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Chapter

NEW TECHNOLOGIES ASSOCIATED TO THE CONSTRUCTION OF NUCLEAR POWER PLANTS

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ABSTRACT

Undoubtedly, energy production and their sustained growth constitute a relevant factor for ensuring the economic and social development of any country. Considering the different available energy sources that the world can use to satisfy the foreseeable increase in energy demand in the coming years, particularly for the production of electricity, at least for the next decades there are only a few realistic options available to reduce further the CO_2 emissions, to satisfy the foreseeable demand of electricity, and to have a secure supply of energy.

One of these options is the use of nuclear energy for electricity generation. If this is true, then, why the public opinion of several countries is against the use of this type of energy sources? One of the reasons is the negative impact of an accident at a nuclear power plant for the human beings and for the environment. The second reason is the possible military uses of certain nuclear installations used for the generation of electricity. The third reason is the nuclear waste generated by nuclear power plants.

To reduce to the minimum the possibility of a nuclear accident it is important to maintain and enhance the safe and reliable operation of the nuclear power reactors. This is an essential priority in the development of a new generation of this type of reactors. Three generations of nuclear power reactors have been used for the production of electricity until now; a four generation is under development by a group of countries. The first generation (Generation I) was advanced in the 1950s and 1960s in the early prototype of nuclear power reactors. The second generation (Generation II) began in the 1970s in the large commercial nuclear power plants; some of these reactors are still operating today. The third generation (Generation III) was developed in the 1990s with a number of evolutionary designs that offer significant advances in safety and economics, and a limited number of this type of reactor has been built, primarily in East Asia. Advances to Generation III are underway, resulting in several (so-called Generation III+) near-term deployable nuclear power reactors that are actively under development and are being considered for deployment in several countries. The European Pressure Reactor (ERP) produced by France is of this type. New nuclear power reactors built between now and 2030 will likely be chosen from these new types of nuclear power reactors. Beyond 2030, the prospect of innovative advances through renewed research and development has stimulated interest worldwide in a fourth generation of nuclear energy systems (Generation IV).

Ten countries have joined together to form the Generation IV International Forum (GIF) to develop future-generation nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, nuclear waste, proliferation, and public perception concerns. The objective for Generation IV nuclear energy systems is to have them available for international deployment about the year 2030, when many of the world's currently operating nuclear power reactors will be at or near the end of their operating licenses.

General Overview

Undoubtedly, energy production and their sustained growth constitute relevant factors for the economic and social progress of any country. For countries in the route of development, such as China, India, the Republic of Korea, Brazil, and South Africa, just to mention only a few ones as examples, the demand of energy increase significantly each year, particularly for the generation of electricity. How to satisfy the increase in the demand of energy in this group of countries without increasing the negative impact in the environment and in the population? The only effective manner to do this is including all types of energy sources in any study to be carried out by the governments of the different countries about the future structure of their energy mix.

During these studies, there are certain factors that should be considered by the national competent authorities and the private sector during the selection of the most economic and convenient structure of the country energy mix. Which are these factors? One of them is the use of fossil fuels for the generation of electricity and their negative impact on the environment. The use of fossil fuels for the generation of electricity is a major and growing contributor to the emission of CO_2, an important element associated with the current climate changes which are affecting several countries. Another factor is the level of the proven reserves of fossil fuels. These reserves are limited and are concentrated in some specific regions, some of them very instable from the political point of view. For some specific energy sources such as oil, the current proven reserves could be depleted in the coming decades. Considering the different options that the countries have in their hands to satisfy their foreseeable increase in their energy demand in the coming years, particularly for the production of electricity, there are only a few realistic options available that can be effectively used for this specific purpose. These options are the following:

1. Increase efficiency in electricity generation and use;
2. Expand use of all available renewable energy sources for the generation of electricity such as wind energy, solar energy, hydro power, biomass, and geothermal energy, among others;
3. Massive introduction of new advanced technology like the capture carbon dioxide emissions technology at fossil-fueled (especially coal) electric generating plants, with the purpose of permanently sequester the carbon produced by these plants in order to reduce CO_2 emission;
4. Increase use of new types of nuclear power reactors that are inherently safe and proliferation risk-free, such as Generation IV nuclear power reactors;
5. Increase energy saving.

Table 1. Nuclear power reactors in operation, under construction or planned by country in 2012

Country	Reactors in operation in 2012		Reactors under construction in 2012		Reactors planned for construction in 2012	
	No.	MW	No.	MW	No.	MW
Argentina[1]	2	935	1	692	0	0
Armenia	1	375	0	0	0	0
Belgium	7	5 927	0	0	0	0
Brazil	2	1 884	1	1 245	0	0
Bulgaria	2	1 906	2	1 906	0	0
Canada	18	12 604	2	1 900	0	0
China[1]	16	11 816	26	26 620	42	34 786
Czech Republic	6	3 766	0	0	0	0
Finland	4	2 736	1	1 600	0	0
France	58	63 130	1	1 600	0	0
Germany	17	20 490	0	0	0	0
Hungary	4	1 889	0	0	0	0
India	20	4 391	7	4 824	0	0
Iran	1	915	0	0	3	2 160
Japan	54	46 934	2	2 650	10[2]	13 192
Korea RO (South)	21	18 751	5	5 560	2	2 680
Mexico	2	1 300	0	0	0	0
Netherlands	1	482	0	0	0	0
Pakistan	3	725	2	630	0	0
Romania	2	1 300	0	0	0	0
Russia	33	23 643	10	8 188	35	34 617
Slovakia	4	1 816	2	782	0	0
Slovenia	1	688	0	0	0	0
South Africa	2	1 830	0	0	0	0
Spain	8	7 567	0	0	0	0
Sweden	10	9 326	0	0	0	0
Switzerland	5	3 263	0	0	0	0
Ukraine	15	13 107	2	1 900	0	0
United Kingdom[2]	19	10 170	0	0	0	0
USA	104	101 465	1	1 165	20	25 724
Vietnam	0	0	0	0	2	2 000
World	448	380 149	65	61 962	114[2]	85 159

[1] Nuclear power reactors operating, under construction and planned in Taiwan are including in the data of China.

[2] Nuclear accident in the Fukushima Daiichi nuclear power plant occurred in Japan in March 2011 can modify this figure.

Source: IAEA.

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[1] The President of Argentina announced, in 2012, that two nuclear power reactors are going to be built in the country in the coming years at a cost of around US$ 6 000 million.

[2] In 2013, the UK government announced the construction of two nuclear power reactors of the third generation plus at an estimates cost of US$ 16 billion.

One of the available energy sources that have proved to be a realistic option from the technological point of view for electricity generation is nuclear energy. However, the use of this type of energy source for the generation of electricity is not a cheap nor and easy option. From the technological point of view the use of nuclear energy for the generation of electricity could be very complicated and costly for many countries, particularly for those with a weak technological development or with limited financial resources available to be invested in the energy sector or with a lack of well-prepared professionals, technicians and high-qualified workers or with a small electrical grid. In comparison to coal fired and natural gas fired power plants, it is true that in many countries nuclear power plants are more expensive to build but less expensive to run, and this is an important characteristic that should be in the mind of national competent authorities during the consideration of the future structure of the country energy mix.

Which is the current situation regarding the use of nuclear energy for the generation of electricity at world level? According to IAEA sources, in 2012 there were 448 nuclear power reactors in operation in 30 countries (31 countries if Taiwan is considered independently from China), with a total capacity of 38 149 MW[3]; 65 nuclear power reactors were under construction, with a capacity of 61 692 MW; and 114 nuclear power reactors have been planned with a capacity of 85 159 MW (See Table….)[4]. There are five nuclear power reactors in long-term shut down in 2011 with a capacity of 2 972 MW and 138 nuclear power reactors permanently shut down with a capacity of 49 152 MW.

The total electricity produced by the 448 nuclear power reactors operating in 30 countries in 2012 was 2 517 980.41 GWh. The number of nuclear power reactors in operation during the period 1980-2011 is shown in Figure 1.

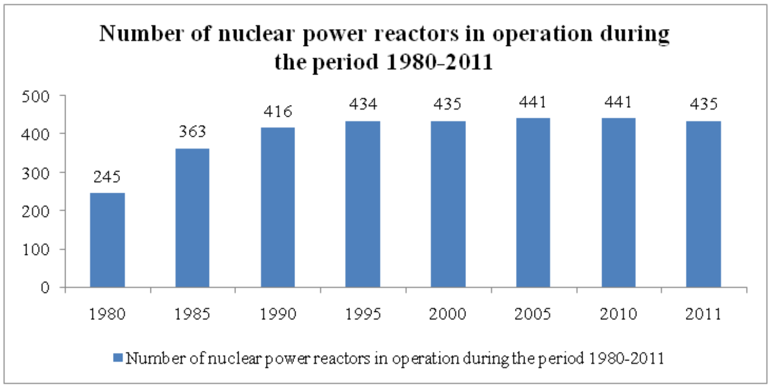
[3] At the end of 2012, and according to the latest IAEA information, there were 437 nuclear power reactors in operation in 31 countries with a net capacity of 372.5 GWh and 66 units under construction. The USA government approved the construction of two nuclear power reactors in Georgia. It is expected that these units enter into operation in 2016.

[4] Several of these nuclear power reactors are not going to be built as consequence of the Fukushima nuclear accident occurred in Japan in March 2011 and the strong public rejection to the use of this type of energy source for the generation of electricity in several countries in the future.

Table 2. Nuclear share per country in June 2012

Country	Number of Reactors in Operation	Nuclear Electricity Supplied (GW/h)	Nuclear Share (%)
Argentina	2	5 893.81	5.0
Armenia	1	2 356.84	33.2
Belgium	7	45 942.28	54.0
Brazil	2	14 794.74	3.2
Bulgaria	2	15 264.14	32.6
Canada	18	88 317.57	15.3
China	16	82 568.66	1.8
Czech Republic	6	26 695.64	33.0
Finland	4	22 265.52	31.6
France	58	423 509.48	77.7
Germany	17	102 311.20	17.8
Hungary	4	14 706.92	43.2
India	20	28 947.67	3.7
Iran, Islamic Republic of	1	97.98	0.0
Japan	54	156 182.14	18.1
Korea, Republic of	21	147 763.46	34.6
Mexico	2	9 313.37	3.6
Netherlands	1	3 917.24	3.6
Pakistan	3	3 843.42	3.8
Romania	2	10 810.98	19.0
Russia	33	162 018.13	17.6
Slovakia	4	14 342.12	54.0
Slovenia	1	5 902.24	41.7
South Africa	2	12 938.54	5.2
Spain	8	55 121.12	19.5
Sweden	10	58 098.43	39.6
Switzerland	5	25 693.89	40.8
Ukraine	15	84 893.98	47.2
United Kingdom	19	62 658.05	17.8
United States of America	104	790 439.33	19.2
Total	448	2 517 980.41	NA

Source: IAEA.



Source: IAEA nuclear power reactors in the world

Figure 1. Number of nuclear power reactors in operation during the period 1980-2011.

From Figure 1 the following can be stated: the number of nuclear power reactors in operation in the world during the period 1995-2000 increased in only one unit; between 2000 and 2010, increased in six units but between 2010 and 2011 decreased in the same number of units; in other words, the number of nuclear power reactors in operation during the period 1995-2011 increased only in one unit[5].

According to Figure 2, the ten countries with the highest participation of nuclear energy in their energy mix in 2012 were the following: France (77.7%), Belgium (54%), Slovakia (54%), Ukraine (47.2%), Hungary (43.2%), Slovenia (41.7%), Switzerland (40.8%), Sweden (39.6%), Republic of Korea (34.6%), and Armenia (33.2%).

The future expansion of the use of nuclear energy for the generation of electricity at world level will depend upon a number of factors. These factors are the following:

- Fossil fuel reserves;
- Fossil fuel prices;
- Energy security concerns;
- Environmental and climate change considerations;
- Nuclear safety concerns;
- Nuclear waste treatment;

[5] In this amount the number of nuclear power reactors shut down and the new nuclear power reactors that entered in operation are included.

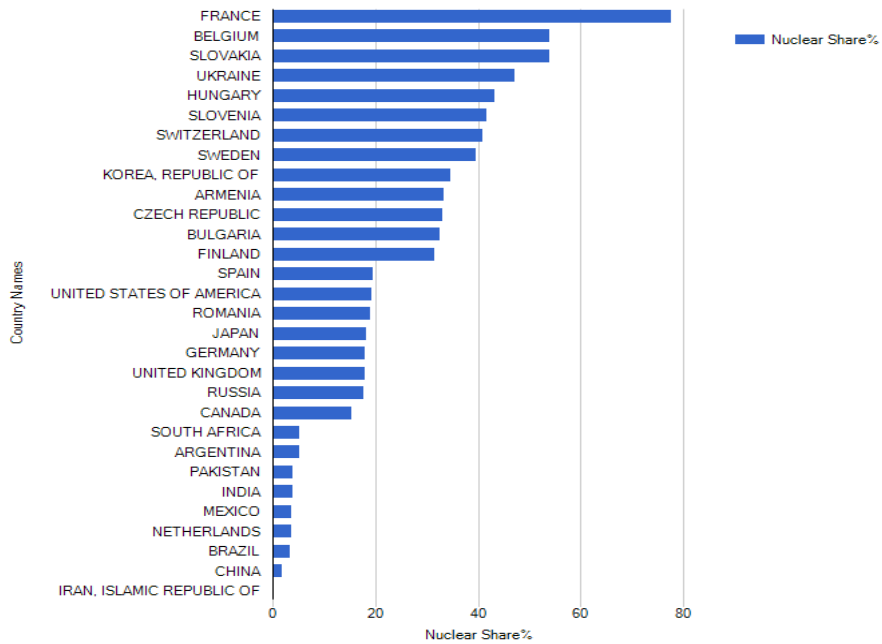
- Cost of the new nuclear technologies associated to new types of nuclear power reactors now under development;
- Public opinion;
- Nuclear proliferation.

Nuclear Safety

A nuclear power programme is a major national undertaking requiring careful planning and preparation, and a major investment in time and human and financial resources. A considerable period of time is indispensable to acquire the necessary competences and a strong safety culture before operating a nuclear power plant. While prime responsibility for the safety operation of a nuclear power plant must rest with the operator, the State has the responsibility, upon committing itself to a nuclear power programme that demands significant investment, to create a robust framework for safety. Establishing a sustainable safety infrastructure is a long process that could cover a period between ten and fifteen years, depending of the characteristics of the country, the type of nuclear power reactor design selected, the manner in which the nuclear power plant is going to be built, the financial resources available to carry out the construction of the nuclear power plant, and the level of the participation of the national industry in the implementation of this phase, among others, and would generally be needed between the consideration of the use of nuclear energy for electricity generation as part of the national energy strategy, and the commencement of operation of the first nuclear power reactor.

According to IAEA SSG-16, the lifetime of a nuclear power plant is divided into five phases from a nuclear safety standpoint:

- Phase 1: this phase is related to the building of the safety infrastructure before a firm decision to introduce a nuclear power programme is adopted by the government. The average duration in the implementation of this phase is between one and three years;
- Phase 2: this phase is related to the safety infrastructure preparatory work for construction of a nuclear power reactor after a policy decision has been taken by the government to introduce a nuclear power programme. The average duration in the implementation of this phase is between three to seven years;



Source: IAEA.

Figure 2. Nuclear share in percentage by country in 2012.

- Phase 3: this phase is related to the safety infrastructure during the construction of the first nuclear power reactor. The average duration in the implementation of this phase is between seven and ten years depending of the type of nuclear power reactor design selected, the manner in which the nuclear power plant is going to be built, the financial resources available to carry out the construction of the nuclear power plant, and the level of participation of the national industry in the implementation of this phase, among others;
- Phase 4: this phase is related to the safety infrastructure during the operation phase of a nuclear power plant. The average duration in the implementation of this phase is between forty and sixty years, depending of the type of nuclear power reactor selected;
- Phase 5: this phase is related to the safety infrastructure during the decommissioning and waste management phases of a nuclear power plant. The average duration in the implementation of this phase is from twenty years to more than hundred years depending of the

nuclear power reactor design selected, and the experience of the country in carry out this type of complex work.

It is important to stress that the government, through their legal system, should establish a national policy for safety well beyond the implementation of the construction phase of the first nuclear power reactor. The regulatory authority, as designated by the government, is charged with the implementation of policies through a regulatory programme or a strategy set forth in its regulations or standards. The government determines also the specific functions of the regulatory authority and the allocation of responsibilities. In addition, the government should adopt laws and policies specifying the responsibilities and functions of different governmental offices in respect of safety and emergency preparedness and response, whereas the regulatory authority establishes a system to provide for effective coordination.

According to the IAEA Safety Fundamentals, the following ten safety principles should be followed by all governments that have decided to introduce a nuclear power programme:

1. The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks;
2. An effective legal and governmental framework for safety, including an independent regulatory authority, must be established and sustained;
3. Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks;
4. Facilities and activities that give rise to radiation risks must yield an overall benefit;
5. Protection must be optimized to provide the highest level of safety that can reasonably be achieved;
6. Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm;
7. People and the environment, present and future, must be protected against radiation risks;
8. All practical efforts must be made to prevent and mitigate nuclear or radiation accidents;
9. Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents;

10. Protective actions to reduce existing or unregulated radiation risks must be justified and optimized[6].

Most of the governments that are considering the introduction of a nuclear power programme are looking for proven existing nuclear technologies rather than developing a specific new design of nuclear power reactors. Nevertheless, it should choose from among various available nuclear technologies, bearing in mind which is the most appropriate technology for the country, taking into account its technological development, the conditions for the transfer of the nuclear technology to be used in the nuclear power plant, the financing of the construction of the nuclear power reactors, among others elements. Such a choice should be made at different times depending on the overall energy policy adopted by the government but, in any case, this policy should emphasize the effective transfer of competence in safety manner to the State[7] (IAEA, SSG-16).

Finally, it is important to stress the following: government should inform all interested parties regarding decisions on the implementation of a nuclear power programme, including the long-term national and international commitments to maintain nuclear safety and the necessity of measures such as establishing new organizations, building new national infrastructure, and making financial provision for radioactive waste and spent fuel management. Information should be provided to the public, industry, news media, non-governmental organizations and neighboring States. After the initial investment for construction of the nuclear power plant, investments are needed for its regular refurbishment, because most equipment is of limited lifetime and should be replaced with new equipment as part of the ageing management programme. Also, technologies have certain design lifetimes, and equipment should be modernized as necessary to ensure the availability of spare parts and to reduce the unplanned shut down.

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[6] For additional information of this important issue see the document IAEA SF-1.

[7] The construction of a nuclear power plant involves numerous contractors, and it is incumbent on the operating organization to ensure that this complex chain of contractors is adequately managed so that the end products are acceptable from a safety standpoint. The responsibility of the operating organization in this respect is the same no matter which option is selected for the nuclear power plant supply contract. The operating organization should verify from the very beginning the quality of equipment and services supplied by the vendor and its subcontractors under contracts of all types, including turnkey and super turnkey projects.

National Energy Plan

In order to ensure the introduction or the expansion of a nuclear power programme in the most efficient and effective manner, the government should adopt a national energy plan specifying the objectives for the national energy policy. Some of the possible objectives of this plan are the following:

- Increased energy independence as much as possible;
- Development of indigenous energy resources as most as possible from the economic point of view;
- Diversification of energy sources;
- Increase energy efficiency;
- Economic optimization of energy and electricity supply;
- Stability of electric grid system;
- Security of electricity supply;
- Availability of energy at prices which support general social and economic development;
- Environmental protection.

