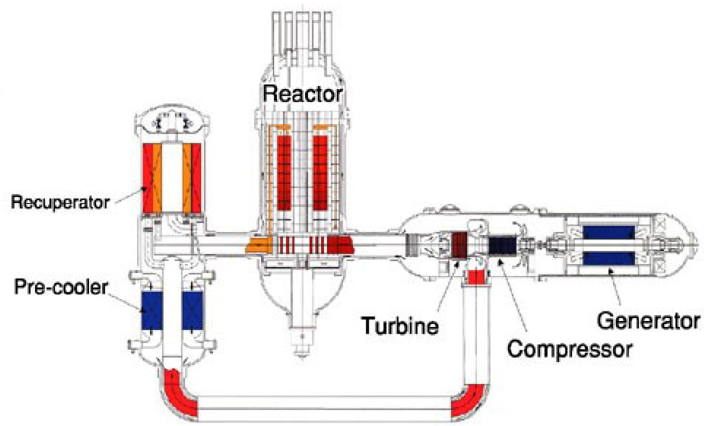
The Japanese version of the high-temperature reactor is based on their operating research test reactor, the HTTR.[49, 50] The HTTR is a 30 MWth prismatic block reactor located at the Oarai Research Center, which started operation in 1999 and was used to test components and fuels up to 950°C. This is considerably higher than current high-temperature reactors, which test to about 750°C, and it's much higher than current light-water reactor coolant temperatures of 310°C. These high temperatures are needed for hydrogen production, which is currently also being researched at the HTTR.

Figure 3.12 HTTR Photo

Source: Japanese Atomic Energy Agency

While much of the nuclear development in Japan has been put on hold, Japan was developing a high-temperature gas turbine reactor using HTTR technology. This prismatic reactor was being designed for 300 MWe.[51] A schematic of the design is shown below.

Figure 3.13 Gas Turbine High Temperature Reactor (GTHTR) 300 Gas Reactor



Source: Japanese Atomic Energy Agency

Table 3.4 Key Technical Parameters of GTHTR 300

Reactor Thermal Power	600 MWth
Reactor Electrical Power	275 MWe
System Pressure (PSIA)	7 MPa
Core Inlet/Outlet Temperature	587°C/850°C
Core Damage Frequency	Not Available
Emergency Safeguards	Passive
Steam Generators	0 (Direct-Cycle Gas Turbine)
Main Coolant Pumps	Turbo Compressor
Core Damage Frequency	Not Available
Refueling Interval	Not Available
Construction Period	Not Available

Source: International Atomic Energy Agency

3.7. Energy Multiplier Module (EM2)

This new reactor plant is a fast reactor that uses helium coolant being developed by General Atomics International. The concept objective is to develop a flexible fuel option that can utilize natural, low-enriched, depleted uranium, thorium, or spent fuel from commercial reactors. The reactor plant is nominally a 265 MWe plant that uses helium gas as a coolant and a direct-cycle gas turbine that produces power with a bottoming organic Rankine cycle that claims thermal efficiencies in the range of 53%. The core outlet temperature is designed to be 850°C. This enables other process heat applications, if needed.[52, 53, 54]

The fuel being proposed is uranium carbide with a composite silicon-coated cladding. For the spent fuel option, the fuel fabrication process starts with spent fuel from which the cladding is extracted. The spent fuel pellets are then pulverized, and an Atomics International Reduction Oxidation (AIROX) process is used to remove fission products.

This pulverized product is then formed into fuel elements. The new design requires fuel elements that are not the traditional GA prismatic graphite blocks but, rather, conventional pin-type fuel rods in a tight lattice configuration, as shown in Figure 3.14 below.

Figure 3.14 EM2 Fuel Assembly

Source: General Atomics

Given the many fuel options, each core needs to be specifically designed for the fuel type. In general, though, the core is a breed-and-burn fuel cycle, which analysis shows allows for a thirty-year core life without refueling or shuffling. This is a difficult target to meet given the material challenges such fuel would have to endure during this time period. Shown in Table 3.5 are the key available technical parameters for the EM2 reactor plant.

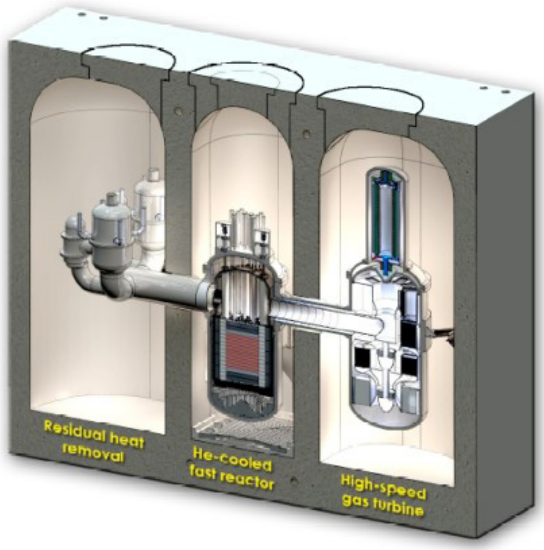
Table 3.5 Key Technical Parameters of EM2

Reactor Thermal Power	500 MWth
Reactor Electrical Power	265 MWe
System Pressure (PSIA)	6 MPa (Helium)
Core Inlet/Outlet Temperature	500°C/850°C
Number of Fuel Assemblies	21 in 17 Stacked Layers
Fuel Assembly Length	Not Available
Emergency Safeguards	Inherently Safe
Steam Generators	Direct-Cycle Gas Turbine
Containment	Confinement (Filtered Vent)
Main Coolant Pumps	2 Helium Gas Circulators
Refueling Interval	30-Year Target as a Waste Burner
Core Damage Frequency	Not Available
Construction Period	Not available

Source: General Atomics

An artist's rendering of the plant is shown in Figure 2.15. The reactor plant is basically in three steel pressure vaults for power production: the reactor vessel (center), the gas turbine power conversion unit (right), and auxiliary and emergency safety systems (left).

Figure 3.15 EM2 Power Block Configuration

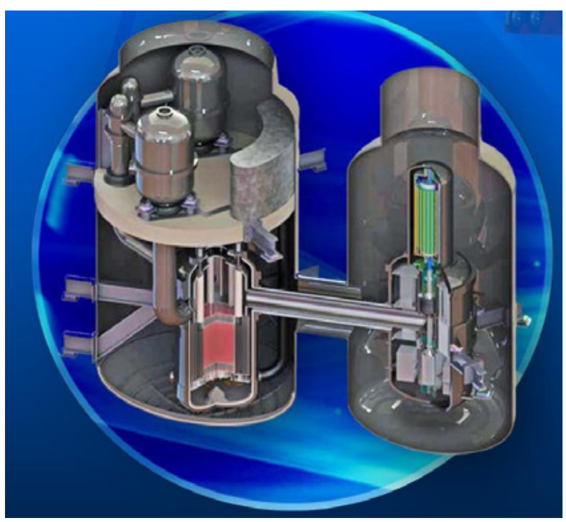


Source: General Atomics

The EM2 overnight capital cost is 3,800/kWe versus 5,000–6,600/kWe for Advanced Light Water Reactor (ALWR) plants. General Atomics has claimed they can build a commercial demonstration plant for about $4 billion in ten years. GA predicts the reactor plant can be made in factory modules and assembled on site in about twenty-one months. They have invested heavily in fuel and cladding materials technology to enable such a prediction.

A more detailed cutaway of the reactor and power conversion system is shown in Figure 3.16.

Figure 3.16 Cutaway of EM2

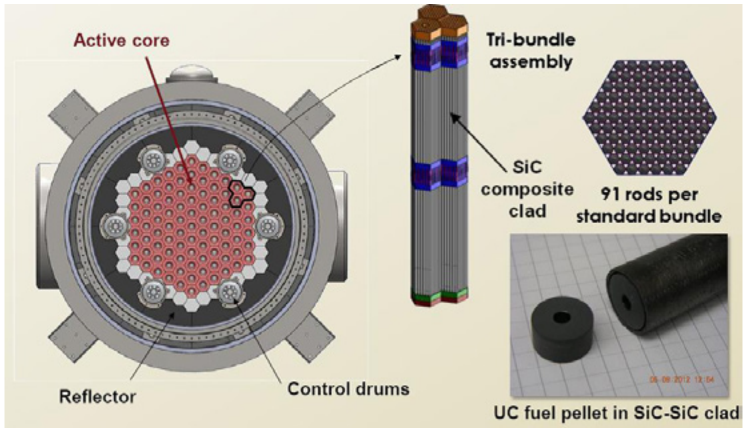


Source: General Atomics

The reactor is a fast reactor, unlike past GA high-temperature gas reactors. This means graphite and other moderating materials need to be eliminated from the design.

The fuel would be configured in a fertile breed-and-burn zone using control drums located on the periphery of the core instead of control rods in the core to control reactivity.

Figure 3.17 EM2 Core Fuel Arrangement



Source: General Atomics

The starter fuel would consist of 12% enriched uranium-235 with a blend of transuranic and mixed oxide fuel from reprocessed spent fuel. Eventually, it would consist of discharged and reprocessed spent fuel from other reactors. The fertile part of the core is designed to be depleted uranium (U-238), natural uranium, spent fuel residues, and thorium. The core design of such a reactor is complex, especially as the designers seek to keep the fuel in the reactor for up to thirty years. The nonproliferation value would be not to have to refuel but to use up the stockpile of spent fuel one would have to reprocess.

In terms of safety systems, GA claims that, due to the very strong negative temperature coefficient, the reactor will shut itself down with increasing temperature. Once shut down, the passive air-cooled residual heat removal system can take care of the decay heat load. In the event of a severe accident or core melt, GA has included a "core catcher" in the design to prevent recriticality.

Due to the deep burn (consumption) of the fuel, most fissile material will be consumed over the life of the plant. That leaves only true fission product wastes to be disposed of, which GA claims is about 3% of that of a light-water reactor.

From a proliferation perspective, the thirty-year fuel cycle is such that no fuel handling would be needed, which reduces proliferation risk considerably.

3.7.1 Nonproliferation

The EM2 has positive proliferation advantages due to its long-life core (up to thirty years). However, to take advantage of this positive feature and provide fuel for the reactor, spent fuel from reactors needs to be reprocessed. The degree of separation of key plutonium isotopes to make the fuel is not clear, but the AIROX process could pose some proliferation risk.

3.7.2 Nuclear Waste

As noted above, the EM2 can be a net waste burner, which leaves about 3% of the nuclear waste from light-water reactors (assuming no reprocessing).[55]

4 Liquid Metal Reactors

A class of reactors that might be important in the future are liquid metal reactors. Interestingly, the first reactor to produce electricity in the United States was the sodium-cooled Experimental Breeder Reactor 1 at the Idaho National Laboratory in 1956. These reactors use either liquid sodium or lead (or lead bismuth) as a coolant. The cores of these reactors use higher energy neutrons than the thermal neutrons used for light-water reactors. For this reason, they are referred to as "fast reactors" and use sodium or lead so as not to slow down neutrons once produced by fission. All use intermediate heat exchangers to transfer the heat generated in the reactor coolant to steam for power production. The unique nature of liquid metal reactors is that they operate at much higher temperatures, which helps increase thermal efficiency of the power production and lower pressure. This makes them easier to build and operate. Additionally, these types of reactors can make more fuel than they consume, if so configured by design of the reactor cores. This "breeding" could become a big advantage in the future should the supply of uranium become tight and the price rise. Sodium-cooled reactors have been built in Russia, France, Japan, and the United States. China is beginning to develop sodium-cooled reactors. Russia has also used lead and lead bismuth reactors for their nuclear submarines. They have abandoned the technology for use in submarines due to difficulties with corrosion due to impurities in the lead coolant.

For liquid metal reactors, the challenge lies in the properties of the coolant. In sodium's case, the liquid sodium is not transparent, and all activities have to be done remotely in the reactor. Tasks, such as refueling, are done without the ability to see the fuel assemblies. In addition, sodium has the unfortunate property of burning in air (moisture), and it reacts violently when in contact with water. The reactor, therefore, requires a cover gas to prevent air intrusion. These challenges make the design more complicated and costly. At present, sodium-cooled reactors are more expensive to build and operate than light-water reactors. The advantage of making more fuel than they consume will only be a factor when the price of uranium get sufficiently high and overcomes the capital cost of construction and operating expense.

Should the sodium- cooled fast reactor be used for breeding, a special "blanket" of uranium- 238, fuel assemblies would be used to make plutonium. To use this plutonium in a recycling mode, either in light-water reactors or as fuel for the fast reactor, the blanket fuel requires reprocessing to extract the still-unused uranium from the reactor core and the plutonium produced in the blanket. While separated plutonium is a proliferation concern, research is under way to avoid creating separated plutonium while still making the by-product useful in light-water reactors or breeder reactors. In both cases, the proliferation threat needs to be monitored to ensure no diversion of this plutonium.

