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The Role of Nuclear in the Future Global Energy Scene

1.1 Introduction

Energy and human life are closely liked. Civilization, present and future, depends on energy to provide the facilities the human race needs.

The world being created today will determine the outcome of a number of issues and conflicting demands, which we only now beginning to identify. Whilst their resolution will fashion the future world, the immediate challenge is to provide enough energy, water and food, to raise the standard of living of the ever-increasing world population without "imperiling our irreplaceable environment".

1.1.1 The Greenhouse Effect

In the last few years global warming, caused by the build up of greenhouse gases, has been the issue on everyone's agenda. As stated by Sir David King, the former UK Government's Chief Scientific Advisor, it is a bigger global threat than terrorism.

Carbon dioxide accounts for half of the human race's contribution to global warming. Carbon emissions have been rapidly increasing since the industrial revolution. In 2002 carbon equivalent emissions from human activity were about 6,500 million tonnes per year with the prediction this would double by 2050. Carbon dioxide emissions come from various sources, such as humans breathing, the natural world and the burning of fossil fuels, either in the generation of electricity or directly in transport. Today, the supply of electricity is responsible for 16% of worldwide carbon dioxide emissions. For the developed world this proportion is greater. For example, in the UK, electricity generation from fossil fuels is responsible for 33% of emissions.

1.1.2 The Global Scene

Powered by countries where rapid economic change is already underway, population, on a global basis, will increase from its present level of 6 billion to over 9 billion by 2025 and in addition the life span of the global population is increasing.

This brings with it a greater degree of urbanization and increased demand for energy. In 1950 only New York had a population of over 10 million. By 2015 there will be 21 cities of more than 10 million, whilst the number with populations between 5 and 10 million will go from 7 to 73. Asia and Africa, currently two-thirds rural, will be half urban by 2025.

Of the 6 billion people in the world only 2 billion have reliable access to electricity; 2 billion have unreliable access, leaving 2 billion with little or no access, of whom one billion live in slums. 2.4 billion people are dependent on wood, crop residues and dung to cook their food.

From 1980 to 2004 total world primary energy demand grew by 54% and to 2030 it is projected to grow at much the same rate (average 1.6% per year, from 469EJ to 716EJ). Electricity growth is even stronger and is projected to almost double from 2004 to 2030 (growing at average 2.6% per year from 17,408 TWh to 33,750 TWh). Due to population growth the increased demand is most dramatic in developing countries.

So what are the sources of fuel that will be used to produce this electricity? The current world demand for electricity is heavily dependent on fossil fuels with coal at 40%, gas 15% and oil 10% whilst hydro is 17% and nuclear is 16% and renewables are minimal. This dependence on fossil fuels will intensify. (Figure 1.1) What is significant is coal produces twice the quantity of CO_2 than does oil or gas whilst hydro and renewables produce far less CO_2 with nuclear producing only 0.4% of that produced by coal.

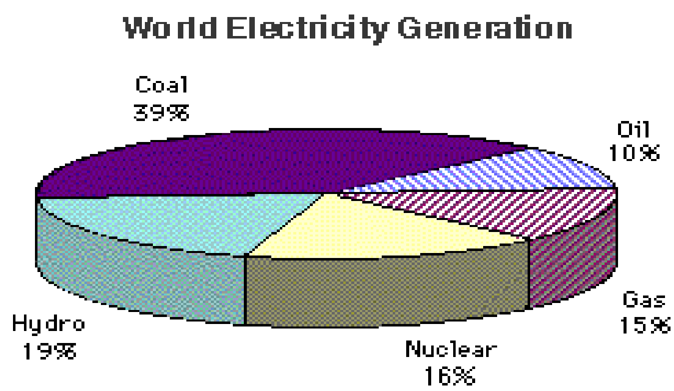


Figure 1.1

Using the UK as an example about 20% of its electricity is currently produced by nuclear power, including that imported from France via a cross channel cable. Had all that power been produced by fossil fuel, a staggering additional 50 million tonnes of carbon dioxide would have been pumped into the atmosphere per year. The saving of CO_2 emissions by the UK's nuclear power stations is equivalent to allowing 120 million people to breathe continuously. It would mean taking 50% of British cars off the road to make equivalent savings.

The Asian nations account for about 30% of the world's coal reserves, China alone has 11% of the total and India 6%. So their energy needs, now and in the long-term future, will come mainly from coal. For example, in the case of China 68.3% of their energy came from indigenous coal in 2005.

No one would deny the developing nations their chance to improve their standard of living. But if they increase energy consumption at the rate suggested, using their indigenous reserves of fossil fuels, emissions of carbon dioxide will rise well above sustainable levels. If this happens, without any reductions elsewhere, the world will destroy its own environment.

Many people, politicians as well as engineers, face the challenge now as to how to convert natural resources into a form of energy which will least affect the environment. Power engineers can produce electricity from a number of sources, none of which are totally environmentally benign. In reality, none of the so-called "benign" energy sources can provide a significant contribution to the global demand for energy. In fact a recent study predicts that renewables, other than hydro, will only contribute 4.4% of the world's electricity by 2030. For example in the UK to meet a Government target of 60% reduction in greenhouse gases, by 2050, assuming no change in technology, the UK should use only 30% of fossil fuel for power generation. So even if nuclear stayed at the present figure of less than 20%, renewables would have to be almost 50%. Clearly an impossible target.

So for the next few decades, there are only a few realistic options for reducing carbon dioxide emissions from electricity generation:

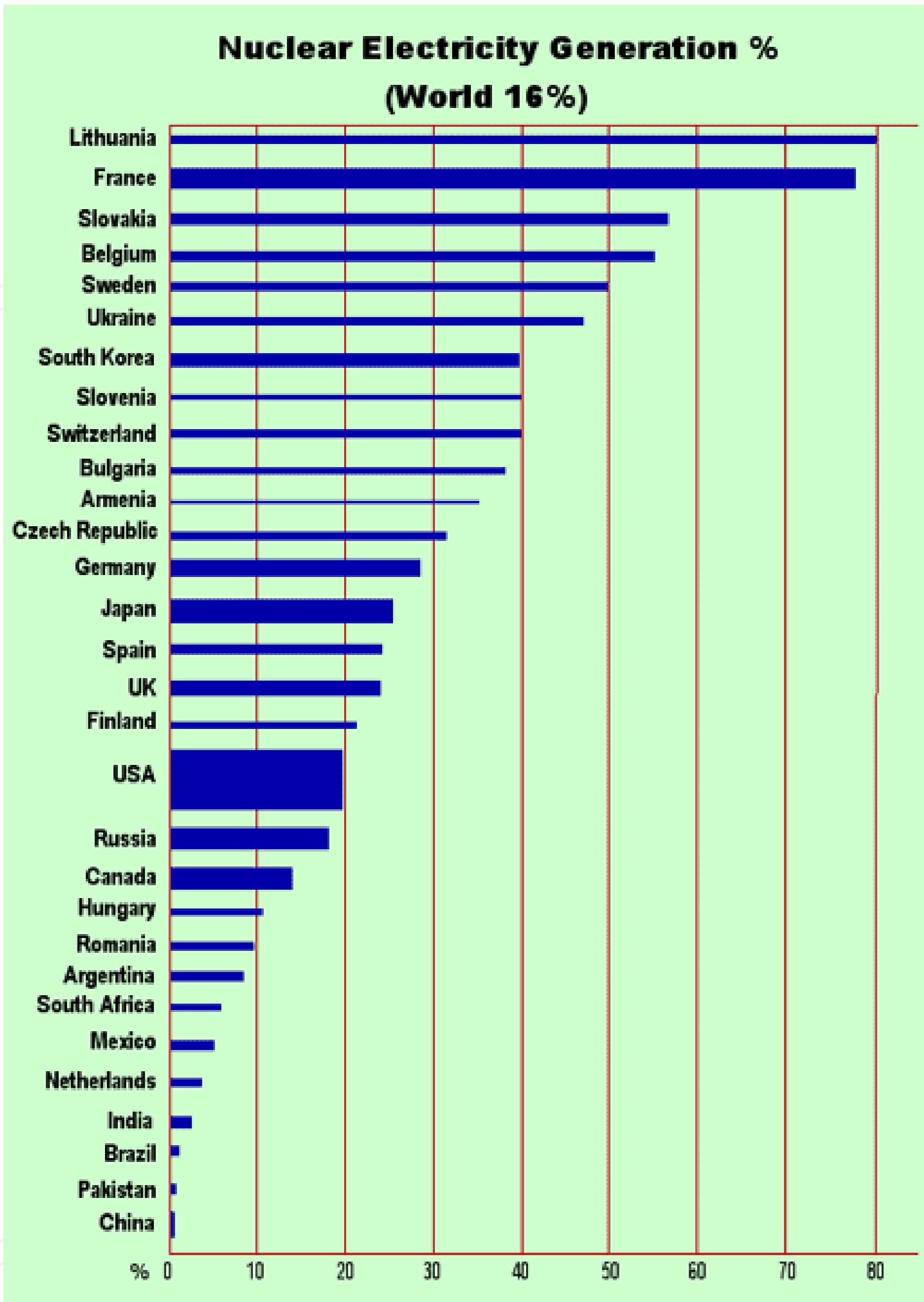
- Expand the use of renewable energy sources such as wind, solar, biomass and geothermal;
- Increase the efficiency of electricity generation and usage;
- Capture carbon dioxide emissions at fossil-fuelled stations and permanently store the carbon dioxide;
- Use of carbon offset permits;
- Increased use of nuclear power;
- Or the rationing of electricity, which would not be very popular.

Against the time scale faced, only nuclear power offers a proven long-term environmentally acceptable solution, able to produce the quantity of energy that the world will need in future.

1.1.3 The Role of Nuclear Today

At this time nuclear provides over 16% of the world's electricity, almost 24% in OECD countries and 35% in the EU.

Many countries have a significant nuclear component in their energy mix, (Figure 1.2) giving them a hedge against imported fuel price increases and also a degree of security of supply. For example France has developed a strong nuclear program, which it will maintain. France currently generates almost 80% of its electricity from nuclear and it has the cheapest electricity in Europe enabling it to export up to 15% of its electricity to its neighbors. The US, with 20% nuclear is, because of its high-energy demand, a major consumer of nuclear electricity.



NOTE: Taiwan has 22% nuclear generation

Figure 1.2 Percentage of Nuclear Generation

Today there are:

- 439 Power Reactors in 30 countries and Taiwan, China, with a total capacity of 371,936MWe
- Supplying 16% of the world's electricity
- 12,500 reactor years of experience
- Three new reactors on line in 2006
- 35 under construction, with a capacity of over 25,000 MWe

- 94 planned with a capacity of 102,000 MWe
- 222 proposed with a capacity of 193,000 MWe
- The world produces as much electricity today from nuclear energy as it did from all the other sources combined in 1960

In the US, incentives for nuclear power have led to statements of interest for 33 new reactors, on 22 sites, with the first reactors planned to be in operation by 2015. In Canada there are plans for 6,000 MWe of new plant.

Argentina and Brazil have declared their intentions to restart their programs and to co-operate in the development of new reactors and fuel cycle capabilities.

South Africa has authorized the construction of 20,000 MWe of plant.

From the current nuclear capacity of 9,000 MWe China is expected to reach 40,000 MWe by 2020 and between 120,000 – 160,000 MWe by 2030. Incredible as it may seem this will only provide 5% of China's needs, fossil fuel will supply 75%. To put it in context, China commissioned 105,000 MWe of new plant (90% fossil); almost double the UK capacity in 2006 alone.

India plans to move from 3,500 MWe today to 21,000MWe by 2020.

Many European countries are building new nuclear, for example Russia, France, Romania, Bulgaria and Finland. The nuclear option is under discussion in at least 30 countries, which currently have no nuclear reactors. Some of this interest is linked to seawater desalination.

1.2 Public Perception

Whilst nuclear plays a significant role in the energy mix today in many countries the public perception of its advantage and disadvantages, and hence future national policy, vary widely across the world. For example, Finland, conscious of the need to be less dependent on imported energy from Russia, recently voted in favor of building a new nuclear station, which is now under construction, with another under consideration.

Developments in the US are also particularly significant. There public opinion is now in favor of nuclear power. On the 8th August 2005 President Bush signed a bill promoting the use of nuclear energy to allow America to produce cleaner energy, to be less reliant on foreign suppliers of fuel and to move closer to building more nuclear power stations by the end of this decade.

A survey of 1,100 people living within 16 km of a nuclear plan in the USA showed that 83% are in favor of nuclear energy, 76% are happy to see a further reactor building on their local site, and 88% are confident of that plant's safety. Employees of electric companies were excluded from the survey. Overall 81% said they felt well informed about their local plant, correlating with an absence of NIMBYism.

The arguments, as seen by the public, against new nuclear build fall into the following heading:

- Economics;
- Disposal of nuclear waste;
- Safety;
- Proliferation; and
- Decommissioning of Nuclear Facilities

1.2.1 Economics of Nuclear Power

For nuclear power plants any cost figures normally include spent fuel management, plant decommissioning and final waste disposal. In contrast coal and gas fired economics take no account of the effects of acid rain or global warming. Only nuclear can claim cost benefits if carbon credits are taken into account as it does not produce any CO_2.

Nuclear decommissioning costs are about 9-15% of the initial capital cost of a nuclear power plant. But when discounted, they contribute only a few percent to the investment costs and even less to the generation cost. In the USA they account for 0.1-0.2 cent/kWh, which is no more than 5% of the cost of the electricity produced.

The back-end of the fuel cycle, including spent fuel storage or disposal in a waste repository, contributes up to another 10% to the overall costs per kWh, less if there is a direct disposal of spent fuel rather than reprocessing. The $26 billion US spent fuel program is funded by a 0.1 cent/kWh levy.

French generation costs, published in 2002 show (EUR cents/kWh): nuclear 3.20, gas 3.05-4.26, coal 3.81-4.57. Nuclear costs benefit from the use of large standardized plants in France.

In addition the cost of nuclear power generation has been dropping over the last decade. This is because of declining fuel (including enrichment), operating and maintenance costs. In general the construction costs of nuclear power plants are significantly higher than for coal- or gas-fired plants because of the need to use special materials, and to incorporate sophisticated safety features and back up control equipment and the longer time scale of construction. These contribute much of the nuclear generation cost but once the plant is built the cost variables are minor, and due to the long life of nuclear plants once the plant is totally depreciated the overall operating costs drop sharply.

In the past long construction periods pushed up financing costs. However today, for example in Asia, construction times have been shorter, for instance the new-generation 1300 MWe Japanese reactors which began operating in 1996 and 1997 were built in a little over four years, and 48 to 54 months is typical projection for plants today. (Figure 1.3)

Figure 1.3

US figures for 2007, published by the Energy Utility Cost Group, showed nuclear utility generating costs averaging 2.866 c/kWh, comprising 1.832 c/kWh operation and maintenance, 0.449 c/kWh fuel and 0.585 c/kWh capital expenditure. US figures from a different source for 2007, published by NEI, gave 1.68 c/kWh for fuel plus O&M.

These figures are for fuel plus operation and maintenance costs only, they exclude capital costs, since these vary greatly among utilities and states, as well as with the age of the plant. (Figure 1.4)

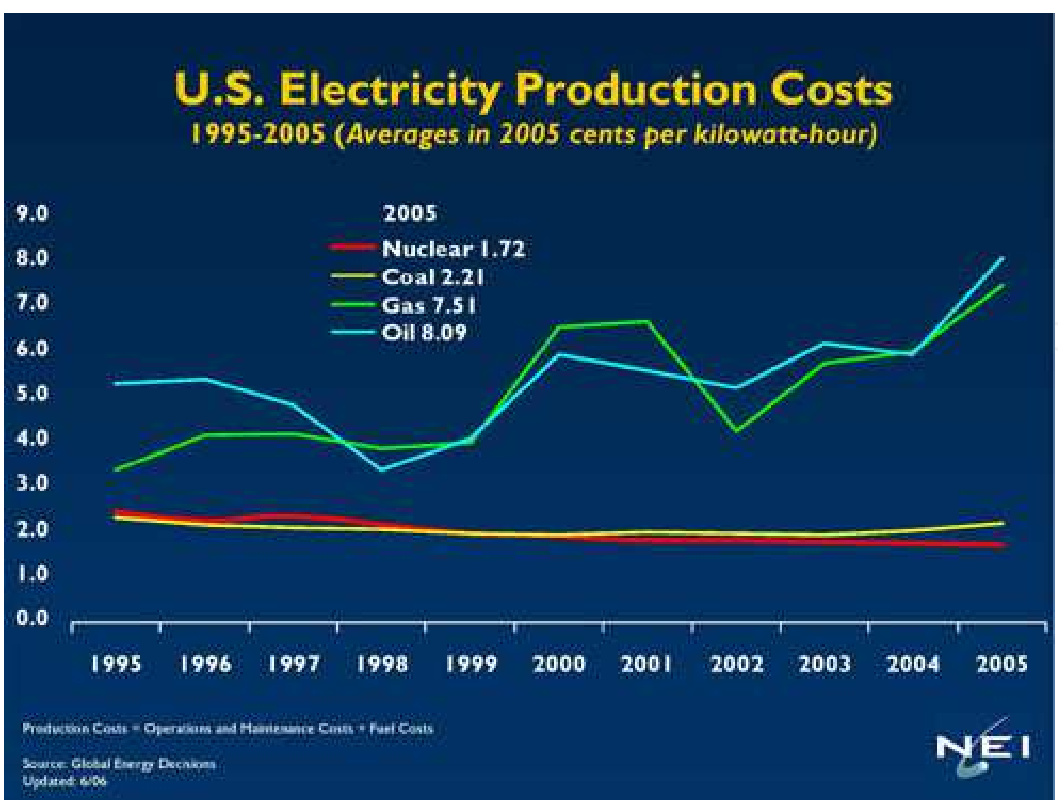


Figure 1.4 US Electricity Production Costs

1.2.1.1 Future Cost Competitiveness

The OECD does not expect investment costs in new nuclear generating plants to rise, as advanced reactor designs become standardized.

Assuming equipment and construction costs increase at the same rate for nuclear and fossil fired plants, the future competitiveness of nuclear power will depend substantially on the additional costs which may accrue to coal generating plants to ensure clean coal generation and the cost of gas for gas-fired plants. It is at present uncertain how the real costs of meeting targets for reducing emissions will be attributed to fossil fuel plants.

Overall, and under current regulatory measures, the OECD expects nuclear to remain economically competitive with fossil fuel generation, except in regions where there is direct access to low cost fossil fuels. In Australia, for example, coal-fired generating plants are close to both the mines supplying them and the main population centers, and large volumes of gas are available on low cost, long-term contracts.

A 2005 OECD comparative study showed that nuclear power had increased its competitiveness over the previous seven years. The principal changes since 1998 were increased nuclear plant capacity factors and rising gas prices. The study did not factor in any costs for carbon emissions from fossil fuel generators, and focused on over one hundred plants able to come on line 2010-15, including 13 nuclear plants. Nuclear overnight construction costs ranged from US$ 1000/kW in Czech Republic to $2500/kW in Japan, and averaged $1500/kW. Coal plants were costed at $1000-1500/kW, gas plants $500-1000/kW and wind capacity $1000-1500/kW.

OECD electricity generating cost projections for year 2010 on - 5% discount rate are shown in Table 20.1

	nuclear	coal	gas
Finland	2.76	3.64	-
France	2.54	3.33	3.92
Germany	2.86	3.52	4.90
Switzerland	2.88	-	4.36
Netherlands	3.58	-	6.04
Czech Rep	2.30	2.94	4.97
Slovakia	3.13	4.78	5.59
Romania	3.06	4.55	-
Japan	4.80	4.95	5.21
Korea	2.34	2.16	4.65
USA	3.01	2.71	4.67
Canada	2.60	3.11	4.00

Source: OECD/IEA NEA 2005.

Table 1.1 US 2003 cents/kWh, Discount rate 5%, 40 year lifetime, 85% load factor.

Nuclear costs were highest by far in Japan. Nuclear is comfortably cheaper than coal in seven of ten countries, and cheaper than gas in all but one. At 10% discount rate (Table 1.2) nuclear ranged 3-5 cents/kWh (except Japan: near 7 cents, and Netherlands), and capital becomes 70% of power cost, instead of the 50% with 5% discount rate. Here, nuclear is again cheaper than coal in eight of twelve countries and cheaper than gas in all but two. Among the technologies analyzed for the report, the new EPR if built in Germany would deliver power at about 2.38 c/kWh - the lowest cost of any plant in the study.

	nuclear	coal	gas
Finland	4.22	4.45	-
France	3.93	4.42	4.30
Germany	4.21	4.09	5.00
Switzerland	4.38	-	4.65
Netherlands	5.32	-	6.26
Czech Rep	3.17	3.71	5.46
Slovakia	4.55	5.52	5.83
Romania	4.93	5.15	-
Japan	6.86	6.91	6.38
Korea	3.38	2.71	4.94
USA	4.65	3.65	4.90
Canada	3.71	4.12	4.36

Source: OECD/IEA NEA 2005.

Table 1.2 US 2003 cents/kWh, Discount rate 10%, 40 year lifetime, 85% load factor.

Based partly on these figures the European Commission in January 2007 published comparative cost estimates for different fuels (Table 1.3):

	2005	Projected 2030 with EUR 20-30/t CO2 cost
Gas CCGT	3.4-4.5	4.0-5.5
Coal - pulverised	3.0-4.0	4.5-6.0
Coal - fluidised bed	3.5-4.5	5.0-6.5
Coal IGCC	4.0-5.0	5.5-7.0
Nuclear	4.0-5.5	4.0-5.5
Wind onshore	3.5-11.0	2.8-8.0
Wind offshore	6.0-15.0	4.0-12.0

Table 1.3 Comparative generating cost in EU – 10% discount rate (EUR)

A 1997 European electricity industry study compared electricity costs from nuclear, coal and gas for base-load plant commissioned in 2005. At a 5% discount rate nuclear (in France and

Spain) at 3.46 cents/kWh (US), was cheaper than all but the lowest-priced gas scenario. However at a 10% discount rate nuclear, at 5.07 c/kWh, was more expensive than all but the high-priced gas scenario. (ECU to US$ @ June '97 rates)

In 1999 Siemens (now Framatome ANP) published an economic analysis comparing combined-cycle gas plants with new designs, including the European Pressurized Water Reactor (EPR) and the SWR-1000 boiling water reactor. Both the 1550 MWe EPR, if built as a series in France/Germany, and the SWR-1000 (with an 8% discount rate) would be competitive with gas-combined cycle, at EUR 2.6 cents/kWh. The current-generation Konvoi plants operating in Germany produce power at 3.0 cents/kWh including full capital costs, falling to 1.5 c/kWh after complete depreciation.

A detailed study of energy economics in Finland published in mid 2000 showed that nuclear energy would be the least-cost option for new generating capacity. The study compared nuclear, coal, gas turbine combined cycle and peat. Nuclear has very much higher capital costs than the others --EUR 1749/kW including initial fuel load, which is about three times the cost of the gas plant. But its fuel costs are much lower, and so at capacity factors above 64% it was the cheapest option.

An August 2003 study (Figure 1.5) put nuclear costs at EUR 2.37 c/kWh, coal 2.81 c/kWh and natural gas at 3.23 c/kWh (on the basis of 91% capacity factor, 5% interest rate, 40 year plant life). With emission trading @ EUR 20/t CO2, the electricity prices for coal and gas increase to 4.43 and 3.92 c/kWh respectively:

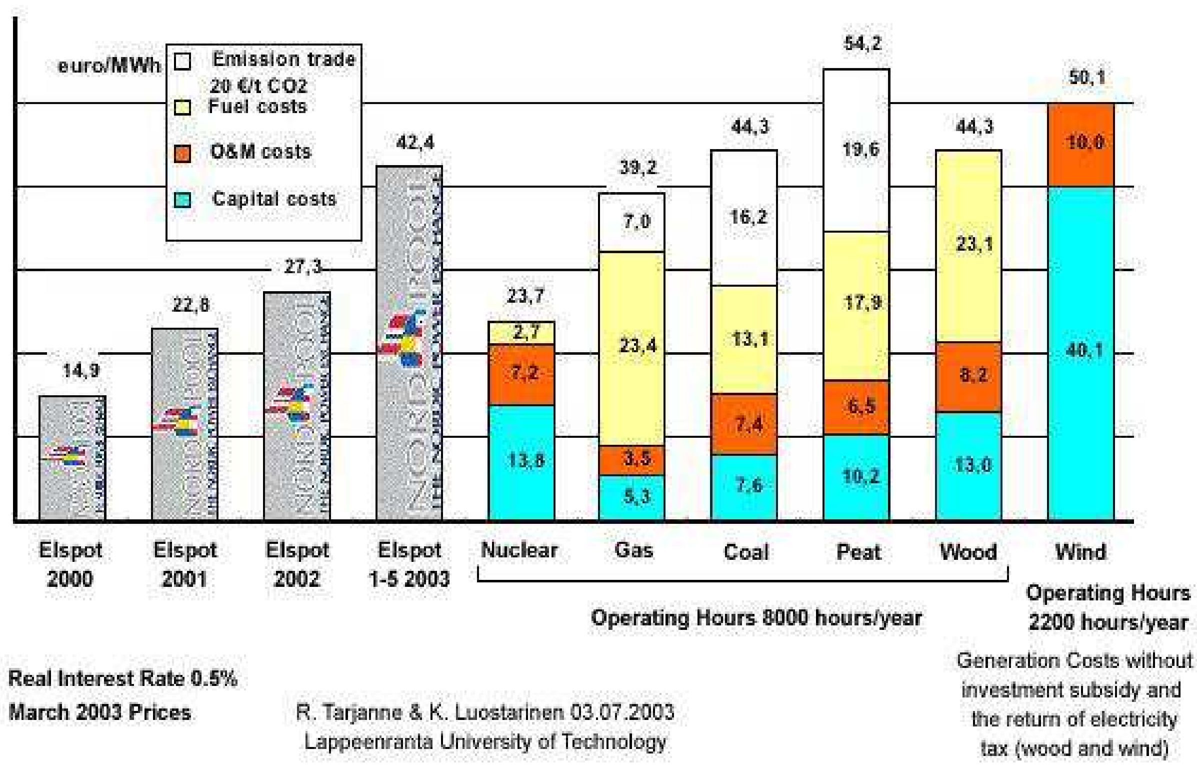


Figure 1.5 Tarjamme and Luostarmen Study 2003

In the middle three bars of Figure 1.5 the relative effects of capital and fuel costs can be clearly seen. The relatively high capital cost of nuclear power means that financing cost and time taken in construction are critical, relative to gas and even coal. But the fuel cost is very much lower, and so once a plant is built its cost of production is very much more predictable than for gas or even coal. The impact of adding a cost for carbon emissions can also be seen.