Experience shows that the time between the adoption of the initial policy decision to consider the introduction of a nuclear power programme by the government, up to the start of operation of the first nuclear power reactor, could be between ten and fifteen years, depending on the type of nuclear power reactor design selected, the technological development of the countries, the type of the agreement reached with the supplier of the nuclear power reactor selected, among others.

One of the main decisions that should be adopted by governments related to the introduction of a nuclear power programme is the establishment of an effective, competent, and independent regulatory authority to oversight all nuclear activities. If the governments decide to establish more than one regulatory authority (e.g. for radiation protection, nuclear safety, environmental protection, and conventional health and safety), effective arrangements should be adopted to ensure that regulatory functions and responsibilities related to the nuclear power programme are properly identified, discharged, and coordinated. The authorization process and the basis for granting an authorization for siting, design, commissioning, operation, and for discharges to the environment should be clearly defined. The regulatory authority needs to develop the capabilities to plan and implement the review and safety assessment activities related to the construction and operation of all nuclear power reactors built in the country throughout its operational life.

Environment and Climate Change Considerations

The increase use of some types of energy sources worldwide for the generation of electricity has become a major environmental concern for the international community. Energy use has environmental impacts at all levels:

- Locally, e.g. through the use of primitive cooking stoves in many developing countries, smog formation in urban areas, and local flooding and resettlement as a result of the construction of new hydro power plants;
- Regionally, through the acid rain caused by emissions of sulphur dioxide and nitrogen oxides;
- Globally, through the contributions of carbon dioxide and methane to the greenhouse effect.

The greenhouse effect and global warming now seem to be the main subject for discussion in several countries from all regions. However, local effects, with potentially negative serious health impacts, concern a large number of people in developing countries and are of the highest priority for these countries, whereas the potential for global climate change, caused to the greatest extent by industrialized countries, is regarded as a problem for those countries. Acid rain, the regional effects of which have been so evident across Europe and the northeastern part of North America, is also having an impact in eastern China and parts of India, among others. This will probably change in the future as significant regional effects over the whole of southern and southeast Asia have been forecast (World Energy Council, 1995). However, it is important to stress that local and regional effects are likely to be much more important in shaping energy policies in most countries than the concerns for global climate change.

MAIN NUCLEAR POWER ACCIDENTS

When safety measures and principles are ignored or are not properly observed by nuclear plant operators, a nuclear accident may occur with serious consequences for the environment and human health. For this reason, safety assessment should be carried out for a nuclear power plant to determine whether an adequate level of safety has been achieved for the plant and whether the safety objectives and safety criteria as specified by the plant designer, the operating organization, and the regulatory body has been met. Safety assessment should be a systematic process throughout the lifetime of the nuclear power plant to identify radiation risks that arise for workers, the public and the environment during normal operation, in anticipated operational occurrences, and in accident conditions (including severe accidents). The aim of safety assessment is to determine whether adequate measures have been taken to control radiation risks to an acceptable level, with account taken of both the prevention of abnormal events and the mitigation of their consequences.

Since 1959, ten major nuclear accidents have been occurred in five countries. These are the following:

- Fukushima, Japan - March 2011;
- Kashiwazaki, Japan - July 2007;
- Mihama, Japan - August 2004;
- Blayais, France - December 1999;
- Tokaimura, Japan - September 1999;
- Tokaimura, Japan - March 1997;
- Chernobyl, Ukraine - April 1986;
- Three Mile Island, USA - March 1979;
- The Urals, USSR - October 1958;
- Windscale, UK – October 1957.

Out of these ten major nuclear accidents, three of them had serious negative consequences for the environment, human health, and public opinion. These accidents, different from each other, are the following:

- Three Miles Island;
- Chernobyl;
- Fukushima.

The first accident occurred during the normal operation of the nuclear power plant; the second accident occurred during a test designed to assess the reactor's safety margin in a particular set of circumstances; and the third accident was the result of an earthquake of magnitude 9 and the tsunami that hit the east coast of Honshu in Japan in March 2011, affecting major equipment in the nuclear power plant, particularly the equipment associated to the safety system of the plant.

Three Miles Island Nuclear Accident

According to Morales Pedraza (2012), the accident at the Three Mile Island Unit 2 (TMI-2) nuclear power plant located near Middletown, Pennsylvania, in the USA, occurred on March 28, 1979. It was the most serious nuclear accident in US commercial nuclear power plant operating history, even though it led to no deaths or injuries to plant workers or members of the nearby community and the negative impact on the environment was minimum. What caused this nuclear accident? The nuclear accident was caused by a sequence of events such as equipment malfunctions, design-related problems and worker errors, which led to a partial meltdown of the TMI-2 unit core but with only very small off-site releases of radioactivity.

The accident began about 4:00 a.m. with a failure in the secondary non-nuclear section of the nuclear power plant. The main feed water pumps stopped running, caused by either a mechanical or electrical failure, which prevented the steam generators from removing heat. First the turbine and then the reactor automatically shut down. Immediately, the pressure in the primary system, which is the nuclear portion of the nuclear power plant, began to increase. In order to prevent that pressure from becoming excessive, the pilot-operated open a valve located at the top of the pressurizer. The valve should have closed when the pressure decreased by a certain amount, but it did not. As a result, cooling water poured out of the stuck-open valve and caused the core of the reactor to overheat.

As coolant flowed from the core through the pressurizer, the instruments available to reactor operators provided confusing information. There was no instrument that showed the level of coolant in the core. Instead, the operators judged the level of water in the core by the level in the pressurizer, and since it was high, they assumed that the core was properly covered with coolant. In addition, there was no clear signal that the pilot-operated relief valve was open. As a result, as alarms rang and warning lights flashed, the operators did not realize that the plant was experiencing a loss-of-coolant accident, and took a series of actions that made conditions worse by simply reducing the flow of coolant through the core.

Source: Photograph courtesy of Ohio Citizen Action.

Figure 3. Three Miles Island nuclear power plant after the accident.

Because adequate cooling was not available, the nuclear fuel overheated to the point at which the long metal tubes which hold the nuclear fuel pellets ruptured and the fuel pellets began to melt. Although the TMI-2 unit suffered a severe core meltdown, the most dangerous kind of nuclear power accident that can occur in a nuclear power reactor, it did not produce the worst-case consequences that nuclear power reactor experts had long feared. In a worst-case accident, the melting of nuclear fuel would lead to a breach of the walls of the containment building and release massive quantities of radiation to the environment. Hopefully, this did not happen in the Three Miles Island nuclear accident.

Undoubtedly, public fear to the use of nuclear energy for the generation of electricity and distrust increased significantly after the Three Mile Island accident and, for this reason, NRC's regulations and oversight became broader and more robust, and management of the nuclear power plants in operation in the country was scrutinized more carefully. The problems identified from careful analysis of the events during those days have led to permanent and sweeping changes in how NRC regulates its licensees which, in turn, has reduced the risk to public health and safety. As result of the Three Miles Island nuclear accident, the construction of new nuclear power reactors in the USA stopped until today.

Chernobyl Nuclear Accident

According to Morales Pedraza (2012), the Chernobyl nuclear accident is the worst nuclear accident ever occurred in a nuclear power plant, considering the area contaminated and the number of countries and people affected. What happened in the Chernobyl nuclear power plant that caused this terrible accident from the environment and human health point of view? Initially, the accident at Unit 4 of the Chernobyl nuclear power plant was considered as resulted from a combination of design and technical deficiencies with a grave operator error. However, in a later report the IAEA put the main cause of the accident to the reactor's design. What really happens? According to WNAO's report, on 25 April prior to a routine shut down, the reactor crew at Unit 4 of the Chernobyl nuclear power plant began preparing for a test to determine how long turbines would spin and supply power to the main circulating pumps following a loss of main electrical power supply. This test had been carried out at Chernobyl nuclear power plant the previous year, but the power from the turbine ran down too rapidly, so new voltage regulator designs were to be tested.

Which was the purpose of the test to be performed in Unit 4 of the Chernobyl nuclear power plant in April 1986? It is well known that nuclear power plants not only produce electricity, they also consume electricity, for example to power the pumps that circulate the coolant. This electricity is usually supplied from the grid. If the source of electricity failed, most reactors are able to derive the required electricity from their own production. However, if the reactor is operating but not producing power, for example when in the process of shutting down, some other sources of supply are required. Generators are generally used to supply the required power, but there is a time delay while they are started. The test carried out at Unit 4 of the Chernobyl nuclear power plant was designed to demonstrate that a coasting turbine would provide sufficient power to pump coolant through the reactor core while waiting for electricity from the diesel generators. The circulation of coolant was expected to be sufficient to give the reactor an adequate safety margin.

In January 1993, the IAEA issued a revised analysis of the Chernobyl nuclear accident, attributing the main root cause to the reactor's design and not to operator error[8]. In 2005, the IAEA and the World Health Organization (WHO) reported that "only 56 people had died directly from the incident, mainly accident workers[9]. They estimated another 4 000 deaths among workers and local residents".

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[8] The IAEA in its 1986 analysis had cited the operators' actions as the principal cause of the accident.

[9] According to WNAO's source, the accident destroyed Unit 4 of the Chernobyl nuclear power plant, killing thirty operators and firemen within three months and several further deaths later. One person was killed immediately and a second died in hospital soon after as a result

Source: Photograph courtesy of Wikimedia Commons (Elena Filatova).

Figure 4. Chernobyl nuclear power plant with the sarcophagus.

After the Chernobyl nuclear accident the pressure of the international community to close nuclear power plants in operation in many countries increased significantly, independently of the type of nuclear power reactors used. In 1995, a memorandum of understanding was signed between the governments of the G-7 countries, the EC, and the Ukraine government, agreeing with the closure of all Chernobyl nuclear power reactors. Based on this memorandum, Unit 2 was shut down in October 1991 after a huge fire in the unit, Unit 1 on November 1996, and Unit 3 in December 2000.

Following the nuclear accident, Unit 4 was encased in a giant concrete sarcophagus (See Figure 4), constructed above the destroyed reactor by hundreds of thousands of soldiers and civilian, including nuclear experts, to prevent further leakage of radioactive material.

However, it is important to stress that the sarcophagus built in 1986 is considered to be unstable and could collapse in the future. A waste management facility began construction in 2001 for the treatment of fuel and other wastes from decommissioned Units 1, 2 and 3. A stabilizing steel structure was extended in December 2006 to spread some of the load on the walls damaged by the explosion. Undoubtedly, the current situation of Unit 4 still represents a serious potential threat to the Ukraine population, if actions are not taken as soon as possible to repair the whole structure of the sarcophagus.

of injuries received. Another person is reported to have died at the time from a coronary thrombosis. Acute radiation syndrome (ARS) was originally diagnosed in 237 people on-site and involved with the clean-up and it was later confirmed in 134 cases. Of these, twenty eight people died as a result of ARS within a few weeks of the accident. Nineteen more subsequently died between 1987 and 2004 but their deaths cannot necessarily be attributed to radiation exposure.

The Fukushima Daiichi Nuclear Accident

The Fukushima Daiichi nuclear accident, considered the second major nuclear accident after Chernobyl, is the third major accident that has been affected the world nuclear industry in the last thirty five years. The accident is the result of a severe climate disaster that was not foreseen that could happen by the construction of the nuclear power plant, killing around 20 000 persons and putting out of service important components of the safety system of the plant. It is important to stress that the type of nuclear power reactors in operation in the Fukushima nuclear power plant was of the boiling water reactor type (see Figure 8) constructed in the 1970s but without the changes introduced in this type of reactor by the USA to modify some failure in the design.

According to the IAEA and the Japan's Nuclear and Industrial Safety Agency, the most relevant events associated to the Fukushima Daiichi nuclear accident are the following: On March 2011 at 06:42 UTC, the IAEA Incident and Emergency Centre (IEC) was activated following notification from the Agency's International Seismic Safety Centre (ISSC) of the earthquake and of the potential for damage at four nuclear power plants located on the north-east coast of Japan as well as the potential for a tsunami. At 8:15 CET on the same day, the IEC received information from its ISSC confirming information about the earthquake of magnitude 9 that hit the east coast of Honshu, Japan. The IEC has received information from the Japan's Nuclear and Industrial Safety Agency that a heightened state of alert has been declared at 11:45 at Fukushima Daiichi nuclear power plant, as result of the earthquake and the tsunami that hit the east coast. A second earthquake of magnitude 6.5 has struck Japan near the coast of Honshu and the Tokai nuclear power plant. As result of these meteorological disasters, four nuclear power plants located on the north-east coast of Japan — Fukushima Daiichi and Fukushima Daini of the Tokyo Electric Power Company (TEPCO), Onagawa (Tohoku Power Company) and Tokai (Japan Atomic Power Company) could be damaged (IAEA GOV/INF/2011/8, 2011).

Japanese authorities have informed the IEC that the earthquake and tsunami have cut the supply of off-site power to the Fukushima Daiichi nuclear power plant. In addition, diesel generators intended to provide back-up electricity to the plant‘s cooling system were disabled by tsunami flooding. At Fukushima Daiichi nuclear power plant, officials have declared a nuclear emergency situation, and at the nearby Fukushima Daini nuclear power plant a heightened alert condition.

Source: Tokyo Electric Power Co.

Figure 5. Fukushima Daiichi nuclear power plant after the accident.

On March 12 at 12:40 UTC, the Japan's Nuclear and Industrial Safety Agency has informed the IEC, that there has been an explosion at the Unit 1 at the Fukushima Daiichi nuclear power plant, and that they are assessing the condition of the reactor core. In addition, there has been an explosion at the Unit 3. The explosion occurred at 11:01 am local time. Unit 1 is being powered by mobile power generators on site, and work continues to restore power to the entire nuclear power plant. There is currently no power via off-site power supply or backup diesel generators being provided to the nuclear power plant. Seawater and boron are being injected into the reactor vessel to cool the reactor.

Due to the explosion on 12 March 2011, the outer shell of the containment building has been lost. Unit 2 is being powered by mobile power generators on site, and work continues to restore power to the entire nuclear power plant. The reactor core is being cooled through reactor core isolation cooling, a procedure used to remove heat from the core. The current reactor water level is lower than normal but remains steady. The outer shell of the containment building was intact at Unit 2 at that time.

According to the information released by the Japanese government, Unit 3 does not have off-site power supply or backup diesel generators providing power to the nuclear power plant. As the high pressure injection system and other attempts to cool the nuclear power reactor core failed, injection of water and boron into the reactor vessel commenced. Water levels inside the reactor vessel increased steadily for a certain amount of time but readings indicating the water level inside the pressure vessel were no longer showing an increase. To relieve pressure, venting of the containment started on 13 March at 9:20 am local time. Planning to reduce the concentration of hydrogen inside the containment building was carried out. The containment building was intact at Unit 3 at that time.

On March 14 at 06:00 UTC, the Japan‘s Nuclear and Industrial Safety Agency has provided further information about the hydrogen explosion that occurred at the Unit 3 at the Fukushima Daiichi nuclear power plant. Another hydrogen explosion occurred at Unit 3 at 11:01 am local time. Six people have been injured as resulted in the explosion. The reactor building exploded but the primary containment vessel was not damaged. The control room of Unit 3 remained operational at that time. At 22:03 local time, Japanese authorities have reported that Unit 2 experienced decreasing coolant levels in the reactor core. Officials have begun to inject sea water into the reactor to maintain cooling of the reactor core. Sea water injections into Units 1 and 3 were interrupted the day before due to a low level in a sea water supply reservoir, but sea water injections were restored at both Units. A fire at Unit 4 occurred at 23:54 UTC and lasted two hours.

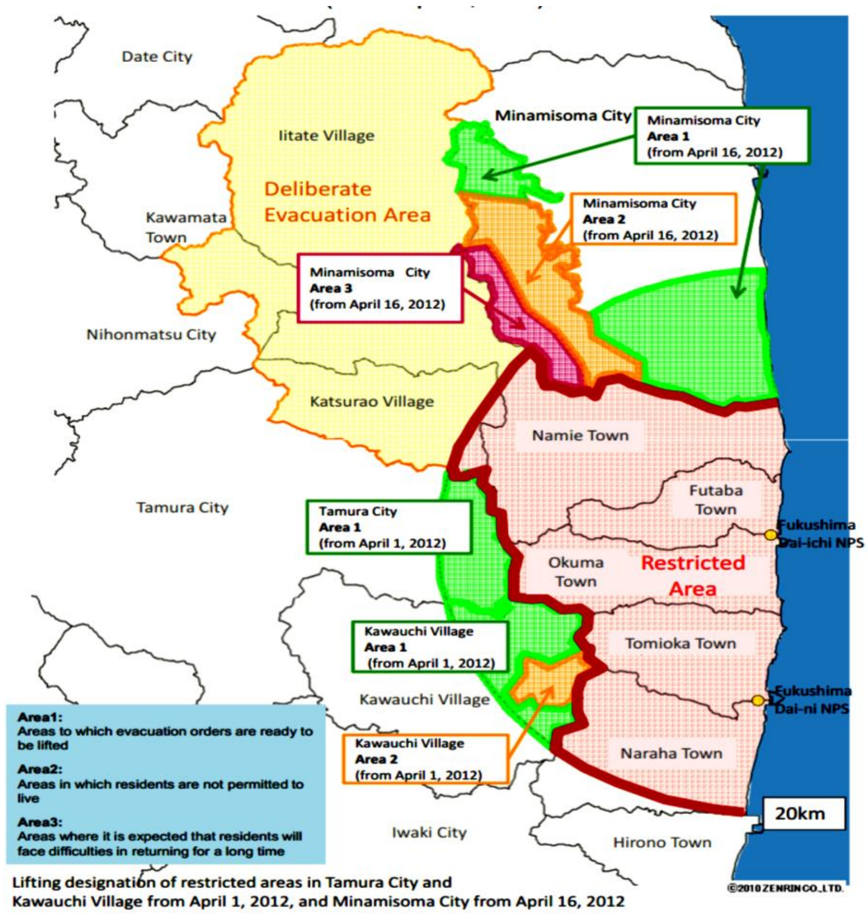
On March 15 at 00:16 UTC, plant operators considered the removal of panels from Units 5 and 6 reactor buildings to prevent a possible build-up of hydrogen in the future. It was a build-up of hydrogen at Units 1, 2 and 3 that led to explosions at the Fukushima Daiichi nuclear power plant. After explosions at both Units 1 and 3, the primary containment vessels of both units are reported to be intact. However, the explosion that occurred at 21:14 UTC on 14 March 2011 at Unit 2 affected the integrity of its primary containment vessel. All three explosions were due to an accumulation of hydrogen gas. Japanese authorities also informed at 04:50 CET that the spent fuel storage pond at the Unit 4 reactor of the Fukushima Daiichi nuclear power plant was on fire and radioactivity was released directly into the atmosphere. Dose rates of up to 400 mSv per hour have been reported at the site. These authorities said that there is a possibility that the fire was caused by a hydrogen explosion. Japanese authorities informed that there has been an explosion at the Unit 2. The explosion occurred at around 06:20 on 15 March 2011 local time. Attempts to return power to the entire Fukushima Daiichi nuclear power plant were also carried out. Japanese authorities reported some casualties to nuclear plant workers. At Fukushima Daiichi nuclear power plant, four workers were injured by the explosion at Unit 1, and there are three other reported injuries in other incidents. In addition, one worker was exposed to higher-than-normal radiation levels that fall below the IAEA guidance for emergency situations.