Lead-cooled reactors are chosen for the fast reactors because lead does not burn in air. However, it has corrosive properties and has potential health risks should vapors be inhaled. The challenge with both types of reactors is keeping the metal in a liquid state when the reactors are shut down since the melting temperatures are quite high.

The following are some highlights of the various designs being used or proposed. At present, Russia is the leader in sodium-cooled reactors, and they have just started up another large electric plant. France has also had two large commercial operational reactors (Phoenix and Super Phoenix), but they were shut down due to operational and political problems. In the United States, General Electric has been developing a smaller version of a sodium-cooled fast reactor called PRISM which they hope to build in the United Kingdom.

For a sustainable nuclear energy future, the ultimate depletion of high-quality U-235 will require the use of breeder reactors that make more fuel than they consume. Such reactors could extend uranium reserves for thousands of years by using what are now stored waste by-products of US enrichment plants (mill tailings of U-238). The biggest impediment at present to deployment of fast reactors is the cost. Since they use liquid metals, the capital costs outweigh the cost of uranium at present. Studies[56] have shown about a 20% differential in capital costs over light-water reactors.

A concern for breeder reactors is proliferation risk since the plutonium fuel produced must be recycled in the reactor. The United States had a design (integral fast reactor)[57] in which a breeder reactor, fuel reprocessing, and fuel fabrication facilities were located at the same site, thus minimizing proliferation risk. While there are international monitoring systems capable of detecting any diversions, countries that desire nuclear weapons are still able to use the fast reactors (or research reactors) to produce the plutonium needed for weapons. Fast reactors can also use reprocessed waste from light-water reactors to eliminate plutonium by fissioning it. This could be the ultimate solution to plutonium disposition. It removes plutonium from the fuel cycle, and therefore, it does not need to be secured or disposed. Currently the excess weapons plutonium is considered for recycling in US nuclear plants, and Russia plans to use reprocessed LWR plutonium waste in their fast reactors.

4.1 Sodium-Cooled Fast Reactors

4.1.1 BN-800

The Russian Federation is the current leader in the development and operation of sodium-cooled fast reactors.[58] Russia believes fast reactors are necessary for the sustainability of nuclear energy. This is due to the future diminishing supplies of uranium-235 needed for thermal reactors. Russia has had a sodium-cooled fast reactor of 300 MWe (BN-600) in operation since 1981. In order to fully utilize fast reactors, the country must have a complete fuel cycle. This includes fuel fabrication and reprocessing to extract the still-usable uranium and plutonium from the used fuel. Russia has such capabilities. Russia has two electric-generating fast reactors currently in operation, including the BN-600 and the BN-800. A new generation of fast reactors is currently in design. These reactors are meant to generate 1,200 MWe. Other countries developing fast reactors with operating demonstration plants include Japan, India, and, most recently, China. France once led the world with the largest sodium-cooled fast reactor in operation, Super Phoenix, but it was shut down in 2000 due to operational problems.

The BN-800 is a 789 MWe reactor designed to produce as much fuel as it consumes during operation. Depending on the core designs, fast reactors can produce more fuel than they consume in the breeding fuel cycle.[59] The BN-800 is a standard design sodium-cooled reactor located in Sverdlovsk, Russia.

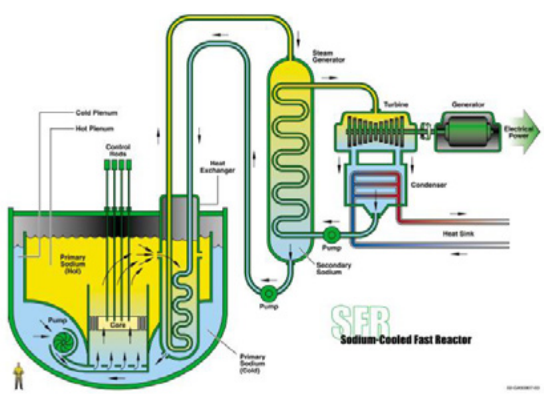
Figure 4.1 Unit 4 of the Beloyarsk Nuclear Power Station in the Sverdlovsk Oblast of Russia

Source: RosAtom

BN-800 is designed not to breed excess plutonium. Rather, it is used as an excess plutonium "burner." This essentially removes plutonium from future proliferation concerns.

The basic design of some sodium-cooled fast reactors is the "pool" type, in which the reactor vessel is submerged in a pool of liquid sodium. This provides a considerable heat sink to remove decay heat. This pool needs to be sealed such that air in not able to come in contact with the coolant or fuel. Sodium is not translucent or transparent, making remote refueling necessary. The conceptual facility schematic is shown below.[60]

Figure 4.2 Schematic of Sodium-Cooled Reactors



Source: RosAtom

As can be seen, two heat exchangers are required to make steam for conventional steam turbines. This is due to the need to keep the radioactive sodium loop in the core separate from the liquid sodium that transfers heat in the second steam generator for the turbines. These additional systems, the complexity of keeping the sodium liquid, and remote operations raise the cost of such reactors to costs that can be more than 10% higher than conventional light-water reactors. The trade-off comes when the price of uranium reaches a level that makes LWRs as expensive as fast reactors to build and operate. Given current market trends in uranium prices, this could be several decades away.

Table 4.1 Technical Parameters of BN-800

Reactor Thermal Power	2,100 MWth
Reactor Electrical Power	880 MWe
System Pressure (PSIA)	0.54 MPa
Core Inlet/Outlet Temperature	354°C/547°C
Number of Fuel Assemblies	Not Available
Fuel Assembly Length	Not Available
Core Damage Frequency	Not Available
Emergency Safeguards	Not Available
Steam Generators	1
Main Coolant Pumps	3
Refueling Interval	140 Days
Construction Period	Not Available

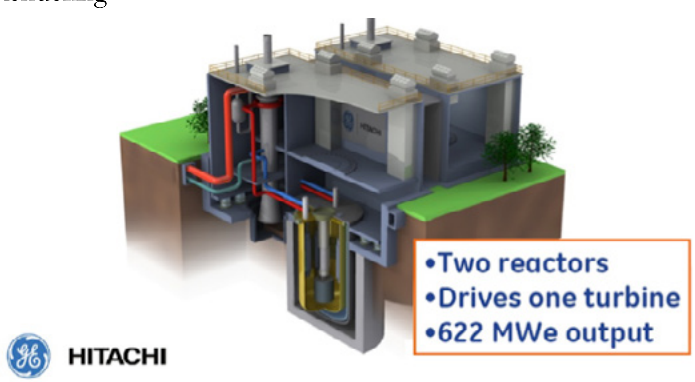
Source: International Atomic Energy Agency

4.1.2 Power Reactor Innovative Small Module (PRISM)

General Electric has been working on sodium-cooled fast reactors since the 1980s. Recently, GE has teamed with Hitachi to develop PRISM, a sodium-cooled reactor designed to consume excess plutonium. This addresses the nuclear waste problem and generates electricity. They have come up with a design in which two small reactors are used to supply heat to steam generators that power a single 622 MWe electric generator.[61, 62, 63] The so-called "block" design is considered modular. It allows for several such blocks to be constructed, providing 1,866 MWe of electric power per plant.

An artist rendering of a GE-Hitachi plant is shown below.

Figure 4.3 PRISM Graphic Rendering



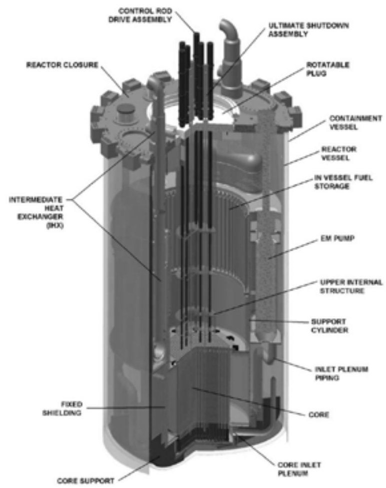
Source: GE/Hitachi

The plant is designed such that it is a net consumer (burner) of fuel. In this case, that fuel is a metallic plutonium and zirconium mixture typical of fast reactor designs. The core is configured like a typical fast reactor. It has a central driver region and an outer blanket. This contains fuel that is bred to make enough fuel during operation to allow the plant to continue to operate for a period of time. The net result is consumption of the plutonium, which is the reason such a reactor is considered to be a means to reduce the risk of proliferation. This is despite using plutonium as a fuel since the plutonium comes from existing stockpiles of plutonium oxide. PRISM is being proposed to the United Kingdom for this purpose.

The value of all sodium-cooled fast reactors is that they are more efficient fuel consumers. Much more energy from the uranium and plutonium fuel can be extracted without the need for additional mining. In a full fuel cycle, including reprocessing and refabrication of used fuel assemblies from light-water reactors or breeder reactors, maximum fuel utilization can occur. That makes this nuclear option sustainable for thousands of years—without relying on the supply of U-235. However, due to proliferation concerns about the use of plutonium at present, such an option is not politically acceptable. This is why GE-Hitachi has chosen to use the reactor to dispose of excess plutonium now in storage.

The PRISM reactor is a pool reactor, much like the BN-800. The reactor vessel is shown below.

Figure 4.4 PRISM Reactor Configuration



Source: GE/Hitachi

The PRISM reactor only contains forty ten-foot uranium oxide metallic fuel assemblies in a nonbreeding configuration. The fuel is expected to stay in the reactor for approximately eight years with about one-third removed every two years. As is typical of sodium-cooled reactors, all operations are remotely handled, including refueling. The key plant characteristics are shown in Table 4.2 for a six-plant configuration that contains three blocks of two reactors each, which is what GE is marketing. Each block (two reactors) will produce 622 MWe using a superheated steam cycle.

Table 4.2 Technical Parameters for PRISM

Reactor Thermal Power	840 MWth per Block (1 Module Has 2 Blocks)
Reactor Electrical Power	311 MWe per Module
System Pressure (PSIA)	Atmospheric
Core Inlet/Outlet Temperature	360°C/499°C
Number of Fuel Assemblies	192
Fuel Assembly Length	.66 to 1.06 m (Depending on Mission)
Core Damage Frequency	Not Available
Emergency Safeguards	Active
Steam Generators	1 per Module
Main Coolant Pumps	4 EM Pumps per Module
Refueling Interval	12–24 Months
Construction Period	Not Available

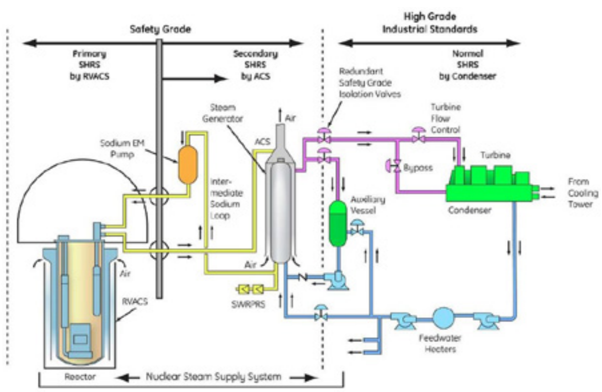
Source: General Electric/Hitachi

The power conversion cycle is shown below. This is typical of fast reactors in general. The goal of GE-Hitachi is to make the power blocks small enough to allow for modularization and enhanced safety. The smaller core size and the large volume of sodium in the guard vessel permit the reactor to be naturally shut down when overheated and cooled using a natural circulation Reactor Vessel Auxiliary Cooling System.

No economic data is available for the cost of the PRISM reactor, but GEH is proposing to build such a plant in the United Kingdom, which will offset the cost of plutonium disposition.[64]

GE is marketing this plant depending on configuration and mission, which can be modified by a different core design within the same vessel envelope. Shown in Figure 4.5 below is the proposed power cycle for a single block.

Figure 4.5 Schematic of PRISM Power Block



HITACHI

Source: GE/Hitachi

4.1.3 TerraPower Traveling-Wave Reactor

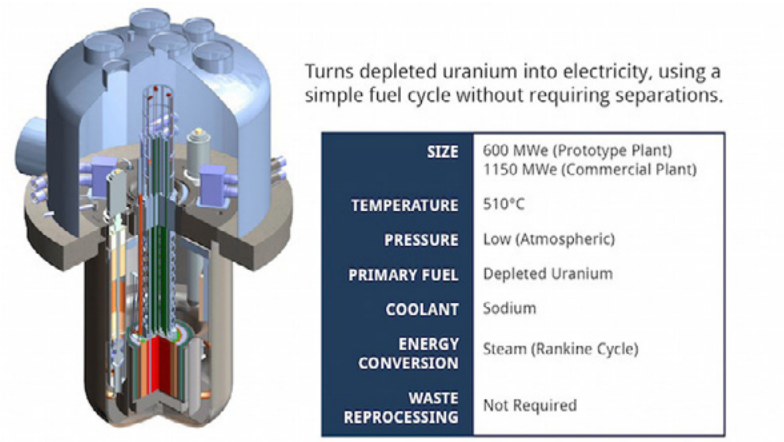
Bill Gates is developing the TerraPower traveling-wave reactor. This will provide a reactor with a long core life without the need for refueling in forty years. The design is a basic pool liquid sodium reactor that utilizes a breed-and-burn core design that makes fuel as it consumes it.[65] The unique core design requires reshuffling every eighteen to twenty-four months to accomplish the forty-year life. A key challenge in this design is to design fuel that can withstand the radiation damage and the sodium environment for forty years. To accomplish this, the fuel is "vented," allowing fission products to escape into the coolant. The gases are removed during operation.