The UK Royal Academy of Engineering carried out an authoritative study in March 2004 on the costs of generating electricity in the UK which took into account capital costs, running costs, fuel, and maintenance costs. Decommissioning costs were assumed to be neutral except in the case of nuclear where these costs were allowed for. In the case of wind, the cost of standby generation was included. For base-load plant, the costs of nuclear were marginally greater than those of combined cycle turbine plant. However, taking into account possible future carbon emission taxes, based on £30 per tonne, nuclear generation became the clear winner.

Also, current designs of nuclear reactors, are being considerably simplified, thereby reducing the capital build times resulting in less financing costs. In addition these new designs produce considerably less waste that in turn reduces back end costs.

1.2.1.2 Nuclear Fuel Costs

From the outset the basic attraction of nuclear energy has been its low fuel costs compared with coal, oil and gas fired plants. Uranium, however, has to be processed, enriched and fabricated into fuel elements, and about half of the cost is due to enrichment and fabrication. Allowances must also be made for the management of radioactive spent fuel and the ultimate disposal of this spent fuel or the wastes separated from it.

In January 2007, the approx. US $ cost to get 1 kg of uranium as UO_2 reactor fuel at likely contract prices (about one third of current spot price) are shown in Table 1.4.

Uranium:	8.9 kg U_3O_8 x $53	472
Conversion:	7.5 kg U x $12	90
Enrichment:	7.3 SWU x $135	985
Fuel fabrication:	per kg	240
Total, approx:		US$ 1787

(If assuming a higher uranium price, say two thirds of current spot price: 8.9 kg x 108 = 961, this gives a total of $2286 or 0.635 c/kWh.)

Table 1.4 At 45,000 MWd/t burn-up this gives 360,000 kWh electrical per kg, hence fuel cost: 0.50 c/kWh.

A Finnish study in 2000 quantified fuel price sensitivity to electricity costs (Figure 1.6)

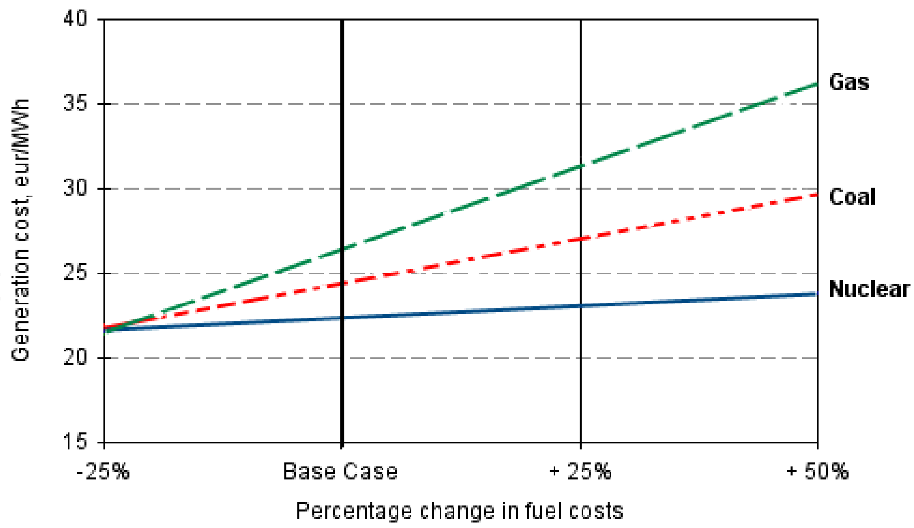


Figure 1.6 The impact of fuel costs on electricity generation costs.
Finland, early 2000

These show that a doubling of fuel prices would result in the electricity cost for nuclear rising about 9%, for coal rising 31% and for gas 66%. These are similar figures to those from the 1992 OECD report. Oil and hence gas prices have already risen significantly since the study, partly reflected in the 2003 figures above.

1.2.2 Disposal of Nuclear Waste

A major concern in the minds of the public is the long-term disposal of nuclear waste. Radioactive wastes arise from many sources – such as:

- Materials and equipment which have become contaminated during the operation of nuclear power stations and the manufacture of nuclear fuel and nuclear weapons;
- Waste arising from reprocessing nuclear fuel after it has been used in a reactor;
- Decommissioning nuclear reactors and other nuclear facilities;
- Use of radioactive materials in university research and medicine;
- Industrial manufacture and use of isotopes for tracing;
- It also arises from coal fired electricity generation and oil exploration.

Every part of the nuclear fuel cycle produces some radioactive waste and the cost of managing and disposing of this 'radwaste' is built into the process. Uranium mining for example generates fine sandy tailings, which contain virtually all the naturally occurring radioactive elements found in the uranium ore.

A large portion of radioactive waste produced from the nuclear fuel cycle has radiation levels similar to, or not much higher than, the natural background level. This waste is relatively easy to deal with. Only a small proportion is highly radioactive and requires isolation from people. The general considerations for classifying radioactive wastes are; a) how long the waste will remain at a hazardous level, b) what the concentration of the radioactive material in the waste and c) whether the waste is heat generating.

The persistence of the radioactivity determines how long the waste requires management. The concentration and heat generation dictate how the waste should be handled. These considerations also result in the disposal methods.

1.2.2.1 Classification of Nuclear Waste

There are several systems of nomenclature in use, but the following is generally accepted:

- Exempt waste, excluded from regulatory control because radiological hazards are negligible.
- Low-level Waste (LLW) contains enough radioactive material to require action for the protection of people, but not so much that it requires shielding in handling or storage.
- Intermediate-level waste (ILW) requires shielding. If it has more than 4000 Bq/g of long-lived (over 30 year half-life) alpha emitters it is categorized as "long-lived" and requires more sophisticated handling and disposal.
- High-level waste (HLW) sufficiently radioactive to require both shielding and cooling, generates >2 kW/m 3 of heat and has a high level of long-lived alpha-emitting isotopes.

Very low level waste or exempt waste. These categories contain negligible amounts of radioactivity and may be disposed of with domestic refuse.

Low-level Waste comprises the bulk of waste from the nuclear fuel cycle. It comprises paper, rags, tools, clothing, and filters etc that contain small amounts of mostly short-lived radioactivity. It does not require shielding during handling and transport and is suitable for shallow land burial. To reduce its volume, these wastes are often compacted or incinerated before disposal. Disposal sites for low-level waste are in operation in many countries. Worldwide they make up 90% of the volume but have only 1% of the total radioactivity of all radioactive wastes.

Intermediate-level Waste contains higher amounts of radioactivity and normally requires shielding. Shielding can be barriers of lead, concrete or water to give protection from penetrating radiation such as gamma rays. Intermediate-level wastes typically comprise resins, chemical sludges and metal fuel cladding, as well as contaminated materials from reactor decommissioning. It may be solidified in concrete or bitumen for disposal. Generally short-lived waste (mainly from reactors) is buried, but long-lived waste (from fuel reprocessing) will be disposed of underground.

High-level Waste (HLW) contains the fission products and transuranic elements generated in the reactor core that are highly radioactive and hot. (Figure 1.7) High-level waste accounts for over 95% of the total radioactivity produced though the actual amount of material is low, 25-30 tonnes of spent fuel, or 3 cubic meters per year of vitrified waste for a typical large nuclear reactor (1000 MWe, light water type), i.e., 2.8% of the total volume of radioactive waste. All the high level waste produced to date in the UK from the military and the civil programs would only fill 4 double decker buses.

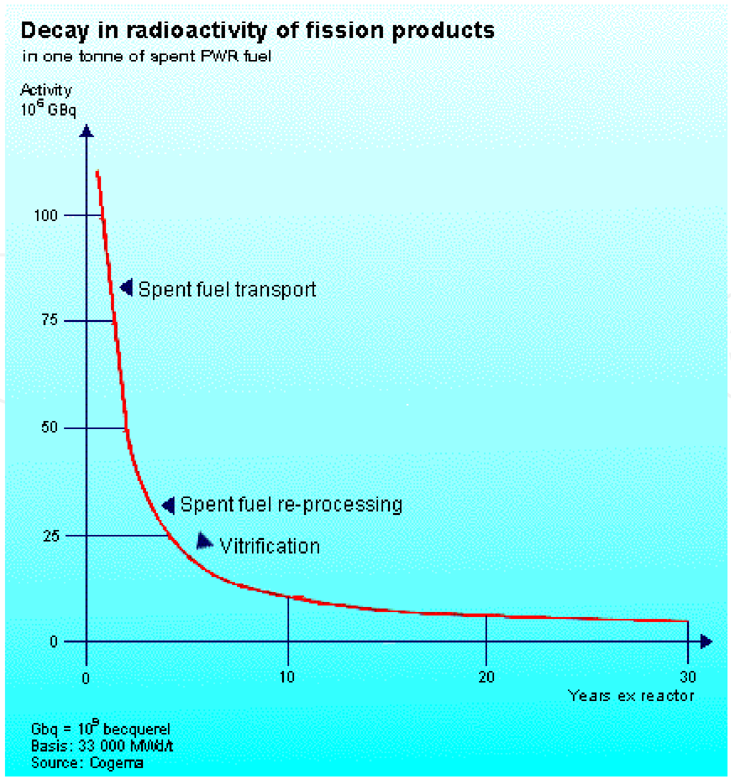


Figure 1.7 Decay in Radioactivity of Fission Products

It is important to realize that the future reactor designs use less fuel hence produce less waste than previous reactors. Ten APR1000 PWR reactors could replace all the UK current reactors. Assuming these reactors operated for 60 years the new waste arising would only be a very small amount to what is already in existence and already safely stored. (Figure 1.8)

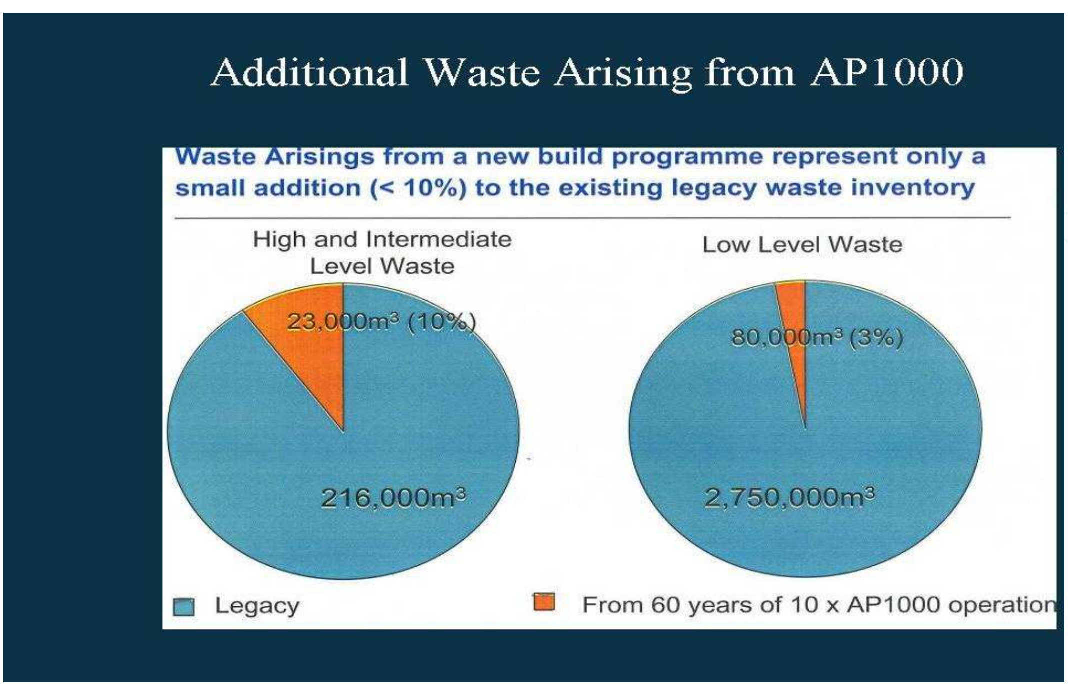


Figure 1.8 UK Waste Arising

1.2.2.2 Management of High Level Waste

There are two types of high level waste, fission products and transuranics separated from the spent fuel and the spent fuel elements themselves from the reactor core when they are not reprocessed. Both types of HLW must be treated prior to disposal. HLW from reprocessing is incorporated into solid blocks of borosilicate glass. This process is known as vitrification. For direct disposal, spent fuel requires encapsulation in containers made, for example, of stainless steel or copper.

For reprocessing when the fission products are first extracted from the spent fuel they are in liquid form, having been dissolved in acid (usually nitric acid). This liquid can be safely retained in stainless steel tanks that are equipped with cooling systems until it is converted into a solid, which is a more convenient material for management, storage, transport and disposal. After drying it is incorporated into molten borosilicate glass that is allowed to solidify inside corrosion resistant canister. Vitrification produces a stable solid that has the high-level waste incorporated its structure.

In either case however there is a cooling period of 20 to 50 years between removal from the reactor and disposal, with the conditioned spent fuel or conditioned HLW being retained in interim storage. This is because the level of radioactivity and heat from the used fuel fall rapidly in these years down to about one thousandth of the level at discharge in 40 years. Such long-term storage facilities may be at one central place as in Sweden or at the reactor site, as in the US. They may again be underwater or dry storage, where circulating air removes the heat generated by the spent fuel. The structure and design of both the building and containers protects the outside world from radiation exposure and the fuel from potential outside hazards.

1.2.2.3 Disposing of High Level Wastes

Final disposal of high level wastes is required in due course but there is no technical or logistical reason why this is urgent. Rather the contrary, the longer HLW waste is in storage, the easier it is to handle safely. HLW is accumulating at about 12,000 tonnes a year worldwide. High-level wastes are highly radioactive for a long time so must be isolated from people for thousands of years while their radiation levels drop.

Geological repositories are planned in stable rock formations in the main countries utilizing nuclear energy. It is the responsibility of each country to dispose of its wastes. Typically a repository will be 500 meters down in rock, clay or salt. The idea is a multiple barrier concept:

- The waste, either as a ceramic oxide (e.g. the spent fuel itself) or through vitrification (separated HLW from reprocessing) is immobilized.
- It is then sealed in a corrosion resistant canister such as stainless steel or copper.
- Finally it is buried in a sold rock formation.

Other means of stabilizing high-level waste are at the research stage. One of the more advanced is a substance called Synroc. This is an advanced ceramic principally comprising three natural titanate minerals which are geo-chemically stable and which together have the capacity to incorporate into their crystal structures nearly all of the elements present in high-level radioactive waste, thereby immobilizing them.

There is an interesting example from nature of long term geologically storage over millions of years. Several nuclear reactors were discovered in 1972 at the Oklo uranium mine in the West African republic of Gabon. The deposit of ore, which contained about 3% U-235, began a self-sustaining chain reaction millions of years ago. Like all reactors, this one created its own high-level waste, up to 5,000 kg of fission products and transuranic elements, which today are found only in, used fuel. The Oklo chain reaction occurred intermittently for more than 500,000 years. Despite its location in a wet, tropical climate, Oklo's uranium deposit and high-level waste has remained securely locked in this natural repository for the past 2000 million years. Many of the waste products stayed where they were created or moved only a few centimeters before decaying into harmless products.

1.2.2.4 Management of Low and Intermediate Waste

The intermediate-level waste (ILW) along with the low-level waste represent some 90% of the total volume of radioactive waste generated during the lifetime of a nuclear power plant. This relatively large volume of long-lived and short-lived ILW contains only about 1% of the total radioactivity. Only a small proportion of the intermediate-level waste remains significantly radioactive for years but all ILW requires shielding when it is handled. Low-level waste (LLW) and short-lived intermediate-level waste is of three kinds:

Process wastes result from the treatment, purification and filtration systems of fluids in direct contact with the parts of the reactor that may be contaminated by radioactivity. These wastes include:

- Filters in the cooling water circuits of the nuclear power plant;
- Resins that trap radioactive materials in the water circuits;
- Radioactive particulates that are retained by air filters installed in the ventilation stacks of nuclear facilities.

Technological wastes arise from the necessary maintenance carried out on a nuclear power plant. Technological waste represents half the volume of LLW and short-lived ILW, but contains little radioactivity.

Solid technological wastes might contain rags, cardboard, plastic sheets, bags, tools and protective clothing. Liquid technological wastes comprise mainly oils, small amounts of lubricants and organic solvents used for decontamination.

Decommissioning wastes occur at the end of a nuclear reactor's life. After the spent fuel is removed the plant is decommissioned and eventually demolished. During this process, large amounts of wastes are generated, though most is not radioactive. About a tenth of it contains some radioactivity up to the intermediate level.

Plant operators make constant efforts to reduce the quantities of waste that are generated. Waste is collected, sorted and then conditioned. The management strategy chosen depends upon the origin and radioactivity level of the waste. LLW, with the lowest concentrations of radioactivity, is usually retained in metal drums, which are often compacted after filling to reduce the volume. Other techniques may also be used to effect volume reduction. These include: melting of metallic waste, incinerating of the combustible parts of waste (whilst retaining the radioactive ash) and super-compacting waste to reduce the total volume further.

Low-level wastes that contain slightly higher radioactivity levels are stabilized by cement or an organic solid (bitumen or resin) and then placed in concrete containers for shielding. Disposal sites for such wastes are in operation in many countries. Typically, these are shallow earth burial sites, which provide a suitable facility to contain the wastes safely. A 1000 MWe nuclear power reactor can be expected to produce around $100m^3$ of low-level waste every year.

1.2.2.5 Long-Lived Intermediate Level Waste

Typically, these wastes arise from dismantled internal structures of the reactor core, which become radioactive after prolonged operation. They also include: the control rods, which regulate the nuclear reaction, the source assemblies, which are used to initiate a nuclear reaction, after new fuel has been loaded, and other rods that limit the reactivity of fresh fuel. ILW is treated and conditioned by incorporating it into cement and then placing it in concrete containers. In some instances, the conditioned waste might subsequently be placed into an additional container, made of metal. Special packages are used for transporting long-lived intermediate level waste. These packages meet internationally approved standards that ensure that the waste is safely contained.

Ultimately long-lived ILW will go to deep geological disposal as with high-level waste.

Sweden has already done this but in most countries, long-lived waste is being safely stored and contained at interim storage facilities. The maintenance of a 1000 MWe nuclear power reactor produces less than 0.5 cubic meters of long-lived ILW each year. If the spent fuel goes for reprocessing, then the cladding from the spent fuel adds an additional 3 cubic meters of ILW.

1.2.2.6 Spent Fuel: Reprocessing and Recycling

Fresh Uranium oxide fuel contains up to 5% U-235. When the fuel reaches the end of its useful life, it is removed from the reactor. At this point it typically contains about 95% U-238, 3% fission products (the residues of the fission reactions) and transuranic isotopes, 1% plutonium and 1% U-235. The plutonium is produced by the neutron irradiation of U-238.

Spent fuel still contains about a quarter of the original fissile U-235 as well as much of the plutonium that has been formed in the reactor. Reprocessing separates out this uranium and plutonium. Several reprocessing facilities, Sellafield in the UK, La Hague in France, and Chelybinsk in Russia are in operation. The wastes left after reprocessing can then be disposed of, while the uranium and plutonium may be recycled for use in a nuclear reactor as mixed oxide (MOX) fuel. This is called the ‘closed fuel cycle’ because the useful ingredients of spent fuel are recycled.

With the recycling option the energy potential can be realized in new nuclear fuel since Pu-239 and U-235 contained in the spent fuel are fissile.

1.2.2.7 Waste from Reprocessing

The reprocessing of spent fuel gives rise to low, intermediate and high level wastes:

High-level waste comprises the non-reusable part of the spent nuclear fuel itself both fission products and transuranic elements other than plutonium. The fission product leftovers are vitrified, i.e. incorporated into glass. Hulls and end fittings from the fuel assemblies are compacted, to reduce the total volume of the waste, and are frequently incorporated into cement before being placed into containers for disposal as ILW.

The major commercial reprocessing plants operating in France and UK also undertake reprocessing for utilities in other countries, notably Japan. Most Japanese spent fuel is reprocessed in Europe, with the vitrified waste and the recovered uranium and plutonium (as MOX) being returned to Japan to be recycled.

1.2.2.8 Recycling

Among the benefits of recycling identified by those countries that are utilizing MOX fuel are conservation of uranium, minimizing the amount of high-level radioactive, reducing reliance on new uranium supply, reducing the fissile plutonium inventory and reduction of spent fuel storage requirements.

1.2.2.9 Plutonium Recycling

Plutonium is recycled through a special fuel fabrication plant to produce mixed oxide (MOX) fuel. MOX fuel is a mixture of plutonium and uranium oxides (formed from natural, depleted or reprocessed uranium). MOX fuel containing 5 to 7% plutonium has characteristics that are similar to uranium oxide based fuel and used as part of a reactor's fuel loading. There are 34 reactors licensed to use MOX fuel across Europe with seventy-five others in the licensing process. Japan for example planned to introduce MOX fuel into twenty of its reactors by the year 2010. It should be noted that plutonium arising from the civil nuclear fuel cycle is not suitable for bombs because it contains far too much of the Pu-240 isotope, due to the length of time the fuel has been in the reactor.

1.2.2.10 Uranium Recycling

Uranium from reprocessing, sometimes referred to as Rep-U, must usually be enriched, and to facilitate this it must first be converted to UF_6.

1.2.3 Safety

Although Chernobyl blemished the image of nuclear energy, the accident's positive legacy is an even stronger system of nuclear safety worldwide. In 1989, the nuclear industry established the World Association of Nuclear Operations (WANO) to foster a global nuclear safety culture. Through private-sector diplomacy, WANO has built a transnational network of technical exchange that includes all countries with nuclear power. Today every nuclear power reactor in the world is part of the WANO system of operational peer review. The aim of WANO's peer-review system standards is set by the UN's International Atomic Energy Agency (IAEA).

Advances in safety practice are unmistakable. At most plants worldwide, reportable safety-related 'events' are near zero. National and international insurance laws assign responsibility to nuclear plant operators. In the US for example, reactor operators share in a 'pooled' private insurance system that has never cost taxpayers a penny.

Today, nuclear power plants have a superb safety record – both for plant workers and the public. In the transport of nuclear material, highly engineered containers – capable of withstanding enormous impact – are the industrial norm. More than 20,000 containers of spent fuel and high-level waste have been shipped safety over a total distance exceeding 30 million kilometers. During the transport of these and other radioactive substances – whether for research, medicine or nuclear – there had never been a harmful radioactive release.

Compare this safety record to other industries such as coal mining, the chemical or transport industries or the risks of smoking or drinking.

1.2.4 Proliferation

Proliferation is a major consideration. Nuclear power entails potential security risks, notably the possible misuse of nuclear facilities to acquire technology or materials as a precursor to the acquisition of a nuclear weapons capability. This is a subject of current major international concern. Fuel cycles that involve the chemical reprocessing of spent fuel to separate weapons-useable plutonium and uranium enrichment technologies are of special significance. An international response is required to reduce the proliferation risk. The response should:

- Re-appraise and strengthen the institutional underpinnings of the International Atomic Energy Agency safe-guards regime, including sanctions;
- Guide nuclear fuel cycle development in ways that reinforce shared non-proliferation objectives.

Civil nuclear power has a role to play in these objectives. The estimated 1500 tonnes of highly enriched uranium from Russia's nuclear weapons could be diluted to supply sufficient PWR fuel for all the world's PWR reactors for 8-9 years whilst plutonium, which represents 95% of energy left in non-reprocessed fuel, can be burned by turning it into mixed oxide fuel again to supply PWR reactors. This is already happening in the US with 174 tonnes of high-enriched uranium and 225 tonnes of Russian material being converted to civil use.

Terrorism cannot be ignored. But nuclear power is not an easy target for terrorists. Reactor core are massively shielded by concrete and computer tests have shown them resistant to 500 mph impacts from aircraft. The only reason for terrorists attacking nuclear power stations would be to prey on fears generated by militant greens rather than produce a lot of dead bodies. Gas and Oil terminals are much more likely targets.

1.2.5 Decommissioning of Nuclear Facilities

To date, 100 mines, 90 commercial power reactors, over 250 research reactors and a number of fuel cycle facilities, have been retired from operation.

At the end of 2005, IAEA reported that eight power plants had been completely decommissioned and dismantled, with the sites released for unconditional use. A further 17 had been partly dismantled and safely enclosed, 31 were being dismantled prior to eventual site release and 30 were undergoing minimum dismantling prior to long-term enclosure.