At Fukushima Daini nuclear power plant, one worker died in a crane operation accident and four others have been injured.

On March 19, Japan‛s Chief Cabinet Secretary Yukio Edano said that sea water injection were carried out at Units 1, 2 and 3 at the Fukushima Daiichi nuclear power plant. Preparations were made to spray water into the used fuel pool at Unit 4, and an unmanned vehicle sprayed more than 1 500 gallons of water over seven hours into the used fuel pool at Unit 3. The situation at the Unit 3 fuel pool was stabilized. Some reactor cooling capacity has been restored at Units 5 and 6 after the installation of generators at those reactors. Progress had been made on a fundamental solution to restore power at the Fukushima Daiichi nuclear power plant, with electricity restored at Units 1 and 2 on March 19 and Unit 3 as early as Sunday.

On March 20 at 2.05 pm GMT, workers on site succeeded in increasing the stability of the Fukushima Daiichi reactor units with Units 5 and 6 now in cold shut down. Pressure built up within Unit 3 but a more significant venting was not seemed necessary at that time. External power has now been connected to Units 5 and 6, allowing them to use their residual heat removal systems and transfer heat to the sea. This has been used to cool the fuel ponds and bring the units to cold shut down status, meaning that water in the reactor system was at less than 100°C. An extended operation to refill the fuel pond took place at Unit 3, with the Hyper Rescue crew spraying for over 13 hours. A similar operation is planned for Unit 4. At Units 1 and 2, external power was restored. Tokyo Electric Power Company (TEPCO) said it would restore functions in the central control room shared by the units so that accurate readings could again be taken from the reactor system. Next, workers checked the condition of the water supply systems to the nuclear power reactors and the used fuel pond. External power for Units 3 and 4 was in place a few days later.

The Japanese authorities have initially classified the accident at Fukushima Daiichi nuclear power plant as a level 4 ―Accident with Local Consequences on the International Nuclear and Radiological Event Scale (INES) of the IAEA. Later on the nuclear accident was classified by the IAEA as level 7 (the same level of the nuclear accident at the Chernobyl nuclear power plant) due to the characteristics of the accident. However, it is important to stress that the radioactive materials liberated as a result of the nuclear accident in the Fukushima nuclear power plant was estimated to be only 10% of the radioactive materials that were released by the nuclear accident in the Chernobyl nuclear power plant.



Source: IAEA.

Figure 6. Affected area.

After the Fukushima Daiichi nuclear accident, the use of nuclear energy for the generation of electricity and its future in Japan have polarized the public opinion, with thousands of protesters[10] demanding its abandonment

[10] Almost 70% of Japanese say their country should reduce its reliance on nuclear energy, in a poll conducted in 2012 as the country's last nuclear power plant went offline. This is a much larger number taking this position than in the weeks following last year's nuclear meltdown at the quake and tsunami-damaged Fukushima Daiichi nuclear power plant. Just 4% of Japanese say the country should expand the use of nuclear power in the coming years.

while some government officials insisting that it remains necessary in order to satisfy, in the most effective ad economic manner, the country energy demand.

As result of the Fukushima Daiichi nuclear accident, a total of 35 units of the country's 54 nuclear power reactors were offline – either damaged, halted by the quake and resulting tsunami or down for routine repairs. The approved programme for the construction of 14 new nuclear power reactors was suspended. Since March 11, Japan has been unable to restart any of its nuclear power reactors that were temporally shut down, scuttled by local opposition and its own meandering policies. That alone has led to nationwide energy shortages, tightening margins for businesses and other activities. But the energy shortages could become more severe in coming months, as the nuclear power reactors that are still operating now come off-line for scheduled tests.

The Ministry of Environment has announced that to clean the areas surrounded the Fukushima Daiichi nuclear power plant that has been contaminated, around 29 million m^3 of contaminated soil has to be removed. Billions of dollars have been approved by the Japanese government for this work as well as for recovering the contaminant area. It is expected that the process of cleaning the contaminated area needs around forty years to be completed. The damage provoked by the nuclear accident in the Fukushima Daiichi nuclear power plant was estimated to be around € 156,500 million. According to Leonid Bolshov, director of the Institute for the Secure Development of Atomic Energy of the Russian Academy of Science, there are two possibilities that can be considered for the clean-up of the Fukushima Daiichi nuclear power plant site and surrounded areas: a) dismantling and burial of all elements and components of the plant; and b) the construction of sarcophagus for each of the nuclear power reactors damage by the nuclear accident.

The main questions that need to be asked now is the following: this type of nuclear accidents can be totally eliminated in the future?; what types of nuclear technologies are under development now that can increase the safety operation of new nuclear power reactors to be constructed in the future with the aim of reducing to the minimum the possibility of a severe nuclear accident? A summary of the newest nuclear technologies under development in several countries are described in the following paragraphs, including the use of advanced construction methods for new nuclear power plants to be built in the future. However, it is important to stress that there is no nuclear technology or any other energy technology that can be 100% secure and, for this reason, all rational measures should be adopted to reduce to the minimum the possibility that a severe nuclear accident could occur in the future.

CURRENT AND FUTURE NUCLEAR TECHNOLOGIES

Types of Fission Nuclear Power Reactors[11]

Nuclear reactors are devices designed to produce and maintain a controlled chain nuclear reaction. There are two different types of nuclear reactors differentiated by their purpose and by their design features. Considering its purpose, they can be classified in two groups: a) nuclear research reactors, and b) nuclear power reactors.

Nuclear research reactors are devices that operate at universities and research institutions in many countries, including in countries where no nuclear power reactors are currently in operation for electricity generation, heat or desalination purposes. This type of reactors is used for multiple purposes, including the production of radiopharmaceuticals, medical diagnosis and therapy, testing materials, and conducting basic research.

Nuclear power reactors are those devices found in nuclear power plants and are used for generating heat mainly for electricity production. However, this type of reactors can be used also for desalination of water and heating. In the form of smaller units, they also power ships.

There are many different types of nuclear power reactors but what is common to all of them is that they produce thermal energy that can be used for its own sake or converted into mechanical energy and ultimately, in the vast majority of cases, into electrical energy. In this type of reactors, the fission of heavy atomic nuclei, the most common of which is uranium-235, produces heat that is transferred to a fluid which acts as a coolant. The heated fluid can be gas, water or a liquid metal. The heat stored by the fluid is then used either directly (in the case of gas) or indirectly (in the case of water and liquid

[11] Fission occurs when a nucleus absorbs a neutron and splits it into two approximately equal parts, known as fission fragments, and ejects several high-velocity fast neutrons in the process. The reactors that use fission to produce heating are called “fission nuclear power reactors”. The fission process concerns only heavy nuclides. It could be spontaneous or a result of nuclear reaction (neutron-induced, or other light or heavier particles-induced). The most important for reactor applications is of course primarily neutron-induced fission reactions and, to less extend spontaneous fission. The international community is also developing the so-called “fusion nuclear power reactors”. Fusion is the process by which two light atomic nuclei combine to form a heavier one. The long-term objective of fusion research is to harness this process to help meet mankind’s future energy needs. It has the potential to deliver large-scale, environmentally benign, safe energy, with abundant and widely available fuel resources. No commercial fusion nuclear power reactors has been produced until today and it is expected, according with the results of the ongoing research in fusion activities, that there will be any of this type of reactors available in the market before 2050.

metals) to generate steam. The heated gas or the steam is then fed into a turbine driving an alternator, which produce the electricity.

Nuclear power reactors can be also classified according to the type of fuel they use to generate heat. These are: a) uranium-fuelled nuclear power reactors; and b) plutonium-fuelled nuclear power reactors.

Uranium–Fuelled Nuclear Power Reactors

Uranium–fuelled nuclear power reactors can be classified in three different groups:

- Pressurized water reactors (PWR[12]), including the pressurized heavy water reactor (PHWR);
- Boiling water reactors (BWR);
- Graphite-moderate gas-cooled nuclear power reactors (GCR).

They are generally available in sizes of about 1 000 MW or greater electrical output. Slightly smaller reactors of 600–700 MW output is also available using water reactor technology. However, if a smaller unit is required due to the capacity of the national grid network, then the available technology is limited, although reactors of 200–400 MW output are being operated and developed by some countries. Several designs are being developed for future applications although a major challenge is to achieve an economic design at a smaller size. High temperature gas cooled reactors (160–270 MW) and several small water cooled reactors are being developed, which may reach design approval during the coming years. In addition, a barge mounted moveable 70 MW output nuclear power plant is currently under construction in Russia.

The only natural element currently used for nuclear fission as fuel is uranium. In the case of the PWRs, the fuel used is dioxide of uranium, and in the case of the PHWRs, the fuel used is the so-called “enriched uranium”. Natural uranium is a highly energetic substance: one kilogram of uranium can generate as much energy as ten tons of oil.

It is also a common practice to classify nuclear power reactors according to the nature of the coolant and the moderator plus, as the need may arise, other design characteristics. The light water reactors category comprises PWR

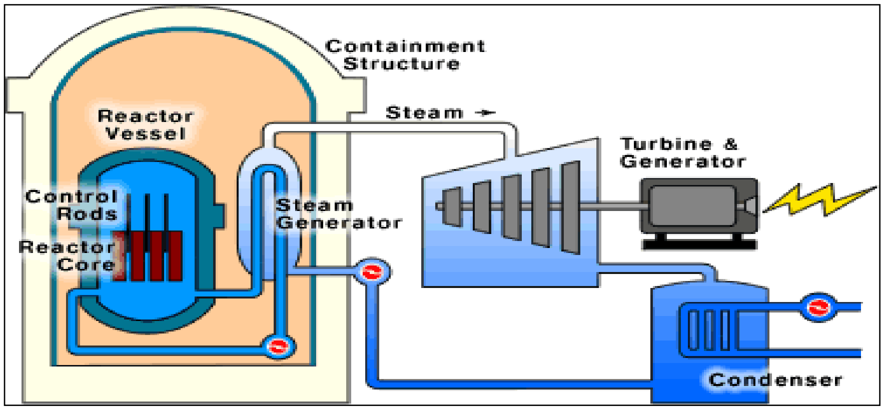
[12] WWER is the PWRs produced in the former Soviet Union, now Russia.

and BWR. Both types of nuclear power reactors use light water as moderator and coolant and enriched uranium as fuel.

The light water reactors operate in the following manner: the light water flows through the nuclear reactor core, a zone containing tens of thousands of long (4 m), thin (1 cm) nuclear fuel rods submerged in a water bath where it picks up the heat generated by the fission of the uranium-235 present in the fuel rods. After the coolant has transferred the heat it has collected to a steam turbine, it is sent back to the reactor core, thus flowing in a loop called the primary circuit. In order to transfer high-quality thermal energy to the turbine, it is necessary to reach temperatures of about 300° C. It is the pressure at which the coolant flows through the reactor core that makes the distinction between PWRs and BWRs.

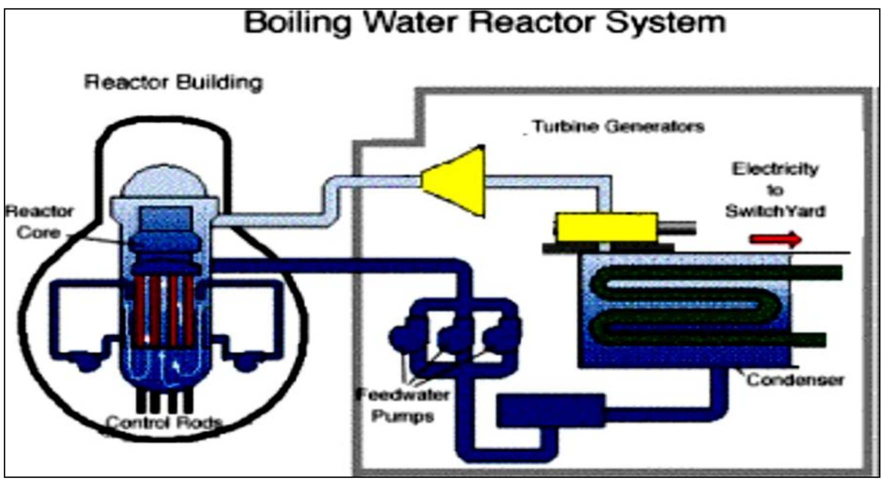
In PWRs, the pressure imparted to the coolant is sufficiently high to prevent it from boiling. The heat drawn from the fuel is transferred to the water of a secondary circuit through heat exchangers. The water on the secondary circuit is transformed into steam, which is fed into a turbine. The fission zone (fuel elements) is contained in a reactor pressure vessel under a pressure of 150 to 160 bar (15 to 16 MPa). The primary circuit connects the reactor pressure vessel to heat exchangers. The secondary circuit side of these heat exchangers is at a pressure of about 60 bar (6 MPa) - low enough to allow the secondary water to boil. The heat exchangers are, therefore, actually steam generators. Via the secondary circuit, the steam is routed to a turbine driving an alternator, which produces the electricity. The steam coming out of the turbine is converted back into water by a condenser after having delivered a large amount of its energy to the turbine. It then returns to the steam generator. As the water driving the turbine (secondary circuit) is physically separated from the water used as reactor coolant (primary circuit) the turbine-alternator set can be housed in a turbine hall outside the reactor building. Safety concepts have been copied from French and German nuclear power reactors, but a new part is the core catcher underneath the reactor tank which, in the event of a full meltdown of the reactor core, prevents it from spreading.

It is important to stress that PWRs are the most common nuclear power reactors operating in different countries around the world (around 60% of the total). There were 270 PWRs in operation in twenty five countries in 2011 with a total net capacity of 248 364 MW. The load factor of the PWRs in 2011 was 81.8% (first place). The USA (69 units or 25.6% of the total) and France (58 units or 21.5% of the total) are the countries with the highest number of PWRs in operation in the world. The main components of the PWRs are shown in Figure 7.



Source: International Nuclear Safety Center, Argonne National Laboratory, USA.

Figure 7. Pressurized water reactors.



Source: International Nuclear Safety Center, Argonne National Laboratory, USA.

Figure 8. Boiling water reactor components.

In the case of the PHWR type, there were 47 units in operation in seven countries in 2011 (17.4% of the total) with a net capacity installed of 23 140 MW. The load factor of the PHWRs in 2011 was 76.6% (third place). Canada (18 units or 38.2% of the total) and India (18 units) are the countries with the highest number of PHWRs in operation in the world in 2011.

In BWRs, the pressure imparted to the coolant is lower than in a PWR to allow it to boil. It is the steam resulting from this process that is fed into the turbine. This basic difference between pressurized and boiling water reactors dictates many of the design characteristics of the two types of light water reactors. Despite their differing designs, it must be noted that the two types of reactors provide an equivalent level of safety. The fission zone of the BWRs is contained in a reactor pressure vessel, at a pressure of about 70 bar (7 MPa). At the temperature reached 290° C approximately, the water starts boiling and the resulting steam is produced directly in the reactor pressure vessel. After the separation of steam and water in the upper part of the reactor pressure vessel, the steam is routed directly to a turbine driving an alternator which produces the electricity. Since the steam produced in the fission zone is slightly radioactive, mainly due to short-lived activation products, the turbine is housed in the same reinforced building as the reactor.

In 2011, there were 84 BWRs in operation in nine countries with a net capacity installed of 77 726 MW. The load factor of this type of reactor in 2011 was 73.7% (fourth place). The USA (35 units or 41.7% of the total) and Japan (26 units or 30.9% of the total) are the two countries with the highest number of BWRs in operation in the world in 2011[13]. The main components of the BWRs are shown in Figure 8.

TYPES OF NUCLEAR POWER REACTORS UNDER DEVELOPMENT AND CONSTRUCTION

PWRs and PHWRs

In France and Germany, AREVA NP has developed a new large PWR type called the “European pressurized water reactor (EPR)” to meet European utility requirements and benefit from economies of scale through a higher power level relative to the latest series of PWRs produced in France (the N4 series) and Germany (the KONVOI series) (IAEA GC (51)/INF/3, 2007). The USA is also working in a design for a large advanced PWR type, the so-called “combustion engineering system 80+” with the purpose of building several units in the country and abroad in the future.

In the Russian Federation, evolutionary versions of the current WWER-1000 (V-320) reactor, the Russian version of the Western PWR type, including the 1 200 MWe AES-2000 and WWER-1000 (V-392) designs have been

[13] It is important to stress that after the Fukushima Daiichi nuclear accident almost all BWRs operating in Japan were shut down.

developed. The first WWER-1000 was connected to the grid at Tianwan, China in 2006. Additional units are under construction in China and India. Two units are planned at Russia’s Novovoronezh site. Russia has also begun development of a larger WWER-1500 design. On July 2009, Russia and Kazakhstan created a joint venture to complete the design of a 200-400 MWe VBER-300 reactor for use in either floating or land-based co-generation power plants (IAEA GC (51)/INF/3, 2007).

The heavy water reactor technology used in the PHWRs was initially developed by the Atomic Energy of Canada Limited’s (AECL’s) from Canada and by Siemens and Kraftwerk Union (KWU) from Germany. In the first case, the type of reactor produced was the so-called “CANDU” reactor. There are several CANDU reactors operating in some countries, such as Canada, Argentina, Romania, among others (See Table 3).

In the second case, the reactor produced is the MZFR reactor[14], the first one built in the Karlsruhe Nuclear Research Center in Germany with a capacity of 65 MW. The MZFR was the type of reactor used as reference for the construction of the first nuclear power reactor in Argentina (Atucha 1) in 1968. It has a pressure vessel, unlike any other existing heavy water reactor, and it now uses slightly enriched (0.85%) uranium fuel, which has doubled the burn-up and consequently reduced operating costs by 40%. Now AECL is producing the advanced CANDU reactor (ACR) design using slightly enriched uranium fuel to reduce the reactor core size, which at the same time reduces the amount of heavy water required to moderate the reactor and allows light water to be used as a coolant.

Table 3. Number of CANDU-6 reactors in operation or under construction outside Canada

Countries	Number of CANDU-6 reactors in operation or under construction outside Canada
Republic of Korea	4
China	2
India	2
Romania	2
Pakistan	1
Argentina	1
Total	12

Source: CEA, 8th Edition, 2008.

[14] Multipurpose research reactor (Mehrzweckforschungreaktor) built by the Karlsruhe Research Center in Germany.

In 2005 and 2006, India connected the first two units using its new 540 MWe PHWR design at Tarapur. India is also designing an evolutionary PHWR with a capacity of 700 MWe.

BWRs

An economic and simplified boiling water reactor (ESBWR – 1400 MWe) has been developed and will be certified in the near future. Just as for the AP-1000 design[15], extensive simplifications have been implemented in this type of reactor as well. For example, the reactor core is cooled by natural circulation, which eliminates the need for coolant pumps. The Dutch nuclear sector has contributed greatly to this design, as the experimental Dodewaard reactor was used as a model for the ESBWR design. Another type of BWRs is the advanced boiling water reactor (ABWR with a capacity of 1 350 MWe) manufactured by General Electric. This design has already been certified in Japan, and four ABWRs are already operating in this country. The first two ABWRs began commercial operation in 1996 and 1997, and two more began commercial operation in 2005 and 2006. Two ABWRs are being constructed in Taiwan.