The demonstration plant produces 1,500 MWth and 600 MWe. A graphic of the plant is in Figure 4.6 shown below.

Figure 4.6 TerraPower Plant Rendering

Source: TerraPower

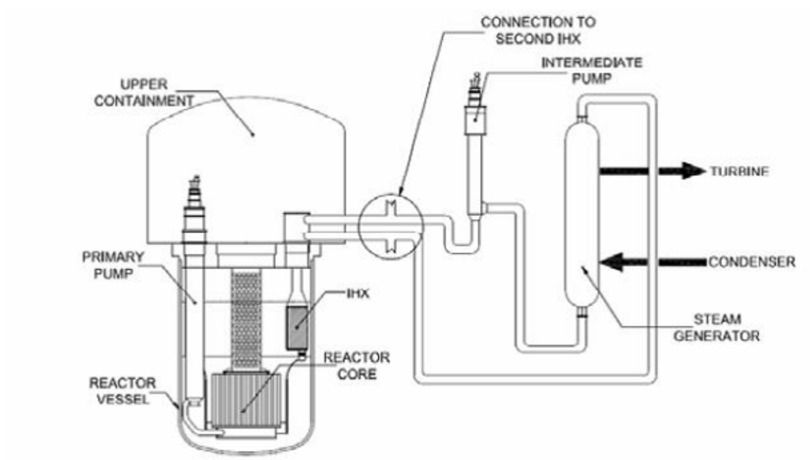
Figure 4.7 TerraPower Reactor Vessel



Source: TerraPower

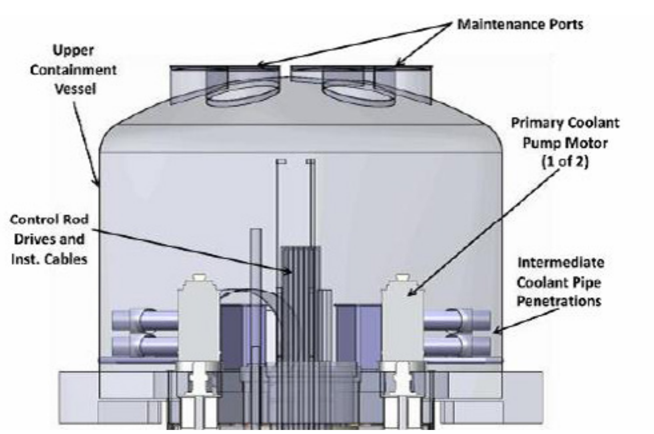
The 600 MWe power conversion system is a standard steam cycle, as shown below.

Figure 4.8 TerraPower Power Conversion System



Source: TerraPower

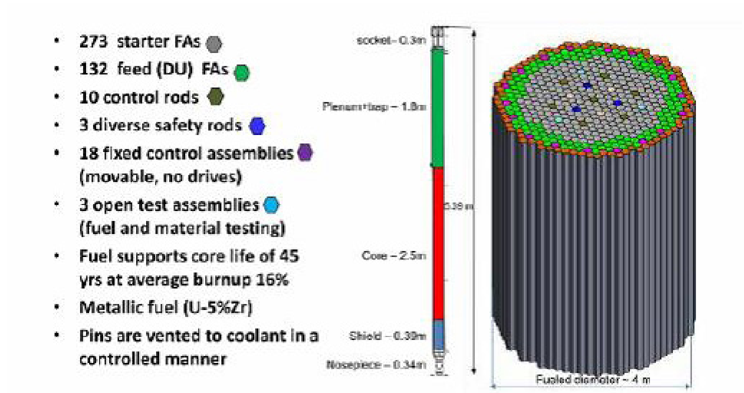
Figure 4.9 TerraPower Containment



Source: TerraPower

As is typical of all pool sodium-cooled fast reactors, all operations must be done remotely due to the need to avoid sodium and air interaction. The fuel shuffling is done automatically and remotely. The core design is complicated since the design has driver assemblies of about 17% enriched uranium fuel and breeding assemblies of depleted uranium, which is mostly U-238. The fuel is a metallic uranium zirconium mixture clad with HT-9. There are 273 starter fuel assemblies and 132 depleted uranium breed assemblies. Over the forty-year life of the core, the breed assemblies will become the drivers as the fuel is shuffled. The fuel assemblies contain 271 wire-wrapped fuel pins, which are 5.6 meters in height.

Figure 4.10 TerraPower Core Arrangement



Source: TerraPower

The plant's key characteristics are shown below.

Table 4.3 TerraPower Technical Parameters

Reactor Thermal Power	1,475 MWth
Reactor Electrical Power	600 MWe
System Pressure (PSIA)	Not Available
Core Inlet/Outlet Temperature	360°C/500°C
Number of Fuel Assemblies	Not Available
Fuel Assembly Length	Not Available
Core Damage Frequency	Not Available
Emergency Safeguards	Not Available
Steam Generators	2
Main Coolant Pumps	2
Core Damage Frequency	Not Available
Refueling Interval	40 Years
Construction Period	36 Months

Source: TerraPower

The present plans call for building the first such plant in China. This is due to the US regulatory system and the extreme difficulty of licensing the first such demonstration reactor. TerraPower has partnered with China National Nuclear Corporation for this project.

The key technical challenges are materials and getting the core to sustain the breed-and-burn cycle efficiently for the expected forty-year core life. Should this design prove technically viable, the need for uranium mining and enriching go down dramatically, and this reduces environmental and proliferation risks. Additionally, since the fuel cycle is "once through," no reprocessing is needed, and the stockpile of depleted uranium can be used for the feed part of the fuel. Depending on future design changes, spent fuel from light-water reactors might be utilized either as driver fuel or feed fuel, but this would require some reprocessing.

No cost numbers have been published, but the price, in terms of capital cost, is likely to be similar to that of other sodium-cooled reactors.

4.2 Lead or Lead-Bismuth Fast Reactors

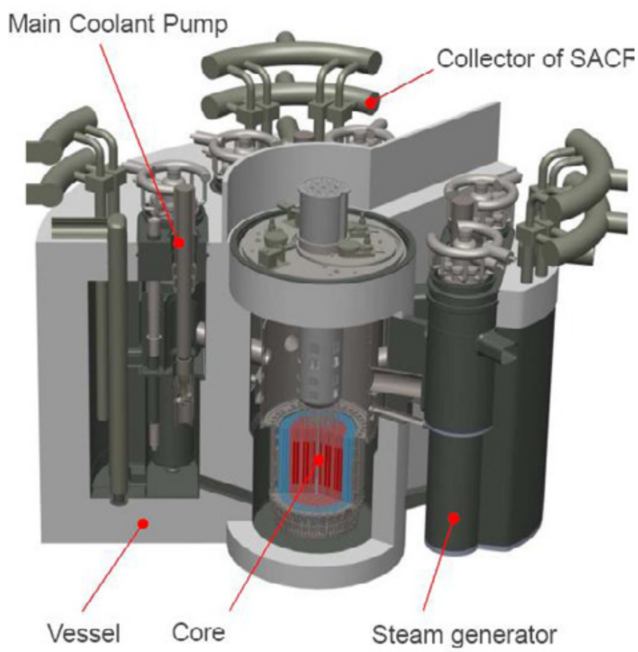
Lead-cooled or lead-bismuth-cooled fast reactors have been developed and used in Russia. The first major application was in the Alpha-class submarines in the 1970s. While technically these reactors have significant advantages over sodium-cooled fast reactors, in that they do not have the chemical reaction challenges sodium poses, they are quite heavy. Lead has a higher melting point, and they are difficult to operate since lead is toxic and, thus, is a hazard to operators. The Russians abandoned lead-cooled reactors for submarines in favor of more conventional pressurized water reactors. The advantage of fast reactors including lead or lead-bismuth is that they are high-power density reactors that allow for smaller reactor cores, which are advantageous for submarines.

The Russians have resurrected the design for land-based applications in a smaller modular form. This is based on their submarine experience using lead-bismuth. The SVBR is the latest version of such a design, and it produces 100 MWe. The Russians are also developing a larger lead-cooled reactor called BREST, which is in the 300 MWe range. The choice between lead and lead-bismuth has several trade-offs. Lead-bismuth has a lower melting temperature (123°C compared to lead's 327°C). This lowers the operating costs since the lead must be heated to those temperatures. However, lead-bismuth, when irradiated, produces polonium, a highly radioactive substance that is extremely hazardous. Additionally, bismuth is in extremely short supply, making the coolant very expensive. The challenge with all lead-based cooling systems is the effect of corrosion on materials, which requires extensive cleanup systems for impurities. Both these reactors are in the research and development phase in Russia.

4.2.1 BREST

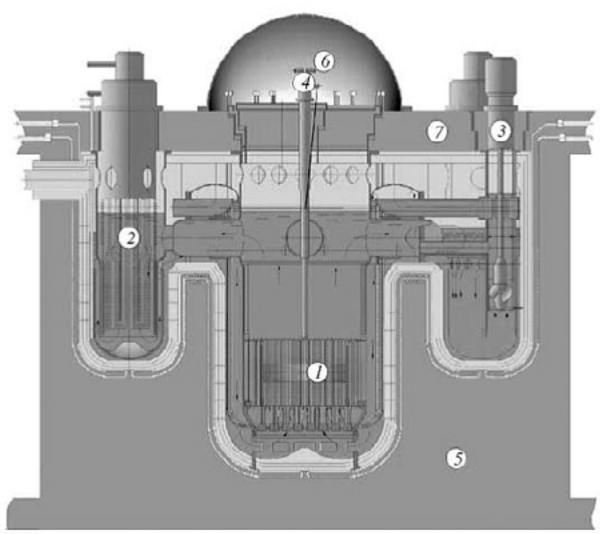
The Russian-developed BREST nuclear plant is a lead-cooled fast reactor designed to breed fuel in a closed fuel cycle system. The Russian Federation is presently designing and testing a 300 MWe version as a demonstration plant for a 1,200 MWe version.[66, 67, 68] Figure 3.11 shows a cutaway of the BREST-300 reactor.

Figure 4.11 Cutaway of BREST-300 Reactor



Source: RosAtom

Figure 4.12 BREST Reactor Configuration



Source: RosAtom

Table 4.4 Technical Parameters of BREST-300

Reactor Thermal Power	700 MWth
Reactor Electrical Power	300 MWe
System Pressure (PSIA)	.003 MPa
Core Inlet/Outlet Temperature	420°C/540°C
Number of Fuel Assemblies	169
Fuel Assembly Length	1.1 m
Core Damage Frequency	$< 10^{-8}$ per Year
Emergency Safeguards	Passive Decay Heat Removal and Immersion in Water
Steam Generators	4
Main Coolant Pumps	4
Core Damage Frequency	< 10−6 per Year
Refueling Interval	5 Years
Construction Period	Not Available

Source: RosAtom

The BREST-300 uses uranium-plutonium nitride fuel with a period of three hundred days between refueling. It has a slightly positive breeding ratio.

The major advantage of using lead over sodium is that lead does not burn in air, and it provides natural shielding of radiation. Lead also has a high boiling point, which is a safety advantage over water-cooled reactors. The higher operating temperature also adds to higher thermal efficiencies. The lack of chemical interactions with water also allows for the elimination of intermediate heat exchangers, simplifying the design and reducing costs. However, the nature of lead requires remote operation for refueling and defueling.

One disadvantage of lead-cooled systems is that they are heavy and require large pumping power for coolant circulation and large structural and system supports. Also, when leaks occur, the lead quickly freezes, making systems inoperable. This might even damage equipment in the process. Lead is also toxic and requires special operating procedures for maintenance and repair.

In terms of safety, the plant design is such that with only two of four emergency core cooling systems in operation, the reactor can be safely cooled down in a station blackout scenario. This avoids a core-melt accident. While this is not as good as some designs of advanced reactors, the presence of lead acts as a heat sink for the heat generated after shutdown.

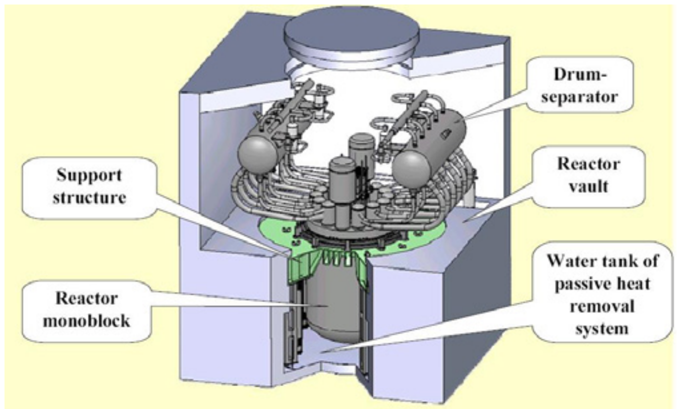
The BREST reactor is designed to operate on a closed fuel cycle, meaning the fuel is reprocessed, and useful uranium and plutonium are returned to the core for recycling. Some might perceive this as a proliferation threat.