The International Atomic Energy Agency has defined three options for decommissioning, the definitions of which have been internationally adopted:

- Immediate Dismantling (or Early Site release/Decon in the US): This option allows for the facility to be removed from regulatory control relatively soon after shutdown or termination of regulated activities. Usually, the final dismantling or decontamination activities begin within a few months or years, depending on the facility. Following removal from regulatory control, the site is then available for re-use.
- Safe Enclosure (or Safestor): This option postpones the final removal of controls for a longer period, usually in the order of 40 to 60 years. The facility is placed into a safe storage configuration until the eventual dismantling and decontamination activities occur.
- Entombment: This option entails placing the facility into a condition that will allow the remaining on-site radioactive material to remain on-site without the requirement of ever removing it totally. This option usually involves reducing the size of the area where the radioactive material is located and then encasing the facility in a long-lived structure such as concrete, that will last for a period of time to ensure the remaining radioactivity is no longer of concern.

There is no right or wrong approach, each having its benefits and disadvantages. National policy determines which approach is adopted. In the case of immediate dismantling (or early site release), responsibility for the decommissioning is not transferred to future generations. The experience and skills of operating staff can also be utilized during the decommissioning program. Alternatively, Safe Enclosure (or Safestor) allows significant reduction in residual radioactivity, thus reducing radiation hazard during the eventual dismantling. The expected improvements in mechanical technique should also lead to a reduction in the hazard and also costs.

In the case of nuclear reactors, about 99% of the radioactivity is associated with the fuel which is removed following a permanent shutdown. Apart from any surface contamination of plant, the remaining radioactivity comes from “activation products” such as steel components that have long been exposed to neutron irradiation. Their atoms are changed into different isotopes such as iron-55, cobalt-60, nickel-63 and carbon-14. The first two are highly radioactive, emitting gamma rays. However, their half-life is such that after 50 years from closedown their radioactivity is much diminished and the risk to workers largely gone.

EDF in France, in particular have a great deal of experience in decommissioning their early nuclear stations.

There are three stages in the Safestor process for decommissioning nuclear power stations:

- **Stage 1** comprises monitored shut down of the installation. Before this level is reached, the power plant is shut down during an initial two to three year period. Non-nuclear equipment and buildings are dismantled. The fuel is unloaded from the reactor and transferred to the reprocessing plant. Finally, all the plant systems are drained down, leaving the power plant "inert". Any residual radioactive material area is contained. By this stage, 99% of the radioactivity has been removed. Although access to the plant is restricted, the equipment is necessary for monitoring of radioactivity is maintained.
- **Stage 2** comprises partial and conditional clearance of the site. This takes around four to five years. The auxiliary systems and fuel handling equipment, which can only be contained for a few years, can be decontaminated before dismantling. The radioactive waste is packaged before dispatch to the storage facility. The part of the plant around the reactor is isolated, contained and placed under surveillance.
- **Stage 3** comprises total and unconditional clearance of the plant site after the third stage of dismantling, which lasts four to five years, and takes place after a forty-year break. The rest of the plant is completely dismantled, and all remaining radioactive materials and equipment are removed. The buildings themselves are dismantled, and the nuclear equipment cut up (using eclectic arc or thermal lance equipment, or by remote control in the case of highly radioactive materials).

Dismantling a reactor produces a considerable amount of materials requiring processing (steel, concrete, pipes, electric cables, etc), in addition to a large quantity of very low active waste, mainly from the final stage of dismantling. Once this phase is completed, the site no longer requires monitoring, and can be returned to use.

1.3 Advantages Of Nuclear Power

So against these concerns what are the advantages of nuclear power, apart from helping to reduce global warming effects?

The UK situation is again an interesting case study as the Government has come to realize the need for security of supply. Currently the generation mix in the UK is 32% coal, 22% nuclear, 38% gas, 4% oil and 4% others and renewables. In other words, a diversified supply.

However, there was a lack of coherent strategy for UK future energy demands and that this is now a major concern not only in the UK but globally. In the UK, demand is increasing by 1 to 1½% per year, coal and nuclear plants are closing down, and the market does not see the certain economic returns required to build new power stations. Yet windmills are being subsidized at £50/60 per MWh at total extra costs to electricity consumers of £30 billion by 2020, more than twice the cost of a 10GW nuclear power program.

Without new power plant, by 2010, standby surplus plant margin will have fallen from a secure position of 25% to a mere 6%. But worse still, by 2020, the UK will be almost totally dependent on imported gas supplies, mainly from Russia, as there are only small amounts of strategic gas and oil reserve within the UK. And these imports will be at the end of a very long supply chain traversing areas of potential political instability giving rise to risks of serious supply shortages and price instability, particularly when Russia is rapidly becoming the major supplier of oil and gas to China, Korea and Japan.

Currently the UK is the highest amongst G8 countries for security of supply because it is largely independent of imported fuels. By 2024 this situation would be completely reversed, the UK would be dependent on imported gas, and so would be the least secure of the G8 countries. The imported gas supply costs are linked to oil prices that are rapidly increasing. On 11th August 2004 UK oil imports exceeded exports for the first time in 11 years. Oil reserves world wide will soon peak, as was so clearly demonstrated by Shell in 2004, and as of June 2008 oil prices had reached $139 a barrel up from $65 in May 2007.

It is difficult to see how a nation such as the UK's, that was totally energy self sufficient, with the exception of uranium ore which is in plentiful supply from stable countries such as Canada and Australia, a nation that was blessed with coal, oil, gas and nuclear, that enabled it to ride through a succession of energy crises, including the oil price increases in 1973, and coal strikes in the early 1980s, allowed itself to be at risk not only on the price of imported energy, that will affect its industrial base, but also has the potential for major blackouts. Also with an average trade deficit of roughly £4 billion a month how would the UK pay for all the gas it would need to import? It is against this background that the Government in the UK decided in 2007/2008 to give a green light for new nuclear construction in the UK. Many other nations also have ongoing nuclear programs to combat such risks and many are now considering the need for a nuclear component in their energy mix.

1.4 Nuclear Power Reactors

1.4.1 Components

The principles for using nuclear power to produce electricity are the same for most types of reactor. The energy released from continuous fission of the atoms of the fuel is harnessed as heat in either a gas or water, and is used to produce steam. The steam is used to drive the turbines that produce electricity.

There are several components common to most types of reactors:

Fuel; usually pellets of uranium oxide (UO_2) arranged in tubes to form fuel rods. The rods are arranged into fuel assemblies in the reactor core. In the case of the Pebble Bed Reactor the fuel is in the form of 60 mm diameter spheres.

Moderator; this is material which slows down the neutrons released from fission so that they cause more fission. It is usually water, but may be heavy water or graphite.

Control rods; these are made with neutron-absorbing material such as cadmium, hafnium or boron, and are inserted or withdrawn from the core to control the rate of reaction, or to halt it. (Secondary shutdown systems involve adding other neutron absorbers, usually as a fluid, to the system.)

Coolant; a liquid or gas circulating through the core so as to transfer the heat from it. . In light water reactors the water moderator functions also as primary coolant. Except in BWRs, there is secondary coolant circuit producing the scheme.

Pressure vessel or pressure tubes; usually a robust steel vessel containing the reactor core and moderator/coolant, but it may be a series of tubes holding the fuel and conveying the coolant through the moderator.

Steam generator; part of the cooling system where the heat from the reactor is used to make steam for the turbine.

Containment; the structure around the reactor core which is designed to protect it from outside intrusion and to protect those outside from the effects of radiation in case of any malfunction inside. It is typically a meter-thick concrete and steel structure.

1.5 The Development History Of Current Nuclear Reactors

Man's understanding of the science of atomic radiation, atomic structure and nuclear fission has developed since 1895 with much of it in the early 1940s. Between 1939 and 1945, development was focused on the atomic bomb. It was Enrico Fermi, at the University of Chicago, took the first major step in the building of the atomic bomb when he supervised the design and assembly of an "atomic pile", a code word for an assembly that in peacetime would become known as a "nuclear reactor".

However, in the course of the developing nuclear weapons, the West and the Soviet Union acquired a range of new technologies and engineers soon realized that the tremendous heat produced by the nuclear fission process could be tapped either for direct use or for generating electricity.

It was also clear that such thermal reactors would allow development of compact long-lasting power sources that could have various applications, especially in powering submarines.

Another type of reactor is the fast breeder reactor that produces more fuel than it uses. It was this type of experimental reactor that first produced a small amount of electricity in December 1951, almost 60 years ago, in the USA.

At that time work in the Soviet Union refined existing thermal reactor designs and developed new ones for commercial energy production.

Their existing graphite-moderated channel-type reactor, for producing plutonium, was modified for heat and electricity generation and in 1954 the world's first nuclear power station began operation, with a design capacity of 5MW. This served as a prototype for other graphite channel reactor designs, including the Chernobyl-type reactor known as an RBMK. (Figure 1.9)

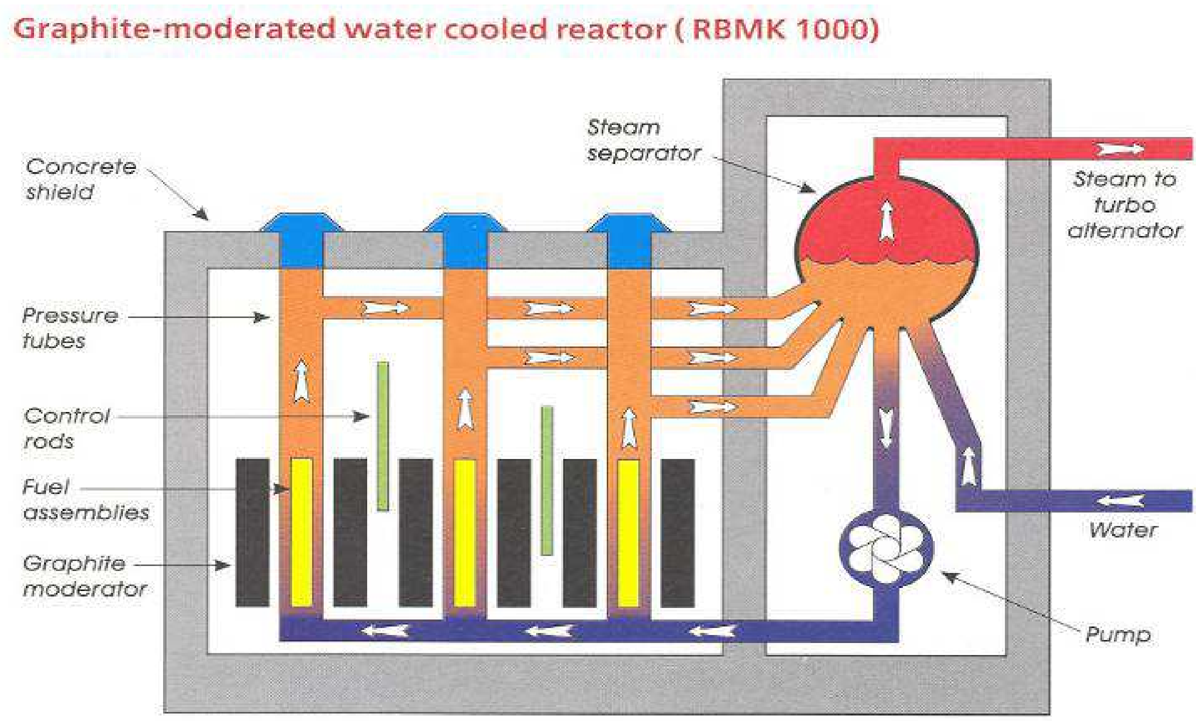


Figure 1.9 RBMK Reactors

In the 1950s the Russians were also developing fast breeder reactors.

In 1964 the first two Soviet commercial nuclear power plants were commissioned, a 100 MW boiling water reactor and a small 210 MW pressurized water reactor, known in Russia as a VVER. The first large RBMK started up in 1973 and the same year saw the commissioning of the first of four small 12 MW boiling water channel-type units for the production of both power and heat.

In the northwest Arctic a slightly bigger VVER, with a rate capacity of 440 MW began operating and this became a standard design. The world's first commercial prototype fast breeder reactor started up in 1972 producing 120 MW electricity and heat to desalinate seawater. A prototype fast neutron reactor started generating 12 MW in 1959. So a vast amount of effort that developed many different designs, took place in Russia.

In 1953 President Eisenhower proposed his "Atoms for Peace" program, which set the course for civil nuclear energy development in the USA.

The main US effort up to that time, under Admiral Rickover, was to develop the Pressurized Water Reactor (PWR) for submarine use. The PWR uses enriched uranium oxide fuel and is moderated and cooled by ordinary light water. (Figure 1.10)

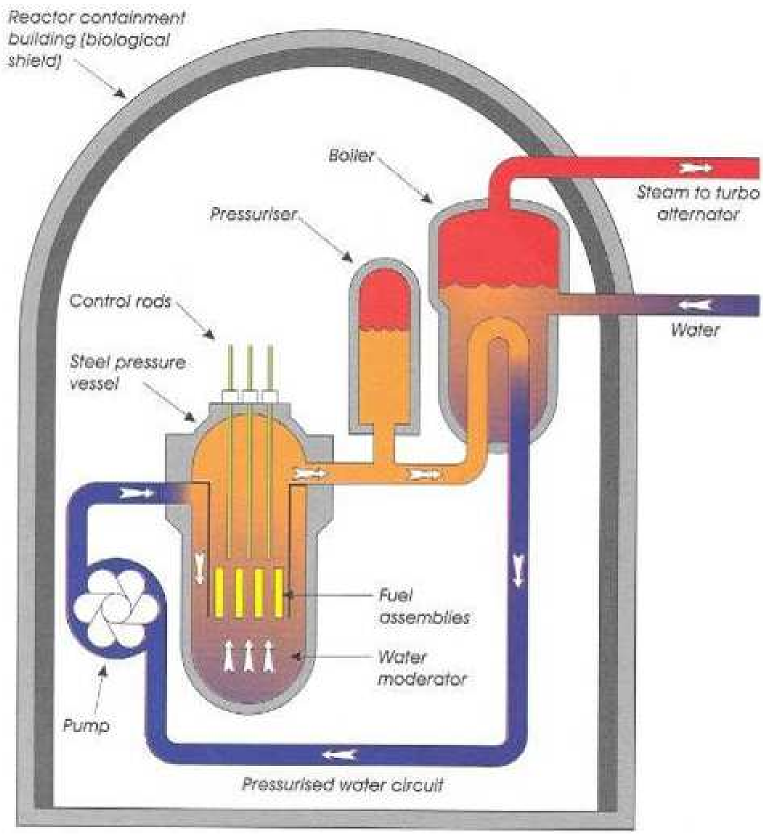


Figure 1.10 Pressurized Water Reactor (CPWR)

The Mark 1 prototype naval reactor started up in March 1953 and the first nuclear-powered submarine, USS Nautilus, was launched in 1954. In 1959 both the USA and the USSR launched their first nuclear-powered surface vessels, ranging from icebreakers to aircraft carriers. The Mark 1 naval reactor led to the building of the 90 MW Shipping Port demonstration PWR reactor, for electricity generation, which started up in 1957 and operated until 1982.

Westinghouse designed the first fully commercial PWR of 250 MW, which started up in 1960 and operated to 1992. Meanwhile the Argonne National Laboratory developed a Boiling Water Reactor (BWR) (Figure 1.11). The first commercial unit, designed by General Electric, was started up in 1960.

By the end of the 1960s international orders were being placed for PWR and BWR reactor units of outputs up to 1,000 MW.

Because, at that time, the USA had a virtual monopoly on uranium enrichment, UK development took a different approach, which resulted in a series of reactors, the Magnox Reactors, fuelled by natural uranium, moderated by graphite and cooled by carbon dioxide. (Figure 1.12)

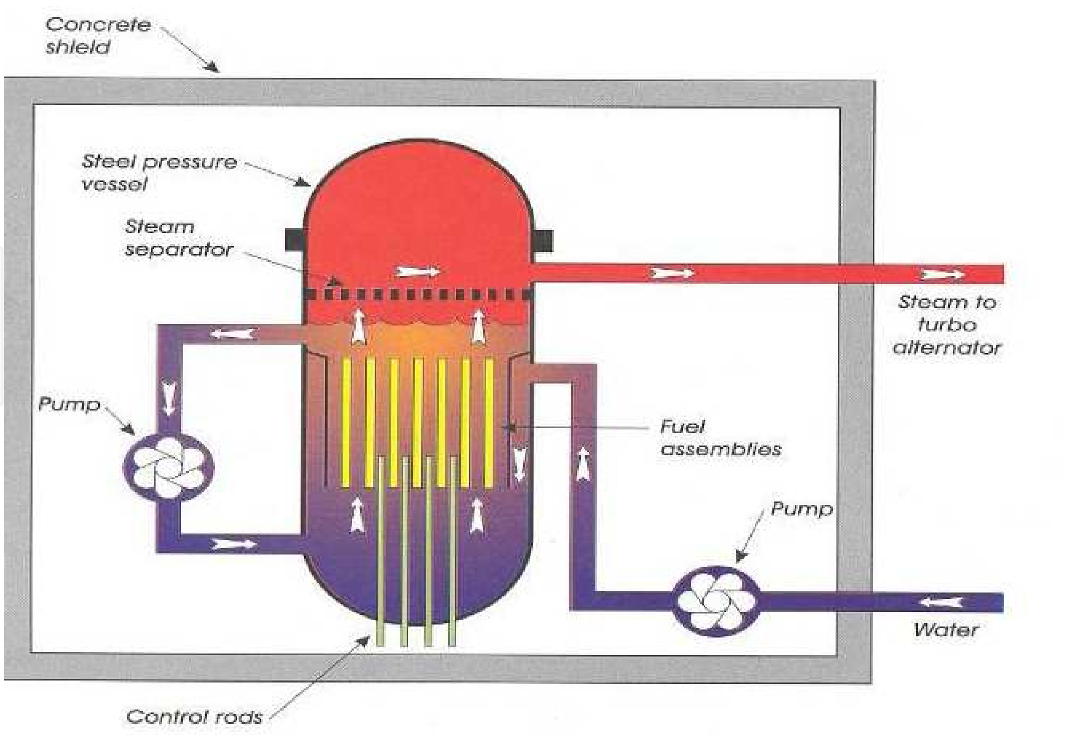


Figure 1.11 Boiling Water reactor (BWR)

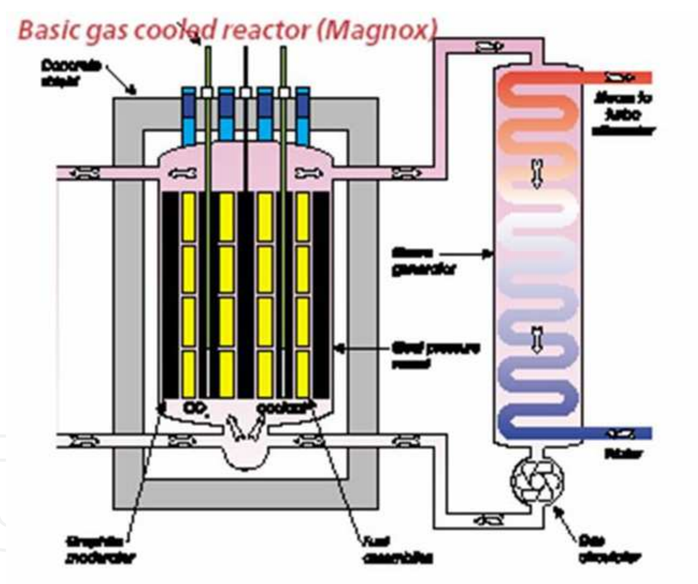


Figure 1.12 Magnox Reactor

The first of these 50 MW Magnox reactors, Calder Hall-1, started up in 1956 and was closed in 2002. A total of 26 Magnox units were built between the 1950s and the 1970s. Eighteen were closed and the remaining 8 are scheduled to be closed by 2011.

However, after 1963, based on the Magnox designs, the UK developed the Advanced Gas Cooled Reactors (AGR). (Figure 1.13) These were to become the backbone of the UK nuclear generation program with 14 AGR reactors providing 8,380 MW.

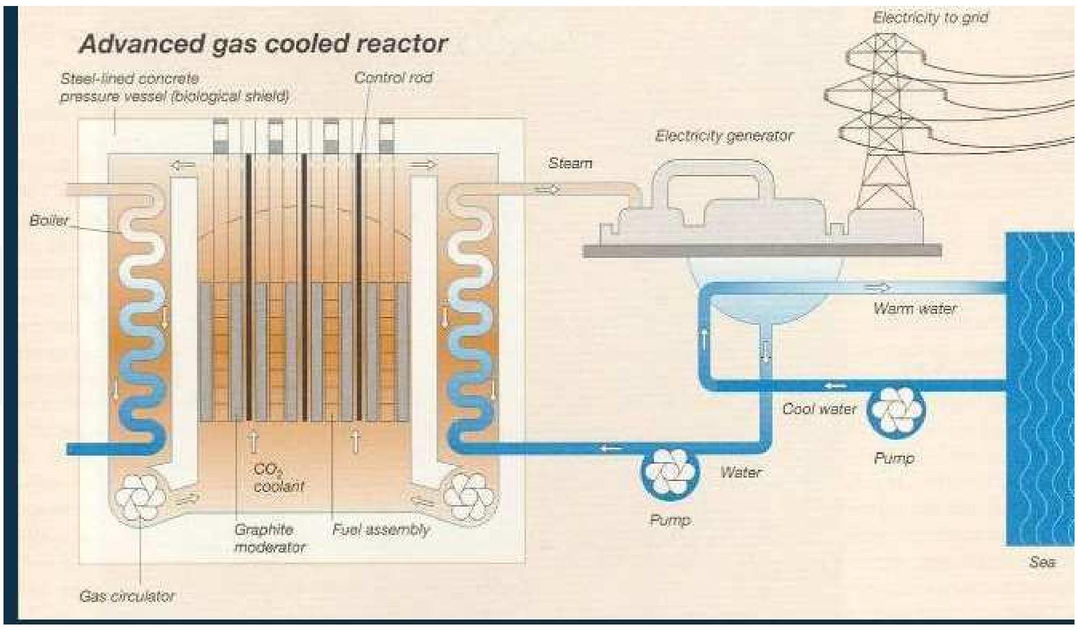


Figure 1.13 Advanced Gas Cooled Reactor (AGR)

Canadian reactor development headed down a different track, using natural uranium fuel and heavy water, both as a moderator and as a coolant. The first CANDU unit started up in 1962 and was followed by 32 more worldwide. (Figure 1.14)

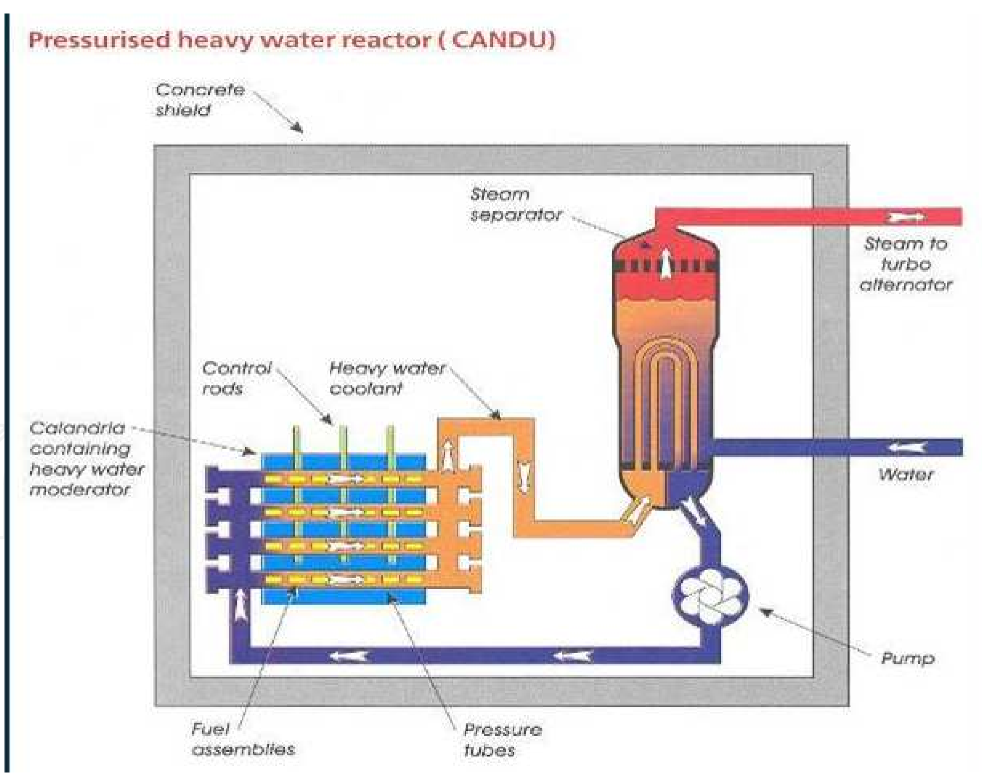


Figure 1.14 CANDU Reactor

France started with a gas-graphite design similar to Magnox, using a different fuel cladding and her first reactor commenced operation in 1956, with commercial models operating from 1959.

France then had the common sense to decide on three successive generations of standardized PWRs.

In addition, many countries built research reactors to provide a source of neutron beans for scientific research and for the production of medical and industrial isotopes.

1.5.1 Nuclear Power Plants in commercial Operation

There are several different types of reactors in operation today as shown in Table 1.5

1.5.2 Nuclear Generating Capacity by Country

As shown in Figure 1.2 the United States has 103 reactors in operation and nuclear generating capacity of 97 GWe, making it the world's leading nuclear nation. Only one reactor, however, has come into operation over the past decade and some smaller, less efficient reactors have closed down. The nuclear share has, however, remained at around 20% of US electricity generation, owing to much better reactor operating performance.

In the remainder of the Americas, Canada stands out with 17 reactors currently in operation and nuclear capacity of 12 GWe. 13% of Canada's electricity generation is nuclear. Elsewhere, Mexico, Brazil and Argentina all have small nuclear programs. South Africa is the only African nation with a small nuclear component in its energy mix. However, it now plans to considerably increase its nuclear generating capacity by the installation of further PWRs or Pebble Bed Reactors.