In Germany, AREVA NP, with international partners from Finland, France, the Netherlands and Switzerland, is developing the basic design of the SWR-1000, an advanced BWR type with passive safety features. A development programme was started in 1991 for ABWR-II with the goal of significantly reducing generation costs, partly through increased power and economies of scale. Commissioning of the first ABWR-II is foreseen in the late 2010s.

In the USA, a large BWR (General Electric's ABWR) was certified in 1997. Westinghouse's AP-600 and AP-1000 designs with passive safety systems were certified in 1999 and 2006 respectively. An international team led by Westinghouse is developing the modular integral 360 MWe international reactor innovative and secure (IRIS) with a core design capable of operating on a four-year fuel cycle. General Electric is designing a large

[15] The AP1000 is the American counterpart of the EPR with a slightly lower capacity (1 100 MWe). The design mainly involves a significant simplification of previous American systems (considerably fewer valves, pumps, and cables, among other components) with further developed passive safety systems, such as emergency heat supply and residual heat removal.

economic simplified boiling water reactor (ESBWR) combining economies of scale with modular passive safety systems (IAEA GC (51)/INF/3, 2007).

Gas Cooled Power Reactors

According to the IAEA and CEA information, there were eighteen operating gas cooled power reactors (GCR)[16] cooled by carbon dioxide plus two test reactors cooled by helium worldwide in 2007. All of these units are located in the UK with a net capacity of 9 034 MWe. The load factor of this type of reactor in 2011 was 68.2% (fifth place). In China, work continues on safety tests and design improvements for the 10 MWth high temperature gas cooled reactor (HTR-10), and plans are in place for the design and construction of the first power reactor prototype (HTR-PM).

The Russian Federation and the USA continue research and development on a 284 MWe gas turbine modular helium reactor (GT-MHR) for plutonium burning. France has an active research and development programme on both thermal as well as fast gas reactor concepts and, in the USA, efforts by the Department of Energy (DOE) continue on the qualification of advanced gas reactor fuel. To demonstrate key technological aspects of gas cooled fast reactors, an experimental reactor in the 50 MWth range is planned for operation around 2017 in France (IAEA GC (51)/INF/3, 2007).

Graphite-moderated gas-cooled nuclear power reactors, formerly operated in France and still operated in the UK, are not built any more in spite of some advantages that this type of reactors have.

[16] These are the so called “Generation III+”. In this type of reactor the uranium nuclear fuel is not contained in rods but in pebbles: spheres the size of tennis balls (See Figure 9). Helium is used as a coolant instead of water. The reactor operates at high temperatures (depending on the type up to around 900° C) and the hot helium gas is used to drive the turbines directly. This design has a much higher efficiency than that of water-cooled reactors: around 41% instead of 34%. In addition to this, these reactors are inherently safe. If the cooling gas is cut off the nuclear reaction will stop automatically. However, the pellets will temporarily continue to heat up and exceed the operating temperature. The pebbles can nevertheless withstand this peak temperature, as a result of which the radioactive material will remain inside the pebbles, even during the worst process disruption. As the reactor must be able to transfer the heat properly to the environment in the event of such a calamity, the reactors have been designed as thin, high columns (large surface area, small volume). This limits the capacity of pebble bed reactors to around 160 MWe. The pebbles can withstand temperatures of up to 1 600° C.

Pressure-Tube Boiling Water Reactors of Russian Design (RBMK)

RBMK type of reactors, which are cooled with light water and moderated with graphite, are now less commonly operating in some former Soviet Union bloc countries. In Russia, there were 15 RBMK in operation in 2011 with a net capacity of 10 219 MW. The load factor of this type of nuclear power reactor in 2011 was 80% (second place). Following the Chernobyl nuclear accident, the construction of this type of reactors outside Russia ceased and the government has decided, in 2010, the closure of all RMBK units in operation in Russia during the coming years.

Other Light Water Reactors

Other light water reactors in the market are the Korean standard nuclear plant (KSNP) series, the Chinese AC-600 design, and the CNP-1 000 for electricity production. China is also developing the QS-600 for electricity production and seawater desalination. Until 2008, eight KSNPs are in commercial operation. Based on the accumulated experience in the operation of the KSNPs, the Republic of Korea is now developing an improved KSNP type of reactor, the so-called “optimized power reactor” (OPR), with the first units planned for commercial operation in the beginning of the 2010s. The Korean next generation of nuclear power reactors, for which development began in 1992, is now named the “advanced power reactor 1400 (APR-1400)” and will be bigger to benefit from economies of scale. The first APR-1400 is scheduled to begin operation before 2013.

The South African Pebble Bed Modular Reactor Company Ltd is developing a 165 MWe pebble bed modular reactor (PBMR), which is expected to be commissioned at the beginning of the current decade. The South African government has allocated initial funding for the project and orders for some lead components have already been made.

In Japan, a 30 MWth high temperature engineering test reactor (HTTR) began operation in 1998, and work continues on safety testing and coupling to a hydrogen production unit. A 300 MWe power reactor prototype is also under consideration. However, after the Fukushima Daiichi nuclear accident the government stopped all development activities related to the design and testing of new nuclear power reactors.

Plutonium-Fuelled Nuclear Power Reactors

Plutonium (Pu) is an artificial element produced in uranium-fuelled nuclear power reactors as a by-product of the chain reaction. It is one hundred times more energetic than natural uranium: one gram of Pu can generate as much energy as one tonne of oil. As it needs fast neutrons in order to fission, moderating materials must be avoided to sustain the chain reaction in the best conditions. The current plutonium-fuelled nuclear power reactors, the so-called "fast breeder reactors", use liquid sodium, which displays excellent thermal properties without adversely affecting the chain reaction.

According to document IAEA GC (51)/INF/3 (2007), in China, the 25 MWe sodium cooled pool experimental fast reactor with a net capacity of 20 MW was connected to the grid in 2001. The next two stages of development will be the construction of a 600 MWe prototype of fast breeder reactor and the construction of a 1 000-1 500 MWe demonstration fast breeder reactor. In India, the fast breeder test reactor (FBTR) has been in operation since 1985 and the 500 MWe prototype fast breeder reactor (PFBR) is now under construction at Kalpakkam. It was scheduled for commissioning initially by the end of 2010 but has been postponed for different reasons.

In Japan, preparatory work began in 2005 on the necessary modifications to the 280 MWe prototype fast breeder MONJU reactor prior to its restart. To develop advanced fuels and materials, as well as technology for minor actinide burning and transmutation, the JOYO reactor, an experimental fast breeder reactor is expected to begin irradiation of oxide dispersion strengthened ferritic steel of uranium-plutonium MOX fuel containing 5% americium, and of MOX containing both neptunium and americium. Regrettably, after the Fukushima Daiichi nuclear accident the government stopped all development activities related to the MONJU and JOYO nuclear power reactors and it is impossible to predict at this stage if the development activities will continue to be supported by the government in the future.

In the Republic of Korea, the Korean Atomic Energy Research Institute has conducted research technology development and design work on the 600 MWe KALIMER-600 advanced fast breeder reactor concept. The conceptual design of this type of reactors was completed in 2006. The KALIMER-600 features a proliferation resistant core without blanket, and a decay heat removal circuit using natural sodium circulation cooling for a large power system. The KALIMER-600 conceptual design, which evolved on the basis of the KALIMER-150 (150 MWe) design, was selected as one of the promising next generation of nuclear power reactor candidates.

BN-600 in Russia is the world's largest operating fast breeder reactor and has now been in operation for twenty-six years. The 800 MWe BN-800 is under construction to commissioning planned for 2012-2013. Russia is also developing various concepts for advanced sodium cooled fast breeder reactors and for heavy liquid metal cooled reactors, specifically the lead cooled BREST-OD-300 and the lead bismuth eutectic cooled SVBR-75/100 systems.

In the USA, within the framework of the Global Nuclear Energy Partnership (GNEP), initial research and developing planning is underway for an advanced burner test reactor (ABTR) to demonstrate actinide transmutation in a fast spectrum, as well as innovative technologies and design features important for subsequent commercial demonstration power plants. Within the Generation IV International Forum, USA activities are focused on gas cooled and lead cooled fast reactors and small modular sodium cooled fast reactors (IAEA GC (51)/INF/3, 2007).

The future fission nuclear power reactors are expected to have, among others, the following advantages over the present generation:

- Lower investment costs and construction times;
- Simpler reactor designs;
- Modular units;
- More passive safety features;
- Low proliferation risks.

Despite of the progress achieved until now in the development of the fusion technology, for the time being, fission technology will be the main nuclear technology used for the construction of new nuclear power reactor designs at least until 2040.

Next Generation of Nuclear Power Reactors

Most of the advanced nuclear power reactor designs available today are evolutionary improvements on previous designs. This situation has the benefit of maintaining proven design features and thus minimizing technological risks improving, at the same time, some important features of current nuclear power reactor designs, on the basis of the lesson learned on past nuclear accidents and the experience gained in the construction of hundreds of nuclear power reactors in different countries. These evolutionary designs generally require little further research and development or confirmatory testing. Examples of commonly utilized elements of evolutionary design for improved economics are:

- Simplified reactor designs;
- Increased reactor power;
- Shortening the construction schedule, reducing the financial charges that accrue without countervailing revenue;
- Standardization and construction in series spreading fixed costs over several units;
- Productivity gains in equipment manufacturing, field engineering and construction;
- Multiple unit construction at a single site;
- Self-reliance and local participation.

Nevertheless, in the long-term, it is important to have more innovative designs that incorporate radical changes and promise significantly shorter construction times and lower capital costs that could help to promote a new era of nuclear power, particularly after the Fukushima Daiichi nuclear accident. Several innovative designs are in the small (< 300 MWe) to medium (300–700 MWe) size range because such designs are more attractive for the introduction of nuclear power in developing countries and for remote locations.

It is important to stress that the majority of the nuclear power reactors today in operation in the world are from the second generation of nuclear power reactors (the so-called "Generation II"). This generation of nuclear power reactors began to be built in the 1970s and is still operating in large commercial power plants in several countries[17]. However, most of the countries expanding their nuclear power programmes are constructing nuclear power reactors of the third generation (the so-called "Generation III"), which are more reliable and with a number of built-in safety features. This generation of nuclear power reactors was developed in the 1990s and incorporates a number of evolutionary designs that offer significant advances in safety and economics[18]. With the purpose of improving the Generation III type of nuclear power reactors, advances design are underway (the so-called "Generation

[17] It is important to stress that the three major nuclear accidents described in the chapter occurred in nuclear power plants built in the 1960s and 1970s.

[18] The third generation of nuclear power reactors is not extremely different to the second generation, but it does include a number of improvements in the field of safety, reliability and cost price of electricity generation. The third generation mainly concerns LWR. Five different manufacturers are marketing LWRs.

III+"), resulting in several near-term deployable plants that is actively under development and are being considered for deployment in several countries. New nuclear power reactors to be built between now and 2030 will likely be chosen using this type of reactor design.

It is important also to note that there is no clear definition of what constitutes a Generation III design, apart from it being designed in the past fifteen years. However, the main common features quoted by the nuclear industry are the following:

- A standardized design to expedite licensing and reduce capital cost and construction time;
- A simpler and more rugged reactor design, making them easier to operate and less vulnerable to operational upsets;
- Higher availability and longer operating life, typically sixty years;
- Reduced possibility of core melt accidents;
- Minimal effect on the environment;
- Higher burn-up to reduce fuel use and the amount of waste;
- Burnable absorbers (poisons) to extend fuel life.

These characteristics are clearly very imprecise and do not define very well what a Generation III reactor is. However, what can be said without any doubt is that Generation III reactors are evolved from existing designs of PWRs, BWRs and CANDU types of nuclear power reactors (Thomas, 2005).

There are a limited number of developing countries particularly interested in the development of commercial nuclear power reactor designs that are smaller than those currently offered on the market[19]. Smaller reactors would reduce the required initial investment and associated infrastructure costs, and they would be better suited to the small electrical grids of most of the developing countries. Innovative small and medium size reactors are under development for all principal reactor lines and some nonconventional

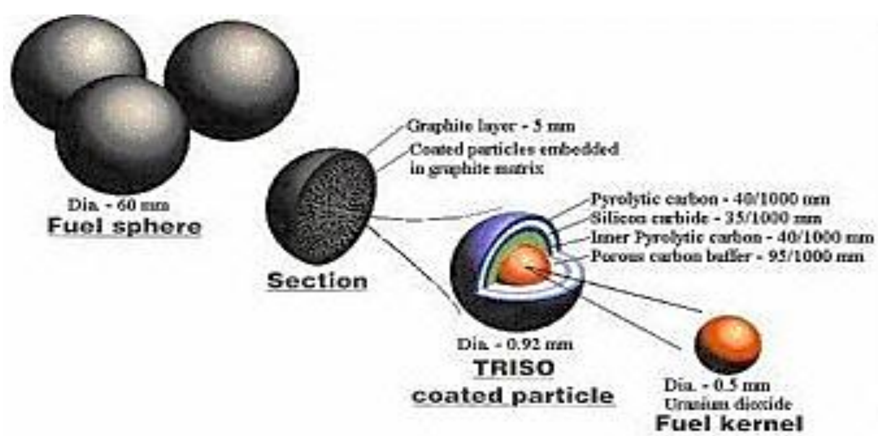
[19] A number of the small and medium size reactor designs are in the category of reactors without on-site refueling. These are reactors designed for infrequent replacement (every 5–25 years) of well contained fuel cassettes in a manner that impedes the clandestine diversion of nuclear fuel material. This category includes factory fabricated and fuelled reactors, and the general expectation is that the supplier country would retain all back end responsibilities for spent fuel and waste. The potential benefits include: possibly lower construction costs in a dedicated facility in the supplier country; lower investment costs and risks for the purchaser, especially if the reactor is leased rather than bought; reduced obligations for spent fuel and waste management; and possibly a higher level of assurance of non-proliferation to the international community (IAEA, 2006).

combinations. More than forty five innovative small and medium size reactor concepts and designs are at different stages of development within national or international research and development programmes, involving both developed and developing countries. Most allow for, or explicitly facilitate, non-electrical applications such as nuclear desalination or hydrogen production. Their target dates for being ready for deployment are before 2030. Some of the many designs in different stages of development are the following:

- The Korean Atomic Energy Research Institute has applied for a construction permit for a one-fifth scale, 65 MWth prototype of a system integrated modular advanced reactor (SMART) which cogenerates electricity while desalinating sea water.
- In the Russian Federation, a barge mounted floating 300 MWth KLT-40S cogeneration plant has been licensed for construction in Severodvinsk. The company manufacturer announced that the project had begun in April of 2007 in Severodvinsk in the White Sea. The ship called "Academic Lomonosov" is a lighter without autonomous propulsion and will be connected to the electric grid near the point in which the ship will be positioned. It will have two reactors KLT-40S with a power of 35 MW each. The reactor core is normally cooled by forced circulation, but the core design relies on convection for emergency cooling. Fuel is uranium aluminum silicide with enrichment levels of up to 20%. The assembly will be carried out in Viliutchinsk, south of the peninsula of Kamchatka. The cost of the project will be about €230 million and the lifetime of the floating nuclear power plant is considered to be fifty years. It is expected that seven units of this type will be built by Russia in the coming decades. China, the Republic of Korea, India, Brazil, Chile, Indonesia, Thailand, and Malaysia have shown some interested in the use of this type of nuclear power reactors for the generation of electricity in the future.
- The 165 MWe South African PBMR is planned for demonstration at full size by 2012-2013. The fuel used in this type of nuclear power reactors is in the form of pebbles instead of rods.
- Several integral PWR designs are well advanced in their development, and some could be available for deployment around 2015–2020. The 335 MWe IRIS design, developed by an international consortium led by Westinghouse Electric Company of USA, is the furthest along in

testing and development. Argentina has started licensing a 27 MWe prototype of the 150 to 300 MWe CAREM design. The first CAREM reactor will be built at the Atucha nuclear power plant site.

- In India, construction is expected to start early in the next decade on the first 300 MWe advanced heavy water reactor, which has been developed for co-generation applications. The reactor is designed to operate with 233 U-Pu-Th fuel; it uses boiling light water as a coolant and heavy water as the moderator. The reactor designer, the Bhabha Atomic Research Centre, is in pre-licensing negotiations with the Atomic Energy Regulatory Body of India.
- In Japan, the Toshiba Corporation, in cooperation with the Central Research Institute of Electric Power Industry and Westinghouse Electric company, is developing a sodium cooled reactor. It has a design power of 10 MWe and a refueling interval of thirty years. Construction of a demonstration reactor and safety tests are planned for the first half of the 2010s. However, the nuclear accident at the Fukushima Daiichi nuclear power plant stopped all activities related with the construction of this type of nuclear power reactor.
- In the USA, two private companies acquired the necessary intellectual property rights to proceed with the design development of two small nuclear power reactors without on-site refueling, and a heat-pipe based Hyperion power module employing uranium-hydride decomposable fuel.



Source: Dutch Research Platform for Sustainable Energy Supply.

Figure 9. Fuel in the form of pebbles.

Table 4. Generation III+ nuclear power reactors

Advanced boiling water reactors	• ABWR II (Advanced boiling water reactor II) • ESBWR (European simplified boiling water reactor) • HC-BWR (High conversion) • SWR-1000 (Siedewasser reactor-1000)
Advanced pressure tube reactor	• ACR-700 (Advanced CANDU reactor 700)
Advanced pressurized water reactors	• AP600 (Advanced pressurized water reactor 600) • AP1000 (Advanced pressurized water reactor 1000) • APR1400 (Advanced power reactor 1400) • APWR+ (Advanced pressurized water reactor plus) • EPR (European pressurized water reactor)[20]
Integral primary system reactors	• CAREM (Argentinean central modular elements) • IMR (International modular reactor) • IRIS (International reactor innovative and secure) • SMART (System-integrated modular advanced reactor)
Modular high temperature gas-cooled reactors	• GT-MHR (Gas turbine-modular high temperature reactor) • PBMR (Pebble bed modular reactor)

Source: DOE (2002).

In summary, sixteen designs could be deployed by 2015 or earlier. These are shown in Table 4 with acronyms or trade names.

However, undoubtedly the future belongs to the fourth generation of nuclear power reactors (the so-called "Generation IV"). This new generation of nuclear power reactors is a revolutionary type of reactors with innovative fuel cycle technologies. The main factors influencing the development of new generation nuclear energy systems in the 21st century will be economics, safety, proliferation resistance, and environmental protection, in addition to improved resource utilization and reduced waste generation. Adding to innovations designed to achieve improved fuel efficiency, there are other

[20] The first EPR, Olkiluoto-3 in Finland, is under construction with commercial operation expected initially to be in 2012. However, there have been a series of delays postponing several times the initial year of initiating operation increasing the construction cost. Also, Électricité de France has started construction of the second EPR at Flamanville, France, with completion anticipated by the beginning of the 2010s but the completion of the construction has been delayed several years for different reasons. AREVA has signed a contract to supply two EPR nuclear power reactors at the Taishan site in China; these are planned for entry into service in 2014. AREVA is also working on a version of the EPR to meet US requirements.

issues which require innovative approaches, including high temperature applications and designs for isolated or remote locations. According to IAEA (2008), specific innovative development approaches that could lead to improvements in efficiency, safety, and proliferation resistance include, among other benefits:

- Long life fuel with very high burn-up;
- Improved fuel cladding and component materials;
- Alternative coolant for improved safety and efficiency;
- Robust and fault tolerant systems;
- High temperature Brayton cycle power conversion[21];
- Thorium fuel design.