4.2.2 SVBR

The SVBR is a 100 MWe Russian-developed design. It is aimed at the small modular reactor market. This design is based on the basic submarine reactor used in the 1970s. An SVBR-type reactor is being designed jointly by Russia's OKB Gidropress, Institute of Physics and Power Engineering (IPPE), and Atomenergoproekt Moscow (AEP).[69, 70, 71] Lead-bismuth coolant was chosen for its lower melting temperature and the Russians' belief in their ability to manage the polonium produced safely. The SVBR developers promote this design for regional power grids to supply electricity, steam for district heating, and industrial applications. They believe the safety of the reactor is such that it could be sited near cities. The designers propose the entire reactor vessel be placed in a water tank. This could provide the necessary decay heat removal in the event of a flow blockage or loss of the main heat removal system. The lead-bismuth is recycled for the life of the reactors, which could be as high as sixty years. Within the same reactor envelope, the core could be fueled with mixed plutonium oxide fuel, either from reprocessed light-water reactors or from excess plutonium from the weapons program. In this mode of operation, the reactor could operate in a self-sustaining closed fuel cycle.

The conceptual layout is in Figure 4.13 shown below.

Figure 4.13 SVBR Reactor Plant Layout

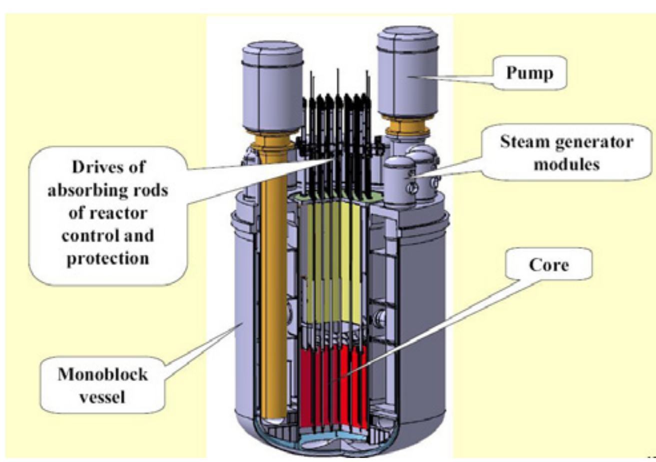


Source: OKB Gidropress, IPPE, and AEP

There are several unique differences in this design that make it attractive for developing nations and for addressing nonproliferation concerns. While the SVBR is a flexible fuel design, the initial application used a uranium oxide core with less than 20% enrichment. The core is designed for a life of seven to eight years, after which the entire core is replaced as a cartridge with the lead bismuth. Additionally, it is an integral reactor with all key components—core, steam generators, and pumps—inside the primary vessel. Third, after the eight years of operation, the entire core of sixty-one fuel assemblies is removed like a cartridge. Fourth, the reactor is small enough to be made modular. The entire reactor vessel and internal components (except the core) can be shipped from the factory.

A more detailed view of the integral reactor design is shown in Figure 4.14 below.

Figure 4.14 SVBR Reactor Vessel



Source: OKB Gidropress, IPPE, and AEP

Table 4.5 SVBR Technical Parameters

Reactor Thermal Power	280 MWth
Reactor Electrical Power	106 MWe
System Pressure (PSIA)	6.7 MPa
Core Inlet/Outlet Temperature	345°C/400°C
Number of Fuel Assemblies	61
Fuel Assembly Length	6 ft.
Core Damage Frequency	$< 10^{-8}$ per Year
Emergency Safeguards	Passive Decay Heat Removal and Immersion in Water
Steam Generators	Not Available
Main Coolant Pumps	2
Core Damage Frequency	Not Available
Refueling Interval	8 Years
Construction Period	Not Available

Source: International Atomic Energy Agency

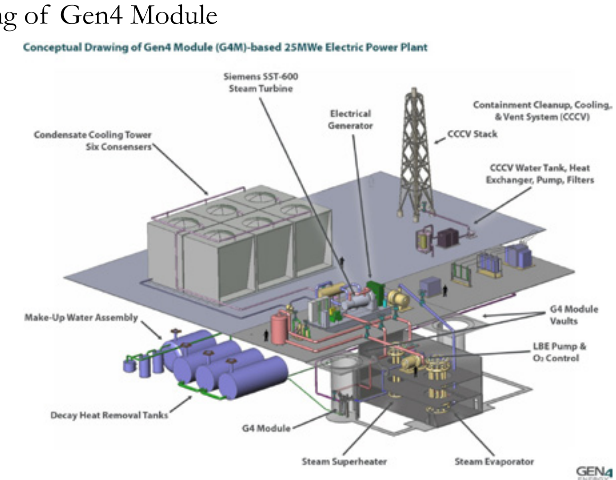
The developers state that the design of the reactor using lead-bismuth permits removal of the intermediate heat exchanger needed in sodium-cooled reactors. It also provides a natural circulation cooling of the core for upsets, which maintains safety function without additional costly backup systems to remove heat. These changes reduce the cost of the plant, but no cost numbers are currently available.

From a nonproliferation standpoint, the fuel cycle is such that access to the fuel would occur only after seven to eight years of operation. Once removed, the fuel cartridge containing lead would be stored in air-cooled vaults, making diversion difficult.

4.2.3 Gen4

The Gen4 reactor is a small liquid metal cooled fast reactor generating 70 MWth of heat and 25 MWe. The design is such that most of it is underground in a containment vault with a ten-year operating cycle. The coolant is a lead-bismuth eutectic.[72, 73] The reactor is designed so that after the ten-year operating cycle, the plant is cooled down for one to four years and then shipped to the shielded factory for defueling and refueling. An artist's rendering of the plant is in Figure 4.15 shown below.

Figure 4.15 Conceptual Drawing of Gen4 Module

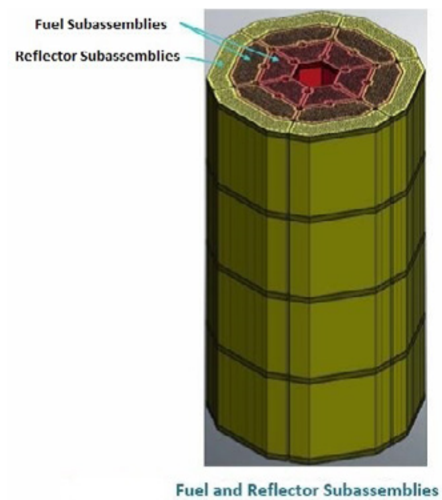


Source: Gen4 Energy

The plant uses a conventional steam cycle. The fuel for the Gen4 is a 19.75% enriched U-235 nitride ceramic fuel. The goal of the designers is to have this plant built in a factory and assembled on site. The outlet temperature of the reactor is about 500°C, which is sent directly to a steam generator.

The reactor core contains eighteen subassemblies, of which twelve are fueled. Six outer assemblies act as reflectors for the neutrons. The diameter of the reactor vessel is 1.6 meters and weighs only four hundred tons when filled with the lead-bismuth. This allows for shipping to the site as an intact, sealed vessel. The length of the fuel assemblies is 2.21 meters. The reason for choosing a fast reactor is that it is much more compact than light-water reactors. This allows for modularity in design and ease of shipment for the same output power.

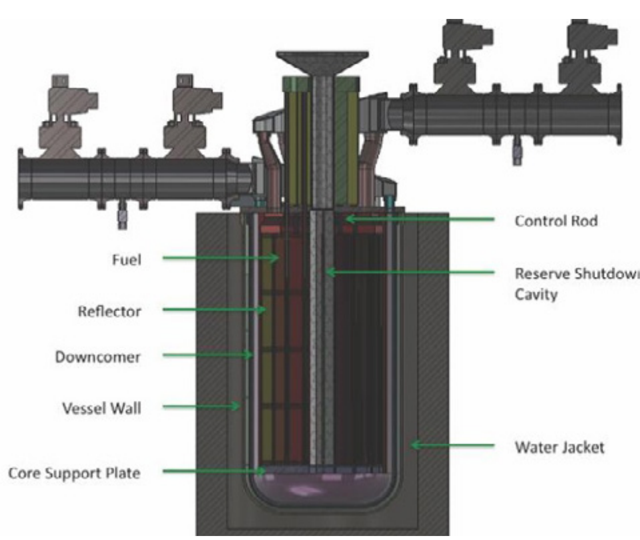
Figure 4.16 Core of Gen4 Design



Source: Gen4 Energy

A more detailed view of the reactor vessel is shown in Figure 4.17 below.

Figure 4.17 Elevation View of Core and Reactor Vessel of Gen4



Source: Gen4 Energy

Table 4.6 Key Technical Parameters of Gen4

Reactor Thermal Power	70 MWth
Reactor Electrical Power	25 MWe
System Pressure (PSIA)	< 0.5 MPa
Core Inlet/Outlet Temperature	400°C/500°C
Number of Fuel Assemblies	12
Fuel Assembly Length	1.7 m (19.75% Enriched)
Core Damage Frequency	Not Available
Emergency Safeguards	Passive
Steam Generators	1
Main Coolant Pumps	1
Refueling Interval	10 Years
Construction Period	Not Available

Source: International Atomic Energy Agency

5 Nuclear Battery

A relatively new development is in the area of a nuclear battery reactor. These reactors are relatively small (in the range of 2 to 25 MWe) and are designed for long-term operation without refueling. The nuclear battery reactors are typically fast spectrum reactors, which use the breed-and-burn fuel cycle to allow for long-term operation. The concept is that after ten to twenty years, the entire reactor vessel and fuel are replaced with a new reactor vessel. This type of reactor is ideal for remote locations and for small grids. Although these designs are still on the drawing board, the most advanced is the Toshiba 4S reactor. The overarching goal for this type of reactor plant is to allow nations to use nuclear energy with less proliferation risk since the reactors are not refueled for ten to twenty years.

5.1 Nonproliferation

As a general rule, nuclear battery reactors are positive responses to proliferation concerns. The attractive feature is that it has a long-life reactor that is not refueled until its operating life of ten to twenty year is complete. Once the operating period is complete, the entire vessel, with the core intact, can be shipped back to the factory for defueling and refueling. This factory can be considered a proliferation risk since it will contain spent fuel and reprocessing or waste disposal processing facilities, depending on the future use of the fuel. It is not clear if the design of such systems has proceeded far enough to outline how such reactors will be safely shipped.

5.2 Nuclear Waste

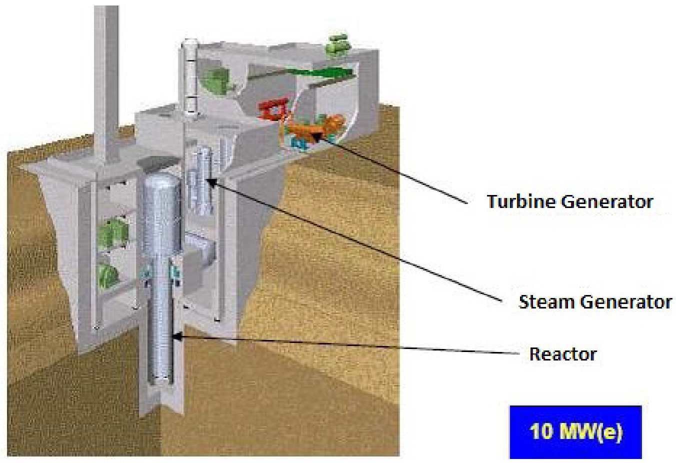
From a nuclear waste perspective, depending on the technology, these reactors have long-life fuel elements that operate for the life of the plant. Simply on that basis, should the spent fuel be disposed of directly and not reprocessed, the waste volume would be lower than light-water reactors. The spent fuel would have to have special handling facilities due to the use of liquid metals that would need to be managed separately. They are also likely to be of higher heat load, making disposal a larger challenge.

5.3 Toshiba 4S

The Toshiba 4S was actively developed for application in Galena, Alaska, to replace costly diesel-generated power.[74] This reactor is a fast reactor cooled by sodium. While two sizes are being offered, 30 MWth and 135 MWth, the design and licensing in the United States were more advanced for the 30 MWth size. US licensing efforts have been suspended because Toshiba was not able to reach an agreement with Galena officials about how to finance the plant.

A graphical representation of the 10 MWe 4S plant is shown in Figure 5.1 below.