Reactor type	**Main Countries**	**Number**	**GWe**	**Fuel**	**Coolant**	**Moderator**
Pressurized Water Reactor (PWR)	US, France, Japan, Russia	264	250.5	Enriched UO_2	water	water
Boiling Water Reactor (BWR)	US, Japan, Sweden	94	86.4	Enriched UO_2	water	water
Pressurized Heavy Water Reactor 'CANDU' (PHWR)	Canada	43	23.6	Natural UO_2	heavy water	heavy water
Gas-cooled Reactor (AGR & Magnox)	UK	18	10.8	Natural U (metal),enriched UO_2	CO_2	graphite
Light Water Graphite Reactor (RBMK)	Russia	12	12.3	enriched UO_2	water	graphite
Fast Neutron Reactor (FBR)	Japan, France, Russia	4	1.0	PuO_2 and UO_2	liquid sodium	none
Other	Russia	4	0.05	Enriched UO_2	water	graphite
	TOTAL	439	384.6			

Table 1.5 Nuclear Power Plants in Commercial Operation

At approaching 80%, France has the highest nuclear share in its electricity generation of any country, with 59 reactors in operation and generating capacity of 63 GWe. Three successive generations of PWRs have been built and the first of a new generation of European Pressurized Water Reactors (EPR) will come into operation around 2012.

Many other European countries have substantial nuclear generating capacity, notably Germany, United Kingdom, Spain, Sweden and Belgium. Within the European Union (EU) as a whole, the nuclear share exceeds 30% of total electricity generation and five of the ten 2004 EU accession states (Czech and Slovak Republics, Hungary, Slovenia and Lithuania) have nuclear power. Finland is building the only new reactor under construction in the EU apart from France.

Japan has 54 nuclear reactors in operation with capacity of 45 GWe providing a nuclear share of around 25%. Nuclear power has become a key element in Japan's energy security and environmental policy, as it has no access to substantial indigenous energy resources. Plans exist for substantial numbers of new reactors in the future.

In Asia, Korea also has a maturing nuclear power sector, but the main growth areas for nuclear are undoubtedly China and India, the biggest developing countries in the world. In both cases, the programs are starting at low bases in terms of shares of total electricity generating capacity but they are targeting nuclear capacities of 40 GWe and 20 GWe by 2020 respectively.

Russia has an important nuclear sector and exports its technology and nuclear materials to many other countries. Its reactor program, however, became stalled at the fall of the Soviet Union and is only now getting back on track. There are currently 31 reactors in operation with generating capacity of 22 GWe, giving a nuclear share of about 17% in total electricity.

Ukraine has substantial nuclear generating capacity and remains close to the Russian industry. The East European countries remain dependent on Soviet-era technology but are gradually breaking away as they enter the EU. Bulgaria and Romania entered the EU in January 2007 and both are interested in adding to their existing stock of reactors.

1.5.3 Nuclear Growth Since 1970

The biggest factor in the continued rise in the quantity of nuclear electricity has, however, been the improved operating performance of nuclear reactors. The United States demonstrates this most strongly, as reactor load factors (showing plant utilization level compared with the theoretical maximum) typically languished in the 60-70% range in the 1980s. The onset of power market liberalization forced reactor operators to improve or go out of business and average load factors in Union States are now around 90%. Other countries had long demonstrated that this is possible and good practice continues to spread, such that world load factors have risen by ten percentage points since 1990.

Over the past five years, world nuclear electricity production has risen by 300 TWh, similar to the output from 40 new nuclear reactors, yet the net increase in the number of reactors has been only 5.

1.6 CURRENT REACTOR TYPES

1.6.1 Light Water Reactors

1.6.1.1 The Pressurized Water Reactor (PWR) (Figure 1.10)

This is the most common reactor type, with over 230 in use for power generation and a further several hundred in naval propulsion. The design originated as a submarine power plant. It uses ordinary water as both coolant and moderator. The design is distinguished by having a primary cooling circuit which flows through the core of the reactor under very high pressure, and a secondary circuit in which steam is generated to drive the turbine.

A PWR has fuel assemblies of 200-300 rods each, arranged vertically in the core, and a large reactor would have about 150-250 fuel assemblies with 80-100 tonnes of uranium.

Water in the reactor core reaches about 325°C; hence it must be kept under about 150 times atmospheric pressure to prevent it boiling. Pressure is maintained by steam in a pressuriser (see diagram). In the primary cooling circuit the water is also the moderator, and if any of it turned to steam the fission reaction would slow down. This negative feedback effect is one of the safety features of the type. The secondary shutdown system involves adding boron to the primary circuit.

The secondary circuit is under less pressure and the water here boils in the heat exchangers that are thus steam generators. The steam drives the turbine to produce electricity, and is then condensed and returned to the heat exchangers in contact with the primary circuit.

1.6.1.2 Boiling Water Reactor (BWR) (Figure 1.11)

This design has many similarities to the PWR, except that there is only a single circuit in which the water is at lower pressure (about 75 times atmospheric pressure) so that it boils in the core at about 285°C. The reactor is designed to operate with 12-15% of the water in the top part of the core as steam, and hence with less moderating effect and thus efficiency there.

The steam passes through drier plates (steam separators) above the core and then directly to the turbines, which are part of the reactor circuit. Since the water around the core of a reactor is always contaminated with traces of radionuclides, it means that the turbine must be shielded and radiological protection provided during maintenance. The cost of this tends to balance the savings due to the simpler design. Most of the radioactivity in the water is very short-lived, so the turbine hall can be entered soon after the reactor is shut down.

A BWR fuel assembly comprises 90-100 fuel rods, and there are up to 750 assemblies in a reactor core, holding up to 140 tonnes of uranium. The secondary control system involves restricting water flow through the core so that steam in the top part means moderation is reduced.

1.6.2 Pressurized Heavy Water Reactor (PHWR or CANDU) (Figure 1.14)

The CANDU reactor design has been developed since the 1950s in Canada. It uses natural uranium (0.7% U-235) oxide as fuel, hence needs a more efficient moderator, in this case heavy water (D_2O).

The moderator is in a large tank called a calandria, penetrated by several hundred horizontal pressure tubes that form channels for the fuel, cooled by a flow of heavy water under high pressure in the primary cooling circuit, reaching 290°C. As in the PWR, the primary coolant generates steam in a secondary circuit to drive the turbines. The pressure tube design means that the reactor can be refueled progressively without shutting down, by isolating individual pressure tubes from the cooling circuit. This ability to refuel on load, as opposed to other reactor types that have to shut down to reload, is a big operating advantage.

A CANDU fuel assembly consists of a bundle of 37 half-meter long fuel rods (ceramic fuel pellets in zircaloy tubes) plus a support structure, with 12 bundles lying end to end in a fuel channel. Control rods penetrate the calandria vertically, and a secondary shutdown system involves adding gadolinium to the moderator. The heavy water moderator circulating through the body of the calandria vessel also yields some heat (though this circuit is not shown on the diagram above).

1.6.3 Advanced Gas Cooled Reactor (AGR) (Figure 1.13)

These are the second generation of British gas-cooled reactors, using graphite moderator and carbon dioxide as coolant. The fuel is a uranium oxide pellet, enriched to 2.5-3.5%, in stainless steel tubes. The carbon dioxide circulates through the core, reaching 650°C and then past steam generator tubes outside it, but still inside the concrete and steel pressure vessel. Control rods penetrate the moderator and a secondary shutdown system involves injecting nitrogen to the coolant.

The AGR was developed from the Magnox reactor (Figure 1.12) also graphite moderated and CO_2 cooled, and a number of these are still operating in UK, albeit they are now planned to progressively close. They use natural uranium fuel in metal form.

1.6.4 Light Water Graphite-Moderated Reactor (RBMK) (Figure 1.9)

This is a Soviet design, developed from plutonium production reactors. It employs long (7 meter) vertical pressure tubes running through graphite moderator, and is cooled by water, which is allowed to boil in the core at 290°C, much as in a BWR. Fuel is low-enriched uranium oxide made up into fuel assemblies 3.5 meters long. With moderation largely due to the fixed graphite, excess boiling simply reduces the cooling and neutron absorption without inhibiting the fission reaction, and a positive feedback problem can arise.

1.6.5 Fast Neutron Reactors

Some reactors (only one in commercial service) do not have a moderator and utilize fast neutrons, generating power from plutonium while making more of it from the U-238 isotope in or around the fuel. While they get more than 60 times as much energy from the original uranium compared with the normal reactors, they are expensive to build and await resource scarcity to come into their own.

1.7 Small Nuclear Power Reactors

As nuclear power generation has become established since the 1950s, the size of reactor units has grown from 60 MWe to more than 1300 MWe, with corresponding economies of scale in operation. At the same time there have been many hundreds of smaller reactors built both for naval use (up to 190 MW thermal) and as neutron sources, yielding enormous expertise in the engineering of small units.

Today, due partly to the high capital cost of large power reactors generating electricity via the steam cycle and the need for nuclear in developing countries where the demand is not high and whose transmission systems are not capable of handling large centralized units of power, there is a move to develop smaller units. These may be built independently or as modules in a larger complex, with capacity added incrementally as required. The IAEA defines "small" as under 300 MWe.

The most prominent modular project is the South African-led consortium developing the Pebble Bed Modular Reactor of 170 MWe. Chinergy is preparing to build a similar unit, the 195 MWe HTR-PM in China. A US-led group is developing another design with 285 MWe modules. Both drive gas turbines directly, using helium as a coolant and operating at very high temperatures. They build on the experience of several innovative reactors in the 1960s and 1970s.

Generally, modern small reactors for power generation are expected to have greater simplicity of design, economy of mass production, and reduced siting costs. Many are also designed for a high level of passive or inherent safety in the event of malfunction. Traditional reactor safety systems are 'active' in the sense that they involve electrical or mechanical operation on command. Some engineered systems operate passively, e.g. pressure relief valves. Both require parallel redundant systems. Inherent or full passive safety depends only on physical phenomena such as convection, gravity or resistance to high temperatures, not on functioning of engineered components.

Some are conceived for areas away from transmission grids and with small loads, others are designed to operate in clusters in competition with large units. The cost of electricity from a 50 MWe unit is estimated by DOE as 5.4 to 10.7 c/kWh (compared with charges in Alaska and Hawaii from 5.9 to 36.0 c/kWh).

Already operating in a remote corner of Siberia are four small units at the Bilibino co-generation plant. These four 62 MWt (thermal) units are an unusual graphite-moderated boiling water design with water/steam channels through the moderator. They produce steam for district heating and 11 MWe (net) electricity each. They have performed well since 1976, much more cheaply than fossil fuel alternatives in the Arctic region.

The US Congress is funding research on both small modular nuclear power plants (assembled on site from factory-produced modules) and advanced gas-cooled designs (which are modular in the sense that up to ten or more units are progressively built to comprise a major power station).

1.7.1 Light Water Reactors

US experience has been of very small military power plants, such as the 11 MWt, 1.5 MWe (net) PM-3A reactor that operated at McMurdo Sound in Antarctica 1962-72, generating a total of 78 million kWh. There was also an Army program for small reactor development and some successful small reactors from the main national program commenced in the 1950s. One was the Big Rock Point BWR of 67 MWe that operated for 35 years to 1997.

Of the following, the first three designs have conventional pressure vessel plus external steam generators (PV/loop design). The others mostly have the steam supply system inside the reactor pressure vessel ('integral' PWR design). All have enhanced safety features relative to current PWRs.

The Russian KLT-40S is a reactor well proven in icebreakers and now proposed for wider use in desalination and, on barges, for remote area power supply. Here a 150 MWt unit produces 35 MWe (gross) as well as up to 35 MW of heat for desalination or district heating (or 38.5 MWe gross if power only). These are designed to run 3-4 years between refueling and it is envisaged that they will be operated in pairs to allow for outages (70% capacity factor), with on-board refueling capability and spent fuel storage. At the end of a 12-year operating cycle the whole plant is taken to a central facility for overhaul and storage of spent fuel. Two units will be mounted on a 20,000 tonne barge.

Although the reactor core is normally cooled by forced circulation, the OKBM design relies on convection for emergency cooling. Fuel is uranium aluminum silicide with enrichment levels of up to 20%, giving up to 4-year refueling intervals.

A larger Russian factory-built and barge-mounted unit (requiring a 12,000 tonne vessel) is the VBER-150, of 350 MW thermal, 110 MWe. It has modular construction and is derived by OKBM from naval designs, with two steam generators. Uranium oxide fuel enriched to 4.7% has burnable poison; it has low burnup (31 GWd/t average, 41.6 GWd/t max) and 8 year refueling interval.

OKBM's larger VBER-300 PWR is a 295 MWe unit, the first of which will be built in Kazakhstan. It was originally envisaged in pairs as a floating nuclear power plant, displacing 49,000 tonnes. As a cogeneration plant it is rated at 200 MWe and 1900 GJ/hr. The reactor is designed for 60-year life and 90% capacity factor. It has four steam generators and a cassette core with 85 fuel assemblies enriched to 5% and 48 GWd/tU burn-up. Versions with three and two steam generators are also envisaged, of 230 and 150 MWe respectively. Also with more sophisticated and higher-enriched (18%) fuel in the core, the refueling interval can be pushed from 2 years out to 15 years with burn-up to 125 GWd/tU. A 2006 joint venture between Atomstroyexport and Kazatomprom sets this up for development as a basic power source in Kazakhstan, then for export.

Another larger Russian reactor is the VK-300 boiling water reactor being developed specifically for cogeneration of both power and district heating or heat for desalination (150 MWe plus 1675 GJ/hr) by the Research & Development Institute of Power Engineering (NIKIET). It has evolved from the VK-50 BWR at Dimitrovgrad, but uses standard components wherever possible, and fuel elements similar to VVER. Cooling is passive, by convection, and all safety systems are passive. Fuel burn-up is 41 GWday/tU. It is capable of producing 250 MWe if solely electrical. In September 2007 it was announced that six would be built at Kola and at Primorskaya in the Far East, to start operating 2017-20.

A smaller OKBM PWR unit under development is the ABV, with 45 MW thermal, 10-12 MWe output. The ABV-6M is said to be 18 MWe. The units are compact, with integral steam generator and enhanced safety. The whole unit of some 600 tonnes will be factory-produced for ground or barge mounting - it would require a 2500 tonne barge. The core is similar to that of the KLT-40 except that enrichment is 16.5% and average burn up 95 GWd/t. Refueling interval is about 8 years, and service life about 50 years.

The CAREM (advanced small nuclear power plant) being developed by CNEA and INVAP in Argentina is a modular 100 MWt /27 MWe pressurized water reactor with integral steam generators designed to be used for electricity generation (27 MWe or up to 100 MWe) or as a research reactor or for water desalination (with 8 MWe in cogeneration configuration). CAREM has its entire primary coolant system within the reactor pressure vessel, self-pressurized and relying entirely on convection. Fuel is standard 3.4% enriched PWR fuel, with burnable poison, and is refueled annually. It is a mature design that could be deployed within a decade.

1.7.2 High Temperature Gas-Cooled Reactors

Building on the experience of several innovative reactors built in the 1960s and 1970s, new high-temperature gas-cooled reactors (HTRs) are being developed which will be capable of delivering high-temperature (up to 950°C) helium either for industrial application via heat exchanger or directly to drive gas turbines for electricity (the Brayton cycle) with almost 50% thermal efficiency possible (efficiency increases 1.5% with each 50°C increment). Technology developed in the last decade makes HTRs more practical than in the past, though the direct cycle means that there must be high integrity of fuel and reactor components.

Fuel for these reactors is in the form of TRISO particles less than a millimeter in diameter. Each has a kernel (c0.5 mm) of uranium oxycarbide, with the uranium enriched up to 20% U-235, though normally less. This is surrounded by layers of carbon and silicon carbide, giving a containment for fission products that is stable to 1600°C or more. With negative temperature coefficient of reactivity (the fission reaction slows as temperature increases) and passive decay heat removal, this makes the reactors inherently safe. They do not require any containment building for safety.

The reactors are sufficiently small to allow factory fabrication, and will usually be installed below ground level.

There are two ways in which these particles are arranged: in blocks - hexagonal 'prisms' of graphite, or in billiard ball-sized pebbles of graphite encased in silicon carbide, each with about 15,000 fuel particles and 9g uranium. There is a greater amount of spent fuel than from the same capacity in a light water reactor. The moderator is graphite.

The Japan Atomic Energy Research Institute's (JAERI) High-Temperature Test Reactor (HTTR) of 30 MW thermal started up at the end of 1998 and has been run successfully at 850°C. In 2004 it achieved 950°C outlet temperature. Its fuel is in 'prisms' and its main purpose is to develop thermo chemical means of producing hydrogen from water.

Based on the HTTR, JAERI is developing the Gas Turbine High Temperature Reactor (GTHTR) of up to 600 MW thermal per module. It uses improved HTTR fuel elements with 14% enriched uranium achieving high burn-up (112 GWd/t). Helium at 850°C drives a horizontal turbine at 47% efficiency to produce up to 300 MWe. The core consists of 90 hexagonal fuel columns 8 meters high arranged in a ring, with reflectors. Each column consists of eight one-meter high elements 0.4 m across and holding 57 fuel pins made up of fuel particles with 0.55 mm diameter kernels and 0.14 mm buffer layer. In each 2-yearly refueling, alternate layers of elements are replaced so that each remains for 4 years.

On the basis of four modules per plant, capital cost is projected at US$ 1300-1700/kWe and power cost about US 3.4 c/kWh.

China's HTR-10**,** a small high-temperature pebble-bed gas-cooled experimental reactor at the Institute of Nuclear & New Energy Technology (INET) at Tsinghua University north of Beijing started up in 2000 and reached full power in 2003. It has its fuel as a 'pebble bed' (27,000 elements) of oxide fuel with average burn up of 80 GWday/t U. Each pebble fuel element has 5g of uranium enriched to 17% in around 8300 particles. The reactor operates at 700°C (potentially 900°C) and has broad research purposes. Eventually it will be coupled to a gas turbine, but meanwhile it has been driving a steam turbine.

Construction of a larger version, the 200 MWe (450 MWt) HTR-PM, was approved in principle in November 2005, with construction starting in 2009. This will have two reactors modules, each of 250 MWt, using 9% enriched fuel (520,000 elements) giving 80 GWd/t discharge burn up. With an outlet temperature of 750ºC the pair will drive a single steam cycle turbine at about 40% thermal efficiency. The size was reduced to 250 MWt from earlier 458 MWt modules in order to retain the same core configuration as the prototype HTR-10 and avoid moving to an annular design like South Africa's PBMR. This Shidaowan demonstration reactor at Rongcheng in Shandong province is to pave the way for an 18-unit (3x6x200MWe) full-scale power plant on the same site at Weihei, also using the steam cycle. Plant life is envisaged as 60 years with 85% load factor.

China Huaneng Group, one of China's major generators, is the lead organization involved in the demonstration unit with 47.5% share; China Nuclear Engineering & Construction (CNEC) will have a 32.5% stake and Tsinghua University's INET 20% - it being the main R&D contributor. Projected cost is US$ 385 million (but later units falling to US$1500/kW with generating cost about 5c/kWh). Start-up is scheduled for 2013. The HTR-PM rationale is both eventually to replace conventional reactor technology for power, and also to provide for future hydrogen production. INET is in charge of R&D, and is aiming to increase the size of the 250 MWt module and also utilize thorium in the fuel. Eventually a series of HTRs, possibly with Brayton cycle directly driving the gas turbines, will be factory-built and widely installed throughout China.

In 2004 the small HTR-10 reactor was subject to an extreme test of its safety when the helium circulator was deliberately shut off without the reactor being shut down. The temperature increased steadily, but the physics of the fuel meant that the reaction progressively diminished and eventually died away over three hours. At this stage a balance between decay heat in the core and heat dissipation through the steel reactor wall was achieved and the temperature never exceeded a safe 1600°C. This was one of six safety demonstration tests conducted then. The high surface area relative to volume, and the low power density in the core, will also be features of the full-scale units (which are nevertheless much smaller than most light-water types).

Between 1966 and 1988, the AVR experimental pebble bed reactor at Juelich, Germany, operated for over 750 weeks at 15 MWe, most of the time with thorium-based fuel. The fuel consisted of about 100,000 billiard ball-sized fuel elements. The thorium was mixed with high-enriched uranium (HEU). Maximum burnups of 150 GWd/t were achieved. It was used to demonstrate the inherent safety of the design due to negative temperature coefficient: the helium coolant flow was cut off and the reactor power fell rapidly.

The 300 MWe THTR reactor in Germany was developed from the AVR and operated between 1983 and 1989 with 674,000 pebbles, over half containing Th/HEU fuel (the rest graphite moderator and some neutron absorbers). These were continuously recycled and on average the fuel passed six times through the core. Fuel fabrication was on an industrial scale. Several design features made the AVR unsuccessful, though the basic concept was again proven. It drove a steam turbine.

An 80 MWe HTR-module was then designed by Siemens as a modular unit to be constructed in pairs. It was licensed in 1989, but was not constructed. This design was part of the technology bought by Eskom in 1996 and is a direct antecedent of PBMR.

South Africa's Pebble Bed Modular Reactor (PBMR) is being developed by a consortium led by the utility Eskom, and drawing on German and previous UK expertise. (Figure 1.15)

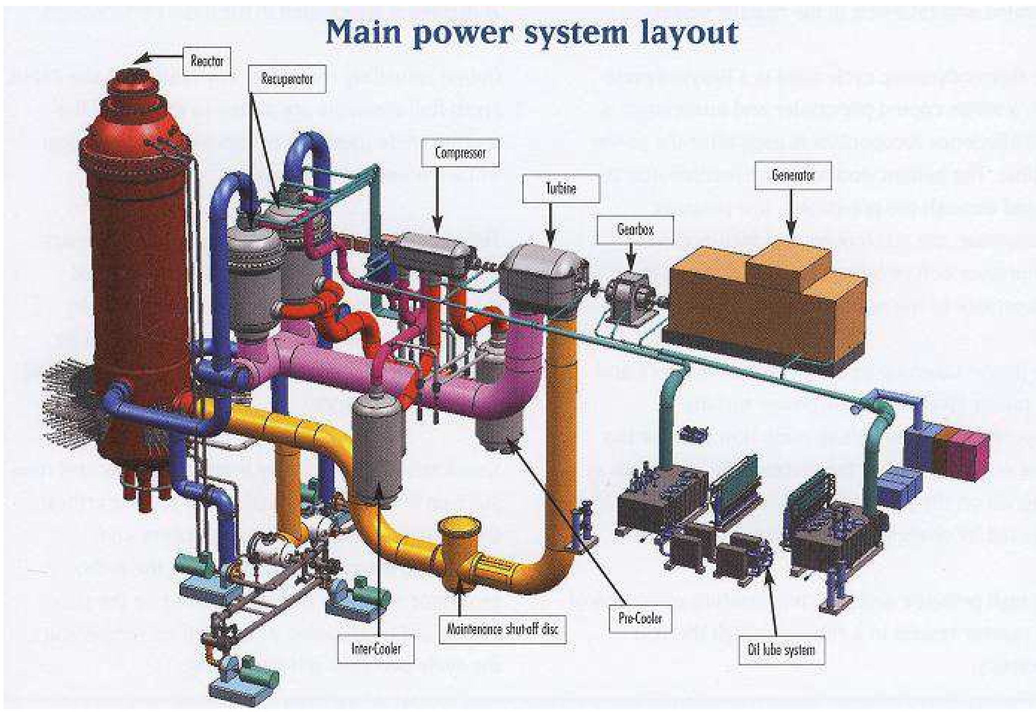


Figure 1.15 Pebble Bed Modular Reactor (PBMR)

It aims for a step change in safety, economics and proliferation resistance. Production units will be 165 MWe. The PBMR will have a direct-cycle gas turbine generator and thermal efficiency about 41%, the helium coolant leaving the bottom of the core at about 900°C. Up to 450,000 fuel pebbles 60 mm diameter, 210 g mass and containing 9g uranium enriched to 10% U-235 recycle through the reactor continuously (about six times each, taking six months) until they are expended, giving an average enrichment in the fuel load of 5% and average burn-up of 80 GWday/t U (eventual target burn-ups are 200 GWd/t). (Figure 1.16)

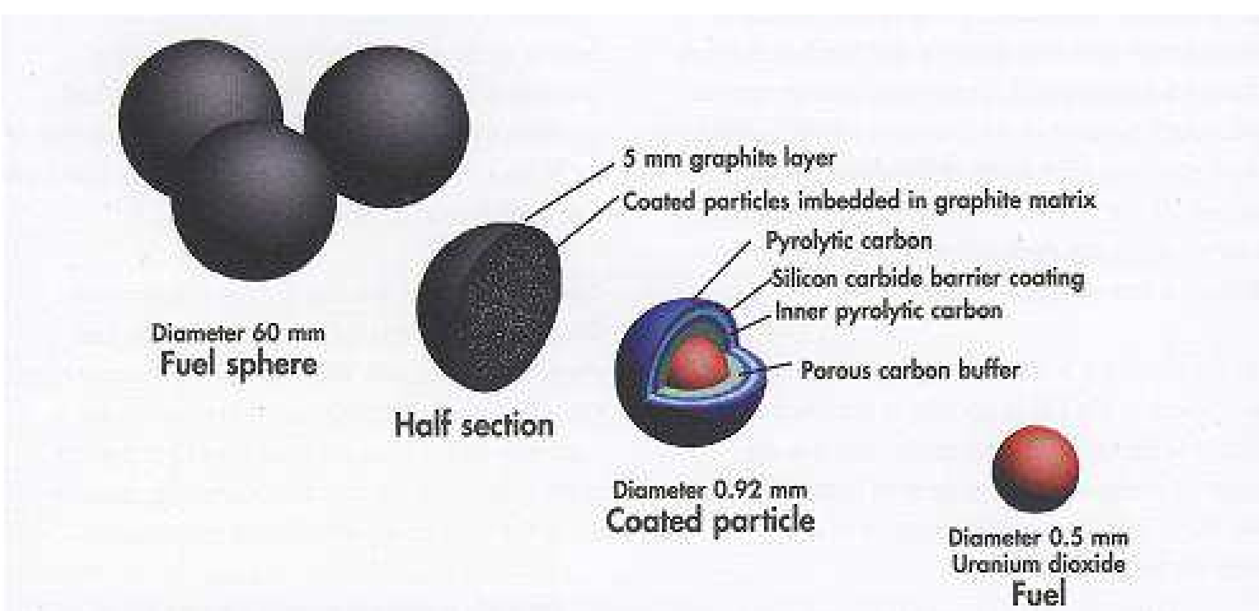


Figure 1.16 Fuel Element Design for PBMR

This means on-line refueling as expended pebbles (which have yielded up to 91 GWd/t) are replaced, giving high capacity factor. The reactor core is lined with graphite and there is a central column of graphite as reflector. Control rods are in the side reflectors and cold shutdown units in the center column.