Why a new generation of nuclear power reactors is needed? The answer is the following: Generation IV initiative is the recognition that the current safety features of Generation III and Generation III+ is not enough to convince public opinion of several countries on the need to use nuclear energy for the generation of electricity in the future, particularly after the nuclear accident in the Fukushima Daiichi nuclear power plant. On the other hand, if the current global nuclear capacity of roughly 400 GWe is maintained, then it will be insufficient to reduce and stabilize CO_2 emissions to the atmosphere in the longer term, particularly due to a foreseeable increase in the energy demand all over the world. The increase in the energy demand in a group of countries such as China, India, South Africa, Brazil, South Korea, and Russia, among others will be high, and the use of different renewable energy sources for electricity production in the coming years will not be enough to satisfy this new demand. For this reason, the international community needs secure sources of energy such as nuclear power, which could deliver the highest power capacity in a manner which would be regarded as long-term sustainable and in the safest possible manner.

[21] The Brayton cycle is used for gas turbines only where both the compression and expansion processes take place in rotating machinery. The Brayton cycle is made up of four internally reversible processes: isentropic compression (in a compressor); constant pressure heat addition; isentropic expansion (in a turbine); and constant pressure heat rejection. All four processes of the Brayton cycle are executed in steady flow devices so they should be analyzed as steady-flow processes.

Generation IV Nuclear Power Reactors

The first fast breeder reactor in the world was Clementine at Los Alamos, USA. The reactor was commissioned in 1946 and used plutonium-239 in metal form as fuel. The experimental breeder reactor (EBR-I) in Idaho, USA, another fast breeder reactor built in this country in 1951, was the first reactor in the world to demonstrate how generation of electricity can be produced by the fission process. EBR-I used high enriched uranium (HEU: + 20% uranium-235) metal containing (+ 90% uranium-235) as fuel. Later in 1962, EBR-I demonstrated breeding of plutonium-239 from uranium-238, for the first time in the world.

According to the energy strategy of the Russian Federation, the government approved a transition from the present water cooled thermal reactors (WWER and RBMK) to fast breeder reactors with a closed fuel cycle during the coming decades. In addition to sodium cooled fast reactors, lead–bismuth cooled fast reactor, namely BREST 300 and BREST 1 200, are being studied. The Russian Federation has accumulated nearly four decades of experience in nuclear submarine reactors cooled with Pb and Pb-Bi alloy and has more than 125 reactor-years' operating experience with sodium cooled fast reactors. The experimental reactors BR-10 and BR-60 and the commercial reactor BN-600 have been extensively used to lay the foundation of sodium cooled fast reactors and its fuel cycle technology. BN-600, the only operating commercial sodium cooled fast reactors in the world today, is in operation since 1982 with a capacity factor exceeding 74%. The design of BN-800 is based on the design features proven in the course of construction and operation of the previous reactor BN-600.

The Dounreay Nuclear Power Development Establishment was started in 1955 primarily to pursue the UK government policy of developing fast breeder reactor technology. The Dounreay experimental fast reactor came on-line in November 1959. The prototype fast reactor of 250 MWe achieved criticality in 1974 and began supplying power in January 1975. The output of the prototype fast reactor was in operation up to 1994 and served as an invaluable test facility for developing advanced fuel and cladding materials that performed satisfactorily up to high burn-up and withstood high neutron dose. Both reactors have been shut down. With regard to the future programme related to fast breeder reactor and accelerator driven systems, the UK has been participating in several develop programmes. The focus of these programmes is on the incineration of Pu in a fast breeder reactor core, and the incineration of minor actinides and long lived fission products. The UK covers the domain of core physics, fuel performance modeling, and fuel cycle modeling.

In France, the first fast breeder reactor, Rapsodie, became operational in 1967 with mixed oxide fuel (MOX). Other fast breeder reactors used by France for the generation of electricity are the Phenix and Super-Phenix. France is firmly committed to nuclear power and has a constant nuclear power production of the current fleet of thermal reactors (PWRs) until about 2025, and thereafter a possible slight decrease (of about 15%) until 2040, followed by a constant supply of power. A license extension of current nuclear power plants is taken into account that Generation III+ reactors (advanced PWRs) would replace retired nuclear power plants of the current generation in 2025, and finally by around 2040 Generation IV reactors would be added. During the past four decades, France has gained extensive industrial scale experience in sodium cooled fast reactor fuel cycle with MOX fuel, including fuel design, fabrication, in-reactor performance, reprocessing and refabricating based on the lessons learned from Rapsodie, Phenix and Super-Phenix fast breeder reactors.

According to U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum (GIF)[22], the main goals for the Generation IV nuclear power reactors are the following:

- **Sustainability:** Generation IV nuclear power reactors will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production. It is expected that this type of nuclear power reactors will minimize and manage their nuclear waste, notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment, and will improve resource utilization and the reduction of nuclear waste generation;
- **Economics:** It is expected that Generation IV nuclear power reactors will have a clear life-cycle cost advantage over other energy sources and will have a level of financial risk comparable to other energy projects;
- **Safety and reliability:** It is expected that Generation IV nuclear power reactor operations will surpass in safety and reliability aspects

[22] The following countries and organizations are members of GIF: EURATOM, France, Japan, the Republic of Korea and the USA.

other nuclear power reactor designs and will have a very low likelihood and degree of reactor core damage. It is expected also that Generation IV nuclear power reactors will eliminate the need for off-site emergency response.

- **Proliferation resistance and physical protection:** It is expected that Generation IV nuclear power reactors will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials for the production of nuclear weapons, and will provide increased physical protection against acts of terrorism.

According to different government sources, it is expected that Generation IV nuclear power reactors may be available for commercial application before 2050.

Future nuclear power reactors must be designed so that during normal operation or anticipated transients safety margins are adequate, accidents are prevented, and off-normal situations do not deteriorate into severe accidents. At the same time, competitiveness requires a very high level of reliability and performance. There has been a definite trend over the years to improve the safety and reliability of nuclear power reactors, particularly after the Three Miles Island, the Chernobyl and Fukushima Daiichi nuclear accidents, reduce the frequency and degree of off-site radioactive releases, and diminish the possibility of significant reactor damage. Generation IV nuclear power reactors must ensure high levels of safety and reliability through further improvements in their designs that are safer and that can reduce the potential for severe accidents and their consequences to the environment and human health to the minimum (DOE, 2002). The achievement of these ambitious goals also requires high human performance and training as a major contributor to the plant availability, reliability, inspectability, and maintainability.

The following are the designs of Generation IV systems already under development on the basis of the set of criteria that have been established:

- Gas cooled fast reactor (GFR);
- Lead cooled fast reactor (LFR);
- Molten salt reactor (MSR);
- Sodium cooled fast reactor (SFR);
- Super critical water cooled reactor (SCWR);
- Very high temperature gas reactor (VHTR).

The above six Generation IV system designs are very different and also present different challenges that need to be solved in the ongoing research and development programmes carried out now in several countries in order to have all, or at least some of them, available in the market as soon as possible[23]. Some of these designs may still need significant additional research and development work before they can be considered ready for the production of electricity. The design and main features of the different Generation IV systems are briefly described in the following paragraphs.

Gas Cooled Fast Reactor (GFR)

The GRF system uses helium coolant operating a Brayton power cycle to generate electricity. The advantaged of the GFR system is its breeding capabilities. Fertile uranium, as well as several other fissile fuels, can be used without the need for neutron moderation. Due to the absorption properties of the fuels responsive to the fast neutron spectrum, fuel can be produced in the reactor over time, ultimately creating more fuel than what was originally installed in the core.

The reactor core of the GFR has been redesign with the aim of accommodating high temperature accident and fast neutron damage. This redesign has certain advantages but also disadvantages. These are the following

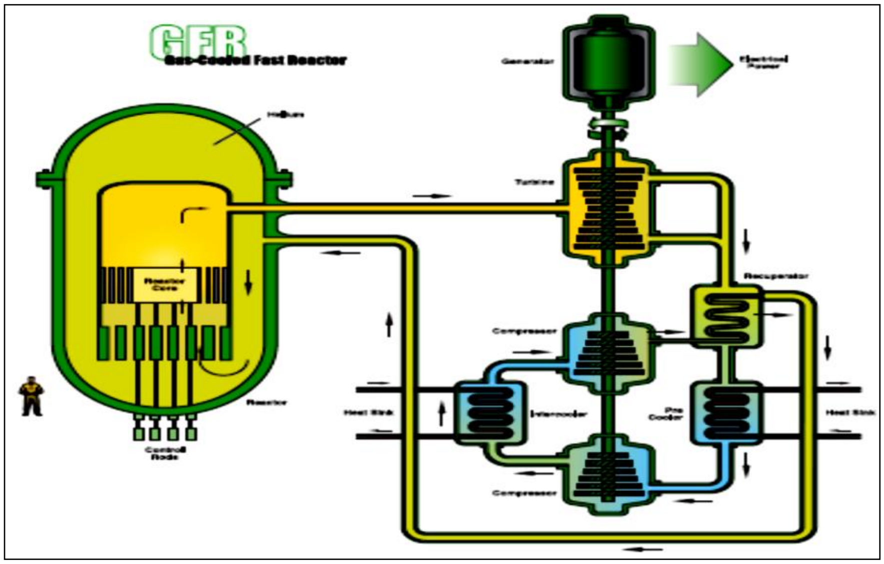
Advantages

- Breeding capabilities;
- Higher power density;
- New fuel design.

Disadvantages

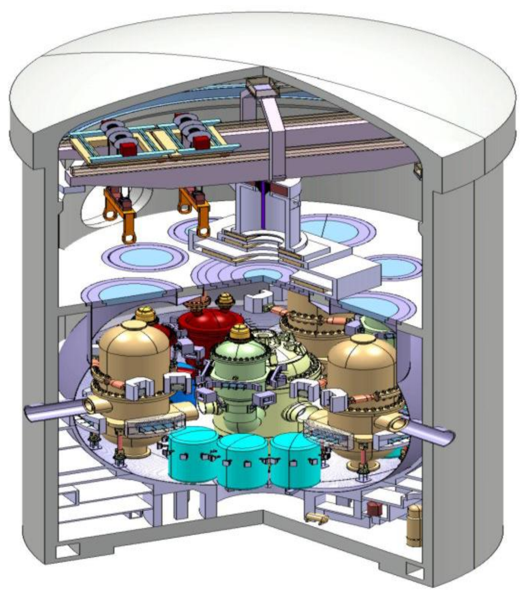
- Low thermal conductivity of helium;
- Fast neutron damage;
- Limited research.

[23] According to several experts' opinion, the SFR is the reference option; LFR and the GFR are the alternative options.



Source: DOE (2002).

Figure 10. Gas-cooled fast reactor system (GFR).



Source: Richard Stainsby from AMEC.

Figure 11. GRF building.

According to DOE (2002), the GFR system features a fast-neutron spectrum and closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle with on-site fuel cycle facilities is envisioned. The fuel cycle facilities can minimize transportation of nuclear materials and will be based on either advanced aqueous, pyro-metallurgical or other dry processing options. The reference reactor for this type of nuclear power reactor design is the 600 MWth (284 MWe) helium-cooled system operating in Russia and the USA with an outlet temperature of 850° C using a direct Brayton cycle gas turbine for high thermal efficiency (net efficiency: 48%). Several fuel forms are being considered for their potential to operate at very high temperatures and to ensure an excellent retention of fission products: composite ceramic fuel, advanced fuel particles, or ceramic clad elements of actinide compounds. Core configurations are being considered based on pin-or plate-based fuel assemblies or prismatic blocks.

According to expert Richard Stainsby, the GFR performance requirements are the following:

- Self-generation of plutonium in the reactor core to ensure uranium resource saving;
- Optional fertile blankets to reduce the proliferation risk;
- Limited mass of plutonium in the reactor core to facilitate the industrial deployment of a fleet of GFRs;
- Ability to transmute long-lived nuclear waste resulting from spent fuel recycling, without lowering the overall performance of the system;
- Favorable economics owing to a high thermal efficiency;
- The proposed safety architecture fits with the objectives considering the following elements: control of reactivity/heat generation by limiting the reactivity swing over the operating cycle; the coolant void reactivity effect is minor; capacity of the system to cool the reactor core in all assumed situations; provision of different systems (redundancy and diversification); a refractory fuel element capable of withstanding very high temperatures (robustness of the first barrier) and confinement of radioactive materials.

The specific challenges of the GFR system are the following:

- The greatest challenge facing the GFR system is the development of robust high temperature refractory fuels and core structural materials;

- This type of reactors must be capable of withstanding the in-core thermal, mechanical, and radiation environment;
- Safety and economic considerations demand a low reactor core pressure drop, which favors high coolant volume fractions;
- Minimizing the plutonium inventory leads to a demand for high fissile material volume fractions;
- Candidate compositions for the fissile compound include carbides, nitrides, as well as oxides;
- Favored cladding materials include: refractory metals and SiC for pin formats and refractory metals and ceramic matrices (e.g. SiC, ZrC, TiN) for dispersion fuels in a plate format;
- High power density, low thermal inertia, poor conduction path and small surface area of the reactor core conspire to prevent conduction cooling;
- A convective flow is required through the reactor core at all times. A natural convection flow is preferred following shut down. This is possible when the circuit is pressurized.. Gas density is too low to achieve enough natural convection. Power requirements for the blower are very large at low pressure;
- The primary circuit must be reconfigured to allow decay heat removal[24]. Main loop(s) must be isolated. Decay heat removal loop(s) must be connected across the core.

The technology base for the GFR system includes the following thermal spectrum gas reactor power plants and a few fast-spectrum gas-cooled reactor designs:

- The HTTR in Japan, which reached full power (30 MWth) using fuel compacts in 1999;
- The HTR-10 in China using pebble fuel.
- A 300 MWth pebble bed modular demonstration power plant designed for deployment in South Africa in the near future;
- A 300 MWth GT-MHR design under development by a consortium of Russian institutes in cooperation with General Atomics.

[24] The reliability of the decay heat removal function is dependent on the reliability of the primary circuit valves.

It is important to stress that the design of the PBMR and GT-MHR reactor systems, fuel, and materials are evolutionary advances of the demonstrated technology already in used in some nuclear power reactors in operation in some countries, except for the direct Brayton-cycle helium turbine, and the implementation of modularity in the plant design.

Finally, the GFR system may benefit from development of the above-mentioned technologies, as well as development of innovative fuel and very-high-temperature materials now under consideration for the VHTR. A phased development path may be drawn from the thermal to the fast-spectrum gas-cooled systems[25]. According to some experts' opinion, it is expected that a conceptual design of an entire GFR prototype system can be developed by 2019. The prototype system is envisioned as an international project that could be placed in operation by 2025 (DOE, 2002).

Lead Cooled Fast Reactor (LFR)

The LFR system uses lead or lead/bismuth as the primary coolant for within the reactor core. It makes use of the fast neutron spectrum and a closed fuel cycle. Small nuclear power reactors can be designed to handle 50 to 150 MWe, medium sized covering 300 to 400 MWe, and even a unit that could generate 1 200 MWe. This type of nuclear power reactor has been designed as a modular configuration, this means that the components making up the plants can me manufactured off site and brought in and pieced together. Each design is rated for anywhere from fifteen through thirty years of operation before any kind of reevaluation or modification upgrade should be done.

The main technical issue of LFR is related to the protection of the integrity of structural materials at high temperature[26]. The thermal cycle that has been therefore purposely selected with 400° C as core inlet temperature - to have sufficient margin above the lead freezing point and to avoid excessive embrittlement of structural material in fast neutron flux - and only 480° C as

[25] It is important to stress that the fast neutron reactor system is the only energy source which generates electricity and breeds its own fuel.

[26] Lead has a high melting point (327.4° C) and a very high boiling point (1 745° C). The high boiling point has a beneficial impact to the safety of the system, whereas the high melting point requires new engineering strategies to prevent freezing of the coolant and blockage of the circulation through the core. Lead is relatively corrosive towards structural materials especially at high temperatures with a consequent necessity to control its purity carefully. Due to its harsh environment coupled with high energy neutrons effects, an accurate choice of materials is required. Among the components, the fuel cladding material is one of the crucial issues.

mean core outlet temperature to mitigate corrosion, and to take advantages in term of creep and reduced thermal shocks in transient conditions. The drawback of such a thermal cycle is the need to increase the coolant flow rate which impacts on the primary system dimensions. This is due to the low lead velocities that can be achieved in order to reduce corrosion and erosion phenomena[27].

Additionally, the use of a coolant with very high density combined with large primary system makes the mechanical design challenging with respect to mechanical loads, particularly to seismic loads. Based on the above mentioned considerations, a large effort has been made to design an innovative primary system as compact as possible, to be accommodated in a short-height reactor vessel, this being a design feature considered basic for a robust design against seismic loads. The result is a type of reactor with very short vessel (around 9 m high), whose feasibility is confirmed by the preliminary mechanical analyses. This result, together with the elimination of the intermediate loop, opens the way to the feasibility of a competitive LFR (Tarantino et al, 2012).

The fuel to be used in this type of nuclear power reactor is uranium with either metal or nitride. The lead leaves the reactor core between 550° C and 800°C. The lead alloy has low neutron absorption and slow down power which facilitates natural circulation. The primary coolant loop operates unpressurized which allows future designs employing passive safety.

The LFR system has been designed specifically for the electricity generation. It can provide cheap and reliable energy because it will either be self-sufficient or can employ the transuranic elements process for refueling. The system also employs a range of new technologies within the plant. They are the following:

- Natural circulation;
- Lift pumps;
- Direct contact heat exchangers;
- Direct contact steam generators.

[27] The technical risk associated with the corrosive behavior of lead does not readily permit, with the present corrosion protection technology based on dissolved oxygen, assurance of the ability to achieve the decades-long lifetime of the high-temperature components normally required for nuclear application. The only possible outcome of this issue has been so far the demonstration of the possibility to remove all the primary system components immersed in lead and their replacement with spare components (Tarantino et al.,2012)

It is important to stress that the use of LFR design will have a number of advantages and disadvantages. These are the following:

Advantages

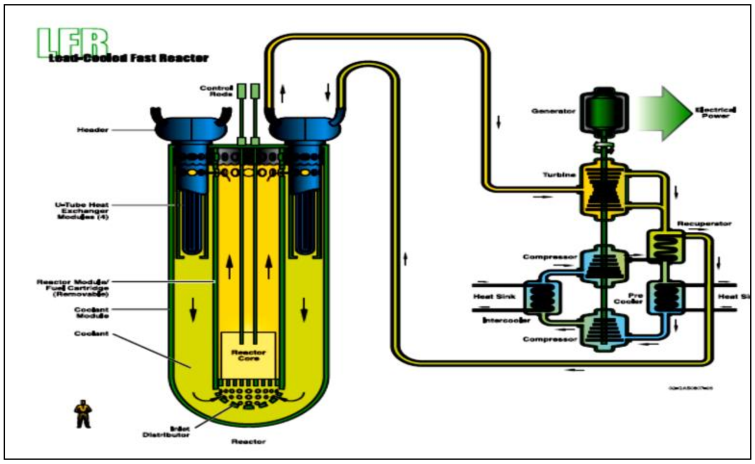
- Operates unpressurized removing potential loss-of-coolant accident;
- Fuel efficiency;
- Capabilities in terms of nuclear materials management (thereby mitigating proliferation risks);
- Design can be manufactured off site and assembled where needed;
- Design can be easily modified to operate with H_2;
- Reduced production of high-level radioactive waste and actinides;
- Each design is rated for anywhere from fifteen through thirty years of operation before any kind of reevaluation or modification upgrade should be done;
- Long refueling interval between ten and twenty years.