Figure 5.1 Artist Rendering of 4S 10 MWe Reactor

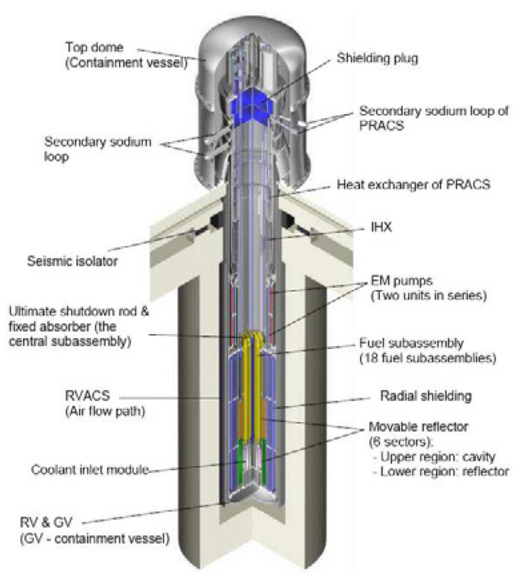


Source: Toshiba

The 4S is a fast reactor using high-energy neutrons. The reason they can operate for twenty to thirty years without core replacement is that the fuel assemblies are very long and have movable side reflectors that follow the reactivity depletion of the core with time. The reflectors reduce the neutron leakage, and this increases the number of neutrons available for fissioning in the active zone of the core.

A cross section of the reactor vessel is shown in Figure 5.2 below.

Figure 5.2 Cutaway of 4S Reactor Vessel



Source: Toshiba

This reactor is an integral reactor in which all important components for conversion of nuclear heat into usable steam for power conversion (intermediate sodium to sodium heat exchanger and electromagnetic pumps) are located in the reactor vessel. The 4S is also defined as a pool-type reactor since it does not have pipes outside the reactor vessel and is surrounded by a sodium pool to assist in decay heat removal in shutdown and accident conditions.

The major design characteristics of the 4S are shown in Table 5.1 below.

Table 5.1 Design Characteristics of 4S

Reactor Thermal Power	30 MWth
Reactor Electrical Power	10 MWe
System Pressure (PSIA)	Not Available
Core Inlet/Outlet Temperature (°F)	Not Available
Number of Fuel Assemblies	18 (19% Enriched U-235)
Fuel Assembly Length	Not Available
Core Damage Frequency	Not Available
Emergency Safeguards	Not Available
Steam Generators	1
Main Coolant Pumps	2 Electromagnetic
Refueling Interval	30 Years
Construction Period	Not Available

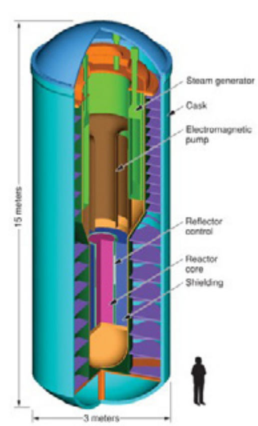
Source: International Atomic Energy Agency

This reactor only has eighteen fuel assemblies with uranium enrichments up to 19%. This allows for a thirty-year lifetime. There are no facilities for refueling at the site, making the reactor a low proliferation risk.

5.4 SSTAR

SSTAR is another battery-type reactor under development in the United States.[75] The size ranges from 10 to 100 MWe, and it uses a liquid metal coolant (sodium, lead, or lead-bismuth). To allow for a long-life core up to thirty years, the reactor is necessarily a fast reactor that incorporates breeding of fuel during operation. The concept calls for a thirty-year operating cycle, and the entire reactor vessel can be transported to and from the site by truck or rail once the thirty-year cycle is complete. A graphic of the reactor is shown in Figure 5.3 below.

Figure 5.3 SSTAR Reactor Graphic



Source: Lawrence Livermore National Laboratory

As can be seen, the reactor is an integral reactor. The pumps, intermediate heat exchanger, and steam generator are in the transportable reactor vessel. The advantage of such a design, from a proliferation point of view, is that it cannot be refueled at the site but must be transported to a special refueling and processing facility.

The key challenges for this and other battery or long-life cores and operating cycles are the material challenges associated with corrosion and erosion of the fuel and internal components. How such reactors will be maintained and repaired is not typically addressed. Lawrence Livermore Laboratory is collaborating with the Central Research Institute for Electric Power Industry (CRIEPI) in Japan to further develop this design. Key technical parameters are shown in Table 5.2.

Table 5.2 Key Technical Parameters of SSTAR

Reactor Thermal Power	45 MWth
Reactor Electrical Power	20 MWe
System Pressure (PSIA)	Natural Circulation Lead Coolant
Core Inlet/Outlet Temperature	420°C/567°C
Number of Fuel Assemblies	18 (19% Enriched U-235)
Fuel Assembly Length	1 m
Core Damage Frequency	Not Available
Emergency Safeguards	Not Available
Steam Generators	0 (Supercritical CO2 Brayton Cycle)
Main Coolant Pumps	0
Refueling Interval	30 Years
Construction Period	Not available

Source: International Atomic Energy Agency

6 Molten Salt Reactors

Molten salt reactors come in two types: those cooled by a molten salt and those that are molten salt fueled (namely a liquid containing the molten fuel). The molten salts used are fluoride based, and some contain lithium, beryllium, sodium, zirconium, or potassium. The goal is to find a salt that is transparent and offers attractive neutronic characteristics but does not absorb many neutrons and is less corrosive. The most common molten salt proposed is FLiBe, 2LiF-BeF2. Early research done on molten salt was done at the Oak Ridge National Laboratory in the 1950s and 1960s. The advantage molten salt reactors offer over liquid metal and light water is that they also operate at essentially atmospheric pressure and at higher temperatures than water. This provides the potential for higher thermal efficiencies. Depending on the molten salt chosen, it can be transparent, which means one can see the fuel being handled—unlike with liquid metals.

For molten-salt-fueled reactors, the fuel is dissolved in with the salt, and the reactor is critical only in the area where there are sufficient reflectors to moderate (or slow down the neutrons), such that fission can take place. At present, these reactors are largely on paper. China is taking the lead on building a small molten-salt-cooled reactor. They have recently approved moving ahead on the design and construction of a small test reactor.

Universities have explored various molten-salt-cooled reactors using graphite pebbles as the fuel. (This is similar to high-temperature gas reactors.) The unique feature of molten salt pebble-bed reactors is that, due to the density of the molten salt, the pebbles float in the salt. This makes online defueling a practical challenge. An additional challenge with molten salts is that they are also corrosive and require high-purity coolant. Research is now under way on corrosion-resistant materials and design features.

The molten salt reactors closest to deployment are at the conceptual design level. The developers at MIT and Berkley are leading the way. Cooperative research programs are being funded by the Department of Energy.[76, 77, 78]

6.1 Nonproliferation

Molten-salt-fueled reactors are unique in that the fuel is dissolved in the salt. To operate such a reactor, online fission product (waste) removal and fuel addition are required. As a thermal reactor, the fuel can be uranium or thorium. Since the fuel is a liquid, it has already been reprocessed to a degree, either from spent fuel or from fresh uranium or thorium. As such, the chemistry of separation poses a proliferation risk at the plant. For the thorium fuel cycle, no plutonium is produced, but U-233 is a necessary by-product of neutron capture, which is needed to sustain the nuclear reaction. U-233, while difficult to handle, is a fissionable weapons-type material. It is more difficult to handle than plutonium, making it less proliferation vulnerable.

The Transatomic design developers argue that the proliferation risk is lower because no new enrichment facilities would be required, and there would be less reprocessing. No actinide separation and reduced spent fuel storage would limit proliferation risk.

Molten-salt-cooled reactors, on the other hand, are similar to other pebble-bed reactors in terms of their proliferation resistance. Typically low-power density-coated TRISO particles, which are difficult to process, and the small quantity of uranium or plutonium contained in each pebble make it a very unattractive target for weapons proliferation.

6.2 Nuclear Waste

For molten-salt-cooled reactors, the waste problem is similar to pebble-bed reactors: high-volume low-heat content spent fuel that can be handled in a repository. This is due to the resistance of graphite to water corrosion and the silicon carbide fuel.

For salt-fueled reactors, the waste issue is not adequately addressed in published works to provide guidance on how such waste will be handled or how the used fuel will be recycled.

6.3 Thorium-Fueled Molten Salt Reactor

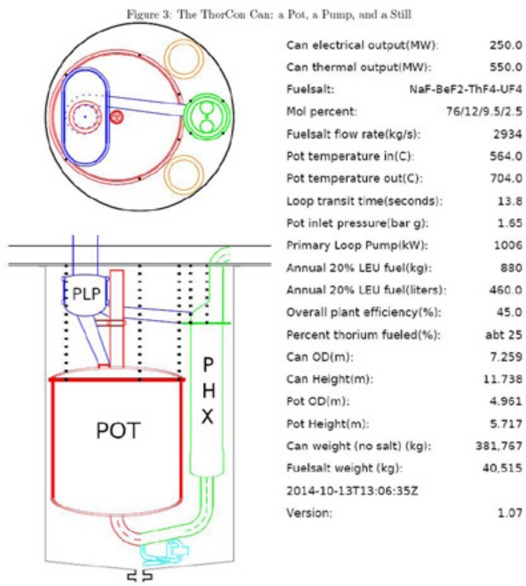
A technology getting a great deal of attention by some is a thorium-fueled light-water or gas reactor. Thorium is a nonfissile isotope that requires a starter core of uranium-235 or uranium-233 bred in a thorium reactor. At present, there are no thorium-fueled reactors in operation, but countries with insufficient uranium resources are looking to thorium as a future fuel for nuclear power plants. Since a thorium reactor is essentially a breeder reactor, a complete fuel cycle capability is needed for reprocessing the fuel in order to provide U-233 for the core of the reactor to sustain a nuclear reaction. Thorium fuel is advertised as more proliferation resistant since the core does not breed plutonium because there is no U-238 in the core. However, the U-233 produced from thorium in the reactor, although difficult to handle from a radiation point of view, is still a material from which nuclear weapons could be made. Advocates additionally point to the advantage of lower levels of high-level waste, but those claims need to be more thoroughly studied if the complete fuel cycle is considered.

6.3.1 ThorCon: Molten-Salt-Fueled Reactor

One such concept for a thorium-fueled molten-salt reactor is called ThorCon. It bases its technology on the Molten-Salt Reactor Experiment (MSRE) tested at the Oak Ridge National Laboratory in the 1960s. The designers claim this reactor is ready to be built and requires no new technology. This is based on the MSRE experience.[79, 80] The basic concept uses canned submodules containing the moderator (in this case, graphite hexagonal blocks). The pumped molten salt flows through this can. The can also contains a pump, a primary heat exchanger, and an off-gas removal system. The salt chosen is a sodium beryllium fluoride salt (NaF-BeF2, NaBe). The claim is that NaBe is available and requires no lithium, which is a neutron absorber. The thorium fluoride is also dissolved in the salt. The ThorCon reactor is a thermal reactor with graphite blocks in the core as a moderator. The expectation is that operators can start the reactor with a U-235 enrichment of less than 5%. The design calls for one or more 250 MWe modules. Each module contains two sealed cans, each housing a 250 MWe primary loop. That includes the reactor (pot), pump, and primary heat exchanger. The concept calls for operating one while the other is in refueling or shutdown mode. This offers additional reliability.

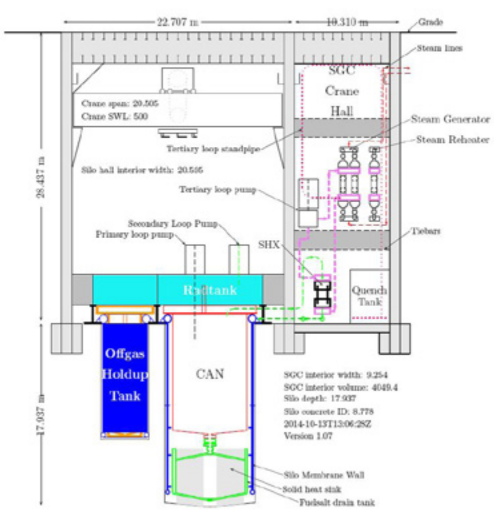
The design is shown in Figure 6.1 below.

Figure 6.1 ThorCon Physical Arrangement



Source: ThorCon Power

Figure 6.2 ThorCon Plant Configuration



Source: ThorCon Power

As can be seen, since the fuel is molten, to shut down the plant in the event of an upset, the fuel is drained from the can into a drain tank, which also has to be cooled by external means.

This design is early in its development cycle. In order to operate without traditional refueling, it needs a means to add fuel (880 kg of 20% enriched uranium) and remove waste produced during the operating cycle. These facilities are not shown in the schematics but will be necessary.