Performance includes great flexibility in loads (40-100%) without loss of thermal efficiency, and with rapid change in power settings. Power density in the core is about one tenth of that in light water reactor, and if coolant circulation ceases the fuel will survive initial high temperatures while the reactor shuts itself down - giving inherent safety. Power control is by varying the coolant pressure and hence flow. Each unit will finally discharge about 35 tonnes/yr of spent pebbles to ventilated on-site storage bins.

The PBMR Demonstration Power Plant (DPP) started construction at Koeberg in 2009 and is expected to achieve criticality in 2013. Eventual construction cost (when in clusters of four or eight units) is expected to be very competitive. Investors in the PBMR project are Eskom, the South African Industrial Development Corporation and Westinghouse. The first commercial units are expected on line soon after the DPP and Eskom has said it expects to order 24, which justify fully commercial fuel supply and maintenance. A contract for the pebble fuel plant at Pelindaba has been let.

Each 210g-fuel pebble contains about 9g U and the total uranium in one fuel load is 4.1 t. MOX and thorium fuels are envisaged. With used fuel, the pebbles can be crushed and the 4% of their volume which is micro spheres removed, allowing the graphite to be recycled. The company says microbial removal of C-14 is possible (also in the graphite reflectors when decommissioning).

In 2006 the PBMR Board formalized the concept of a higher-temperature PBMR Process Heat Plant (PHP) with reactor output temperature of 950°C. The first plants are envisaged for 2016 and the applications will be oil sands production, petrochemical industry (process steam), steam methane reforming for hydrogen and eventually thermo chemical hydrogen production. This design will be submitted to US Department of Energy as a candidate Next-Generation Nuclear Plant.

A design certification application to the US Nuclear Regulatory Commission was considered in 2008, with approval expected in 2012, opening up world markets.

A larger US design, the Modular Helium Reactor (MHR, formerly the GT-MHR), will be built as modules of up to 600 MWt. In its electrical application each would directly drive a gas turbine at 47% thermal efficiency, giving 280 MWe. It can also be used for hydrogen production (100,000 t/yr claimed) and other high temperature process heat applications. The annular core consists of 102 hexagonal fuel element columns of graphite blocks with channels for helium coolant and control rods. Graphite reflector blocks are both inside and around the core. Half the core is replaced every 18 months. Burn-up is up to 220 GWd/t, and coolant outlet temperature is 850°C with a target of 1000°C.

The MHR is being developed by General Atomics in partnership with Russia's OKBM, supported by Fuji (Japan) and Areva NP. Initially it will be used to burn pure ex-weapons plutonium at Seversk (Tomsk) in Russia. A burnable poison such as Er-167 is needed for this fuel. The preliminary design stage was completed in 2001, but the program to construct a prototype in Russia seems to have languished since. Areva is working separately on a version of this called Antares.

The development timeline was for a prototype to be constructed in Russia 2006-09 following regulatory review there.

A smaller version of this, the Remote-Site Modular Helium Reactor (RS-MHR) of 10-25 MWe has been proposed by General Atomics. The fuel would be 20% enriched and refueling interval would be 6-8 years.

A third full-size HTR design is Areva's Very High Temperature Reactor (VHTR) being put forward by Areva NP. It is based on the MHR and has also involved Fuji. Reference design is 600 MW (thermal) with prismatic block fuel like the MHR. Target core outlet temperature is 1000°C and it uses and indirect cycle, possibly with a helium-nitrogen mixes in the secondary system. This removes the possibility of contaminating the generation or hydrogen production plant with radionuclides from the reactor core.

HTRs can potentially use thorium-based fuels, such as HEU or LEU with Th, U-233 with Th, and Pu with Th. Most of the experience with thorium fuels has been in HTRs. General Atomics say that the MHR has a neutron spectrum is such and the TRISO fuel so stable that the reactor can be powered fully with separated transuranic wastes (neptunium, plutonium, americium and curium) from light water reactor used fuel. The fertile actinides enable reactivity control and very high burn-up can be achieved with it - over 500 GWd/t - the Deep Burn concept and hence DB-MHR design. Over 95% of the Pu-239 and 60% of other actinides are destroyed in a single pass.

The three larger HTR designs, with the AHTR described below, are contenders for the US Next-Generation Nuclear Plant.

A small US HTR concept is the Adams Atomic Engines 10 MWe direct simple Brayton cycle plant with low-pressure nitrogen as the reactor coolant and working fluid, and graphite moderation. The reactor core will be a fixed, annular bed with about 80,000 fuel elements each 6 cm diameter and containing approximately 9 grams of heavy metal as TRISO particles, with expected average burn-up of 80 GWd/t. The initial units will provide a reactor core outlet temperature of 800°C and a thermal efficiency near 25%. Limiting coolant flow controls power output. A demonstration plant is proposed for completion by 2011 with series production by 2014.

1.7.3 Liquid Metal Cooled Fast Reactors

Fast neutron reactors have no moderator, a higher neutron flux and are normally cooled by liquid metal such as sodium, lead, or lead-bismuth, with high conductivity and boiling point. They operate at or near atmospheric pressure and have passive safety features (most have convection circulating the primary coolant). Automatic load following is achieved due to the reactivity feedback - constrained coolant flow leads to higher core temperature that slows the reaction. Primary coolant flow is by convection. They typically use boron carbide control rods.

The Encapsulated Nuclear Heat Source (ENHS) is a liquid metal-cooled reactor concept of 50 MWe being developed by the University of California. The core is at the bottom of a metal-filled module sitting in a large pool of secondary molten metal coolant that also accommodates the 8 separate and unconnected steam generators. There is convection circulation of primary coolant within the module and of secondary coolant outside it. Outside the secondary pool the plant is air-cooled. Control rods would need to be adjusted every year or so and load-following would be autonomous. The whole reactor sits in a 17-meter deep silo. Fuel is a uranium-zirconium alloy with 13% U enrichment (or U-Pu-Zr with 11% Pu) with a 15-20 year life. After this the module is removed, stored on site until the primary lead (or Pb-Bi) coolant solidifies, and it would then be shipped as a self-contained and shielded item. A new-fuelled module would be supplied complete with primary coolant. The ENHS is designed for developing countries and is highly proliferation-resistant but is not yet close to commercialization.

A related project is the Secure Transportable Autonomous Reactor – STAR being developed by Argonne under the leadership of Lawrence Livermore Laboratory (DOE). It a lead-cooled fast neutron modular reactor with passive safety features. Its 400 MWt. size means it can be shipped by rail and cooled by natural circulation. It uses U-transuranic nitride fuel in a cassette that is replaced every 15-20 years. The STAR-LM was conceived for power generation, running at 578°C and producing 180 MWe.

STAR-H2 is an adaptation for hydrogen production, with reactor heat at up to 800°C being conveyed by a helium circuit to drive a separate thermo chemical hydrogen production plant, while lower grade heat is harnessed for desalination (multi-stage flash process). Any commercial electricity generation then would be by fuel cells, from the hydrogen. Its development is further off.

A smaller STAR variant is the Small Sealed Transportable Autonomous Reactor - SSTAR, being developed in collaboration with Toshiba and others in Japan (see 4S four paragraphs below). It has lead or Pb-Bi cooling, runs at 566°C and has integral steam generator inside the sealed unit, which would be installed below ground level. Conceived in sizes 10-100 MWe, main development is now focused on a 45 MWt/ 20 MWe version as part of the US Generation IV effort. After a 20-year life without refueling, the whole reactor unit is then returned for recycling the fuel. The core is one-meter diameter and 0.8m high. SSTAR will eventually be coupled to a Brayton cycle turbine using supercritical carbon dioxide. Prototype envisaged 2015.

For all STAR concepts, regional fuel cycle support centers would handle fuel supply and reprocessing, and fresh fuel would be spiked with fission products to deter misuse. Complete burn up of uranium and transuranics is envisaged in STAR-H2, with only fission products being waste.

Japan's LSPR is a lead-bismuth cooled reactor of 150 MWt /53 MWe. Fuelled units would be supplied from a factory and operate for 30 years, then be returned. Concept intended for developing countries.

A small-scale design developed by Toshiba Corporation in cooperation with Japan's Central Research Institute of Electric Power Industry (CRIEPI) and funded by the Japan Atomic Energy Research Institute (JAERI) is the 5 MWt, 200 kWe Rapid-L, using lithium-6 (a liquid neutron poison) as control medium. It would have 2700 fuel pins of 40-50% enriched uranium nitride with 2600°C melting point integrated into a disposable cartridge. The reactivity control system is passive, using lithium expansion modules (LEM), which give, burn up compensation, partial load operation as well as negative reactivity feedback. As the reactor temperature rises, the lithium expands into the core, displacing an inert gas. Other kinds of lithium modules, also integrated into the fuel cartridge, shut down and start up the reactor. Cooling is by molten sodium, and with the LEM control system, reactor power is proportional to primary coolant flow rate. Refueling would be every 10 years in an inert gas environment. Operation would require no skill, due to the inherent safety design features. The whole plant would be about 6.5 meters high and 2 meters diameter.

The Super-Safe, Small & Simple - 4S 'nuclear battery' system is being developed by Toshiba and CRIEPI in Japan in collaboration with STAR work and Westinghouse in USA. It uses sodium as coolant (with electromagnetic pumps) and has passive safety features, notably negative temperature and void reactivity. The whole unit would be factory-built, transported to site, installed below ground level, and would drive a steam cycle. It is capable of three decades of continuous operation without refueling. Metallic fuel (169 pins 10mm diameter) is uranium-zirconium enriched to less than 20% or U-Pu-Zr alloy with 24% Pu for the 10 MWe version or 11.5% Pu for the 50 MWe version. Steady power output over the core lifetime is achieved by progressively moving upwards an annular reflector around the slender core (0.68m diameter, 2m high in the 10 MWe version, 1.2m diameter and 2.5m high in the 50 MWe version) at about one millimeter per week. Burn up will be 34,000 MWday/t. After 14 years a neutron absorber at the center of the core is removed and the reflector repeats its slow movement up the core for 16 more years. Burn up will be 34,000 MWday/t. In the event of power loss the reflector falls to the bottom of the reactor vessel, slowing the reaction, and external air circulation gives decay heat removal. A further safety device is a neutron absorber rod that can drop into the core. After 30 years the fuel would be allowed to cool for a year, then it would be removed and shipped for storage or disposal.

Both 10 MWe and 50 MWe versions of 4S are designed to automatically maintain an outlet coolant temperature of 550°C - suitable for power generation with high temperature electrolytic hydrogen production. Plant cost is projected at US$ 2500/kW and power cost 5-7 cents/kWh for the small unit - very competitive with diesel in many locations. The design has gained considerable support in Alaska and toward the end of 2004 the town of Galena granted initial approval for Toshiba to build a 4S reactor in that remote location. A pre-application NRC has been underway with a view to application for design certification in 2009 and construction and operating license (COL) application by 2012. Its design is sufficiently similar to PRISM - GE's modular 150 MWe liquid metal-cooled inherently-safe reactor which went part-way through US NRC approval process for it to have good prospects of licensing. Toshiba plans a worldwide marketing program to sell the units for power generation at remote mines, desalination plants and for making hydrogen. Eventually it expects sales for hydrogen production to outnumber those for power supply. The L-4S is Pb-Bi cooled version of 4S.

The Hyperion reactor is a small self-regulating hydrogen-moderated and potassium-cooled reactor fuelled by powdered uranium hydride. A US design certification application is possible in 2012.

A significant fast reactor prototype was the EBR-II, a fuel recycle reactor of 62 MWt at Argonne which used the pyrometallurgically refined spent fuel from light water reactors as fuel, including a wide range of actinides. The objective of the program is to use the full energy potential of uranium rather than only about one percent of it. It is shut down and being decommissioned. An EBR-III of 200-300 MWe was proposed but not developed.

Russia has experimented with several lead-cooled reactor designs, and has used lead-bismuth cooling for 40 years in its submarine reactors. Pb-208 (54% of naturally-occurring lead) is transparent to neutrons. A significant Russian design is the BREST fast neutron reactor, of 300 MWe or more with lead as the primary coolant, at 540°C, and supercritical steam generators. The core sits in a pool of lead at near atmospheric pressure. It is inherently safe and uses a U+Pu nitride fuel. No weapons-grade Pu can be produced (since there is no uranium blanket), and spent fuel can be recycled indefinitely, with on-site facilities. A pilot unit is being built at Beloyarsk and 1200 MWe units are planned.

A smaller and newer Russian design is the Lead-Bismuth Fast Reactor (SVBR) of 75-100 MWe. This is an integral design, with the steam generators sitting in the same Pb-Bi pool at 400-480°C as the reactor core, which could use a wide variety of fuels. The unit would be factory-made and shipped as a 4.5m diameter, 7.5m high module, then installed in a tank of water that gives passive heat removal and shielding. A power station with 16 such modules is expected to supply electricity at lower cost than any other new Russian technology as well as achieving inherent safety and high proliferation resistance. (Russia built 7 Alfa-class submarines, each powered by a compact 155 MWt Pb-Bi cooled reactor, and 70 reactor-years operational experience was acquired with these.)

1.7.4 Molten Salt Reactors

During the 1960s the USA developed the molten salt breeder reactor concept as the primary back-up option for the fast breeder reactor (cooled by liquid metal) and a small prototype MSR Experiment (8 MW) operated at Oak Ridge over four years. There is now renewed interest in the concept in Japan, Russia, France and the USA, and one of the six generation IV designs selected for further development is the MSR.

In the Molten Salt Reactor (MSR) the fuel is a molten mixture of lithium and beryllium fluoride salts with dissolved enriched uranium, thorium or U-233 fluorides. The core consists of unclad graphite moderator arranged to allow the flow of salt at some 700°C and at low pressure. Heat is transferred to a secondary salt circuit and thence to steam. It is not a fast reactor, but with some moderation by the graphite is epithermal (intermediate neutron speed). The fission products dissolve in the salt and are removed continuously in an on-line reprocessing loop and replaced with Th-232 or U-238. Actinides remain in the reactor until they fission or are converted to higher actinides which do so. A full-size 1000 MWe MSR breeder reactor was designed but not built. In 2002 a Thorium MSR was designed in France with a fissile zone where most power would be produced and a surrounding fertile zone where most conversion of Th-232 to U-233 would occur.

The FUJI MSR is a 100 MWe design operating as a near-breeder and being developed internationally by a Japanese, Russian and US consortium.

The attractive features of this MSR fuel cycle include: the high-level waste comprising fission products only, hence shorter-lived radioactivity; small inventory of weapons-fissile material (Pu-242 being the dominant Pu isotope); low fuel use (the French self-breeding variant claims 50kg of thorium and 50kg U-238 per billion kWh); and safety due to passive cooling up to any size.

The Advanced High-temperature Reactor (AHTR) is a larger reactor using a coated-particle graphite-matrix fuel like that in the GTMHR (see above section) and with molten fluoride salt as primary coolant. While similar to the gas-cooled HTR it operates at low pressure (less than 1 atmosphere) and higher temperature, and gives better heat transfer than helium. The salt is used solely as coolant, and achieves temperatures of 750-1000°C while at low pressure. This could be used in thermo chemical hydrogen manufacture. Reactor sizes of 1000 MWe/2400 MWt are envisaged, with capital costs estimated at less than $1000/kW.

Molten fluoride salts are a preferred interface fluid between the nuclear heat source and any chemical plant. The aluminum smelting industry provides substantial experience in managing them safely. The hot molten salt can also be used with secondary helium coolant generating power via the Brayton cycle.

1.7.5 Modular Construction

The IRIS developers have outlined the economic case for modular construction of their design (about 330 MWe), and the argument applies similarly to other smaller units. They point out that IRIS with its size and simple design is ideally suited for modular construction. The economy of scale is replaced here with the economy of serial production of many small and simple components and prefabricated sections. They expect that construction of the first IRIS unit will be completed in three years, with subsequent reduction to only two years.

Site layouts have been developed with multiple single units or multiple twin units. In each case, units will be constructed so that there is physical separation sufficient to allow construction of the next unit while the previous one is operating and generating revenue. In spite of this separation, the plant footprint can be very compact so that a site with three IRIS single modules providing 1000 MWe is similar or smaller in size than one with a comparable total power single unit.

Eventually IRIS is expected to have a capital cost and production cost comparable with larger plants. But any small unit such as this will potentially have a funding profile and flexibility otherwise impossible with larger plants. As one module is finished and starts producing electricity, it will generate positive cash flow for the next module to be built. Westinghouse estimates that 1000 MWe delivered by three IRIS units built at three year intervals financed at 10% for ten years require a maximum negative cash flow less than $700 million (compared with about three times that for a single 1000 MWe unit). For developed countries small modular units offer the opportunity of building as necessary, for developing countries it may be the only option, because their electric grids cannot take 1000+ MWe single units.

1.7.6 Floating Nuclear Power Plants

Apart from over 200 nuclear reactors powering various kinds of ships, Rosatom in Russia has set up a subsidiary to supply floating nuclear power plants ranging in size from 70 to 600 MWe. These will be mounted in pairs on a large barge, which will be permanently moored where it is needed to supply power and possibly some desalination to a shore settlement or industrial complex. The first will have two 40 MWe reactors based on those in icebreakers and will operate at Severodvinsk, in the Archangel region. Gazprom will use five of the next seven for offshore oil and gas field development and for operations on the Kola and Yamal peninsulas. One is for Pevek on the Chukotka peninsula, another for Kamchatka region, both in the far east of the country. Further Far East sites being considered are Yakutia and Taimyr. Electricity cost is expected to be much lower than from present alternatives.

The Russian KLT-40S is a reactor well proven in icebreakers and now proposed for wider use in desalination and, on barges, for remote area power supply. Here a 150 MWt unit produces 35 MWe (gross) as well as up to 35 MW of heat for desalination or district heating. These are designed to run 3-4 years between refueling and it is envisaged that they will be operated in pairs to allow for outages, with on-board refueling capability and used fuel storage. At the end of a 12-year operating cycle the whole plant is taken to a central facility for overhaul and removal of used fuel. Two units will be mounted on a 20,000 tonne barge. A larger Russian factory-built and barge-mounted reactor is the VBER-150, of 350 MW thermal, 110 MWe. The larger VBER-300 PWR is a 325 MWe unit, originally envisaged in pairs as a floating nuclear power plant, displacing 49,000 tonnes. As a cogeneration plant it is rated at 200 MWe and 1900 GJ/hr.

1.8 Advanced Nuclear Power Reactors

The nuclear power industry has been developing and improving reactor technology for more than five decades and is starting to build the next generations of reactors to fill orders now materializing.

Several generations of reactors are commonly distinguished. Generation I reactors were developed in 1950-60s, and outside the UK none are still running today. Generation II reactors are typified by the present US fleet and most in operation elsewhere. Generation III (and 3+) are the Advanced Reactors discussed in this section. The first are in operation in Japan and others are under construction or ready to be ordered. Generation IV designs are still on the drawing board and will not be operational before 2020 at the earliest.

About 85% of the world's nuclear electricity is generated by reactors derived from designs originally developed for naval use. These and other second-generation nuclear power units have been found to be safe and reliable, but they are being superseded by better designs.

Reactor suppliers in North America, Japan, Europe, Russia and South Africa have a dozen new nuclear reactor designs at advanced stages of planning, while others are at a research and development stage. Fourth-generation reactors are at concept stage. Third-generation reactors have:

- a standardized design for each type to expedite licensing, reduce capital cost and reduce construction time,
- a simpler and more rugged design, making them easier to operate and less vulnerable to operational upsets,
- higher availability and longer operating life - typically 60 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.

The greatest departure from second-generation designs is that many incorporate passive or inherent safety features which require no active controls or operational intervention to avoid accidents in the event of malfunction, and may rely on gravity, natural convection or resistance to high temperatures.

Advanced Thermal Reactors being marketed in 2008 are given in Table 1.6.

Traditional reactor safety systems are 'active' in the sense that they involve electrical or mechanical operation on command. Some engineered systems operate passively, e.g. pressure relief valves. They function without operator control and despite any loss of auxiliary power. Both require parallel redundant systems. Inherent or full passive safety depends only on physical phenomena such as convection, gravity or resistance to high temperatures, not on functioning of engineered components.

Country and Developer	Reactor	Size MWe	Design Progress	Main Features (improved safety in all)
US-Japan (GE-Hitachi, Toshiba)	ABWR	1300	Commercial operation in Japan since 1996-7. In US: NRC certified 1997, FOAKE.	Evolutionary design. More efficient, less waste. Simplified construction (48 months) and operation.
USA (Westinghouse)	AP-600 AP-1000 (PWR)	600 1100	AP-600: NRC certified 1999, FOAKE. AP-1000 NRC certification 2005.	Simplified construction and operation. 3 years to build. 60-year plant life.
France-Germany (Areva NP)	EPR US-EPR (PWR)	1600	Future French standard. French design approval. Being built in Finland. US version developed.	Evolutionary design. High fuel efficiency. Low cost electricity.
USA (GE)	ESBWR	1550	Developed from ABWR, under certification in USA	Evolutionary design. Short construction time.
Japan (Utilities, Mitsubishi)	APWR US-APWR EU-APWR	1530 1700 1700	Basic design in progress, planned for Tsuruga US design certification application 2008.	Hybrid safety features. Simplified Construction and operation.
South Korea (KHNP, derived from Westinghouse)	APR-1400 (PWR)	1450	Design certification 2003, First units expected to be operating c 2012.	Evolutionary design. Increased reliability. Simplified construction and operation.
Germany (Areva NP)	SWR-1000 BWR)	1200	Under development, pre-certification in USA	Innovative design. High fuel efficiency.
Russia (Gidropress)	VVER-1200 (PWR)	1200	Replacement for Leningrad and Novovoronezh plants	High fuel efficiency.
Russia (Gidropress)	V-392 (PWR)	950-1000	Two being built in India, Bid for China in 2005.	Evolutionary design. 60-year plant life.
Canada (AECL)	CANDU-6 CANDU-9	750 925+	Enhanced model Licensing approval 1997	Evolutionary design. Flexible fuel requirements. C-9: Single stand-alone unit.
Canada (AECL)	ACR	700 1080	undergoing certification in Canada	Evolutionary design. Light water cooling. Low-enriched fuel.
South Africa (Eskom, Westinghouse)	PBMR	170 (module)	prototype due to start building (Chinese 200 MWe counterpart under const.)	Modular plant, low cost. High fuel efficiency. Direct cycle gas turbine.
USA-Russia et al (General Atomics - OKBM)	GT-MHR	285 (module)	Under development in Russia by multinational joint venture	Modular plant, low cost. High fuel efficiency. Direct cycle gas turbine.

Table 1.6 Advanced Thermal Reactors Being Marketed in 2008

1.8.1 Licensing

Many of the future designs are larger than predecessors. Increasingly they involve international collaboration. Certification of designs is on a national basis, and is safety-based. In Europe there are moves towards harmonized requirements for licensing.

However, in Europe reactors may also be certified according to compliance with European Utilities Requirements (EUR). These are basically a utilities' wish list of some 5000 items needed for new nuclear plants. Plants certified as complying with EUR include Westinghouse AP1000, Gidropress' AES-92, Areva's EPR, GE's ABWR, Areva's SWR-1000, and Westinghouse BWR 90.

In the USA a number of reactor types have received Design Certification (see below) and others are in process: ESBWR from GE-Hitachi, US EPR from Areva and US-APWR from Mitsubishi. Early in 2008 the NRC said that beyond these three, six pre-application reviews would get underway until about 2010. These include IRIS from Westinghouse, PBMR from Eskom and 4S from Toshiba as well as Canada's ACR and General Atomics' GT-MHR apparently

Longer term, NRC expected to focus on the Next-Generation Nuclear Plant (NGNP) for the USA - essentially the Very High Temperature Reactor (VHTR) among the Generation IV designs.

Two major international initiatives have been launched to define future reactor and fuel cycle technology, mostly looking further ahead than the main subjects of this chapter.

Generation IV International Forum (GIF) is a US-led grouping set up in 2001 that has identified six reactor concepts for further investigation with a view to commercial deployment by 2030.

The IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) is focused more on developing country needs, and initially involved Russia rather than the USA, though the USA has now joined it. It is now funded through the IAEA budget.

At the commercial level, by the end of 2006 three major Western-Japanese alliances had formed to dominate much of the world reactor supply market:

- Areva with Mitsubishi Heavy Industries(MHI) in a major project and subsequently in fuel fabrication,
- General Electric with Hitachi,
- Westinghouse had become a 77% owned subsidiary of Toshiba (with Shaw group 20%).