Disadvantages

- Requires a great deal of research and development to become available in the market[28];
- New fuel needs to be analyzed for performance specification;
- Component design places new risks on outside manufacturers.

According to DOE (2002) and the International Forum Generation IV, the LFR system can be used as a burner to consume actinides from spent LWR fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of molten lead as a relatively inert coolant. In terms of sustainability, lead is abundant and hence available, even in case of deployment of a large number of LFR systems. More importantly, as with other fast systems, fuel sustainability is greatly enhanced by the conversion capabilities of the LFR fuel cycle.

[28] The needed research activities are identified and described in the System Research Plan adopted in 2008 by the LFR Provisional System Steering Committee. It is expected that in the future, the required efforts could be organized into four major areas of collaboration. These areas are: system integration and assessment; lead technology and materials; system and component design; and fuel development (Tarantino et al, 2012).



Source: DOE (2002).

Figure 12. Lead-cooled fast reactor system (LFR).

It is important to single out that the LFR system was primarily envisioned for missions in electricity and hydrogen production and for actinide management as well. Given its research and development needs in the areas of fuels, materials, and corrosion control, a two-step process leading to industrial deployment of the LFR system has been envisioned: by 2025 for reactors operating with relatively low primary coolant temperature and low power density; and by 2035 for more advanced designs.

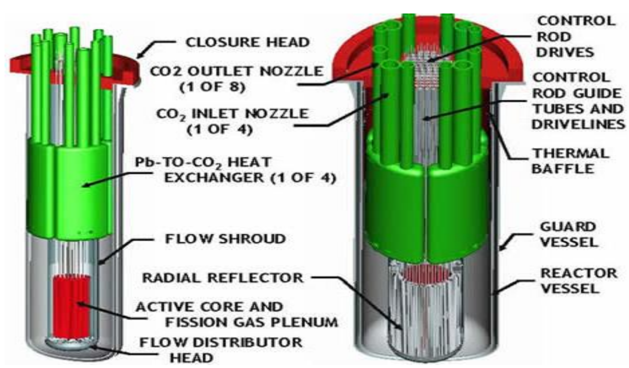
The preliminary evaluation of the LFR concepts considered by the LFR Provisional System Steering Committee (PSSC) covers their performance in the areas of sustainability, economics, safety and reliability, and proliferation resistance and physical protection. The LFR concepts that are currently being designed are two pool-type reactors:

- The small secure transportable autonomous reactor (SSTAR), developed in the USA;
- The European lead-cooled system (ELSY), developed by the EC.

The SSTAR system is a small factory-built turnkey plant operating on a closed fuel cycle with very long refueling interval (fifteen to twenty years or more) cassette core or replaceable reactor module. The current reference design for the SSTAR system in the USA is a 20 MWe natural circulation reactor concept with a small shippable reactor vessel (See Figure 13). Specific features of the lead coolant, the nitride fuel containing transuranic elements, the fast spectrum core, and the small size combine to promote a unique approach to achieve proliferation resistance, while also enabling fissile self-sufficiency, autonomous load, following simplicity of operation, reliability, transportability, as well as a high degree of passive safety features. Conversion of the core thermal power into electricity at a high plant efficiency of 44% is accomplished utilizing a supercritical carbon dioxide Brayton cycle power converter.

The initial design of ELSY system is almost complete (See Figure 14). The ELSY core is made up of 162 open square fuel assemblies arranged in three radial zones with different levels of plutonium enrichment: 56 fuel assemblies in the inner zone with a plutonium enrichment of 14%; 50 fuel assemblies in the intermediate zone with a plutonium enrichment of 17%; and 56 fuel assemblies in the outer zone with a plutonium enrichment of 19.9%. The fuel cycle management tentatively adopted is five years fuel residence time and the refueling of 25% of the core each 1.25 years. The fuel assemblies consist of 428 fuel pins arranged in a 21x21 square lattice.

The next step in its development is the research and development testing of several design innovations, in order to start with confidence the detailed engineering design of a reduced-scale demonstration facility. The ELSY reactor is rated at 600 MWe. This mid-size rating is the result of the fact that plants of the order of several hundred MWe are most economically attractive for addition to the European interconnected grids. In addition, a larger plant would require an increase mass of the lead coolant and would entail increased mechanical loads on the reactor vessel and its supporting structure.



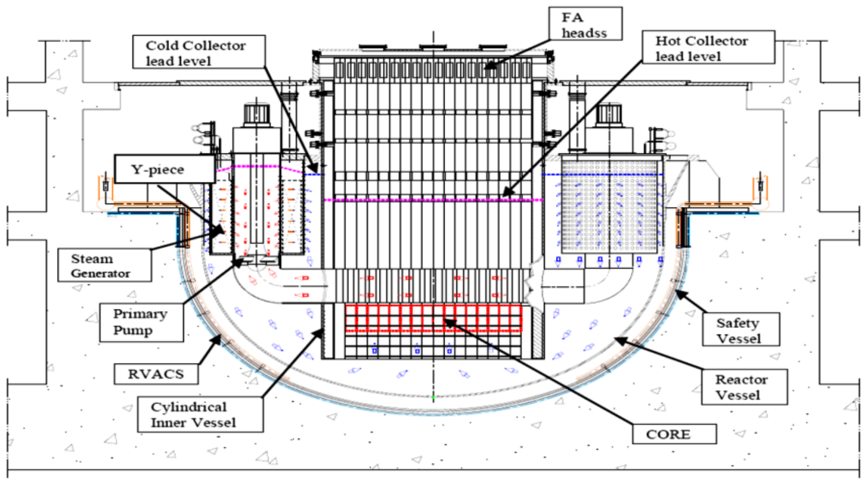
Source: International forum Generation IV.

Figure 13. Conceptual 20 MWe (45 MWth) SSTAR system.

Table 5. Key design data of SSTAR and ELSY systems

Parameters	SSTAR	ELSY
Power (MWe)	19.8	600
Conversion Ratio	~1	~1
Thermal efficiency (%)	44	42
Primary coolant	Lead	Lead
Primary coolant circulation (at power)	Natural	Forced
Primary coolant circulation for direct heat removal (DHR)	Natural	Natural
Core inlet temperature (° C)	420	400
Core outlet temperature (° C)	567	480
Fuel	Nitrides	MOX (Nitrides)
Fuel cladding material	Si-Enhanced Ferretic/Martensitic Stainless Steel	T91 (Aluminized)
Peak cladding temperature (° C)	650	550
Fuel pin diameter (mm)	25	10.5
Active core dimensions Heigh/equivalent diameter (m)	0.976/1.22	0.9/4.32
Primary pumps	-	Mechanical, integrated in the SG
Working fluid	Supercritical CO_2 at 20 MPa, 552°C	Water-superheated steam at 18 MPa, 450°C
Primary/secondary heat transfer system	Four Pb-to-CO_2 HXs	Eight Pb-to-H_2O SGs
Direct heat removal (DHR)	Reactor Vessel Air Cooling System + Multiple Direct Reactor Cooling Systems	Reactor Vessel Air Cooling System + Four Direct Reactor Cooling Systems + Four Secondary Loops Cooling Systems

Source: International forum Generation IV.



Source: Tarantino and others (2012).

Figure 14. ELSY reference configuration at the end of 2008.

The choice of a mid-size reactor power suggested the use of forced circulation to shorten the reactor vessel thereby avoiding excessive coolant mass and alleviating mechanical loads on the reactor vessel. Thanks to the favorable neutron characteristics of lead, the fuel rods have been spaced further apart than in the case of previous fast-neutron cores that were built. This and the innovative steam generators with flat spirals tube bundle enable the design of a low pressure loss primary loop. The needed pump head, in spite of the higher density of lead could, therefore, be kept low (on the order of two bars) with reduced requirement of pumping power.

The reactor module is designed to be factory-fabricated and then transported to the plant site. The reactor is cooled by natural convection and sized between 120 and 400 MWth, with a reactor outlet coolant temperature of 550° C, possibly ranging up to 800° C, depending upon the success of the materials research and development. The system is specifically designed for distributed generation of electricity and other energy products, including hydrogen and potable water.

The technologies employed in the development of this type of reactor are extensions of those currently available from the Russian Alpha class submarine Pb-Bi alloy-cooled reactors, from the integral fast reactor metal alloy fuel recycle and re-fabrication development, and from the advanced liquid metal reactor passive safety and modular design approach[29]. Existing ferritic stainless steel and metal alloy fuel, which are already significantly developed for sodium fast breeder reactors, are adaptable to Pb-Bi cooled reactors at reactor outlet temperatures of 550° C (DOE, 2002).

Finally, it is important to stress the following: the LFR research and development plan incorporates two tracks of improvement leading to a single joint demonstration facility by 2020. Separate designs for a small transportable LFR with a long core life and a moderate-sized LFR will be researched in the demonstration facility.

The LFR system under consideration offer great promise in terms of the potential for providing cost effective, simple and robust fast breeder reactor concepts that are essential to long-term sustainability of the nuclear energy option. Recent efforts, particularly in the development of the ELSY system, have gone a long way toward verifying the advantages of lead cooled systems. Clearly, additional work needs to be done, but overall, the prospects continue to appear very positive. ELSY system aims at demonstrating the possibility to design a fast breeder reactor using simple engineered technical features, whilst fully complying with the Generation IV goals of sustainability, economics, safety, proliferation resistance and physical protection (Tarantino et al, 2012).

[29] Research and design on the use of the lead-bismuth eutectic (LBE) alloy as a coolant for nuclear reactors was initiated in the early 1950s in Russia for military submarine propulsion. The first LBE- cooled nuclear submarine was put into operation in 1963 and in total fifteen units has been built including three land system reactors, plus one replacement reactor for submarines. However, LBE has two major drawbacks. The first one is represented by bismuth transmutation into highly radioactive polonium by neutron capture, which limits the access to the reactor and requires extensive use of robotic systems. The second one deals with re-crystallization: LBE undergoes expansion in the solid state which can damage the mechanical structures in case of freezing. In addition LBE has shown other inconveniences such as formation of solid impurities, black dust and macroscopic slag with consequent potential for filter and pipe occlusions, loop malfunctions, and cover gas piping blockage. Recent experiences acquired by ENEA have shown that this does not occur with pure lead (IAEA, 2011). For this reason, most of the civil reactor projects developed in the past years are based on pure lead as coolant. Among them, BREST-300 and BREST-1200 have been launched in Russia; ELSY and its evolutions European lead-cooled fast reactor (ELFR) and the advanced lead fast reactor (European Demonstrator LFR-Demo ALFRED) have been proposed in the framework of European projects, and SSTAR in USA. LBE is mainly reserved to experimental reactors because of the lower freezing temperature when compared to lead and for the large power density that can be obtained even at low operating temperature (Tarantino et al., 2012).

Table 6. LFR potential performance against the four goal areas and the eight goals for Generation IV

Generation IV Goal Areas	Goals for Generation IV Nuclear Energy Systems	Goals achievable via	
		Inherent features of lead	**Specific engineered solutions**
Sustainability	Resource utilization.	• Lead is a low moderating medium. • Lead has low absorption cross-section. • This enables a core with fast neutron spectrum even with a large coolant fraction.	• Conversion ratio close to 1.
	Waste minimization and management.		• Great flexibility in fuel loading including homogeneously diluted minor actinides.
Economics	Life cycle cost.	• Lead does not react with water. • Lead does not burn in air. • Lead has a very low vapor pressure. • Lead is inexpensive.	• Reactor pool configuration. • No intermediate coolant loops. • Compact primary system. • Simple design of the reactor internals. • Supercritical water (high efficiency).
	Risk to capital (Investment protection).		• Small reactor size. • Potential for in-vessel replaceable components. • Long refueling cycle.
Safety and reliability	Operation will excel in safety and reliability.	Lead as: • Very high boiling point. • Low vapor pressure. • High shielding capability for gamma radiation. • Good fuel compatibility and fission product retention.	• Primary system at atmospheric pressure. • Low coolant ΔT between core inlet and outlet.
	Low likelihood and degree of core damage .	Lead as: • Good heat transfer characteristics.	• Large fuel pin pitch. • Natural circulation cooling (small system).

Table 6. (Continued)

Generation IV Goal Areas	Goals for Generation IV Nuclear Energy Systems	Goals achievable via	
		Inherent features of Lead	**Specific engineered solutions**
		• High specific heat and thermal expansion coefficient. • Core with inherent negative reactivity feedback.	• Decay heat removal (DHR) in natural circulation. • Primary pumps in the hot collector (moderate - or large - size system) DHR coolers in the cold collector.
	No need for offsite emergency response.	• Lead density is close to that of fuel (considerably reduced risk of re-criticality in case of core melt). • Lead retains released fission products	
Proliferation resistance and physical protection	Unattractive route for diversion of weapon-usable material.	• Lead system neutronic enables long core life.	• Small system features sealed, long-life core. • Use of a MOX fuel containing minor actinides increases proliferation resistance.
	Increased physical protection against acts of terrorism.	• Primary coolant chemically compatible with air and water operating at ambient pressure.	• Simplicity in design. • Independent, redundant and diversified DHR loops. • No use of reactive or flammable coolant materials.

Source: International forum Generation IV.

Table 7. Summary of key issues, proposed strategies and research and development needs

General issue	Specific issue	Proposed strategy and needs in research and development
Lead technology	Lead purification Oxygen control.	Technology for the purification of large quantities of lead to be confirmed. Extend oxygen control technology to pure lead for pool reactors.
Materials resistant to corrosion on lead	Material corrosion.	Selection of a low core outlet temperature for initial reactor design. Development of new materials for service at temperature up to 650° C.
	Reaction vessel corrosion.	Vessel temperature limited by design to about 400° C.
	Fuel cladding.	15-15 Ti. Selection of aluminized surface treated steel for cladding.
	Reactor internals.	Materials protected by oxygen control.
	Heat removal.	Confirmation of the suitability of aluminized steels for steam generator to avoid lead pollution and heat transfer degradation.
	Pump impeller.	Test of innovative materials at high lead speed.
Potentially high mechanical loading	Earthquake.	Reactor building built with 2D - seismic isolators + short vessel design.
	SGTR accident.	Prevention by design of: • Steam entrainment into the core. • Reactor vessel pressurization. • Pressure wave propagation across the primary system.
Main safety functions	Diversified, reliable, and redundant DHR.	Use of both atmospheric air and pool water.
	Diversified, reliable, and redundant reactor shut down system.	Confirmation of operation of diversified solutions is needed.

Table 7. (Continued)

General issue	Specific issue	Proposed strategy and needs in research and development
Special operation	Refueling.	Innovative solutions are proposed for ELSY with refueling machine operation in gas.
	ISI and repair.	Reduction by design of the need for ISI.
		Operation of device at ~400°C in lead need to be verified.
Fuel and core design	Fuel selection.	Use of MOX for LFR short –term deployment. MA bearing fuel and high-burn fuels to be developed in synergy of SFR.
	Lead-fuel interaction.	To be assessed.
	Failed fuel detection.	New solutions need to be investigated.
	Needs of appropriate computer codes.	Verification and validation of new CFD codes, thermic hydraulic SC and neutronic codes for LFR applications. Development verification and validation of correlations and models that deals with lead chemical behavior.

Source: Tarantino and others (2012).

Molten Salt Reactor (MSR)

MSR system was first developed in the late 1940s and 1950s for aircraft propulsion. The aircraft reactor experiment (ARE) in 1954 demonstrated high temperatures (815° C) and established benchmarks in performance for a circulating fluoride molten salt system. The MSR experiment demonstrated many features, including:

- A lithium/ beryllium fluoride salt;
- Graphite moderator;
- Stable performance;
- Off-gas systems;
- Use of different fuels, including uranium-235, uranium-233, and plutonium.

A detailed 1 000 MWe engineering conceptual design of a MSR system was developed. Many issues relating to the operation of MSRs as well as the stability of molten salt fuel and its compatibility with graphite and Hastelloy N[30] were already resolved (DOE, 2002). Significant progress was achieved in 2009 in the development of the MSR system. This included:

- Development of MSFR pre-conceptual design and performance analysis of MSFR potential for starting with plutonium and minor actinides from PWRs wastes;
- Laboratory scale processing of Ni-W-Cr alloys was recently demonstrated. The alloys were found to have acceptable workability and very good high temperature hardness (Auger et al., 2009). The whole potentialities of these kinds of materials as well as Hastelloy N[30] have yet to be tested and characterized over the full range of temperatures and in the presence of the fluoride salts;
- Corrosion tests of Ni-based alloys (Fabre et al., 2009 and Ignatiev et al., 2008a);
- Better understanding of the PuF_3 solubility in various carrier salts by means of thermochemical modeling (Beneš et al., 2009);

[30] Hastelloy N[30] is a nickel-base alloy that was invented at Oak Ridge National Laboratories in the USA as a container material for molten fluoride salts. It has good oxidation resistance to hot fluoride salts in the temperature range between 704° C and 871° C.

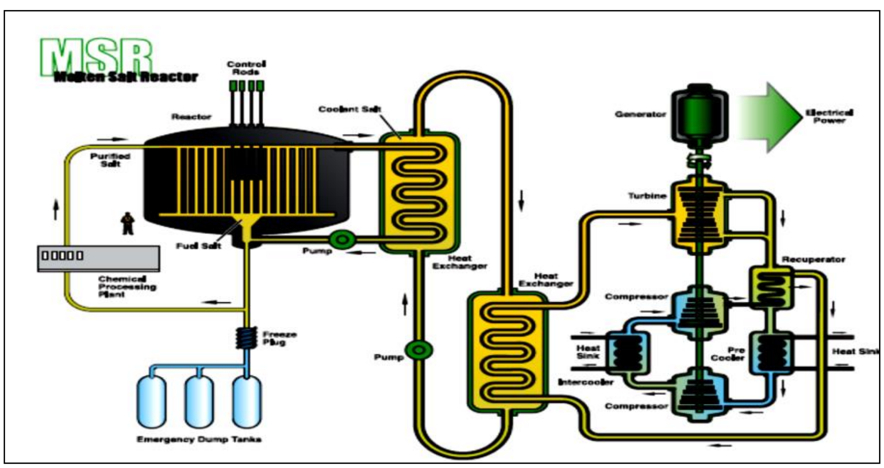
- The material property database for molten and liquid salts was extended through experiments and theoretical calculations. New experimental facilities were and continue to be developed;
- Significant improvement of fuel salt clean-up scheme;
- The optimal core configuration and salt composition of a moderated MSR system that maximize the power density while keeping the self-breeding capabilities were found. New breeding gain definitions were developed that account for the unique behavior of the MSR system. Some preliminary studies on the salt composition were published in 2008 (Nagy et al., 2008);
- Better understanding of the transmutation capabilities, dynamics and safety-related parameters, for fertile and fertile-free fuel concepts (Ignatiev et al., 2008b);
- Demonstration of fluoride-cooled high-temperature reactor (FHR) performance and safety;
- Construction of a fluoride salt test loop was initiated in the USA;
- An FHR component test plan was completed in the USA (Holcomb et al, 2009). The test plan provides a roadmap to the major technical demonstrations required to enable a test scale FHR to be built;
- Construction of a surrogate material compact integral effect test apparatus in support of a test scale FHR was initiated. The new apparatus is intended to demonstrate the coupled thermal hydraulics response of FHRs to transients including loss of heat sink and loss of forced circulation.
- Criticality tests for the assessment of FHR fuel and core behavior in the USA and the Czech Republic were carried out successfully in these two countries.