Table 6.1 ThorCon Key Technical Parameters (per Can)

Reactor Thermal Power	550 MWth
Reactor Electrical Power	250 MWe
System Pressure (PSIA)	1.65 bar (NaF-BeF_2-ThF_4-UF_4 Molten Salt Fuel)
Core Inlet/Outlet Temperature	564°C/704°C
Number of Fuel Assemblies	No Fuel Assemblies but 18 Graphite Blocks for Fuel to Flow Through (19% Enriched U-235)
Fuel Assembly Length	5.7 m
Core Damage Frequency	Fuel Is Already Molten
Emergency Safeguards	Drain Tank to Control Criticality
Steam Generators	1
Main Coolant Pumps	1
Refueling Interval	Online
Construction Period	Not Available

Source: Thorcon Power

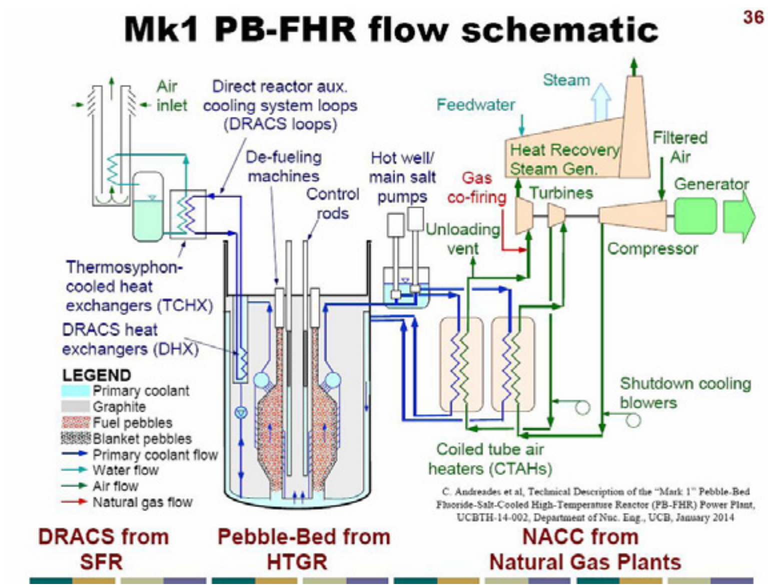
Several key features of this reactor are that it operates at a low pressure, but high temperatures which increase thermal efficiency. The developers believe the cans can be fabricated in a factory and shipped to the site for installation into the silos. The expectation is that the cans will have to be replaced every eight years as intact cans (with a new can installed replacing the old one). The cost projections at this early stage predict a capital cost of less than $800/kWe and an operation cost of $0.03 to $0.05/kWh. These estimates were made in 2014, which was quite early in the design cycle.

6.4 Fluoride Salt-Cooled High-Temperature Reactors

The fluoride salt-cooled high-temperature reactor (FHR) is a reactor being researched at MIT and UC Berkeley.[81, 82] The fundamentals of the technology are based on using a transparent molten salt, FLiBe (Li2BeF4); microsphere TRISO-coated particles, which were developed as part of the high-temperature gas reactor program; pebble-shaped fuel, which is smaller (3 versus 6 cm) than what's used in pebble-bed reactors; and a power conversion system that uses gas turbines. The advantage of this technology is that it uses a low-pressure primary coolant, and it reaches outlet temperatures of 600°C to capture higher thermal efficiencies.

A schematic of the current version of the design is in Figure 5.3 shown below.

Figure 6.3 Pebble Molten-Salt-Cooled Reactor Schematic



Source: MIT and UC Berkeley

The design poses some interesting technical challenges since the pebbles float in the molten salt, and defueling the plant does not rely on gravity. The core design has an annular core with control rods inserted in the center annulus. It is a large pool reactor that requires large heat loads to keep the molten salt from freezing at 460°C. While the inlet temperature rise is only 100°C, the outlet temperature of 700°C helps thermal efficiency for the power cycle. The advantage of lower pumping power and higher heat capacity of the molten salt over helium is counterbalanced by the need to maintain a high-temperature pool of the molten salt for operation and for during shutdowns.

Key technical information is provided below.

Table 6.2 Key Technical Parameters for FHR

Reactor Thermal Power	263 MWth
Reactor Electrical Power	242 MWe (with Gas Cofiring)
System Pressure (PSIA)	Not Available
Core Inlet/Outlet Temperature	600°C/700°C
Number of Fuel Assemblies	Not Available
Fuel Assembly Length	Not Available
Core Damage Frequency	< 10^{-8} per Year
Emergency Safeguards	Passive Decay Heat Removal and Immersion in Water
Steam Generators	0 (Gas Turbines with Cofiring)
Main Coolant Pumps	2
Refueling Interval	Can occur while reactor is online
Core Damage Frequency	Fuel Is Already Molten
Construction Period	Not Available

Source: MIT and UC Berkeley

The power conversion cycle proposed is a Brayton cycle gas turbine, made by General Electric, that generates 242 MWe. China is interested in this product and is presently working on the design of a 10 MWth demonstration plant by 2017 and a 100 MWth design by 2022.

6.5 Transatomic

Transatomic Power is pursuing a new and innovative design built on the principles of the molten salt reactors developed at Oak Ridge.[83] Former MIT students are developing this reactor. They are implementing innovations in the use of zirconium hydride moderator rods instead of graphite and a lithium fluoride salt. While the liquid-fueled reactor is being promoted as a nuclear waste burner (spent fuel from light-water reactors), it can also operate on a low-enriched uranium or thorium fuel cycle.

The power rating for the plant is 1,250 MWth with an electrical output of 520 net MWe. The key characteristics of the design are shown in Table 6.3

Table 6.3 Transatomic Technical Parameters

Reactor Thermal Power	1,250 MWth
Reactor Electrical Power	550 MWe
System Pressure (PSIA)	1.65 bar (NaF-BeF2-ThF4-UF4 Molten Salt Fuel)
Core Inlet/Outlet Temperature	650°C Outlet, LiF-(Act)F4
Number of Fuel Assemblies	Not Available
Fuel Assembly Length	Not Available
Core Damage Frequency	Fuel Is Already Molten
Emergency Safeguards	Drain Plug
Steam Generators	1
Main Coolant Pumps	2
Refueling Interval	Online
Construction Period	Not Available

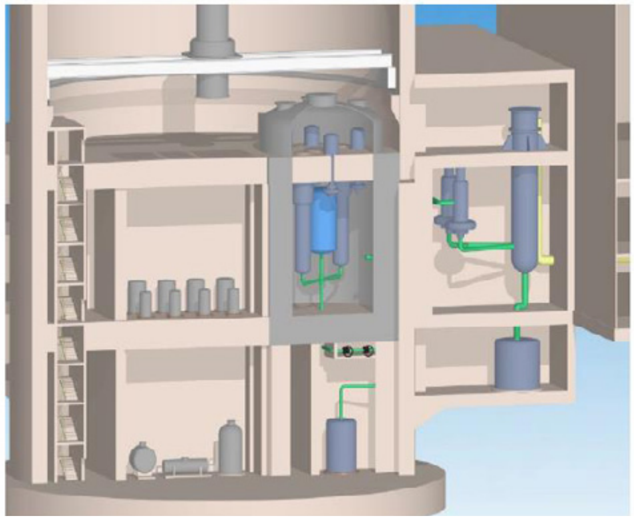
Source: Transatomic Power

The design uses conventional MSR features in that the plant utilizes intermediate heat exchangers to create steam for a conventional steam cycle. The intermediate molten salt loops contain nonradioactive lithium potassium sodium fluoride salts (LiF-KF-NaF).

Additionally, the design employs a low-pressure system with freeze plugs that open if the reactor happens to overheat. This stops the nuclear reaction. The interesting key to this design is that it uses a breeding neutronic cycle by controlling the neutron spectrum in the core. This allows for fissioning fuel in the central region and breeding plutonium in the undermoderated external region. The mixing of both fuel regions during recirculation handles the equivalent shuffling required in a fixed-fuel breeder reactor. While the graphite in the ThorCon reactor needs to be replaced approximately every four years, the zirconium hydride moderator rods also need to be replaced in a somewhat easier manner.

Figure 6.4 below shows a schematic of the conceptual power plant.

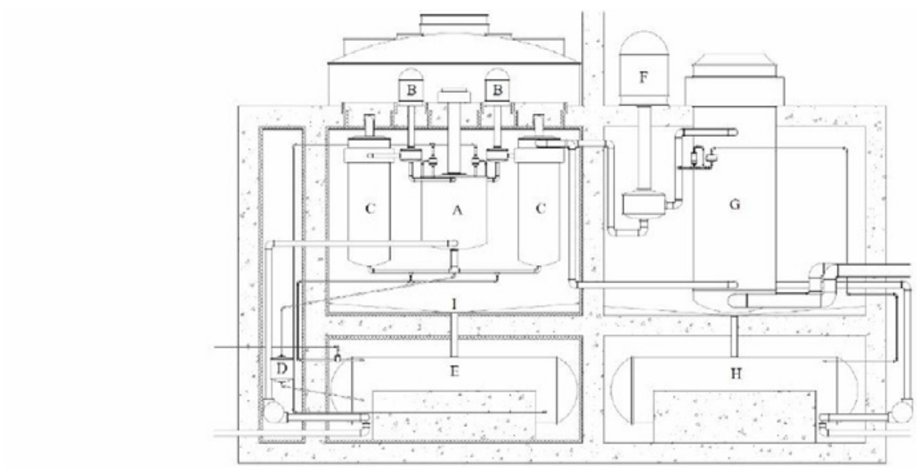
Figure 6.4 Conceptual Layout of Transatomic Plant



Source: Transatomic

A schematic of the reactor design is shown below.

Figure 6.5 Simplified Reactor Schematic



Source: Transatomic

The nuclear island also contains a fission product removal system and fuel addition system. This continuously operating system maintains constant fuel inventory and removes waste during operation. In this manner, the reactor avoids the need to overload the system with fuel and improves the consumption of the uranium fuel to very high burnup levels. This is possible without concern about fuel cladding damage, which might occur in fixed-fuel reactors.

The fuel for the design can either be uranium or spent fuel from light-water reactors. The spent fuel would require shipment to a processing plant, removal of the cladding, and then dissolving of the uranium oxide fuel and fission products into the molten salt. This obviously increases the cost of the fuel, but a reduction in disposal costs (by reducing size and number of high-level waste repositories) could offset the expense.

The operating temperature of the TAP plant is also high (650°C) but well below the boiling point of the molten salt (1,200°C). This high outlet temperature allows for thermal efficiencies in the 44% range.

The biggest challenges of molten salt reactors are the corrosive properties of the molten salt and the production of tritium, which must be removed.

From a safety perspective, should the reactor overheat, the neutronics would cause the reactor to shut down, and the freeze valves would also open and cause the fuel to drain into a noncritical drain tank. As with all liquid-fueled reactors, all the lines need to be electrically heated to prevent the fuel from freezing at 500°C.

The developers quote an nth-of-a-kind capital cost of $2 billion for 520 net MWe. Busbar costs were not provided. In order to utilize molten salt fuel, processing or reprocessing of spent fuel would be required. That would raise the operating and capital costs since separate fuel handling facilities would be required. The plan of the developers is that some of these expenses will be offset by lower spent fuel disposal charges, but that remains to be seen, given the responsibility for disposal is federal and not commercial.

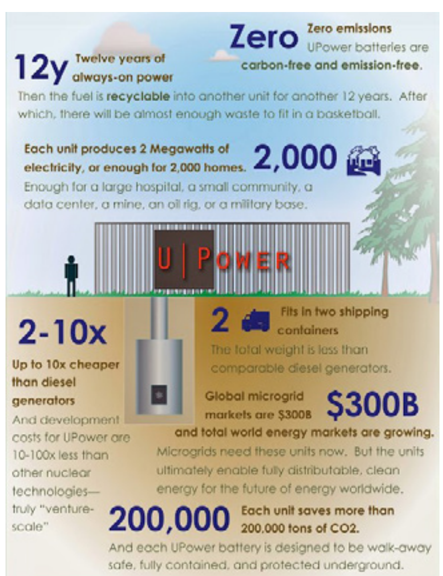
The Transatomic technology is a novel application of previous molten salt fuel designs, but it is early in the development cycle. There is no regulatory basis for licensing this or other molten salt reactors, but past developments and operation of MSRs at Oak Ridge provide optimism about technical success.

7. Small Grid: Local Applications

7.1 UPower: Oklo, Inc.

A new and innovative design of microreactors is the 2 MWe UPower (now called Oklo) plant.[84] This plant is designed for small industrial applications, malls, and remote locations. The plant is designed for a twelve-year operating cycle. It can burn nuclear waste from light-water reactors. The plant is designed to be a metal block containing metallic fuel in a heat pipe configuration that uses liquid sodium. The power conversion system is not finalized, but consideration is being given to organic Rankine cycle, steam, or supercritical C O2. A pictorial representation is in Figure 7.1 (shown below).

Figure 7.1 UPower Microreactor



Source: UPower

The design is such that the nuclear plant can fit into a standard shipping container. Two additional containers would house the power conversion system. With mass manufacturing of these small modules, designers claim they can produce electricity for $0.03/kWh. While the design is only in very preliminary stages, they have received venture capital funding to move the design forward.

There is insufficient technical information available publicly to put together a table of key parameters.

7.1.1 Nonproliferation

Given the limited design information available publicly, it is hard to characterize the proliferation risk, except to say that, since it claims to use nuclear waste from light-water reactors, this means that some form of reprocessing is needed, and that poses a proliferation problem. The analyzed twelve-year life of the reactor without refueling is a positive attribute.