Then in March 2008 Toshiba signed a technical cooperation agreement on civil nuclear power with Russia's Atomenergoprom- the single vertically-integrated state holding company for Russia's nuclear power sector created in 2007. This could lead to a "strategic partnership" and include designing and engineering of commercial nuclear power plants, as well as manufacturing and maintenance of large equipment.

1.8.2 Light Water Reactors

In the USA, the federal Department of Energy (DOE) and the commercial nuclear industry in the 1990s developed four advanced reactor types. Two of them fall into the category of large "evolutionary" designs that build directly on the experience of operating light water reactors in the USA, Japan and Western Europe. These reactors are in the 1300-megawatt range.

One is an advanced boiling water reactor (ABWR) derived from a General Electric design. Two examples built by Hitachi and two by Toshiba are in commercial operation in Japan, with another under construction there and two in Taiwan. Four more are planned in Japan and another two in the USA. Though GE and Hitachi have subsequently joined up, Toshiba retains some rights over the design. Both GE-Hitachi and Toshiba (with NRG Energy in USA) are marketing the design.

The other type, System 80+, is an advanced pressurized water reactor (PWR), which was ready for commercialization but is not now being promoted for sale. Eight System 80 reactors in South Korea incorporate many design features of the System 80+, which is the basis of the Korean Next Generation Reactor program, specifically the APR-1400 that is expected to be in operation soon after 2010 and marketed worldwide.

The US Nuclear Regulatory Commission (NRC) gave final design certification for both in May 1997, noting that they exceeded NRC "safety goals by several orders of magnitude". The ABWR has also been certified as meeting European requirements for advanced reactors.

Another, more innovative US advanced reactor is smaller - 600 MWe - and has passive safety features (its projected core damage frequency is nearly 1000 times less than today's NRC requirements). The Westinghouse Advanced Passive 600, AP-600, gained NRC final design certification in 1999.

These NRC approvals were the first such generic certifications to be issued and are valid for 15 years. As a result of an exhaustive public process, safety issues within the scope of the certified designs have been fully resolved and hence will not be open to legal challenge during licensing for particular plants. US utilities will be able to obtain a single NRC license to both construct and operate a reactor before construction begins.

Separate from the NRC process and beyond its immediate requirements, the US nuclear industry selected one standardized design in each category - the large ABWR and the medium-sized AP-600, for detailed first-of-a-kind engineering (FOAKE) work. The US$ 200 million program was half funded by DOE. It means that prospective buyers now have firm information on construction costs and schedules.

The Westinghouse AP-1000, scaled-up from the AP-600, received final design certification from the NRC in December 2005 - the first generation 3+ type to do so. It represents the culmination of a 1300 man-year and $440 million design and testing program. In May 2007

Westinghouse applied for UK generic design assessment (pre-licensing approval) based on the NRC design certification, and expressing its policy of global standardization. The application was supported by utilities including E.ON.

Overnight capital costs are projected at $1200 per kilowatt and modular design will reduce construction time to 36 months. The 1100 MWe AP-1000 generating costs are expected to be below US$ 3.5 cents/kWh and it has a 60-year operating life. It has been selected for building in China (4 units) and is under active consideration for building in Europe and USA, and is capable of running on a full MOX core if required.

General Electric has developed the ESBWR of 1390 MWe with passive safety systems, from its ABWR design. Originally the European Simplified Boiling Water Reactor, this is now known as the Economic & Simplified BWR (ESBWR) and a 1560 MWe version is at preliminary stage of NRC design certification in the USA, so that design approval was expected at the end of 2008, with formal certification 12 months later. It is favored for early US construction and could be operational in 2014. It uses 4.2% enriched fuel and has a design life of 60 years.

Another US-origin but international project which is a few years behind the AP-1000 is the International Reactor Innovative & Secure (IRIS). Westinghouse is leading a wide consortium developing it as an advanced 3rd Generation project. IRIS is a modular 335 MWe pressurized water reactor with integral steam generators and primary coolant system all within the pressure vessel. It is nominally 335 MWe but can be less, e.g. 100 MWe. Fuel is initially similar to present LWRs with 5% enrichment and burn-up of 60,000 MWd/t with fuelling interval of 3 to 3.5 years, but is designed ultimately for 10% enrichment and 80 GWd/t burn-up with an 8-year cycle, or equivalent MOX core. The core has low power density. IRIS could be deployed in the next decade, and US design certification is at pre-application stage. Multiple modules are expected to cost US$ 1000-1200 per kW for power generation, though some consortium partners are interested in desalination, one in district heating.

In Japan**,** the first two ABWRs, Kashiwazaki Kariwa-6 & 7, have been operating since 1996 and are expected to have a 60-year life. These GE-Hitachi-Toshiba units cost about US$ 2000/kW to build, and produce power at about US 7c/kWh. Two more started up in 2004 & 2005. Future ABWR units are expected to cost US$ 1700/kW. Several of the 1350 MWe units are under construction and planned in Japan and Taiwan.

To complement this ABWR Hitachi-GE has completed systems design for three more of the same type - 600, 900 and 1700 MWe versions of the 1350 MWe design. The smaller versions will have standardized features that reduce costs. Construction of the ABWR-600 is expected to take 34 months - significantly less than the 1350 MWe units.

Mitsubishi's large APWR (1538 MWe) - advanced PWR - was developed in collaboration with four utilities (Westinghouse was earlier involved). The first two are planned for Tsuruga. It is simpler, combines active and passive cooling systems to greater effect, and has over 55 GWd/t fuel burn-up. It will be the basis for the next generation of Japanese PWRs.

The US-APWR will be 1700 MWe, due to higher thermal efficiency (39%) and has 24 month refueling cycle and target cost of $1500/kW. US design certification application was in January 2008 with approval expected in 2011. The first units may be built for TXU at Comanche Peak near Dallas, Texas. In March 2008 MHI submitted the same design for EUR certification, as EU-APWR**.**

The Atmea joint venture has been established by Areva NP and Mitsubishi Heavy Industries to develop an 1100 MWe (net) three-loop PWR with extended fuel cycles, 37% thermal efficiency and the capacity to use mixed-oxide fuel only. Fuel cycle is 12-24 months and the reactor has load-following capability. They expect to have this ready for license application by 2010. The reactor is regarded as mid-sized relative to other generation III units and will be marketed primarily to countries embarking upon nuclear power programs.

In South Korea, the APR-1400 Advanced PWR design has evolved from the US System 80+ with enhanced safety and seismic robustness and was earlier known as the Korean Next-Generation Reactor. Design certification by the Korean Institute of Nuclear Safety was awarded in May 2003. The first of these 1450 MWe reactors will be Shin-Kori-3 & 4, expected to be operating about 2012. Fuel has burnable poison and will have up to 60 GWd/t burn-up. Projected cost is US$ 1400 per kilowatt, falling to $1200/kW in later units with 48-month construction time. Plant life is 60 years.

In Europe, several designs are being developed to meet the European Utility Requirements (EUR) of French and German utilities, which have stringent safety criteria. Areva NP (formerly Framatome ANP) has developed a large (1600 and up to 1750 MWe) European pressurized water reactor (EPR), which was confirmed in mid 1995 as the new standard design for France and received French design approval in 2004. It is derived from the French N4 and German Konvoi types and is expected to provide power about 10% cheaper than the N4. It will operate flexibly to follow loads, have fuel burn-up of 65 GWd/t and the highest thermal efficiency of any light water reactor, at 36%. It is capable of using a full core load of MOX. Availability is expected to be 92% over a 60-year service life. It has four separate, redundant safety systems rather than passive safety.

The first EPR unit is being built at Olkiluoto in Finland, the second at Flamanville in France. A US version, the US-EPR, is undergoing review in USA with intention of a design certification application in 2007. It is now known as the Evolutionary PWR (EPR). Overnight capital cost is quoted as $2400 per kilowatt, levelised over the first four units. Albeit due to delays in the construction program the costs of Olkiluoto have risen sharply.

Together with German utilities and safety authorities, Areva NP (Framatome ANP) is also developing another evolutionary design, the SWR 1000, a 1200-1290 MWe BWR with 60-year design life. The design was completed in 1999 and US certification was sought, but then deferred. As well as many passive safety features, the reactor is simpler overall and uses high-burn up fuels enriched to 3.54%, giving it refueling intervals of up to 24 months. It is ready for commercial deployment and the prospects of that will be helped by a 2008 agreement with Siemens and the major German utility E.On (Siemens built the Gundremmingen plant on which the design is based, for E.On).

Toshiba has been developing its evolutionary advanced BWR (1500 MWe) design, originally BWR 90+ from ABB then Westinghouse, working with Scandinavian utilities to meet EUR requirements.

In Russia, several-advanced reactor designs have been developed - advanced PWR with passive safety features.

Gidropress late-model VVER-1000 units with enhanced safety (AES 92 & 91 power plants) are being built in India and China. Two more are planned for Belene in Bulgaria. The AES-92 is certified as meeting EUR.

A third-generation standardized VVER-1200 reactor of 1150-1200 MWe is an evolutionary development of the well-proven VVER-1000 in the AES-92 plant, with longer life, greater power and efficiency. The lead units will be built at Novovoronezh II, to start operation in 2012-13 followed by Leningrad II for 2013-14. An AES-2006 plant will consist of two of these OKB Gidropress reactor units expected to run for 50 years with capacity factor of 90%. Capital cost is said to be US$ 1200/kW and construction time 54 months. They have enhanced safety including that related to earthquakes and aircraft impact with some passive safety features, double containment and core damage frequency of $1x10^{-7}$.

Atomenergoproekt say that the AES-2006 conforms to both Russian standards and European Utilities Requirements (EUR).

The VVER-1500 model was being developed by Gidropress. It will have 50-60 MWd/t burn-up and enhanced safety. Design was expected to be complete in 2007 but this schedule has slipped in favor of the evolutionary VVER-1200.

OKBM's VBER-300 PWR is a 295-325 MWe unit developed from naval power plants and was originally envisaged in pairs as a floating nuclear power plant. It is designed for 60-year life and 90% capacity factor. It now planned to develop it as a land-based unit with Kazatomprom, with a view to exports, and the first unit will be built in Kazakhstan.

Canada has had two designs under development that are based on its reliable CANDU-6 reactors, the most recent of which are operating in China.

The CANDU-9 (925-1300 MWe) was developed from this also as a single-unit plant. It has flexible fuel requirements ranging from natural uranium through slightly-enriched uranium, recovered uranium from reprocessing spent PWR fuel, mixed oxide (U & Pu) fuel, direct use of spent PWR fuel, to thorium. It may be able to burn military plutonium or actinides separated from reprocessed PWR/BWR waste. A two-year licensing review of the CANDU-9 design was successfully completed early in 1997, but the design has been shelved.

Some of the innovation of this, along with experience in building recent Korean and Chinese units, was then put back into the Enhanced CANDU-6 - built as twin units - with power increase to 750 MWe and flexible fuel options, plus 4.5 year construction and 60-year plant life (with mid-life pressure tube replacement). This is under consideration for new build in Ontario.

The Advanced Candu Reactor (ACR), a 3rd generation reactor, is a more innovative concept. While retaining the low-pressure heavy water moderator, it incorporates some features of the pressurized water reactor. Adopting light water cooling and a more compact core reduces capital cost, and because the reactor is run at higher temperature and coolant pressure, it has higher thermal efficiency.

The ACR-700 design was 700 MWe but is physically much smaller, simpler and more efficient as well as 40% cheaper than the CANDU-6. But the ACR-1000 of 1080-1200 MWe is now the focus of attention by AECL. It has more fuel channels (each of which can be regarded as a module of about 2.5 MWe). The ACR will run on low-enriched uranium (about 1.5-2.0% U-235) with high burn-up, extending the fuel life by about three times and reducing high-level waste volumes accordingly. It will also efficiently burn MOX fuel, thorium and actinides.

Regulatory confidence in safety is enhanced by a small negative void reactivity for the first time in CANDU, and utilizing other passive safety features as well as two independent and fast shutdown systems. Units will be assembled from prefabricated modules, cutting construction time to 3.5 years. ACR units can be built singly but are optimal in pairs. They will have 60 year design life overall but require mid-life pressure tube replacement.

ACR is moving towards design certification in Canada, with a view to following in China, USA and UK. In 2007 AECL applied for UK generic design assessment (pre-licensing approval). The first ACR-1000 unit is expected to be operating in 2016 in Ontario.

The CANDU X or SCWR is a variant of the ACR, but with supercritical light water coolant (e.g. 25 MPa and 625ºC) to provide 40% thermal efficiency. The size range envisaged is 350 to 1150 MWe, depending on the number of fuel channels used. Commercialization envisaged after 2020.

India is developing the Advanced Heavy Water reactor (AHWR) as the third stage in its plan to utilize thorium to fuel its overall nuclear power program. The AHWR is a 300 MWe reactor moderated by heavy water at low pressure. The calandria has 500 vertical pressure tubes and the coolant is boiling light water circulated by convection. Each fuel assembly has 30 Th-U-233 oxide pins and 24 Pu-Th oxide pins around a central rod with burnable absorber. Burn-up of 24 GWd/t is envisaged. It is designed to be self-sustaining in relation to U-233 bred from Th-232 and have a low Pu inventory and consumption, with slightly negative void coefficient of reactivity. It is designed for 100-year plant life and is expected to utilize 65% of the energy of the fuel.

Once it is fully operational, each AHWR fuel assembly will have the fuel pins arranged in three concentric rings arranged:

Inner: 12 pins Th-U-233 with 3.0% U-233,
Intermediate: 18 pins Th-U-233 with 3.75% U-233,
Outer: 24 pins Th-Pu-239 with 3.25% Pu.

The fissile plutonium content will decrease from an initial 75% to 25% at equilibrium discharge burn-up level.

1.8.3 High Temperature Gas-Cooled Reactors

These reactors use helium as a coolant that at up to 950ºC drives a gas turbine for electricity and a compressor to return the gas to the reactor core. Fuel is in the form of TRISO particles less than a millimeter in diameter. Each has a kernel of uranium oxycarbide, with the uranium enriched up to 17% U-235. This is surrounded by layers of carbon and silicon carbide, giving a containment for fission products that is stable to 1600°C or more. These particles may be arranged: in blocks - hexagonal 'prisms' of graphite or in billiard ball-sized pebbles of graphite encased in silicon carbide.

As described earlier South Africa's Pebble Bed Modular Reactor (PBMR) is being developed by a consortium led by the utility Eskom, and drawing on German expertise. It aims for a step change in safety, economics and proliferation resistance. Production units will be 165 MWe. They will have a direct-cycle gas turbine generator and thermal efficiency about 42%. Up to 450,000 fuel pebbles recycle through the reactor continuously (about six times each) until they are expended, giving an average enrichment in the fuel load of 4-5% and average burn-up of 90 GWday/t U (eventual target burn-ups are 200 GWd/t). This means on-line refueling as expended pebbles are replaced, giving high capacity factor. The pressure vessel is lined with graphite and there is a central column of graphite as reflector. Control rods are in the side reflectors and cold shutdown units in the central column.

Performance includes great flexibility in loads (40-100%), with rapid change in power settings. Power density in the core is about one tenth of that in light water reactor, and if coolant circulation ceases the fuel will survive initial high temperatures while the reactor shuts itself down - giving inherent safety. Each unit will finally discharge about 19 tonnes/yr of spent pebbles to ventilated on-site storage bins.

Overnight construction cost (when in clusters of eight units) is expected to be US$ 1000/kW and generating cost below 3 US cents/kWh. Investors in the PBMR project are Eskom, the South African Industrial Development Corporation and Westinghouse. A demonstration plant is due to be built in 2007 for commercial operation in 2010.

A larger US design, the Gas Turbine - Modular Helium Reactor (GT-MHR), will be built as modules of 285 MWe each directly driving a gas turbine at 48% thermal efficiency. The cylindrical core consists of 102 hexagonal fuel element columns of graphite blocks with channels for helium and control rods. Graphite reflector blocks are both inside and around the core. Half the core is replaced every 18 months. Burn-up is about 100,000 MWd/t. It is being developed by General Atomics in partnership with Russia's Minatom, supported by Fuji (Japan). Initially it will be used to burn pure ex-weapons plutonium at Tomsk in Russia. The preliminary design stage was completed in 2001.

1.8.4 Fast Neutron Reactors

Several countries have research and development programs for improved Fast Breeder Reactors (FBR), which are a type of Fast Neutron Reactor. These use the uranium-238 in reactor fuel as well as the fissile U-235 isotope used in most reactors.

About 20 liquid metal-cooled FBRs have already been operating, some since the 1950s, and some supply electricity commercially. About 290 reactor-years of operating experience have been accumulated.

Natural uranium contains about 0.7 % U-235 and 99.3 % U-238. In any reactor the U-238 component is turned into several isotopes of plutonium during its operation. Two of these, Pu 239 and Pu 241, then undergo fission in the same way as U 235 to produce heat. In a fast neutron reactor this process is optimized so that it can 'breed' fuel, often using a depleted uranium blanket around the core. FBRs can utilize uranium at least 60 times more efficiently than a normal reactor.

They are however expensive to build and could only be justified economically if uranium prices were to rise to pre-1980 values, well above the current market price. For this reason research work on the 1450 MWe European FBR has almost ceased. Closure of the 1250 MWe French Superphenix FBR after very little operation over 13 years also set back developments.

In the UK, the Dounreay Fast Reactor started operation in 1959 using sodium-potassium coolant. The much larger Prototype Fast reactor that operated for 20 years until the UK Government withdrew funding followed this.

Research continues in India. At the Indira Gandhi Center for Atomic Research a 40 MWt fast breeder test reactor has been operating since 1985. In addition, the tiny Kamini there is employed to explore the use of thorium as nuclear fuel, by breeding fissile U-233. In 2004 construction of a 500 MWe prototype fast breeder reactor started at Kalpakkam. The unit is expected to be operating in 2010, fuelled with uranium-plutonium carbide (the reactor-grade Pu being from its existing PHWRs) and with a thorium blanket to breed fissile U-233. This will take India's ambitious thorium program to stage 2, and set the scene for eventual full utilization of the country's abundant thorium to fuel reactors.

Japan plans to develop FBRs, and its Joyo experimental reactor that has been operating since 1977 is now being boosted to 140 MWt. The 280 MWe Monju prototype commercial FBR was connected to the grid in 1995, but was then shut down due to a sodium leak.

The Russian BN-600 fast breeder reactor has been supplying electricity to the grid since 1981 and has the best operating and production record of all Russia's nuclear power units. It uses uranium oxide fuel and the sodium coolant delivers 550°C at little more than atmospheric pressure. The BN 350 FBR operated in Kazakhstan for 27 years and about half of its output was used for water desalination. Russia plans to reconfigure the BN-600 to burn the plutonium from its military stockpiles.

Construction has started at Beloyarsk on the first BN-800, a new larger (880 MWe) FBR from OKBM with improved features including fuel flexibility - U+Pu nitride, MOX, or metal, and with breeding ratio up to 1.3. It has much enhanced safety and improved economy - operating cost is expected to be only 15% more than VVER. It is capable of burning 2 tonnes of plutonium per year from dismantled weapons and will test the recycling of minor actinides in the fuel.

Russia has experimented with several lead-cooled reactor designs, and has used lead-bismuth cooling for 40 years in reactors for its 7 Alfa class submarines. Pb-208 (54% of naturally-occurring lead) is transparent to neutrons. A significant new Russian design is the BREST fast neutron reactor, of 300 MWe or more with C, and supercritical steam generators. It lead as the primary coolant, at 540 is inherently safe and uses a high-density U+Pu nitride fuel with no requirement for high enrichment levels. No weapons-grade plutonium can be produced (since there is no uranium blanket - all the breeding occurs in the core). The initial cores can comprise Pu and spent fuel - hence loaded with fission products, and radio logically 'hot'. Subsequently, any surplus plutonium, which is not in pure form, can be used as the cores of new reactors. Used fuel can be recycled indefinitely, with on-site reprocessing and associated facilities. A pilot unit is planned for Beloyarsk and 1200 MWe units are proposed.

In the USA, GE was involved in designing a modular 150 MWe liquid metal-cooled inherently safe reactor - PRISM. GE and Argonne have also been developing an advanced liquid-metal fast breeder reactor (ALMR) of over 1400 MWe, but both designs at an early stage were withdrawn from NRC review. No US fast neutron reactor has so far been larger than 66 MWe and none has supplied electricity commercially.

The Super-PRISM is a GE advanced reactor design for compact modular pool-type reactors with passive cooling and decay C, heat removal. Modules are 1000 MWt and operate at higher temperature - 510 than the original PRISM. The pool-type modules contain the complete primary system with sodium coolant. The Pu & DU fuel can be oxide or metal, but minor actinides are not removed in reprocessing so that even fresh fuel is intensely radioactive and hence resistant to misappropriation. The fission products are removed in reprocessing and resultant wastes are shorter-lived than usual. Fuel stays in the reactor six years, with one third removed every two years. The commercial plant concept uses six reactor modules to provide 2280 MWe, and the design meets Generation IV criteria including generation cost of less than 3 cents/kWh.

Korea's KALIMER (Korea Advanced Liquid Metal Reactor) is a 600 MWe pool type sodium-cooled fast reactor designed to operate at over 500ºC. It has evolved from a 150 MWe version. It has a transmuter core, and no breeding blanket is involved. Future development of KALIMER as a Generation IV type is envisaged.

In the USA Mitsubishi Heavy Industries (MHI) is involved with a consortium to develop the Advanced Recycling Reactor, a fast reactor that will burn actinides with uranium and plutonium. This will be based on MHI's Japan Standard Fast reactor concept, though with breeding ration less than 1:1. In this connection MHI has also set up Mitsubishi FBR Systems (MFBR).

1.8.5 Accelerator-Driven Systems

A recent development has been the merging of accelerator and fission reactor technologies to generate electricity and transmute long-lived radioactive wastes.

A high-energy proton beam hitting a heavy metal target produces neutrons by spallation. The neutrons cause fission in the fuel, but unlike a conventional reactor, the fuel is sub-critical, and fission ceases when the accelerator is turned off. The fuel may be uranium, plutonium or thorium, possibly mixed with long-lived wastes from conventional reactors.

Many technical and engineering questions remain to be explored before the potential of this concept can be demonstrated.

1.9 Generation IV Nuclear Reactors

The Generation IV International Forum (GIF) was initiated in 2000 and formally chartered in mid 2001. It is an international collective representing governments of countries where nuclear energy is significant now and also seen as vital for the future. They are committed to joint development of the next generation of nuclear technology. Led by the USA, Argentina, Brazil, Canada, France, Japan, South Korea, South Africa, Switzerland, and the UK are members of the GIF, along with the EU. Russia and China were admitted in 2006.

After some two years' deliberation, GIF (then representing ten countries) late in 2002 announced the selection of six reactor technologies that they believe represent the future shape of nuclear energy. These are selected on the basis of being clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while being resistant to diversion of materials for weapons proliferation and secure from terrorist attacks. They will be the subject of further development internationally.

In addition to selecting these six concepts for deployment between 2010 and 2030, the GIF recognized a number of International Near-Term Deployment advanced reactors available before 2015.

Most of the six systems employ a closed fuel cycle to maximize the resource base and minimize high-level wastes to be sent to a repository. Three of the six are fast reactors and one can be built as a fast reactor, one is described as epithermal, and only two operate with slow neutrons like today's plants.

Only one is cooled by light water, two are helium-cooled and the others have lead-bismuth, sodium or fluoride salt coolant. The latter three operate at low pressure, with significant safety advantage. The last has the uranium fuel dissolved in the circulating coolant. Temperatures range from 510°C to 1000°C, compared with less than 330°C for today's light water reactors, and this means that four of them can be used for thermo chemical hydrogen production.

The sizes range from 150 to 1500 MWe (or equivalent thermal), with the lead-cooled one optionally available as a 50-150 MWe "battery" with long core life (15-20 years without refueling) as replaceable cassette or entire reactor module. This is designed for distributed generation or desalination.

At least four of the systems have significant operating experience already in most respects of their design, which may mean that they can be in commercial operation well before 2030.

In February 2005 five of the participants signed an agreement to take forward the R&D on the six technologies. The USA, Canada, France, Japan and UK agreed to undertake joint research and exchange technical information.

While Russia was not initially part of GIF, one design corresponds with the BREST reactor being developed there, and Russia is now the main operator of the sodium-cooled fast reactor for electricity - another of the technologies put forward by the GIF.

India is also not involved with the GIF but is developing its own advanced technology to utilize thorium as a nuclear fuel. A three-stage program has the first stage well established, with Pressurized Heavy Water Reactors (PHWRs, elsewhere known as CANDUs) fuelled by natural uranium to generate plutonium. Then Fast Breeder Reactors (FBRs) use this plutonium-based fuel to breed U-233 from thorium, and finally advanced nuclear power systems will use the U-233. The spent fuel will be reprocessed to recover fissile materials for recycling. The two major options for the third stage, while continuing with the PHWR and FBR programs, are an Advanced Heavy Water Reactor and sub critical Accelerator-Driven Systems.

Closely related to GIF is the Multinational Design Evaluation Program (MDEP) set up in 2005, led by the OECD Nuclear Energy Agency and involving the IAEA. It aims to develop multinational regulatory standards for design of Gen IV reactors. The US Nuclear Regulatory Commission (NRC) has proposed a three-stage process culminating in international design certification for these. Ten countries are involved so far: Canada, China, Finland, France, Japan, Korea, Russia, South Africa, UK, USA, but others which have or are likely to have firm commitments to building new nuclear plants may be admitted. In September 2007 the NRC called for countries involved in development of Gen IV reactors to move to stage 3 of design evaluation, which means developing common design requirements so that regulatory standards can be harmonized. NRC has published its draft design requirements.