In a MSR system, the fuel is dissolved in a fluoride salt coolant. Prior MSR systems were mainly considered as thermal-neutron-spectrum graphite-moderated concepts. Since 2005, research and development has focused on the development of fast-spectrum MSR concepts (MSFR) combining the generic assets of fast neutron reactors (extended resource utilization and waste minimization) to those relating to molten salt fluorides as fluid fuel and coolant (favorable thermal-hydraulic properties, high boiling temperature, and optical transparency). In addition, MSFRs exhibit large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors (Mathieu et al, 2009). MSFR systems have been recognized as a long-term alternative to solid-fuelled fast neutron systems with unique potential (negative feedback coefficients, smaller fissile inventory, easy in-service inspection and simplified fuel cycle, among others.).

Taking advantage of technology available since the 1960s, the MSR system has been designed for a plethora of uses. From commercial power plants to nuclear powered bomber aircraft, the MSR system has the advantage of low pressure operation with higher core heat transfer. This allows for a reduced reactor size with fewer pumps and pipes operating at higher efficiencies. There are two proposals for the MSR designs:

- Molten salt fueled reactors;
- Molten salt cooled reactors.

The chemical characteristics of molten salts demand constant reprocessing and purification. Fluoride salts react with water, creating hydrofluoric acid, which is incredibly corrosive. The reprocessing is advantageous in that it removes fission products, increasing the neutron economy of the reactor. The safety advantages (retention of fission products, lower risk of explosion, and less risk of departure from nucleate boiling), combined with the higher efficiencies associated with higher operating temperatures, encourages the new design proposals. The main advantages and disadvantages of the MSR system are the following:



Source: DOE (2002).

Figure 15. Molten salt reactor system (MSR).

Advantages

- Allows for small reactor size;
- Technology is researched and proven;
- Higher operating temperatures;
- Can use simple two fluid fuel processing without the “plumbing problem”;
- Very strongly negative fuel salt coefficients;
- Blanket will also have negative temperature/void coefficient as it acts as a partial reflector;
- Ease of graphite core fabrication (and replacement if necessary);
- Ease of modeling and prototyping;
- Fissile inventory of 400 kg per GWe or even lower is possible;
- Chemical retention of fission products.

Disadvantages

- High corrosion potential;
- Unknown material required for corrosion resistance.

General benefits of the MSR system are the following:

- Salts have a high boiling point and operate at low pressure;
- Fuel salt at the lowest pressure of the circuit, which is the opposite of a LWR;
- Volatile fission products continuously removed and stored, including xenon;
- Low fissile inventory;
- Very high thermal efficiency;
- Ability to use closed thorium cycle;
- Only consume 800 kg thorium per GWe/year;
- Transuranic waste production extremely low;
- Much lower long term radio-toxicity.

The main problems associated to the MSR system are the following:

- Limited graphite lifetime (four years);

- Fuel processing hindered by chemical similarity of thorium and rare earth fission products;
- Problem with temperature reactivity coefficient recently discovered;
- Possible improvements of the MSR system but at the expense of lower conversion ratios;
- Graphite pebbles as moderator: removes need for flux flattening; can go to smaller to higher power core; pyro-lytic coatings for increased safety;
- Carrier salt switch; $NaF-BeF_2$ low cost, low melting point; $NaF-ZrF_4$ low cost, no tritium production;
- Graphite free "tank of salt" core: retain thermal spectrum by having very low fuel concentration and let the carrier salt act as moderator (Be, Li, F).

According to DOE (2002), the MSR system features an epithermal to thermal neutron spectrum and a closed fuel cycle tailored to the efficient utilization of plutonium and minor actinides. A full actinide recycle fuel cycle is envisioned in this type of reactor. In the MSR system, the fuel is a circulating liquid mixture of sodium, zirconium, and uranium fluorides. The molten salt fuel flows through graphite core channels, producing a thermal spectrum.

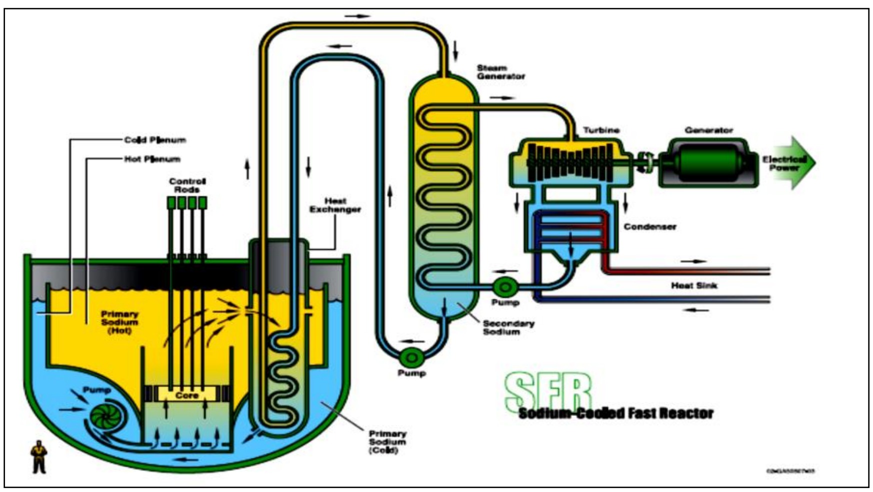
The heat generated in the molten salt is transferred to a secondary coolant system through an intermediate heat exchanger, and then through another heat exchanger to the power conversion system. Actinides and most fission products form fluorides in the liquid coolant.

The homogenous liquid fuel allows addition of actinide feeds with variable composition by varying the rate of feed addition. There is no need for fuel fabrication. The reference plant has a power level of 1 000 MWe. The system operates at low pressure and has a coolant outlet temperature above 700° C, affording improved thermal efficiency.

Sodium Cooled Fast Reactor (SFR)

The core of an SFR is smaller than the core of a typically water cooled reactor of comparable power. The core of a SFR consists of a central core, containing the fuel assemblies in triangular or hexagonal array and an outer region with radial blankets and radial shields. SFR fuel elements have a fissile material enrichment that is much higher than in a thermal reactor. The small experimental reactors like EBR I, EBR II, BR 10, and DFR have fissile material enrichment as high as 90% or more, while medium and large cores like Phenix and Super-Phenix, would have fuel with fissile material in the range of 20–25% and 15–20% respectively. The high fissile material investment necessitates the fuel to operate at a much higher burn-up level compared to that of LWRs. Accordingly, refueling times are longer.

The SFR operates in the following manner: the sodium coolant in the primary heat transport system of an SFR becomes radioactive (by neutron activation) and hence a secondary sodium coolant circuit is needed. The fission heat energy is transferred by primary sodium in an intermediate heat exchanger to a secondary sodium coolant in either a ‘loop’ or a ‘pool’ configuration. The hot non-radioactive secondary sodium is used to generate steam in another heat exchanger. The temperature of sodium leaving the reactor is approximately 550° C, which is substantially higher than that of water cooled reactors (300–330° C). Unlike water cooled reactors, SFRs do not require pressurization to keep the coolant in a liquid state because of the high boiling point of sodium (882° C). The outlet coolant pressure in the SFR system is near atmospheric and consequently the reactor vessel need not be as thick as that for a typical LWR system.



Source: US DOE, 2002.

Figure 16. Sodium-cooled fast reactor system (SFR).

From the inception of nuclear energy, the important role of the SFR system and its fuel cycle has been recognized for the long-term sustainability of nuclear power. The two recent international projects on the development of advanced and innovative nuclear energy systems, namely, the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the Generation IV International Forum (GIF), have also identified the importance of fast breeder reactors and their fuel cycle in the 21st century. Both projects are aimed at the selection of design concepts and promotion of development of advanced nuclear power technologies, which may set the basis for sustainable growth of the power industry and make it possible to develop nuclear power in the 21st century.

The key specifications of the SFR fuel cycle system are high average core burn-up, low decontamination reprocessing process, and minor actinides bearing fuel. These issues contribute to achieve economic competitiveness, reduction of environmental burden, and enhancement of nuclear non-proliferation, among others.

Recently, a feasibility study on commercializing fast breeder reactor and associated fuel cycle has been completed. A loop type SFR of 1 500 MWe has emerged with MOX as reference fuel. An advanced aqueous reprocessing and a simplified process based on cold pelletization of MOX fuel are considered most promising. Metallic fuel is considered to have potential merit to improve the core performance of the SFR.

Before the Fukushima Daiichi nuclear accident, the switchover from the LWR to the SFR was planned to be completed by the Japanese government by the end of this century but this projection has suffered important changes and perhaps will not be materialized due to a strong public opinion against the use of nuclear energy for electricity generation in the country in the future.

India is pursuing a three stage, self-reliant and indigenous nuclear power programme, linking the fuel cycles of PHWR, SFR, and self-sustaining thorium-232– uranium-233 reactor systems for judicious utilization of modest uranium but vast thorium resources. SFR is the center stage of the future Indian nuclear power programme.

China is focusing on sodium cooled, pool type, inherently safe SFR system with UO_2 (HEU) as reference, and MOX and U-Pu-Zr as advanced fuels. In the next phase, China prototype fast reactor (CPFR) of 600 MWe has been planned by 2020. The possibility of China modular fast reactor (CMFR) of 300 MWe is also being considered. In the third phase, China demonstration fast reactor (CDFR) of 1 000–1 500 MWe is likely to be constructed in 2025. The China commercial fast reactor (CCFR) is likely to be operational by 2035.

In the area of the SFR fuel cycle activities, China is constructing a medium size reprocessing plant and a laboratory-size MOX fuel production line. Later, there are plans to build industrial size reprocessing and MOX fuel fabrication plants.

The Korean government launched a ten year programme in 2007 for the development of a conceptual design of a SFR system. The programme is being conducted by the Fast Reactor Technology Development Group at KAERI under the third national mid- and long-term nuclear research and development programme. The basic research and development efforts have been directed towards the development of the advanced fast reactor concept KALIMER-600 (Korea advanced liquid metal reactor with a capacity of 600 MWe). This type of reactor system features a proliferation resistant core without blanket, and a decay heat removal circuit using natural sodium circulation cooling for a large power system. The KALIMER-600 design will serve as a starting point for meeting the Generation IV technology goals of sustainability, safety and reliability, economics, proliferation resistance and physical protection. In December 2008, the government authorized the long-term SFR development plan, and the construction of an advanced SFR demonstration nuclear power plant by 2028 in association with the pyro-process technology development in three phases:

- First phase (up to–2011): development of an advanced SFR design concept;
- Second phase (2012–2017): standard design of an advanced SFR demonstration power plant;
- Third phase (2018–2028): construction of an advanced SFR demonstration power plant.

The SFR development will be extended to the commercialization phase with its initialization around 2050. The advantages and disadvantages of the SFR system are the following:

Advantages

- Advantages of every other fast breeder reactor;
- Two primary fuel cycle technologies;
- Recovers and recycles 99.9% of the actinides;
- Inherently low decontamination factor of the product;
- Never separates plutonium at any stage;

- Achieves thermal efficiency of 40%.

Disadvantages

- Expensive;
- Fuel system research and development still needed;
- Overall system research is still needed to verify passive safety systems and component design;
- Sodium catches fire and explodes when in contact with water or air.

The fissile material concentration and in turn the burn-up of the SFR fuel is much higher than that of the LWR fuel and depends mainly on the extent of radiation damage of the fuel assembly structural materials, including the cladding tube and duct tube.

The key to the commercial success of the SFR fuel cycle lies in developing plutonium based fuels that would:

- Operate safely to high burn-up without failure;
- Be simple and safe to manufacture economically on an industrial scale;
- Be easy to reprocess, adapting the established aqueous or pyro-electrolytic processes;
- Breed and burn plutonium efficiently from uranium-238 and burn minor actinides;
- Breed uranium-233 if thorium-232 is used in blanket;
- Be amenable to proliferation resistance.

The needed research activities are included into four main areas. These areas are the following:

- System integration and assessment;
- System and component design;
- Lead technology and materials;
- Fuel development.

Based on the experience in the past five decades, the following conclusions are drawn on the status and further development of SFR fuels:

- MOX is the reference fuel for the SFR system. The use of MOX fuel in nuclear power reactors has attained maturity in France, the UK, and Japan, where industrial scale fabrication, large irradiation database both as driver fuel and as experimental fuel pins to high burn-up, and industrial scale reprocessing have been demonstrated.
- Metallic fuel is an advanced fuel for the SFR system and is very efficient from the point of view of high breeding ratio and low doubling time. Metallic fuel, in combination with pyro-electrolytic reprocessing and injection casting is very promising for SFR systems with co-location of reactor, fuel fabrication and reprocessing facility. Metallic fuels are easy to manufacture remotely on an industrial scale.
- The radiation damage of fuel structural materials is a major challenge of high burn-up SFR fuels. Further improvement is underway with oxide dispersion strengthened alloy. If one single effort were to be chosen that contributed to the successful development of fast reactor fuels, it would be the cladding and duct development programmes from several countries.
- There is a need for international database and collaborative research on out-of-pile and in-pile property evaluation and irradiation-testing of MOX, metallic fuels and fuel structural materials like modified austenitic steel, ferritic-martensitic alloys, including HT-9 and oxide dispersion, strengthened steels. International collaboration is also needed for effective utilization of the very few SFRs that will be in operation in the world in the near future, namely, BOR 60, BN-600, and FBTR for development of advanced fuels and fuel assembly structural materials.

Finally, it is important to stress the following: the research and development and industrial activities on the SFR system and its fuel cycle began to decline from the late 1980s for a number of reasons. Firstly, the nuclear accidents at Three Mile Island and Chernobyl nuclear power plants, in quick succession, slowed down the growth of nuclear power. As a result, the demand and spot price of uranium started to fall rapidly and instead of the projected shortage, uranium remained abundantly available and relatively cheap. Secondly, fast breeder reactors were not found at that time to be economically competitive with thermal nuclear power reactors. Thirdly, opposition to breeding and recovery of plutonium from spent fuel, from a weapons proliferation viewpoint, forced some countries to suspend their fast reactor fuel development programme.

The SFR system will comes in two different sizes according to the energy output desired. The smaller size will accommodate 150 to 600 MWe while the larger size handles 600 to 1 500 MWe. Both of these setups use sodium as the moderator. Sodium is a heavier material, and when neutrons collide with sodium atoms, they do not lose as much energy as happen when water is use as coolant. This is a main advantage to using this kind of reactor. The small unit uses the fuel mixed with metal alloys while the larger uses the uranium-plutonium oxide.

The SFR's development is mainly determined by its fuel material developments. It is said to be the most realizable Generation IV reactor and could be used for electricity production in the USA in the 2020s. SFR system is the most technologically developed of the six Generation IV systems included in this chapter. SFRs have been built and operated in France, Japan, Germany, the United Kingdom, Russia, and the United States. Demonstration plants ranged from 1.1 MWth (at EBR-I in 1951 in the USA) to 1 200 MWe (at Super-Phenix in 1985 in France). As a benefit of these previous investments in technology, the majority of the research and development needs presented for the SFR are performance-related. With the exception of passive safety assurance, there are few viability issues with regard to the reactor systems. The fuel options for the SFR system are MOX and metal. Both are highly developed as a result of many years of work in several national reactor development programmes (DOE, 2002).

Supercritical Water Cooled Reactor (SCWR)

The SCWRs are promising advanced systems because of their high thermal efficiency (i.e., about 45% vs. about 33% efficiency for current LWRs) and considerable plant design simplification. SCWRs are basically LWRs operating at higher pressure and temperatures with a direct once-through cycle. Operation above the critical pressure eliminates coolant boiling, so the coolant remains single-phase throughout the system. Thus, the need for recirculation and jet pumps, pressurizer, steam generators, steam separators and dryers is eliminated. The main mission of the SCWR is generation of low-cost electricity. It is built upon two proven technologies:

- LWRs, which are the most commonly deployed nuclear power reactors in the world;
- Supercritical fossil-fired boilers, a large number of which is also in use around the world.

The SCWR concept is being investigated by thirty two organizations in thirteen countries. The SCWR uses water as the primary coolant. Because the water is at such a high pressure, it will never experience a phase change when heating up or cooling off. As the water leaves this reactor core, it will be 500° C. As one can see in Figure 17, the water is incredibly hot, and under such high pressures, it will never boil. This means the water will not change phase. The fuel that will be responsible for reacting and heating the water is low-enriched uranium.

The cycle that this nuclear power reactor design uses really is very simplistic. It has no need for steam generators or steam evaporators like its PWR, BWR cousins. It has no recirculation pump, and has half the number of steam lines. One of the other greatest aspects about this type of nuclear power reactor is the incredible small size of the reactor itself, smaller than both the PWRs and BWRs.

The main advantages and disadvantages of the SFR system are the following:

Advantages

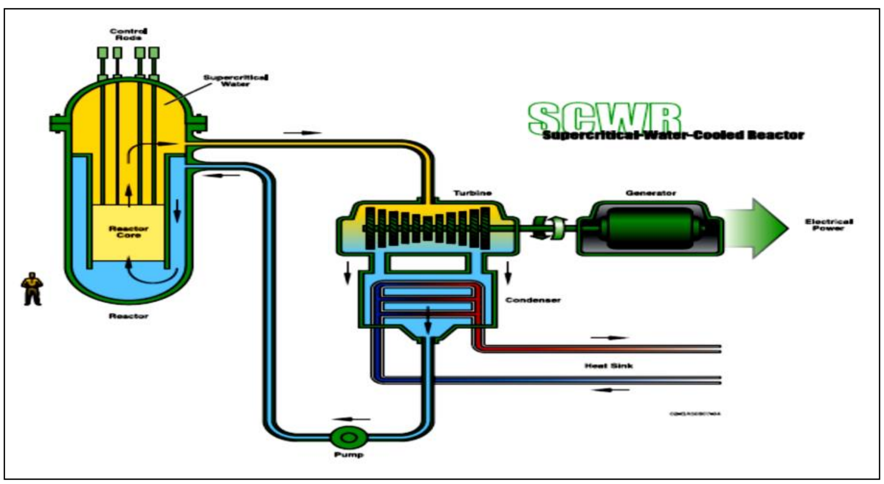
- 45% thermal efficiency;
- Able to use majority of PWR and BWR reactor vessel components;
- No need for jet pumps, pressurizers or dryers;
- Next logical improvements to the PWR and BWR configurations;

Disadvantages

- Great deal of research needs to be done on supercritical water;
- High pressure of water creates greater chance of loss of coolant accident;
- High pressure and temperatures require the use of special materials (unlike the traditional LWRs). These materials are neutron absorbers and thus enriched fuel will have to be used. New analysis of capable pipes and components should be carried out in order to probe their resistant to high pressure and temperatures.