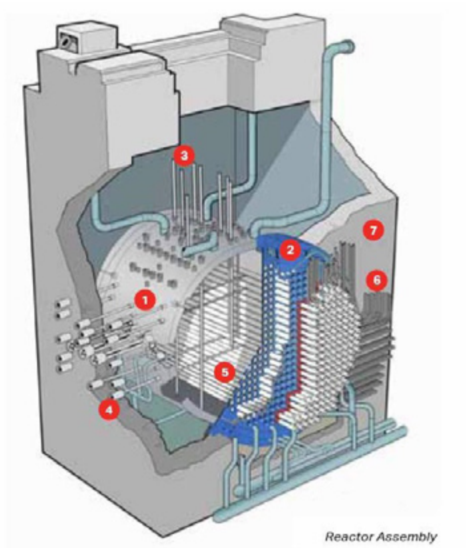
7.1.2 Nuclear Waste

Here again, the claim that the waste from this reactor would fit into a wastebasket cannot be verified since it is not known what processing is required or what the final isotopic composition of the fuel would be. Unless this information is provided, it is not possible to verify claims made.

8 Heavy-Water Reactors

Heavy-water reactors use deuterium instead of hydrogen as the atom to make up the water molecule. Deuterium is rare and expensive. It would have to be made especially for this application. The leading country in development of heavy-water reactors is Canada.[85] The reason for choosing heavy water as a coolant is that the reactor can use natural uranium, which is only 0.7% enriched in uranium-235. This is fissionable when compared to what a light-water reactor needs (about 4% enriched to become critical in a reactor). Choosing heavy-water reactors avoids the costly and complicated process of enriching uranium to 4% or 5% and is a nonproliferation plus. However, the reactor does produce plutonium since most of the fuel is U-238. The reactor design is quite different than conventional light-water reactors in that the fuel is contained in bundles loaded in horizontal pressurized tubes. These reactors are large pressurized heavy-water calandria types that are refueled while operating (Figure 8.1).

Figure 8.1 Schematic of CANDU Reactor

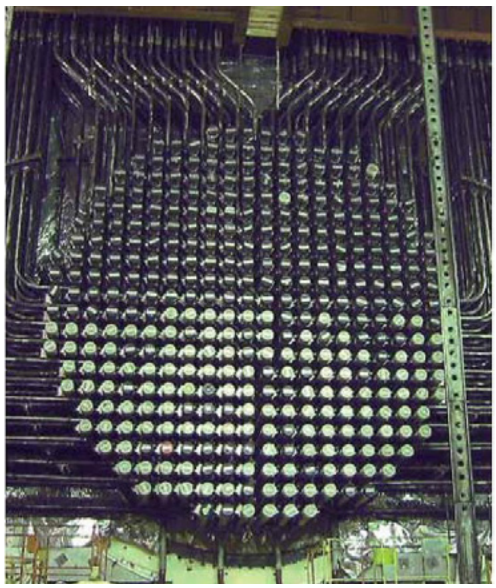


Source: CANDU Energy

The calandria contains 380 pressure tubes into which fuel assemblies are horizontally placed. Pressurized heavy water is circulated in these tubes, or channels, and this cools the fuel. The calandria itself is a large low-pressure vessel that houses the coolant and fuel tubes. It also contains heavy water to help moderate (slow down) the neutrons and act as a reflector.

The face of the calandria and a typical fuel element are shown in more detail in figure 8.2 and figure 8.3, respectively. Twelve of these fuel elements are inserted per fuel channel during operation.

Figure 8.2 Face of Calandria Reactor



Source: CANDU Energy

Figure 8.3 CANDU Fuel Element

Source: CANDU Energy

Because heavy water is not an efficient moderator of neutrons, the cores have to be considerably larger than in light-water reactors of the same power level. Canada has been developing an Advanced Candu Reactor (ACR) 6 to address concerns about CANDU-6's large size and moderator temperature coefficients being positive. By slightly enriching the uranium, however, this does take away some of the advantages of avoiding enrichment plants.

The challenge for these reactors is that they can also produce a significant amount of plutonium, and with online refueling, they pose a proliferation threat. Heavy-water reactors are deployed in many nations of the world, including Canada, China, Pakistan, Argentina, South Korea, India, and Romania. The United States has not licensed a heavy-water reactor to date.

Table 8.1 CANDU Reactors Operating in the World

Country	Type of Reactor	Units	Net Capacity (MWe)
Argentina	CANDU	1	600
Canada	CANDU	19	13,513
China	CANDU	2	1,280
India	CANDU and CANDU derived	2; 16	277; 3,480
Pakistan	CANDU	1	125
Romania	CANDU	2	1,305
South Korea	CANDU	4	2,579

Source: International Atomic Energy Agency

Figure 8.4 Bruce Power 4 Units

Source: Canadian Nuclear Association

8.1. Enhanced CANDU 6 (EC6)

The current version of the CANDU design is the Enhanced CANDU 6, which is based on the basic CANDU design but has improvements in safety features.[86] The technical data of the EC6 is summarized below.

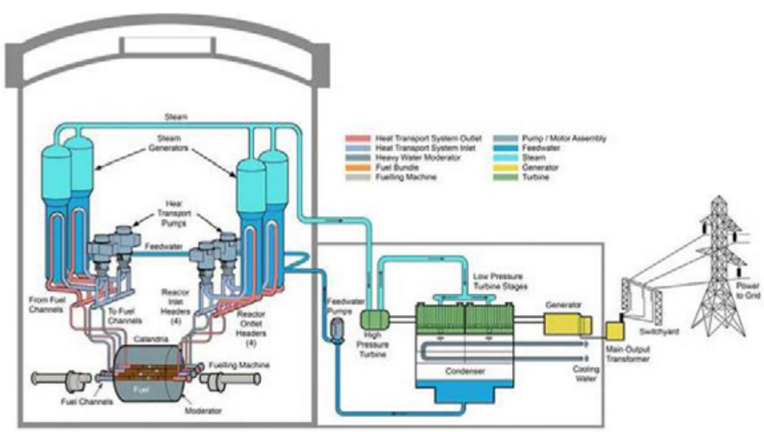
Table 8.2 Enhanced CANDU 6 Operating Parameters

Reactor Thermal Power	2,084 MWth
Reactor Electrical Power	740 MWe
System Pressure (PSIA)	10.09 MPa
Core Inlet/Outlet Temperature	265°C/310°C
Number of Fuel Assemblies	12 × 380 = 4,560
Fuel Assembly Length	.5 m per Bundle
Emergency Safeguards	Active
Steam Generators	4
Main Coolant Pumps	4
Core Damage Frequency	Not Available
Refueling Interval	Online Refueling
Construction Period	55 Months

Source: CANDU Energy

The power cycle of heavy-water reactors is similar to a conventional pressurized reactor in which the heavy water heated in the reactor is sent to a steam generator that contains ordinary water. This is allowed to boil to create steam for the power turbines. A schematic of the plant and power conversion system is shown in Figure 8.5 below.

Figure 8.5 EC6 Power Plant Schematic



Source: CANDU Energy

What is interesting about CANDU reactors is that they can be refueled online by simply pushing fuel assemblies horizontally on the face of the calandria and dropping them out of the back and into a spent fuel pool. This does present a proliferation concern. It allows fuel to be discharged at burnups that is more favorable in plutonium isotopes for weapons. This type of refueling also has distinct advantages. It does not require shutdowns for refueling, which can extend from weeks to months, as can be seen in light-water reactors.

The EC6 is targeted to achieve construction completion, from first concrete to in-service date, in fifty-five months. A second unit is to follow six months later. The cost of the deuterium is about 11% of the capital cost of the CANDU reactor plants.

9 Conclusion

While the utilities and governments of several countries in Europe, the United States, and Japan have shown little interest in new nuclear plants, other nations, companies, and independent entrepreneurs are working to advance the technology. The survey of new reactor designs presented in this report highlights the number of options available for countries interested in pursuing nuclear power in their energy mix. Decision-makers have many choices, based on fifty years of nuclear development. It just depends on the needs of the nation. New, innovative reactor designs that aim to overcome some of the traditional issues around nuclear energy are on the drawing board.

Conventional light-water reactors are now incorporating more passive safety features to avoid dependency on electric power for emergency functions. In addition, many new technologies are being developed that use liquid metal coolants, such as sodium, lead, or lead-bismuth. Molten-salt-cooled reactors use fluoride salts; molten-salt-fueled reactors use uranium dissolved in the molten salt; high-temperature helium-cooled gas reactors, both pebble and prismatic, use helium. Even fast neutron spectrum reactors use gas coolants. All these are being developed worldwide. More innovative reactor designs that could boost usable power and efficiency by using a variety of materials for their coolants and moderation—without presenting a safety threat—are being put forward, as this report has highlighted. Other innovations include plants that consume nuclear waste from light-water reactors, microreactors that can serve industrial parks and remote areas, and small nuclear battery-type reactors that don't have to be refueled for ten to thirty years. These are all exciting new technologies currently on the drawing board.

While it is recognized that nuclear energy offers a path to help address global climate change, it currently appears to receive little credit or value in the marketplace. To bring many of these technologies to market would be expensive. Funds are needed to support the research and development. In addition, there are many regulatory obstacles, which stem largely from the lack of knowledge by regulators about the new technologies. Most existing regulations do not apply to non-water-based nuclear plants. This is where much of the innovation is taking place. To be successful, new regulatory regimes will be needed to enable deployment of these new, and arguably safer, reactor plants and systems.

Designers are focused on increasing modularity in design to reduce the cost, to shorten the construction schedule, and to improve the quality of construction. Small modular reactors are being developed to address the very high capital cost of large plants. However, small reactors, unless they can take advantage of economies of production versus economies of scale, will have a higher cost of power than conventional large plants.

Despite all these innovations in design simplification and modularity, the up-front cost of building large reactors remains very high compared to fossil alternatives, such as oil, coal, and natural gas. While the capital costs are high, fuel and operating costs are relatively low. Unlike fossil fuels, they are also predictable. In addition, nuclear plants are being designed for at least sixty years of operation, making them a more long-term investment in comparison to solar and wind installations. Thus, pursuing nuclear energy requires a long-term vision for a nation's electricity supply.

It is important to note that when making cost comparisons, nuclear plants do not receive financial incentives similar to those of other clean-energy sources, such as solar or wind. Extending clean energy credits is needed to offset the high capital cost of the nuclear energy development. This should come either in the form of a carbon tax or production tax credits. Another critical requirement is a predictable and stable financial and regulatory regime to encourage new investment. A means to address the financial risk during early stages of the development cycle through demonstration is needed. The Secretary of the Energy Advisory Board (SEAB) recently prepared a report for the Secretary of the US Department of Energy. It was entitled "Report on the Future of Nuclear Power".[87] In it, the task force concludes that, for the United States, "there is no shortcut to reestablish a vigorous U.S. nuclear power initiative that could be a major carbon-free generation. To be successful, such an initiative will take time, significant public resources, restricted electricity markets, and sustained and skilled management attention. If the nation wishes to have a significant nuclear power option in the 2030–2050 time period, it must undertake the measures recommended in the Task Force report."

Many of the recommendations of the SEAB report were cited as needs in this report. This includes funding and support for nuclear innovation and reactor demonstration, a revised regulatory system and approach, credit for the clean air opportunities nuclear offers, power market restructuring to credit the production of base load power, and a larger, more focused, reorganized role of government support of nuclear power development and deployment.

From the standpoint of proliferation, commercial nuclear plants are generally considered a low risk. The spent fuel is difficult to reprocess. Separating the weapons materials from the waste stream is also difficult, and, when separated, the resulting plutonium is not ideal for nuclear weapons. Proliferation risks come instead from uranium enrichment plants and reprocessing facilities. To date, nation-states are the only entities capable of building and operating such facilities. This is due to their size and complexity and access to raw materials, such as uranium and spent fuel. Notwithstanding some speculation to the contrary, this is likely to persist as a barrier to non-state actor proliferation in the future. It would be easier for terrorists or criminals to simply steal or purchase nuclear weapons.

The decision to develop nuclear weapons is a political one, and only international pressure can help prevent their development. These efforts have not always been successful. At present, the threats posed by Iran and North Korea are serious and are being internationally addressed. The United Nations has created the International Atomic Energy Agency to monitor nuclear facilities and identify diversions. There are no technological solutions to prevent weapons proliferation—only detection.

New nuclear designs, such as battery-type long-life reactors, do offer possible solutions to proliferation since the reactors are not refueled during operation, and the entire reactor and core are shipped to a central facility for refueling and refurbishment. To maximize the value of the uranium fuel and what is now considered waste, such as spent fuel and depleted uranium currently in storage, fast spectrum reactors could be an option for the long-term sustainability of nuclear energy. Fast reactors can be designed without recycling and breeding excess plutonium, addressing proliferation concerns. The challenge for fast reactors is the capital and operating cost. They are more difficult to operate and to maintain due to the use of liquid metals.

Despite proliferation concerns, there are factors that could make fast reactors attractive to some countries. They offer the potential for electricity supply for thousands of years, consuming what is now waste from enrichment facilities in the United States and other nations.