1.9.1 Generation IV International Forum Reactor Technologies

Gas-cooled fast reactors; like other helium-cooled reactors which have operated or are under development, these will be high-temperature units - 850°C, suitable for power generation, thermo chemical hydrogen production or other process heat. For electricity, the gas will directly drive a gas turbine (Brayton cycle). Fuels would include depleted uranium and any other fissile or fertile materials. Spent fuel would be reprocessed on site and all the actinides recycled to minimize production of long-lived radioactive wastes.

Lead-cooled fast reactors; liquid metal (Pb or Pb-Bi) cooling is by natural convection. Fuel is depleted uranium metal or nitride, with full actinide recycle from regional or central reprocessing plants. A wide range of unit sizes is envisaged, from factory-built "battery" with 15-20 year life for small grids or developing countries, to modular 300-400 MWe units and large single plants of 1400 MWe. Operating temperature of 550°C is readily achievable but 800°C is envisaged with advanced materials and this would enable thermo chemical hydrogen production.

This corresponds with Russia's BREST fast reactor technology that is lead-cooled and builds on 40 years experience of lead-bismuth cooling in submarine reactors. Its fuel is U+Pu nitride. More immediately the GIF proposal appears to arise from two experimental designs: the US STAR and Japan's LSPR, these being lead and lead-bismuth cooled respectively.

Molten salt reactors; the uranium fuel is dissolved in the sodium fluoride salt coolant which circulates through graphite core channels to achieve some moderation and an epithermal neutron spectrum. Fission products are removed continuously and the actinides are fully recycled, while plutonium and other actinides can be added along with U-238. Coolant temperature is 700°C at very low pressure, with 800°C envisaged. A secondary coolant system is used for electricity generation, and thermo chemical hydrogen production is also feasible.

During the 1960s the USA developed the molten salt breeder reactor as the primary back-up option for the conventional fast breeder reactor and a small prototype was operated. Recent work has focused on lithium and beryllium fluoride coolant with dissolved thorium and U-233 fuel. The attractive features of the MSR fuel cycle include: the high-level waste comprising fission products only, hence shorter-lived radioactivity; small inventory of weapons-fissile material (Pu-242 being the dominant Pu isotope); low fuel use (the French self-breeding variant claims 50kg of thorium and 50kg U-238 per billion kWh); and safety due to passive cooling up to any size.

Sodium-cooled fast reactors; this builds on more than 300 reactor-years experienced with fast neutron reactors over five decades and in eight countries. It utilizes depleted uranium in the fuel and has a coolant temperature of 550°C enabling electricity generation via a secondary sodium circuit, the primary one being at near atmospheric pressure. Two variants are proposed: a 150-500 MWe type with actinides incorporated into a metal fuel requiring pyrometallurgical processing on site, and a 500-1500 MWe type with conventional MOX fuel reprocessed in conventional facilities elsewhere.

Early in 2008, the USA, France and Japan signed an agreement to expand their cooperation on the development of sodium-cooled fast reactor technology. The agreement relates to their collaboration in the Global Nuclear Energy Partnership, aimed at closing the nuclear fuel cycle through the use of advanced reprocessing and fast reactor technologies, and seeks to avoid duplication of effort.

Supercritical water-cooled reactors. This is a very high-pressure water-cooled reactor that operates above the thermodynamic critical point of water to give a thermal efficiency about one third higher than today's light water reactors from which the design evolves. The supercritical water (25 MPa and 510-550°C) directly drives the turbine, without any secondary steam system. Passive safety features are similar to those of simplified boiling water reactors. Fuel is uranium oxide, enriched in the case of the open fuel cycle option. However, it can be built as a fast reactor with full actinide recycle based on conventional reprocessing. Most research on the design has been in Japan.

Very high-temperature gas reactors; these are graphite-moderated, helium-cooled reactors, based on substantial experience. The core can be built of prismatic blocks such as the Japanese HTTR and the GTMHR under development by General Atomics and others in Russia, or it may be pebble bed such as the Chinese HTR-10 and the PBMR under development in South Africa, with international partners. Outlet temperature of 1000°C enables thermo chemical hydrogen production via an intermediate heat exchanger, with electricity cogeneration, or direct high-efficiency driving of a gas turbine (Brayton cycle). There is some flexibility in fuels, but no recycle. Modules of 600 MW thermal are envisaged.

1.9.2 INPRO

As well as the GIF, another program with similar aims is coordinated by the IAEA. This is the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). It was launched in 2001 and has 22 members including Russia, aiming "to support the safe, sustainable, economic and proliferation-resistant use of nuclear technology to meet the global energy needs of the 21st century." It does this by examining issues related to the development and deployment of Innovative Nuclear Energy Systems (INS) for sustainable energy supply.

One of the case studies in phase 1 of INPRO was undertaken by Russia on its BN-800 fast reactor, though the emphasis was on the methodology rather than the technology. Nevertheless, fast reactor systems will feature in further INPRO work.

1.9.3 Global Nuclear Energy Partnership (GNEP)

This concept, announced in 2006, builds on earlier US work with the Integral Fast Reactor (IFR) project and international work on fast reactors. Its main thrust in to counter proliferation concerns, but will have the effect of much greater resource utilization as well.

It envisages fabrication and leasing of fuel for conventional reactors, with the used fuel being returned to fuel supplier countries and pyro-processed to recover uranium and actinides, leaving only fission products as high-level waste. The actinide mix is then burned in on-site fast reactors.

1.10 The Hydrogen Economy

1.10.1 Nuclear Energy and Hydrogen Production

Hydrogen can be burned in a normal internal combustion engine, and some test cars are thus equipped. Trials in aircraft have also been carried out. However, its main use is likely to be in fuel cells.

A fuel cell is conceptually a refuelable battery, making electricity as a direct product of a chemical reaction. But where the normal battery has all the active ingredients built in at the factory, fuel cells are supplied with fuel from an external source. They catalyze the oxidation of hydrogen directly to electricity at relatively low temperatures and the claimed theoretical efficiency of converting chemical to electrical energy to drive the wheels is about 60% (or more). However, in practice about half that has been achieved, except for the higher-temperature solid oxide fuel cells – 46%.

Like electricity, hydrogen is an energy carrier (but not a primary energy source). As oil becomes more expensive, hydrogen may replace it as a transport fuel and in other application. This development becomes more likely as fuel cells are developed, with hydrogen as the preferred fuel. If gas also becomes expensive, or constraints are put on carbon dioxide emissions, non-fossil sources of hydrogen will become necessary.

Similar to electricity, hydrogen for transport use will tend to be produced near where it is to be used. This will have major geo-political implications as industrialized countries become less dependent on oil and gas from distant parts of the world.

While a growing hydrogen economy already exists, linked to the worldwide chemical and refining industry, a much greater one is in sight. With new uses for hydrogen as a fuel, the primary energy demand for its production may approach that for electricity production.

Certainly the most exciting future prospect is that of cogeneration of electricity and hydrogen, so called “hydricity”. If we are to address the problem of global warming then we have to reduce carbon emissions from both the electricity generation and transport sectors. The use of hydrogen, via fuel cells for transportation, has great potential. Here co generating nuclear reactors can provide the answer to reducing CO_2 production from generation and transportation. It is then possible that nuclear’s role in hydrogen production may evolve in several stages, such as the use of nuclear heat to assist steam reforming of natural gas, then the high-temperature electrolysis of steam, using heat and electricity from nuclear reactors, then finally, high-temperature thermo chemical production using nuclear heat from high temperature reactors. So because reactor types, such as the VHTR, can produce hydrogen as a by-product of electricity production these could be a potential source of reasonably priced hydrogen, created without emitting carbon dioxide, which can be used to fuel the hydrogen powered vehicles that are already being developed. This is a long-term solution.

The advent of the hydrogen economy also has another incidental benefit for the nuclear industry in that it can stabilize the economics by shielding them from the problems in the market place due to it being a base load generator. This is because, whilst electricity is sold in a simple commodity market, it cannot be stored. However, the hydrogen economy solves this problem, because whilst a nuclear generator operates continuously any excess power, at times of light load, can be diverted into electrolysis plants to generate hydrogen. Incidentally this is why pumped storage and nuclear are natural partners.

Thus whilst a VHTR plant will generate electricity and hydrogen 24 hours per day, hydrogen production can be increased at night by producing hydrogen rather than electricity. There will also be the advantage that electricity can be sold according to market price. This is a complete reversal of the situation today and has the potential to permanently enhance the economics for nuclear power generation.

1.11 The Nuclear Fuel Cycle

The various activities associated with the production of electricity from nuclear reactions are referred to collectively as the nuclear fuel cycle. The nuclear fuel cycle starts with the mining of uranium and ends with the disposal of nuclear waste. With the reprocessing of used fuel as an option for nuclear energy, the stages form a true cycle (Figure 1.17).

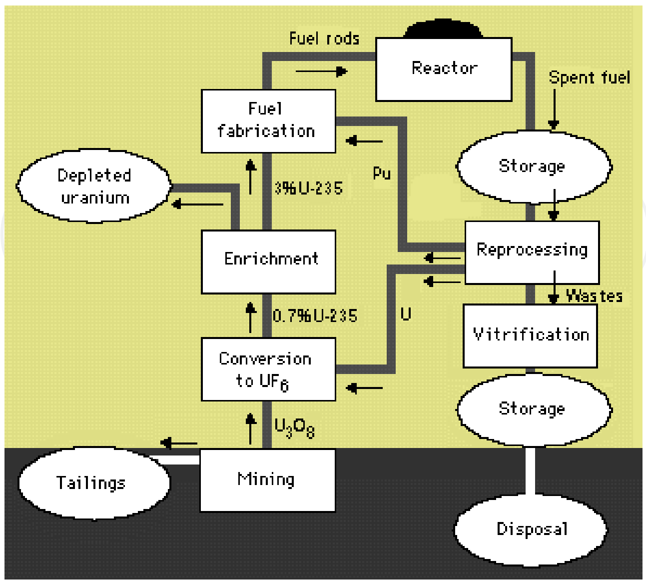


Figure 1.17 The Nuclear Fuel Cycle

1.11.1 Uranium

Uranium is a slightly radioactive metal that occurs throughout the earth's crust. It is about 500 times more abundant than gold and about as common as tin. It is present in most rocks and soils as well as in many rivers and in seawater. It is, for example, found in concentrations of about four parts per million (ppm) in granite, which makes up 60% of the earth's crust. In fertilizers, uranium concentration can be as high as 400 ppm (0.04%), and some coal deposits contain uranium at concentrations greater than 100 ppm (0.01%). Most of the radioactivity associated with uranium in nature is in fact due to other minerals derived from it by radioactive decay processes, and which are left behind in mining and milling.

There are a number of areas around the world where the concentration of uranium in the ground is sufficiently high that extraction of it for use as nuclear fuel is economically feasible. Such concentrations are called ore.

1.11.2 Uranium Mining

Both excavation and in situ techniques are used to recover uranium ore. Excavation may be underground and open pit mining.

In general, open pit mining is used where deposits are close to the surface and underground mining is used for deep deposits, typically greater than 120 m deep. Open pit mines require large holes on the surface, larger than the size of the ore deposit, since the walls of the pit must be sloped to prevent collapse. As a result, the quantity of material that must be removed in order to access the ore may be large. Underground mines have relatively small surface disturbance and the quantity of material that must be removed to access the ore is considerably less than in the case of an open pit mine.

An increasing proportion of the world's uranium now comes from in situ leaching (ISL), where oxygenated groundwater is circulated through a very porous orebody to dissolve the uranium and bring it to the surface. ISL may be with slightly acid or with alkaline solutions to keep the uranium in solution. The uranium is then recovered from the solution as in a conventional mill.

The decision as to which mining method to use for a particular deposit is governed by the nature of the orebody, safety and economic considerations.

In the case of underground uranium mines, special precautions, consisting primarily of increased ventilation, are required to protect against airborne radiation exposure.

1.11.3 Uranium Milling

Milling, which is generally carried out close to a uranium mine, extracts the uranium from the ore. Most mining facilities include a mill, although where mines are close together, one mill may process the ore from several mines. Milling produces a uranium oxide concentrate that is shipped from the mill. It is sometimes referred to as 'yellowcake' and generally contains more than 80% uranium. The original ore may contain as little as 0.1% uranium.

In a mill, uranium is extracted from the crushed and ground-up ore by leaching, in which either a strong acid or a strong alkaline solution is used to dissolve the uranium. The uranium is then removed from this solution and precipitated. After drying and usually heating it is packed in 200-litre drums as a concentrate.

The remainder of the ore, containing most of the radioactivity and nearly all the rock material, becomes tailings, which are emplaced in engineered facilities near the mine (often in mined out pit). Tailings contain long-lived radioactive materials in low concentrations and toxic materials such as heavy metals; however, the total quantity of radioactive elements is less than in the original ore, and their collective radioactivity will be much shorter-lived. These materials need to be isolated from the environment.

1.11.4 Conversion

The product of a uranium mill is not directly usable as a fuel for a nuclear reactor. Additional processing, generally referred to as enrichment, is required for most kinds of reactors. This process requires uranium to be in gaseous form and the way this is achieved is to convert it to uranium hex fluoride, which is a gas at relatively low temperatures.

At a conversion facility, uranium is first refined to uranium dioxide, which can be used as the fuel for those types of reactors that do not require enriched uranium. Most is then converted into uranium hex fluoride, ready for the enrichment plant. It is shipped in strong metal containers. The main hazard of this stage of the fuel cycle is the use of hydrogen fluoride.

1.11.5 Enrichment

Natural uranium consists, primarily, of a mixture of two isotopes (atomic forms) of uranium. Only 0.7% of natural uranium is "fissile", or capable of undergoing fission, the process by which energy is produced in a nuclear reactor. The fissile isotope of uranium is uranium 235 (U-235). The remainder is uranium 238 (U-238).

In the most common types of nuclear reactors, a higher than natural concentration of U-235 is required. The enrichment process produces this higher concentration, typically between 3.5% and 5% U-235, by removing over 85% of the U-238. Separating gaseous uranium hex fluoride into two streams, one being enriched to the required level and known as low-enriched uranium does this. The other stream is progressively depleted in U-235 and is called 'tails'.

There are two enrichment processes in large-scale commercial use, each of which uses uranium hex fluoride as feed: gaseous diffusion and gas centrifuge. They both use the physical properties of molecules, specifically the 1% mass difference, to separate the isotopes. The product of this stage of the nuclear fuel cycle is enriched uranium hex fluoride, which is reconverted to produce enriched uranium oxide.

1.11.6 Fuel Fabrication

Reactor fuel is generally in the form of ceramic pellets. These are formed from pressed uranium oxide that is sintered (baked) at a high temperature (over 1400°C). The pellets are then encased in metal tubes to form fuel rods, which are arranged into a fuel assembly ready for introduction into a reactor. The dimensions of the fuel pellets and other components of the fuel assembly are precisely controlled to ensure consistency in the characteristics of fuel bundles.

In a fuel fabrication plant great care is taken with the size and shape of processing vessels to avoid criticality (a limited chain reaction releasing radiation). With low-enriched fuel criticality is most unlikely, but in plants handling special fuels for research reactors this is a vital consideration.

1.11.7 Uranium Requirements

Table 1.7 is a review of current uranium requirements.

Concerns are being expressed about the uranium requirement for future nuclear generation.

The uranium resource is sustainable, with adequate known resources being continuously replenished at least as fast as they are being used. The essential dynamic is the strength of market forces when the market is constantly evolving through advances in human knowledge and the technologies of exploration, mining, and resource utilization. Depletion of today’s known uranium resources will be more than counterbalanced by replenishment from new discoveries, technical progress and possible substitution.

Country and Developer	Reactor	Size MWe	Design Progress	Main Features (improved safety in all)
US-Japan (GE-Hitachi, Toshiba)	ABWR	1300	Commercial operation in Japan since 1996-7. In US: NRC certified 1997, FOAKE.	Evolutionary design. More efficient, less waste. Simplified construction (48 months) and operation.
USA (Westinghouse)	AP-600 AP-1000 (PWR)	600 1100	AP-600: NRC certified 1999, FOAKE. AP-1000 NRC certification 2005.	Simplified construction and operation. 3 years to build. 60-year plant life.
France-Germany (Areva NP)	EPR US-EPR (PWR)	1600	Future French standard. French design approval. Being built in Finland. US version developed.	Evolutionary design. High fuel efficiency. Low cost electricity.
USA (GE)	ESBWR	1550	Developed from ABWR, under certification in USA	Evolutionary design. Short construction time.
Japan (Utilities, Mitsubishi)	APWR US-APWR EU-APWR	1530 1700 1700	Basic design in progress, planned for Tsuruga US design certification application 2008.	Hybrid safety features. Simplified Construction and operation.
South Korea (KHNP, derived from Westinghouse)	APR-1400 (PWR)	1450	Design certification 2003, First units expected to be operating c 2012.	Evolutionary design. Increased reliability. Simplified construction and operation.
Germany (Areva NP)	SWR-1000 BWR)	1200	Under development, pre-certification in USA	Innovative design. High fuel efficiency.
Russia (Gidropress)	VVER-1200 (PWR)	1200	Replacement for Leningrad and Novovoronezh plants	High fuel efficiency.
Russia (Gidropress)	V-392 (PWR)	950-1000	Two being built in India, Bid for China in 2005.	Evolutionary design. 60-year plant life.
Canada (AECL)	CANDU-6 CANDU-9	750 925+	Enhanced model Licensing approval 1997	Evolutionary design. Flexible fuel requirements. C-9: Single stand-alone unit.
Canada (AECL)	ACR	700 1080	undergoing certification in Canada	Evolutionary design. Light water cooling. Low-enriched fuel.
South Africa (Eskom, Westinghouse)	PBMR	170 (module)	prototype due to start building (Chinese 200 MWe counterpart under const.)	Modular plant, low cost. High fuel efficiency. Direct cycle gas turbine.
USA-Russia et al (General Atomics - OKBM)	GT-MHR	285 (module)	Under development in Russia by multinational joint venture	Modular plant, low cost. High fuel efficiency. Direct cycle gas turbine.

Sources: Reactor data: WNA to 23/05/08.
IAEA- for nuclear electricity production & percentage of electricity (% e) 5/07.
WNA: Global Nuclear Fuel Market (reference scenario) - for U.

Table 1.7 World Nuclear Power Reactors 2006-2008 and Uranium Requirements

In addition, a huge increase in efficiency is readily possible through the technological step to fast neutron reactors. This option – unique among mineral resources – offers the nuclear industry a special kind of insurance against future resource shortage.

It may therefore be fairly concluded that uranium supplies will be more than adequate to fuel foreseeable expansions of nuclear power.

1.12 Thorium as a Nuclear Fuel

Thorium, as well as uranium, can be used as a nuclear fuel. Although not fissile itself, thorium-232 (Th-232) will absorb slow neutrons to produce uranium-233 (U-233), which is fissile (and long-lived). Hence like uranium-238 (U-238) it is fertile.

In one significant respect U-233 is better than uranium-235 and plutonium-239, because of its higher neutron yield per neutron absorbed. Given a start with some other fissile material (U-235 or Pu-239), a breeding cycle similar to but more efficient than that with U-238 and plutonium (in normal, slow-neutron reactors) can be set up. However, there are also features of the neutron economy that counter this advantage. In particular Pa-233 is a neutron absorber that diminishes U-233 yield. The Th-232 absorbs a neutron to become Th-233 that quickly beta decays to protactinium-233 and then more slowly to U-233. The irradiated fuel can then be unloaded from the reactor, the U-233 separated from the thorium, and fed back into another reactor as part of a closed fuel cycle.

Over the last 30 years there has been interest in utilizing thorium as a nuclear fuel since it is more abundant in the Earth's crust than uranium. Also, all of the mined thorium is potentially useable in a reactor, compared with the 0.7% of natural uranium, so some 40 times the amount of energy per unit mass might theoretically be available (without recourse to fast breeder reactors).

A major potential application for conventional Pars involves fuel assemblies arranged so that a blanket of mainly thorium fuel rods surrounds a more-enriched seed element containing U-235 that supplies neutrons to the sub critical blanket. As U-233 is produced in the blanket it is burned there. This is the Light Water Breeder Reactor concept that was successfully demonstrated in the USA in the 1970s.

It is currently being developed in a more deliberately proliferation-resistant way. The central seed region of each fuel assembly will have uranium enriched to 20% U-235. The blanket will be thorium with some U-238, which means that any uranium chemically separated from it (for the U-233) is not useable for weapons. Spent blanket fuel also contains U-232, which decays rapidly and has very gamma-active daughters creating significant problems in handling the bred U-233 and hence conferring proliferation resistance. Plutonium produced in the seed will have a high proportion of Pu-238, generating a lot of heat and making it even more unsuitable for weapons than normal reactor-grade Pu.

A variation of this is the use of whole homogeneous assembles arranged so that a set of them makes up a seed and blanket arrangement. If the seed fuel is metal uranium alloy instead of oxide, there is better heat conduction to cope with its higher temperatures. Seed fuel remains three years in the reactor, blanket fuel for up to 14 years.

Since the early 1990s Russia has had a program to develop a thorium-uranium fuel, which more recently has moved to have a particular emphasis on utilization of weapons-grade plutonium in a thorium-plutonium fuel.

The thorium-plutonium fuel claims four advantages over MOX: proliferation resistance, compatibility with existing reactors - which will need minimal modification to be able to burn it and the fuel can be made in existing plants in Russia. In addition, a lot more plutonium can be put into a single fuel assembly than with MOX, so that three times as much can be disposed of as when using MOX. The spent fuel amounts to about half the volume of MOX and is even less likely to allow recovery of weapons-useable material than spent MOX fuel, since less fissile plutonium remains in it. With an estimated 150 tonnes of weapons plutonium in Russia, the thorium-plutonium project would not necessarily cut across existing plans to make MOX fuel.

1.12.1 Thorium R&D History

The use of thorium-based fuel cycles has been studied for about 30 years, but on a much smaller scale than uranium or uranium/plutonium cycles. Basic research and development has been conducted in Germany, India, Japan, Russia, the UK and the USA. Test reactor irradiation of thorium fuel to high burnups has also been conducted and several test reactors have either been partially or completely loaded with thorium-based fuel.

Noteworthy experiments involving thorium fuel include the following, the first three being high-temperature gas-cooled reactors:

- Between 1967 and 1988, the AVR (Atom Versuchs Reaktor) experimental pebble bed reactor at Julich, Germany, operated for over 750 weeks at 15 MWe, about 95% of the time with thorium-based fuel. The fuel used consisted of about 100 000 billiard ball-sized fuel elements. Overall a total of 1360 kg of thorium was used, mixed with high-enriched uranium (HEU). Maximum burnups of 150,000 MWd/t were achieved.
- Thorium fuel elements with a 10:1 Th/U (HEU) ratio were irradiated in the 20 MWth Dragon reactor at Winfrith, UK, for 741 full power days. Dragon was run as an OECD/Euratom cooperation project, involving Austria, Denmark, Sweden, Norway and Switzerland in addition to the UK, from 1964 to 1973. The Th/U fuel was used to 'breed and feed', so that the U-233 formed replaced the U-235 at about the same rate, and fuel could be left in the reactor for about six years.
- General Atomics' Peach Bottom high-temperature, graphite-moderated, helium-cooled reactor (HTGR) in the USA operated between 1967 and 1974 at 110 MWth, using high-enriched uranium with thorium.
- In India, the Kamini 30 kWth experimental neutron-source research reactor using U-233, recovered from ThO_2 fuel irradiated in another reactor, started up in 1996 near Kalpakkam. The reactor was built adjacent to the 40 MWt Fast Breeder Test Reactor, in which the ThO_2 is irradiated.

- In the Netherlands, an aqueous homogenous suspension reactor has operated at 1MWth for three years. The HEU/Th fuel is circulated in solution and reprocessing occurs continuously to remove fission products, resulting in a high conversion rate to U-233.
- There have been several experiments with fast neutron reactors.

1.12.2 Thorium Power Reactors

Much experience has been gained in thorium-based fuel in power reactors around the world, some using high-enriched uranium (HEU) as the main fuel:

- The 300 MWe THTR (Thorium High-Temperature Reactor) reactor in Germany was developed from the AVR and operated between 1983 and 1989 with 674,000 pebbles, over half containing Th/HEU fuel (the rest graphite moderator and some neutron absorbers). These were continuously recycled on load and on average the fuel passed six times through the core. Fuel fabrication was on an industrial scale.
- The Fort St Vrain reactor was the only commercial thorium-fuelled nuclear plant in the USA, also developed from the AVR in Germany, and operated 1976 - 1989. It was a high-temperature (700°C), graphite-moderated, helium-cooled reactor with a Th/HEU fuel designed to operate at 842 MWth (330 MWe). The fuel was in micro spheres of thorium carbide and Th/U-235 carbide coated with silicon oxide and pyrolytic carbon to retain fission products. It was arranged in hexagonal columns ('prisms') rather than as pebbles. Almost 25 tonnes of thorium was used in fuel for the reactor, and this achieved 170,000 MWd/t burn-up.
- Thorium-based fuel for Pressurized Water Reactors (PWRs) was investigated at the Shippingport reactor in the USA using both U-235 and plutonium as the initial fissile material. It was concluded that thorium would not significantly affect operating strategies or core margins. The light water breeder reactor (LWBR) concept was also successfully tested here from 1977 to 1982 with thorium and U-233 fuel clad with Zircaloy using the 'seed/blanket' concept.
- The 60 MWe Lingen Boiling Water Reactor (BWR) in Germany utilized Th/Pu-based fuel test elements.

1.12.3 Emerging Advanced Thorium Reactor Concepts

Concepts for advanced reactors based on thorium-fuel cycles include:

- Light Water Reactors; with fuel based on plutonium oxide (PuO_2), thorium oxide (ThO_2) and/or uranium oxide (UO_2) particles arranged in fuel rods.
- High-Temperature Gas-cooled Reactors (HTGR) of two kinds: pebble bed and with prismatic fuel elements.