According to DOE (2002), SCWR system features two fuel cycle options: the first is an open cycle with a thermal neutron spectrum reactor; the second is a closed cycle with a fast-neutron spectrum reactor and full actinide recycle. Both options use a high-temperature, high-pressure, water-cooled reactor that operates above the thermodynamic critical point of water (22.1 MPa, 374° C) to achieve a thermal efficiency approaching 45%. The fuel cycle for the thermal option is a once-through uranium cycle. The fast-spectrum option uses central fuel cycle facilities based on advanced aqueous processing for actinide recycle. The fast-spectrum option depends upon the materials' research and development success to support a fast-spectrum reactor.

In either option, the reference nuclear power reactor has a 1 700 MWe power level, an operating pressure of 25 MPa, and a reactor outlet temperature of 550° C. Passive safety features similar to those of the simplified boiling water reactor are incorporated. Owing to the low density of supercritical water, additional moderator is added to thermalize the core in the thermal option. Note that the balance-of-plant is considerably simplified because the coolant does not change phase in the reactor.



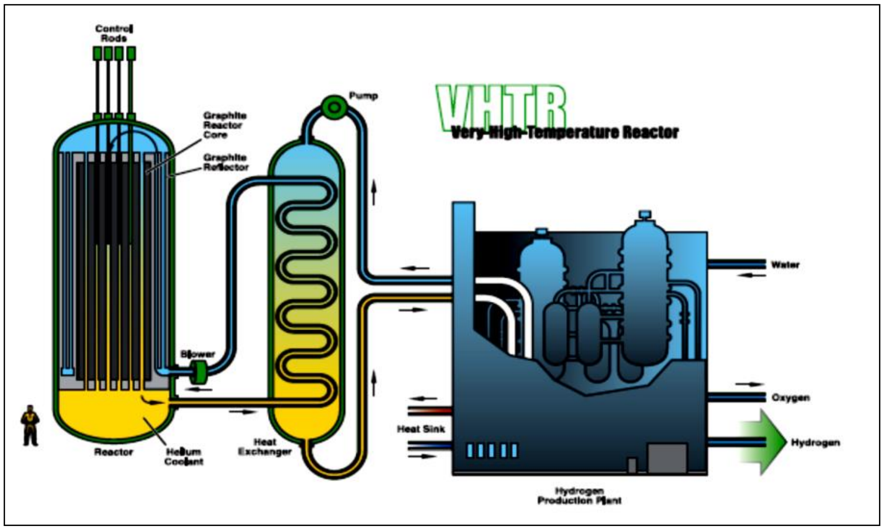
Source: US DOE, 2002.

Figure 17. Supercritical-water-cooled reactor system (SCWR).

Much of the technology base for the SCWR can be found in the existing LWRs and in commercial supercritical-water-cooled fossil-fired power plants in operation in some countries. However, there are some relatively immature areas. There have been no prototypes SCWRs built and tested. For the reactor primary system, there has been very little in-pile research done on potential SCWR materials or designs, although some SCWR in-pile research has been done for defense programmes in Russia and the United States. Limited design analysis has been underway over the past ten to fifteen years in Japan, Canada, and Russia. For the balance of plant, there has been development of turbine generators, piping, and other equipment extensively used in supercritical-water-cooled fossil-fired power plants. The SCWR may have some success in adopting portions of this technology base (DOE, 2002).

Based on the research work carried out in the USA for the SCWR programme the following can be concluded:

- The SCWRs can make substantial use of existing LWR technology. For example, the design and materials of the SCWR reactor pressure vessel and containment are similar to the PWRs and BWRs, respectively;
- The SCWRs can achieve high thermal efficiencies making extensive use of available supercritical fossil plant technology in the balance of plant. For example, the materials and design of the power conversion cycle as well as the start-up and shut down procedures and equipment can be drawn from fossil power plants with only minor modifications;
- Based on preliminary one-dimensional analyses, the SCWRs appears to be stable with respect to thermal-hydraulic and thermal/nuclear oscillations because of its relatively low coolant reactivity feedback coefficient;
- The importance of the loss of feed water as a key abnormal event has been recognized. The design of a suitable high-pressure high-capacity fast-acting auxiliary feed water system will be a major challenge in proving the viability of the SCWRs;
- Limited corrosion and stress-corrosion testing of traditional stainless steels in high-temperature water has shown that finding materials that would perform satisfactorily in the SCWRs environment will be a challenge. However, classes of materials with promising mechanical properties and corrosion resistance have been identified and will be tested (Buongiorno and MacDonald, 2003).



Source: US DOE, 2002.

Figure 18. Very-high-temperature gas reactor system (VTHR).

Very High Temperature Gas Reactor (VHTR)

According to DOE (2002), VHTR system uses a thermal neutron spectrum and a once-through uranium cycle. The VHTR system is primarily aimed at relatively faster deployment of a system for high temperature process heat applications, such as coal gasification and thermochemical hydrogen production, with superior efficiency. The typical VHTR design employs a helium coolant operated under the Brayton power cycle with the intension of achieving working temperatures in excess of 1 000° C. The initial design was developed in the 1980s and has proven its capabilities in several countries. South Africa's PBMR is one such design that offers several advantageous characteristics. Thorium-232, in conjunction with uranium-235, is encased in a graphite shell. The fuel design allows constant recycling/reprocessing/refueling of the reactor core, eliminating a significant percentage of shut down losses associated with rod framework designs. The helium gas provides a nonreactive cooling medium with a low nuclear cross section that can either directly drive a turbine for power production or can be directed to process heat to increase the efficiency of hydrogen gas production to an economical level. The gas coolant adds another level of redundancy to the safety of the system.

In the event of fuel cladding failure, radioactivity will be contained within the core without the possibility of transport via coolant. Other coolants have been considered, but have not been tested.

The main advantages and disadvantages of the VHTR system are the following:

Advantages

- The use of helium as coolant;
- The use of thorium and uranium as fuel;
- Higher electric power efficiencies (over 50%);
- Hydrogen production.

Disadvantages

- Fuel may not be rotated out when it needs to be;
- Low thermal conductivity of helium;
- Limited research.

Following are the major areas of research and development to be focused in the future:

- High temperature fuels – kernels and coatings (FP retention, integrity and higher burn-up);
- Metallic components for structural power conversion;
- Fuel element design – minimum temperature drop between fuel and coolant;
- Air/water ingress effects on core/other materials;
- Control materials for elevated temperatures;
- Heat exchanger, recuperator materials and design, and vessel materials.

The reference reactor concept has a 600 MWth helium cooled core based on either the prismatic block fuel of the gas turbine–modular helium reactor (GT-MHR) or the pebble fuel of the PBMR. The primary circuit is connected to a steam reformer/steam generator to deliver process heat. The VHTR system has coolant outlet temperatures above 1000° C. It is intended to be a high-efficiency system that can supply process heat to a broad spectrum of high temperature and energy-intensive, nonelectric processes.

The system may incorporate electricity generation equipment to meet cogeneration needs. The system also has the flexibility to adopt uranium/plutonium fuel cycles and offer enhanced waste minimization. The VHTR requires significant advances in fuel performance and high temperature materials, but could benefit from many of the developments proposed for earlier prismatic or PBMR system.

The VHTR system evolves from HTGR experience and extensive international databases that can support its development. The basic technology for the VHTR has been well established in former HTGR power plants, and is being advanced in concepts such as the GT-MHR and PBMR. The implementation of a 30-MWth HTTR project in Japan[31] had the intention to demonstrate the feasibility of reaching outlet temperatures up to 950° C coupled to a heat utilization process. The HTR-10 will be used in China to demonstrate that the system could generate electricity and can be used for co-generation at a power level of 10 MWth. The former projects in Germany provided data relevant to VHTR development. The coupling of this technology will be demonstrated in large scale in the HTTR programme but still needs complementary research and development for market introduction (DOE, 2002).

Associated Costs to Generation IV Nuclear Power Reactors

A summary of the cost associated with the different stages of the development of the six possible new nuclear power reactor systems are shown in document DOE (2002). It is important to stress that not all of the six types of nuclear power reactor designs mentioned in this chapter could be ready for use in the coming decades. This will depend on the evolution of the research works, the cost of the construction of the prototype reactor, and the decision of the interested countries on which types of systems will be used for the construction of the prototype selected for the stage of demonstration.

The VHTR, successor to the Chinese and South-African test reactors looks promising and its development will be completed first. The Idaho National Laboratory in the USA wishes to have a demonstration reactor linked to a hydrogen production plant operational in 2015.

[31] The nuclear accident in the Fukushima Daiichi nuclear power plant occurred in March 2011 in Japan stopped all research activities on this type of reactor.

Finally, it is important to stress the following: the MSR is the type of system that requires more resources for research and development activities (US$ 1 000 million) before the system could be ready for commercial use, followed by LFR (US$ 990 million) and GFR (US$ 940 million).

NEW FRAMEWORK FOR THE NUCLEAR FUEL CYCLE

A global expansion of the use of nuclear energy for electricity generation would likely drive a corresponding increase in the demand for nuclear fuel and nuclear fuel cycle services in several countries. In order to satisfy this increase without increasing the number of enrichment and reprocessing facilities in different countries, the IAEA has proposed the creation of a new multinational framework for the nuclear fuel cycle with the purpose of satisfying the foreseeable increase in the demand of these services and to impede nuclear proliferation. However, this is not an easy goal to achieve because there are different opinions about the right of a State to develop their own national nuclear fuel cycle and, at the same time, there are no clear signals that countries with enrichment and reprocessing facilities will support a multilateral approach to a nuclear fuel cycle in which their facilities will be involved.

The concept of a multilateral low enriched uranium (LEU) supply bank is not a new one, and has in fact been discussed in past decades without any agreement on how to proceed to the establishment of such bank. Undoubtedly, assurances of supply of nuclear fuel, including nuclear fuel reserves (or banks), could provide countries confidence in obtaining nuclear fuel for their nuclear power programmes for peaceful purposes, when needed, and protect them against disruption of supply for political reasons. The risk of such disruptions could possibly dissuade countries from initiating or expanding current nuclear power programmes or create vulnerabilities in the security of fuel supply that might in turn drive governments to invest in national uranium enrichment capabilities and/or reprocessing facilities with possible additional proliferation risks. Thus, multilateral approach to the nuclear fuel cycle, in general, have the potential to facilitate peaceful use of nuclear energy while providing the international community with additional assurance that the sensitive parts of the nuclear fuel cycle are less vulnerable to misuse for non-peaceful purposes.

There are several proposals related to the multinational framework for the nuclear fuel cycle that are under consideration by the international community. Among these proposals are the following:

- The Nuclear Threat Initiative (NTI plan);
- The establishment of a joint enrichment facility at the pre-existing Angarsk Electrolysis Chemical Complex, located at Angarsk in Russia, which is already a manufacturer of LEU;
- A German plan for a multilateral uranium enrichment plant under the auspices of the IAEA. The plant would be financed by countries who would act as buyers of the plant's nuclear fuel.

However, to be successful in the creation of a multinational framework for the nuclear fuel cycle it should be developed in stages. The first stage would be to establish mechanisms to assure the supply of nuclear fuel to those countries with a nuclear power programme but without enrichment facilities. States should have confidence that they would be able to obtain nuclear fuel in a predictable and stable manner over the longer term, and will not be subject to economic pressure or political discrimination that could affect the normal supply of nuclear fuel, when needed. While a well-functioning market is likely to ensure this, a back-up mechanism could add further confidence by helping to protect against political disruptions or economic pressure against a State without enrichment facilities. Such a mechanism will also make less likely the spread of sensitive nuclear fuel cycle facilities all over the world.

The second stage is the identification of enrichment facilities in a particular country (or countries) that is willing to allow the IAEA to use them to provide services on a commercial basis to any country, with the purpose of supporting their nuclear power programme without any kind of political discrimination or exclusion but under certain conditions acceptable for the international community. Among these conditions are:

- The State concerned should be a State party of the NPT[32] and of the CTBT[33];
- The State concerned should be a member of the IAEA and should have in force a full scope safeguards agreement and has ratified the IAEA Additional Protocol.

[32] Treaty on the Non-Proliferation of Nuclear Weapons (TNP).
[33] Comprehensive Nuclear-Test-Ban Treaty CTBT).

- The State concerned should not be under international sanctions for the violation of their commitments and obligations related to non-proliferation of nuclear weapons.

Advanced Construction Methods for New Nuclear Power Plants

In addition to the design of new nuclear power reactors, there are several advanced methods for the construction of new nuclear power plants now in use in a limited scale that could reduce the negative consequences of a nuclear accident and diminish construction cost and time. These are, among others, the following:

- Open top installation;
- Modularization with prefabrication and pre-assembly;
- Advanced welding techniques;
- Steel plate reinforced concrete and slip-forming;

Open Top Installation

According to IAEA and other sources, constraints on installing major components inside a nuclear power reactor and containment building can have a major impact on the construction schedule and the overall cost of the construction of a nuclear power plant. In the past, the walls of the nuclear power reactor and containment building were constructed with temporary openings to allow the entry of large equipment. In open top installation method, the reactor and containment building is built with a temporary roof with an opening through which major pieces of equipment, such as the reactor vessel and steam generators can be lowered into position using very heavy lift cranes. Once the equipment is placed and containment building is being finished, them the temporary roof is replaced by a permanent containment dome.

The open top installation method has been used successfully with modularization to shorten construction schedules. It is true that the use of very heavy lift cranes add additional construction costs, but these are more than compensated for by the shortened construction time. Very heavy lift cranes also add to planning requirements as it is vital to ensure that they are strategically placed to conduct multiple lifting activities, including the installation of heavy equipment in other buildings of the plant or to provide lifting capabilities in case that two units are being built concurrently next to each other.

Modularization with Prefabrication and Pre-assembly

According to IAEA and other sources, prefabrication and pre-assembly of modules are construction techniques used in many industries for the construction of different types of plants. A module is an assembly consisting of multiple components such as structural elements, piping, valves, tubing, conduits, cable trays, reinforcing bar mats, instrument racks, electrical panels, supports, ducting, access platforms, ladders and stairs. Modules may be fabricated at a factory or in a workshop at the plant site, and multiple modules can be fabricated while the civil engineering work is progressing at the site in preparation for receiving the modules. This reduces site congestion, improves accessibility for personnel and materials, and can shorten the construction schedule and construction cost. It can also significantly reduce on-site workforce requirements.

Modularization also facilitates mass production of modules in the event that several nuclear power reactors are being built at the same time. Mass production reduces production times and labor requirements and costs. Modularization makes it easier to assure a controlled production environment, with associated improvements in quality and efficiency. It makes it possible to manufacture modules before the site is available, and, in the case of concrete, it facilitates the use of accelerated curing techniques.

The decision to apply a modular approach for the construction of a nuclear power reactor should be made in the conceptual design stage, and then it must be followed throughout the implementation of the project. This allows equipment to be designed to conveniently fit into a module, and for modules to be sized to match the capacity of very heavy lift cranes and transport routes to the site.

Modularization also affects testing procedures as many components can be initially tested at the fabrication facility to help eliminate potential faults before formal post-installation tests at the construction site. Other impacts of modularization are:

- The need to complete the total nuclear power plant design before fabricating modules;
- The need for factories or workshops to fabricate modules;
- Earlier expenditures on engineering, materials and components for fabricating modules;
- The need for expensive heavy lift cranes;
- The costs of transporting modules.

Source: IAEA.

Figure 19. Installing a steam generator at Qinshan 3-1 in China using the open top installation method.

Source: IAEA.

Figure 20. Installing the upper drywell super large scale module at Kashiwazaki Kariwa-7 in Japan.

Source: IAEA.

Figure 21. Steel plate reinforced concrete and slip-forming.

Advanced Welding Techniques

According to IAEA and other sources, nuclear power plant construction involves numerous welds to connect both components of structures and components of pressurized systems. It also involves weld cladding, which refers to one metal being deposited onto the surface of another to improve its performance characteristics. Quality welding is both crucial and time consuming, and techniques to increase the rate at which weld metal can be deposited while maintaining high quality can reduce construction times and costs. Recent advanced welding technologies that meet this objective include gas metal arc welding, gas tungsten arc welding, and submerged arc welding. In addition, automatic welding equipment that makes it easier to weld in narrow spaces can further decrease construction times. Automatic welding equipment has been used to weld titanium tubes to condenser tube sheets at Tarapur-3 in India and to weld piping at Kashiwazaki Kariwa-7 in Japan.

In order to ensure the correct use of advanced welding technologies, a systematic inspection of this activity should be carried out by the national regulatory authority during the whole construction period of the nuclear power plant. The inspectors selected to carry out all inspection activities during the construction of a nuclear power plant should be duly prepared and trained and must be certificated using ISO standards as reference.

Steel Plate Reinforced Concrete and Slip-Forming

According to IAEA and other sources, reinforced concrete is used in the foundations of nuclear power plants and in structures such as reactor containments, auxiliary buildings, turbine buildings and spent fuel storage areas. Conventionally reinforced concrete is fabricated in place using reinforcing bars (‘rebar’) with external forms to frame the structure prior to pouring the concrete. The time required to place the reinforcing bars and to construct and remove the forms into which the concrete is poured is considerable. It is a major part of the construction schedule.

Steel plate reinforced concrete is an alternative to conventionally reinforced concrete and can be used for most floors and walls (Omoto, 2002). The concrete is placed between permanent steel plate forms with welds to tie the steel plates, rebar and tie-bars together. The forms can include any necessary penetrations and piping runs. Because of structural credit for the steel plate–concrete combination, the amount of rebar may be reduced, and because the steel plate structure can be self-supporting, reinforced concrete sections can be modularized and prefabricated off-site, followed by placement and welding on site.

CONCLUSION

One of the available energy sources for the generation of electricity that has proved that can supply the power that a country need at any time, in the amount desired, in a clean manner, and when is required, is nuclear energy. However, the use of nuclear energy for the generation of electricity is not an easy and cheap option and in some countries faces a strong rejection of the public opinion. From the technological point of view, the use of nuclear energy for the generation of electricity could be very difficult alternative for many countries, particularly for those with a weak technological development, limited financial resources, lack of qualified personnel and relative small electrical grid.

Most of the advanced nuclear power reactor designs available today are evolutionary improvements on previous designs. These evolutionary designs generally require little further research and development or confirmatory testing. In the longer term, more innovative designs that incorporate radical changes and promise significantly shorter construction times and lower capital costs could help to promote a new era of nuclear power reactors, particularly after the Fukushima Daiichi nuclear accident.

It is important to stress that the majority of the nuclear power reactors today in operation in the world are from the second generation of nuclear power reactors built in the 1970s. However, most of the countries expanding their nuclear power programmes are constructing nuclear power reactors of the third generation, which are more reliable and with a number of built-in safety features. Advances to third generation of nuclear power reactors are underway, resulting in several near-term deployable plants that is actively under development and are being considered for deployment in several countries such as France, China and Finland, just to mention a few ones. New nuclear power reactors to be built between now and 2030 will likely be chosen using this type of reactor design (Generation III +).

Undoubtedly, the future belongs to the fourth generation of nuclear power reactors (Generation IV). This new generation of nuclear power reactors is a revolutionary type of reactors with innovative fuel cycle technologies. In addition to innovations designed to achieve improved fuel efficiency, there are other issues which require innovative approaches, including high temperature applications and designs for isolated or remote locations. The Generation IV system designs are very different from older systems and also present different challenges that need to be solved in the ongoing research and development programmes in order to have all, or at least most of them, available in the market as soon as possible.

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