There are several designs that use waste from light-water reactors as fuel, but they do require reprocessing. Reprocessing plants, like nuclear reactors, can be inspected and monitored for diversion by the IAEA. Thorium fuel is also being considered as an alternative fuel in order to avoid production of any plutonium. However, U-233 is an attractive weapons material that comes from the thorium fuel after reprocessing. While nuclear nonproliferation concerns are serious, they can be best addressed by political, not technological, solutions.

In an attempt to summarize nuclear options for a decision-maker, table 9.1 was prepared in order to address the key attributes of nuclear technology choices. The table condenses and compares the information the safety; fuel cycle and waste; economics; licensing and experience base; and proliferation resistance of the included reactor designs. Comparing such different technologies and sizes is, at best, subjective. However, using current light-water reactors in operation as a reference, it is possible to make some judgments. The rankings for each of the criteria are explained in the notes below the table. However, the following is a summary:

For safety risk, the ranking is based on the inherent safety of the design. The ideal technology is one that does not pose a risk of meltdown. High-temperature gas reactors, among others, offer this capability. In addition, the more passive the emergency response, the lower the safety risk. These emergency response systems rely on gravity to provide cooling rather than active systems that require electric power.

For fuel cycle and waste, the ranking is simply based on whether the spent fuel is disposed of directly or reprocessed. Clearly, this is a difficult criterion to simplify since, with reprocessing, the amount of waste is significantly reduced. However, reprocessing creates the risk of proliferation, especially if plutonium or U-233 is separated for recycling. There are reprocessing cycles under development that do not separate plutonium for recycling, but these have yet to be implemented—either in the reprocessing step or in the reactor utilization of the unseparated fuel. This rating does not assume unseparated plutonium. Thus, while direct disposal is preferred from a fuel cycle perspective in the proliferation context, it does not reduce the amount of nuclear waste that must be disposed.

For economics, the reference standard is a light-water reactor of about 1,200 MWe. The costs referenced are country specific, as shown in figure 2.2. While Chinese costs are not included in this table, China builds nuclear reactors at significantly lower cost than in Europe and in the United States. Their costs are more in line with the Korean estimates shown in that figure. While the Chinese price might be significantly lower than the US price, when one compares the time to complete the construction of the plants in Korea and China, the differences are significant (with China building plants four to five years faster). This indicates there are likely ways to reduce US costs.

Competition for new nuclear plants can also drive down costs, as witnessed by Abu Dhabi's choice of the Korean APWR-1400 for their four new nuclear units. These will be built at a reported 20% to 30% lower cost than other competitors. What is clear from the available economic analyses is that the capital costs of new nuclear plants has risen dramatically in the last decade, making some of these plants unaffordable for utilities to finance, given the size of the plant and the investment needed. The answer for some vendors has been to increase the size of the plants in order to reduce the cost per kilowatt-hour for the electricity. In response to the high initial capital cost outlay, smaller modular reactors are being designed that are within the financial reach of utilities. However, despite the lower investment, these smaller modular reactors, unless they can capture the economies of mass production, will be more expensive per kilowatt-hour.

Nuclear battery or long-life reactors are small modular reactors whose target market is largely remote areas and developing nations with small electric grids that cannot support large plants and those lacking a long history of reactor operations. Their cost per kilowatt-hour is high but typically lower than fossil alternatives. In evaluating economics, one must consider applications and target markets.

In the category of experience base and licensing, clearly light-water and heavy-water reactors have the most experience and are much easier to license than new, innovative technologies. This becomes a huge hurdle for developers who need to finance research and development as well as the education for the regulators of the new technologies. Thus, while the new technologies might provide substantial safety and application advantages, such as nuclear waste consumption, the burden of bringing them to market requires substantial government and long-term investor support. The SEAB report provides several suggestions to enable a new nuclear era for the United States. The key recommendation is to establish a quasi government corporation to cost share with the private sector in order to advance several promising technologies in four stages of development and efficiently bring one or more of these promising technologies to market. This will take a political agreement and strong executive leadership, which might not be possible in the United States. However, that could work in other countries.

In the area of nonproliferation, this too is a complex criterion to effectively evaluate. Simply throwing away the used fuel is the easiest short-term solution to avoid reprocessing, but it comes at the expense of higher nuclear waste volume. At present, there is no place in the United States or Japan for disposal, given the cancellation of the Yucca Mountain geological repository. In further evaluating the proliferation risks, one must consider the importance of international monitoring and inspection to prevent diversion at all stages of the fuel cycle including enrichment of uranium, reprocessing methodologies, and disposal. While there have only been a few nations that have chosen to develop nuclear weapons, they did so out of their perceived national need, independent of whether they had the benefit of a commercial nuclear power enterprise. Thus, the proliferation risk is more dependent on the nation than on the technology. Table 9.1 recognizes that reprocessing poses additional proliferation risk. All efforts should be directed to reduce the national desire for nuclear weapons. With adequate inspections and monitoring and effective penalties for violations, the benefits of recycling nuclear waste could be achieved.

The bottom line for decision-makers regarding new nuclear plants for their nations is how they view the global climate challenge. Should they consider it a serious threat, there are many developing nuclear technologies to choose from that meet the nation's need and perception of the proliferation threat. While not all reactors under development worldwide are summarized in this report, the nuclear plants presented represent the types of plants being considered.

To assist the decision-maker in assessing the timing of the availability of summarized nuclear options, table 9.2, table 9.3, and table 9.4 highlight what might be available now, in ten years, and in twenty to twenty-five years, should the technologies continue to be developed through demonstration stages. The timeline for availability of the plants is largely a personal judgment of the author. It's based on the needed research and development and the licensing challenges each technology type brings. It should be noted that bringing new nuclear technologies to market will take ten or more years and require a long-term vision, as well as sustained funding and commitments from the government and private sector. Such risk sharing will be needed if new nuclear technologies are going to contribute to global clean air goals.

In order to enable these developing technologies, a fair playing field is needed. There must be financial incentives and support, which means giving nuclear the same kinds of clean-air subsidies and regulatory treatment provided to solar and wind. In addition, nations need to change the current regulatory systems to acknowledge the inherent safety of the new designs, which would allow for more efficient and cost-effective deployment.

The real question for decision-makers is whether a clean energy source, such as nuclear, should be a part of their nations' energy strategies. It's then a matter of how to create policy and regulatory systems that encourage development.

Table 9.1 Comparison of Reactor Technologies on Key Parameters

Generation III Light-Water Reactors

Name	Power Level (MWe)	Coolant	Safety Risk	Fuel Cycle/ Waste	Economics	Experience Base; Licensing	Nonproliferation
AP-1000	1,115 PWR	water	Low	medium	medium	limited; licensed	medium
ESBWR	1,600 BWR	water	low	medium	medium	limited; licensed	medium
APR-1400	1,450 PWR	water	medium	medium	medium	limited; licensed	medium
EPR PWR	1,660 PWR	water	medium	medium	medium	limited; licensed	medium
VVER-1200	1,198 PWR	water	medium	medium	medium	limited; licensed	medium
ABWR	1,371 BWR	water	medium	medium	medium	operating	medium
APWR	1,530 PWR	water	medium	medium	medium	limited; licensed	medium
CPR-1000	1,030 PWR	water	medium	medium	medium	operating	medium
ACPR-1000	1,150 PWR	water	low	medium	medium	limited; licensed	medium
HPR-1000	1,150 PWR	water	medium	medium	medium	limited; licensed	medium
CAP-1400	1,500 PWR	water	medium	medium	medium	limited; licensed	medium

Small Modular Reactors

SMART	100 PWR	water	low	medium	high	limited; licensed	medium
NuScale	50 PWR	water	low	medium	high	minus; not licensed	medium
SMR-160	160 PWR	water	low	medium	high	minus; not licensed	medium
KLT-40	40 PWR	water	medium	medium	high	limited; licensed	medium
West 225	225 PWR	water	low	medium	high	minus; not licensed	medium
CAREM	27 PWR	water	low	medium	high	limited; licensed	medium

High-Temperature Gas Reactors

HTR-PM	200 HTR PEBBLE	helium	very low	high	high	some; licensed	low
X-energy	45 HTR PEBBLE	helium	very low	high	high	minus; not licensed	low
GT-MHR	300 BLOCK	helium	very low	high	high	minus; not licensed	low
ANTARES	300 BLOCK	helium	very low	high	high	minus; not licensed	low
GT-MHR	300 BLOCK	helium	very low	high	high	minus; not licensed	low
EM2	265 FAST GAS	helium	medium	low	high	minus; not licensed	higher

Nuclear Battery

4S	10 FAST	sodium	low	low	high	minus; not licensed	low
SS	20 FAST	liquid metal	low	low	high	minus; not licensed	low

Small Grid

UPower (Oklo)	2 HEAT PIPES	sodium	not available	not available	not available	minus; not licensed	not available
SS	20 FAST	liquid metal	low	low	high	minus; not licensed	low

Molten Salt

ThorCon	250 MOLTEN SALT FUEL	NaBe (thorium)	medium	medium	high	minus; not licensed	medium
FHR	242 MOLTEN SALT	FLiBe	medium	medium	high	minus; not licensed	medium
Transatomic	520 MOLTEN SALT FUEL	lithium fluoride	medium	low	high	minus; not licensed	higher

Liquid Metal

BN-800	880 FAST	sodium	medium	low	high	good; licensed	higher
PRISM	622 FAST	sodium	medium	low	high	minus; not licensed	higher
TerraPower	600 FAST	sodium	medium	low	high	minus; not licensed	higher
BREST	300 FAST	lead	medium	low	high	minus; not licensed	higher
SVBR	100 FAST	lead-bismuth	medium	low	high	minus; not licensed	higher
Gen4	25 FAST	lead-bismuth	medium	medium	high	minus; not licensed	medium

Heavy Water

Enhanced CANDU 6	750 PHWR	heavy water	medium	medium	medium	medium; licensed	higher

Keys:

Safety Risk: Rating depends on degree of passive or inherent safety in design relative to standard LWRs as medium risk and the results of the PSA, if available. (Lower core damage frequency.)

Fuel Cycle: Rating depends on volume of waste produced (high volume is high) or if fuel cycle calls for recycling of separated plutonium (high) or if reprocessing ends up consuming nuclear waste (low).

Economics: Rating depends on reactor size. This is due to economies of scale neglecting the high cost of individual plant construction, which is important for many responsible nuclear power investments.

These ratings do not include fuel or operating and maintenance costs, and these can be significant compared to capital costs for small modular reactors, which require refueling in two to four years. For reactors that have not been built, the ratings are best estimates.

Experience Base: Rating depends on whether plant is operating. Most new technologies do not have operating plants, but some are under construction.

Proliferation: Rating depends on whether reprocessed separated plutonium is used, but this is not a complete metric for risk. It is more country dependent. Proliferation risk is best mitigated by international monitoring of the technologies, and it's best judged by evaluating the nature of the countries seeking to develop nuclear power.

Table 9.2 Options to Build Now

Name	Country	Power Level (Mwe)
AP 1000	US	1115 PWR
ESBWR	US	1600 BWR
APR 1400	Korea	1450 PWR
EPR PWR	France	1660 PWR
VVER 1200	Russia	1198 PWR
ABWR	US/Japan	1371 BWR
APWR	Japan	1530 PWR
CPR 1000	China	1030 PWR
ACPR 1000	China	1150 PWR
HPR 1000	China	1150 PWR
CAP 1400	China	1500 PWR
E Candu 6	Canada	750 PHWR
HTR-PM	China	200 HTGR
KLT-40	Russia	40 PWR
BN 800	Russia	800 SFR
BREST	Russia	300 LFR

Table 9.3 Options to Build in Ten Years

SMART	Japan	100 PWR
NuScale	US	50 PWR
SMR 160	US	160 PWR
West 225	US	225 PWR
CAREM	Argentina	27 PWR

High Temperature Gas Reactors

HTR-PM	China	200 HTR PEBBLE
X-Energy	USA	45 HTR PEBBLE
ANTARES	France	300 BLOCK
GT-MHR	Japan	300 BLOCK

Table 9.4 Options to Build in Twenty to Twenty-Five Years

Nuclear Battery

4-S	Japan	10 FAST	Sodium
S Star	USA	20 FAST	liquid metal

Liquid Metal

PRISM	USA	622 FAST	Sodium
Terrapower	USA	600 FAST	Sodium
SVBR	Russia	100 FAST	Lead/bismuth
GEN 4	USA	25 FAST	Lead/bismuth

Gas

EM2	USA	265 FAST	Helium Gas

NOTES

1. Though plutonium is now man-made, 1.7 billion years ago, during the early stages of the earth's formation, there was sufficient U-235 that had not decayed away (about 4% to 5%). This was sufficient to cause a natural nuclear fission in the Oklo mine in Gabon. This fissioning lasted for a few hundred thousand years, creating plutonium from the fission neutron interaction with U-238.

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5. It should be noted that current reactors have sought and obtained permission for extending their operating licenses to sixty years, with consideration being given to further extensions up to eighty years, depending on the economics of doing so.

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