  Gas Turbine-Modular Helium Reactor (GT-MHR); research on HTGRs in the USA led to a concept using a prismatic fuel. The use of helium as a coolant at high temperature, and the relatively small power output per module (600 MWth), permit direct coupling of the MHR to a gas turbine (a Brayton cycle), resulting in generation at almost 50% thermal efficiency. The GT-MHR core can accommodate a wide range of fuel options, including HEU/Th, U-233/Th and Pu/Th. The use of HEU/Th fuel was demonstrated in the Fort St Vrain reactor (see above).

Pebble-Bed Modular reactor (PBMR) - Arising from German work the PBMR was conceived in South Africa and is now being developed by a multinational consortium. It can potentially use thorium in its fuel pebbles.

- Molten salt reactors (MSR) - This is an advanced breeder concept, in which the fuel is a molten mixture of lithium and beryllium fluoride salts with dissolved enriched uranium, thorium or U-233 fluorides. The core consists of unclad graphite moderator arranged to allow the flow of salt at some 700°C and at low pressure. Heat is transferred to a secondary salt circuit and thence to steam. It is not a fast reactor, but with some moderation by the graphite is epithermal (intermediate neutron speed). The fission products dissolve in the salt and are removed continuously in an on-line reprocessing loop and replaced with Th-232 or U-238. Actinides remain in the reactor until they fission or are converted to higher actinides which do so. The MSR was studied in depth in the 1960s, but is now being revived because of the availability of advanced technology for the materials and components.
- There is now renewed interest in the MSR concept in Japan, Russia, France and the USA, and one of the six generation IV designs selected for further development is the MSR. In 2002 a Thorium MSR was designed in France with a fissile zone where most power would be produced and a surrounding fertile zone where most conversion of Th-232 to U-233 would occur.
- Advanced Heavy Water Reactor (AHWR); India is working on this, and like the Canadian ACR the 300 MWe design is light water cooled. The main part of the core is sub critical with Th/U-233 oxide and Th/Pu-239 oxide, mixed so that the system is self-sustaining in U-233. The initial core will be entirely Th-Pu-239 oxide fuel assemblies, but as U-233 is available, 30 of the fuel pins in each assembly will be Th-U-233 oxide, arranged in concentric rings. It is designed for 100-year plant life and is expected to utilize 65% of the energy of the fuel. About 75% of the power will come from the thorium.
- CANDU-type reactors; AECL is researching the thorium fuel cycle application to enhanced CANDU-6 and ACR-1000 reactors. With 5% plutonium (reactor grade) plus thorium high burn-up and low power costs are indicated.
- Plutonium disposition; today MOX (U,Pu) fuels are used in some conventional reactors, with Pu-239 providing the main fissile ingredient. An alternative is to use Th/Pu fuel, with plutonium being consumed and fissile U-233 bred. The remaining U-233 after separation could be used in a Th/U fuel cycle.

Much development work is still required before the thorium fuel cycle can be commercialized, and the effort required seems unlikely while (or where) abundant uranium is available.

1.13 Nuclear Fusion Power

Fusion powers the sun and stars as hydrogen atoms fuse together to form helium, and matter is converted into energy. Hydrogen, heated to very high temperatures changes from a gas to a plasma in which the negatively charged electrons are separated from the positively charged atomic nuclei (ions). Normally, fusion is not possible because the positively charged nuclei naturally repel each other. But as the temperature increases the ions move faster, and they collide at speeds high enough to overcome the normal repulsion. The nuclei can then fuse, causing a release of energy.

In the sun, massive gravitational forces create the right conditions for this, but on Earth they are much harder to achieve. Fusion fuel - different isotopes of hydrogen - must be heated to extreme temperatures of over ten million degrees Celsius, and must be kept dense enough, and confined for long enough (at least one second) to trigger the energy release. The aim of the controlled fusion research program is to achieve "ignition" which occurs when enough fusion reactions take place for the process to become self-sustaining, with fresh fuel then being added to continue it.

1.13.1 Basic Fusion Technology

With current technology, the reaction most readily feasible is between the nuclei of the two heavy forms (isotopes) of hydrogen - deuterium (D) and tritium (T). Each D-T fusion event releases 17.6 MeV (2.8×10^{-12} joule, compared with 200 MeV for a U-235 fission). Deuterium occurs naturally in sea water (30 grams per cubic meter), which makes it very abundant relative to other energy resources. Tritium does not occur naturally and is radioactive, with a half-life of around 12 years. It can be made in a conventional nuclear reactor, or in the present context, bred in a fusion system from lithium. Lithium is found in large quantities (30 parts per million) in the Earth's crust and in weaker concentrations in the sea. While the D-T reaction is the main focus of attention, long-term hopes are for a D-D reaction, but this requires much higher temperatures.

In a fusion reactor, the concept is that neutrons will be absorbed in a blanket containing lithium that surrounds the core. The lithium is then transformed into tritium and helium. The blanket must be thick enough (about 1 meter) to slow down the neutrons. This heats the blanket and a coolant flowing through it then transfers the heat away to produce steam that can be used to generate electricity by conventional methods. The difficulty has been to develop a device that can heat the D-T fuel to a high enough temperature and confine it long enough so that more energy is released through fusion reactions than is used to get the reaction going.

At present, two different experimental approaches are being studied: fusion energy by magnetic confinement (MFE) and fusion by inertial confinement (ICF). The first method uses strong magnetic fields to trap the hot plasma. The second involves compressing a hydrogen pellet by smashing it with strong lasers or particle beams.

1.13.2 Magnetic Confinement (MFE)

In MFE, hundreds of cubic meters of D-T plasma at a density of less than a milligram per cubic meter are confined by a magnetic field at a few atmospheres pressure and heated to fusion temperature.

Magnetic fields are ideal for confining plasma because the electrical charges on the separated ions and electrons mean that they follow the magnetic field lines. The aim is to prevent the particles from coming into contact with the reactor walls as this will dissipate their heat and slow them down. The most effective magnetic configuration is toroidal, shaped like a thin doughnut, in which the magnetic field is curved around to form a closed loop. For proper confinement, this toroidal field must have superimposed upon it a perpendicular field component (a poloidal field). The result is a magnetic field with force lines following spiral (helical) paths, along and around which the plasma particles are guided. There are several types of toroidal confinement system, the most important being tokamaks, stellarators and reversed field pinch (RFP) devices.

In a tokamak, the toroidal field is created by a series of coils evenly spaced around the torus-shaped reactor, and the poloidal field is created by a strong electric current flowing through the plasma. In a stellarator the helical lines of force are produced by a series of coils which may themselves be helical in shape. But no current is induced in the plasma. RFP devices have the same toroidal and poloidal components as a tokamak, but the current flowing through the plasma is much stronger and the direction of the toroidal field within the plasma is reversed.

In tokamaks and RFP devices, the current flowing through the plasma also serves to heat it to a temperature of about 10 million degrees Celsius. Beyond that, additional heating systems are needed to achieve the temperatures necessary for fusion. In stellarators, these heating systems have to supply all the energy needed.

The tokamak (*toroidalnya kamera ee magnetnaya katushka* - torus-shaped magnetic chamber) was designed in 1951 by Soviet physicists Andrei Sakharov and Igor Tamm. Tokamaks operate within limited parameters outside which sudden losses of energy confinement (disruptions) can occur, causing major thermal and mechanical stresses to the structure and walls. Nevertheless, it is considered the most promising design, and research is continuing on various tokamaks around the world, the two largest being the Joint European Torus (JET) in the UK and the tokamak fusion test reactor (TFTR) at Princeton in the USA.

Research is also being carried out on several types of stellarator. The biggest of these, the Large Helical Device at Japan's National Institute of Fusion Research, began operating in 1998. It is being used to study of the best magnetic configuration for plasma confinement. At Garching in Germany, plasma is created and heated by electromagnetic waves, and this work will be progressed in the W7-X stellerator, to be built at the new German research center in Greifswald. Another stellarator, TJ-II, is under construction in Madrid, Spain. Because stellarators have no toroidal current there are no disruptions and they can be operated continuously. The disadvantage is that, despite the stability, they do not confine the plasma so well.

RFP devices differ from tokamaks mainly in the spatial distribution of the toroidal magnetic field, which changes sign at the edge of the plasma. The RFX machine in Padua is used to study the physical problems arising from the spontaneous reorganization of the magnetic field, which is an intrinsic feature of this configuration.

1.13.3 Inertial Confinement (ICF)

In ICF, which is a newer line of research, laser or ion beams are focused very precisely onto the surface of a target, which is a sphere of D-T ice, a few millimeters in diameter. This evaporates or ionizes the outer layer of the material to form a plasma crown that expands generating an inward-moving compression front or implosion that heats up the inner layers of material. The core or central hot spot of the fuel may be compressed to one thousand times its liquid density, and ignition occurs when the core temperature reaches about 100 million degrees Celsius. Thermonuclear combustion then spreads rapidly through the compressed fuel, producing several times more energy than was originally used to bombard the capsule. The time required for these reactions to occur is limited by the inertia of the fuel (hence the name), but is less than a microsecond. The aim is to produce repeated micro explosions.

Recent work at Osaka in Japan suggests that 'fast ignition' may be achieved at lower temperature with a second very intense laser pulse through a millimetre-high gold cone inside the compressed fuel, and timed to coincide with the peak compression. This technique means that fuel compression is separated from hot spot generation with ignition, making the process more practical.

So far most inertial confinement work has involved lasers, although their low energy makes it unlikely that they would be used in an actual fusion reactor. The world's most powerful laser fusion facility is the NOVA at Lawrence Livermore Laboratory in the US, and declassified results show compressions to densities of up to 600 times that of the D-T liquid. Various light and heavy ion accelerator systems are also being studied, with a view to obtaining high particle densities.

1.13.4 Cold Fusion

In 1989, spectacular claims were made for another approach, when two researchers, in USA and UK, claimed to have achieved fusion in a simple tabletop apparatus working at room temperature. Other experimenters failed to replicate this "cold fusion", however, and most of the scientific community no longer considers it a real phenomenon. Nevertheless, research continues. Cold fusion involves the electrolysis of heavy water using palladium electrodes on which deuterium nuclei are said to concentrate at very high densities.

1.13.5 Fusion History

Today, many countries take part in fusion research to some extent, led by the European Union, the USA, Russia and Japan, with vigorous programs also under way in China, Brazil, Canada, and Korea. Initially, fusion research in the USA and USSR was linked to atomic weapons development, and it remained classified until the 1958 Atoms for Peace conference in Geneva. Following a breakthrough at the Soviet tokamak, fusion research became big science in the 1970s. But the cost and complexity of the devices involved increased to the point where international co-operation was the only way forward.

In 1978, the European Community (with Sweden and Switzerland) launched the JET project in the UK. JET produced its first plasma in 1983, and saw successful experiments using a D-T fuel mix in 1991. In the USA, the PLT tokamak at Princeton produced a plasma temperature of more than 60 million degrees in 1978 and D-T experiments began on the Tokamak Fusion Test Reactor (TFTR) there in 1993. In Japan, experiments have been carried out since 1988 on the JT-60 Tokamak.

1.13.6 ITER

In 1985, the Soviet Union suggested building a next generation tokamak with Europe, Japan and the USA. Collaboration was established under the auspices of the International Atomic Energy Agency (IAEA). Between 1988 and 1990, the initial designs were drawn up for an International Thermonuclear Experimental Reactor (ITER) with the aim of proving that fusion could produce useful energy. The four parties agreed in 1992 to collaborate further on Engineering Design Activities for ITER (ITER is both an acronym, and means 'a path' or 'journey' in Latin). Canada and Kazakhstan are also involved through Euratom and Russia respectively.

Six years later, the ITER Council approved the first comprehensive design of a fusion reactor based on well-established physics and technology with a price tag of US$ 6 billion. Then the USA decided pull out of the project, forcing a 50% reduction in costs and a redesign. The result was the ITER - Fusion Energy Advanced Tokomak (ITER- FEAT) - expected to cost $3 billion but still achieve the targets of a self-sustaining reaction and a net energy gain. The energy gain is unlikely to be enough for a power plant, but it will demonstrate feasibility (Figure 1.18).

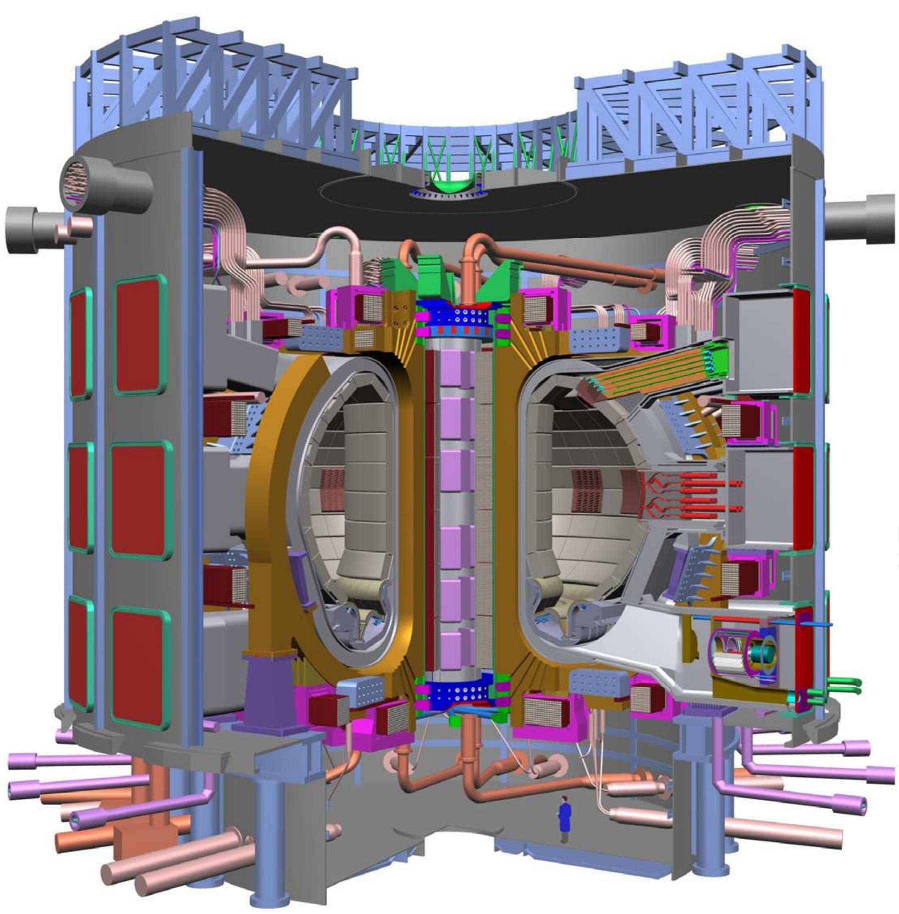


Figure 1.18 International Tokamak Experimental Reactor (ITER)

In 2003 the USA rejoined the project and China also announced it would do so. After deadlocked discussion, the six partners agreed in mid 2005 to site ITER at Cadarache, in southern France. The deal involved major concessions to Japan, which had put forward Rokkasho as a preferred site. The EU and France will contribute half of the EUR 12.8 billion total cost, with the other partners - Japan, China, South Korea, USA and Russia - putting in 10% each. Japan will provide a lot of the high-tech components, will host a EUR 1 billion materials testing facility and will have the right to host a subsequent demonstration fusion reactor. The total cost of the 500 MWt ITER comprises about half for the ten-year construction and half for 20 years of operation.

In November 2006 China, India, Japan, Russia, South Korea, the USA and the European Union - signed the ITER implementing agreement. The French President praised the attempt to "tame solar fire to meet the challenge of ecological energy".

1.13.7 Assessing Fusion Power

The use of fusion power plants could substantially reduce the environmental impacts of increasing world electricity demands since, like nuclear fission power, they would not contribute to acid rain or the greenhouse effect. Fusion power could easily satisfy the energy needs associated with continued economic growth, given the ready availability of fuels. There would be no danger of a runaway fusion reaction as this is intrinsically impossible and any malfunction would result in a rapid shutdown of the plant.

However, although fusion generates no radioactive fission products or transuranic elements and the unburned gases can be treated on site, there would a short-term radioactive waste problem due to activation products. Some component materials will become radioactive during the lifetime of a reactor, due to bombardment with high-energy neutrons, and will eventually become radioactive waste. The volume of such waste would be similar to that due to activation products from a fission reactor. The radiotoxicity of these wastes would be relatively short-lived compared with the actinides (long-lived alpha-emitting transuranic isotopes) from a fission reactor.

There are also other concerns, principally regarding the possible release of tritium into the environment. It is radioactive and very difficult to contain since it can penetrate concrete, rubber and some grades of steel. As an isotope of hydrogen, it is easily incorporated into water, making the water itself weakly radioactive. With a half-life of 12.4 years, tritium remains a threat to health for about 125 years after it is created, as a gas or in water. It can be inhaled, absorbed through the skin or ingested. Inhaled tritium spreads throughout the soft tissues and tritiated water mixes quickly with all the water in the body. Each fusion reactor could release significant quantities of tritium during operation through routine leaks, assuming the best containment systems. An accident could release even more. This is one reason why long-term hopes are for the deuterium-deuterium fusion process, dispensing with tritium.

While fusion power clearly has much to offer when the technology is eventually developed, the problems associated with it also need to be addressed if is to become a widely used future energy source. Much will change before fusion power is commercialized, including the development of new materials.

1.14 Nuclear Energy And Seawater Desalination

It is estimated that one fifth of the world's population does not have access to safe drinking water, and that this proportion will increase due to population growth relative to water resources. The worst affected areas are the arid and semiarid regions of Asia and North Africa. Wars over access to water, not simply energy and mineral resources, are conceivable. Fresh water is a major priority in sustainable development. Where it cannot be obtained from streams and aquifers, desalination of seawater or mineralized groundwater is required.

Most desalination today uses fossil fuels, and thus contributes to increased levels of greenhouse gases. Total world capacity is approaching 30 million m^3/day of potable water, in some 12,500 plants. Half of these are in the Middle East. The largest produces 454,000 m^3/day.

Desalination is energy-intensive. Reverse osmosis needs about 6 kWh of electricity per cubic meter of water (depending on its salt content), while other techniques require heat at 70-130°C and use 25-200 kWh/m^3. A variety of low-temperature heat sources may be used, including solar energy. The choice of process generally depends on the relative economic values of fresh water and particular fuels.

Small and medium sized nuclear reactors are suitable for desalination, often with cogeneration of electricity using low-pressure steam from the turbine and hot seawater feed from the final cooling system. The main opportunities for nuclear plants have been identified as the 80-100,000 m^3/day and 200-500,000 m^3/day ranges.

The feasibility of integrated nuclear desalination plants has been proven with over 150 reactor-years of experience, chiefly in Kazakhstan, India and Japan.

The BN-350 fast reactor at Aktau, in Kazakhstan, successfully produced up to 135 MWe of electricity and 80,000 m^3/day of potable water over some 27 years, about 60% of its power being used for heat and desalination. The plant was designed as 1000 MWt but never operated at more than 750 MWt, but it established the feasibility and reliability of such cogeneration plants. (In fact, oil/gas boilers were used in conjunction with it, and total desalination capacity through ten MED units was 120,000 m^3/day.)

In Japan, some ten desalination facilities linked to pressurized water reactors operating for electricity production has yielded 1000-3000 m^3/day each of potable water, and over 100 reactor-years of experience have accrued. MSF was initially employed, but MED and RO have been found more efficient there. The water is used for the reactors' own cooling systems.

India has been engaged in desalination research since the 1970s and in 2002 set up a demonstration plant coupled to twin 170 MWe nuclear power reactors (PHWR) at the

Madras Atomic Power Station, Kalpakkam, in southeast India. This Nuclear Desalination Demonstration Project is a hybrid reverse osmosis / multi-stage flash plant, the RO with 1800 m^3/day capacity and the higher-quality MSF 4500 m^3/day. They incur a 4 MWe loss in power from the plant.

Much relevant experience comes from nuclear plants in Russia, Eastern Europe and Canada where district heating is a by-product.

Large-scale deployment of nuclear desalination on a commercial basis will depend primarily on economic factors. The UN's International Atomic Energy Agency (IAEA) is fostering research and collaboration on the issue, and more than 20 countries are involved.

One obvious strategy is to use power reactors which run at full capacity, but with all the electricity applied to meeting grid load when that is high and part of it to drive pumps for RO desalination when the grid demand is low.

South Korea has developed a small nuclear reactor design for cogeneration of 90 MWe of electricity and potable water at 40,000 m^3/day. The 330 MWt SMART (System integrated Modular Advanced Reactor) reactor (an integral PWR) has a long design life and needs refueling only every 3 years. The feasibility of building a cogeneration unit employing MSF desalination technology for Madura Island in Indonesia is being studied. Another concept has the SMART reactor coupled to four MED units, each with thermal-vapor compressor (MED-TVC) and producing total 40,000 m^3/day.

Spain is building 20 RO plants in the southeast to supply over 1% of the country's water.

In the UK, a 150,000-m3/day RO plant is proposed for the lower Thames estuary, utilizing brackish water.

In India plants delivering 45,000 m^3/day are envisaged, using both MSF and RO desalination technology.

China is looking at the feasibility of a nuclear seawater desalination plant in the Yantai area producing 160,000 m^3/day by MED process, using a 200 MWt reactor.

Russia has embarked on a nuclear desalination project using dual barge-mounted KLT-40 marine reactors (each 150 MWt) and Canadian RO technology to produce potable water.

Pakistan is continuing efforts to set up a demonstration desalination plant coupled to its KANUPP reactor (125 MWe PHWR) near Karachi and producing 4500 m^3/day.

Tunisia is looking at the feasibility of a cogeneration (electricity-desalination) plant in the southeast of the country, treating slightly saline groundwater.

Morocco has completed a pre-project study with China, at Tan-Tan on the Atlantic coast, using a 10 MWt heating reactor which produces 8000 m^3/day of potable water by distillation (MED).

Egypt has launched a feasibility study of a cogeneration plant for electricity and potable water at El-Dabaa, on the Mediterranean coast.

Algeria is considering a 150,000-m^3/day MSF desalination plant for its second-largest town, Oran (though nuclear power is not a prime contender for this).

A 200,000 m^3/day MSF desalination plant was designed for operation with the Bushehr nuclear power plant in Iran in 1977, but appears to have lapsed due to prolonged construction delays.

Argentina has also developed a small nuclear reactor design for cogeneration or desalination alone - the 100 MWt CAREM (an integral PWR).

Large-scale deployment of nuclear desalination on a commercial basis will depend primarily on economic factors. One obvious strategy is to use power reactors which run at full capacity, but with all the electricity applied to meeting grid load when that is high and part of it to drive pumps for reverse osmosis desalination when the grid demand is low.

There are now a large number of prospective projects, most of which have requested technical assistance from IAEA under its technical cooperation project on nuclear power and desalination. This was initiated in 1998 with a review of reactor designs intended for coupling with desalination systems as well as advanced desalination technologies. This program is expected to enable further cost reductions of nuclear desalination.

1.15 Acknowledgements

This Chapter has been prepared by Dr. Hawley, Vice-Chancellor of The World Nuclear University and Chairman of Berkeley Resources Ltd, Welsh Power Group Ltd and Lister Petter Investment Holdings Ltd and a Non-Executive Director of Colt Telecom Group SA. He has been an Advisory Director to HSBC Bank plc, Managing Director of CA Parsons and NEI plc, CEO of Nuclear Electric and British Energy, a Board Member of Rolls-Royce plc and Chairman of several companies, including Taylor Woodrow plc and until 2004 an Advisory Director to HSBC Bank plc. He is an acknowledged international expert on power generation, nuclear energy and the environment. He is the author of many books and papers on aspects of power generation and dielectrics.
Much of the information contained in this chapter has been extracted from the World Nuclear Association website Information Papers section http://www.world-nuclear.org and ably edited by Ian Hore-Lacy to whom I owe grateful thanks.
Deep thanks are given to Michelle Brider who has so ably turned all Dr Hawley's scribbling into this chapter.

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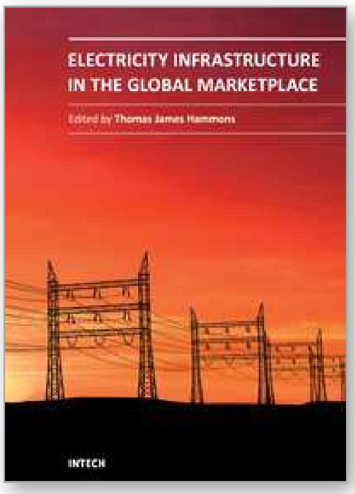

Electricity Infrastructures in the Global Marketplace
Edited by

ISBN 978-953-307-155-8
Hard cover, 802 pages
Publisher InTech
Published online 27, June, 2011
Published in print edition June, 2011

This book discusses trends in the energy industries of emerging economies in all continents. It provides the forum for dissemination and exchange of scientific and engineering information on the theoretical generic and applied areas of scientific and engineering knowledge relating to electrical power infrastructure in the global marketplace. It is a timely reference to modern deregulated energy infrastructure: challenges of restructuring electricity markets in emerging economies. The topics deal with nuclear and hydropower worldwide; biomass; energy potential of the oceans; geothermal energy; reliability; wind power; integrating renewable and dispersed electricity into the grid; electricity markets in Africa, Asia, China, Europe, India, Russia, and in South America. In addition the merits of GHG programs and markets on the electrical power industry, market mechanisms and supply adequacy in hydro-dominated countries in Latin America, energy issues under deregulated environments (including insurance issues) and the African Union and new partnerships for Africa's development is considered.

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