6

Generation Technologies through the Year 2025

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6.1 Fossil Fuels

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6.1.1 Introduction

The generation of electric power from fossil fuels has seen continuing, and in some cases dramatic technical advances over the last 20–30 years. Technology improvements in fossil fuel combustion have been driven largely by the need to reduce emissions, by the need to conserve fossil fuel resources, and by the economics of the competitive marketplace. The importance of fossil fuel-fired electric generation to the world is undeniable—more than 70% of all power in the U.S. is fossil fuel-based and worldwide the percentage is higher, and growing. Today most large power plants worldwide burn coal though many generating companies are adding natural gas plants, particularly where the cost of gas-fired generation, and the long-term supply of gas, appear favorable. This chapter reviews the current status and likely future deployment of competing generation technologies based on fossil fuels.

It is likely, particularly in the developed world, that gas turbine-based plants will continue to be added to the new generation market in the immediate future. The most advanced combustion turbines now achieve more than 40% lower heating value (LHV) efficiency in simple cycle mode and greater than 50% efficiency

in combined cycle mode. In addition, combustion turbine/combined cycle (CT/CC) plants offer siting flexibility, swift construction schedules, and capital costs between $400/kW and $800/kW. These advantages, coupled with adequate natural gas supplies (though new wells and pipelines will be needed in the United States) and the assurance, in the longer term, of coal gasification backup, have made this technology one important choice for green field and repowered plants in the U.S. and in Europe.

However, fossil steam pulverized coal (PC) plants are dominant in the expanding nations of the East such as China and India. In fact for the developing world there is good reason why the coal-fired power plant may still be the primary choice for many generation companies. Fuel is plentiful and inexpensive, and sulfur dioxide scrubbers have proved to be more reliable and effective than early plants indicated. In fact up to 99% SO_2 removal efficiency is now possible. Removal of nitrogen oxides is also well advanced with over 95% removal possible using selective catalytic reduction (SCR). Ways to remove mercury are currently under study, and the issue of carbon dioxide control and sequestration from fossil plants is receiving renewed attention as ways to control global warming are pursued. Combustion of coal currently occurs in three basic forms, direct combustion of PC, combustion of coal in a suspended bed of coal and inert matter, and coal gasification.

The *pulverized coal plant*, the most common form of coal combustion, has the capability for much improved efficiency even with full flue gas desulfurization (FGD), ferritic materials technology now having advanced to the point where higher steam pressures and temperatures are possible. Advanced supercritical PC plants are moving ahead commercially, particularly in Japan and Europe. Even higher steam conditions for PC plants, perhaps using nickel-based superalloys, are under study.

Worldwide the application of *atmospheric fluidized bed combustion* (AFBC) plants has increased, and such plants offer reductions in both SO_2 and NO_x while permitting the efficient combustion of vast deposits of low-rank fuels such as lignites. Since the early 1990s, AFBC boiler technology has become established worldwide as a mature, reliable technology for the generation of steam and electric power—with its advantage of in-furnace SO_2 capture with limestone. In fact, the major impetus in the widespread deployment of this relatively new boiler technology since the mid-1980s has been its resemblance to a conventional boiler with the added capability for in-situ SO_2 capture, which could eliminate or reduce the need for flue-gas desulfurization.

Coal gasification power plants are operating at the 250–300 MW level. Much of the impetus came from the U.S. DOE clean coal program where two gasification projects are in successful commercial service. Large gasification plants for power are also in operation in Europe. Gasification with combined cycle operation not only leads to minimum atmospheric (SO_2 and NO_x) and solid emissions, but also provides an opportunity to take advantage of new gas turbine advances. With the rapid advances now being introduced in combustion turbine technology, the coal gasification option is seen as a leading candidate for new plant construction within the first half of the 21st century. Several new gasification plants are planned for the U.S. within the 2010–2020 time frame.

6.1.2 Fuels for Electric Power Generation in the United States

The Energy Information Administration lists more than 500 GW of fossil steam units in the U.S. Coal-fired units dominate with 1400 units capable of generating over 300 GW. All told, fossil-steam plants generate more than 70% of all electric energy in the country, Figure 6.2, and these aging units, on average more than 35 years old, will remain the foundation of the power industry for the immediate future, and certainly through the next 10 years.

The U.S. electric power industry burns about $30 billion worth of fossil fuels each year, accounting for 70%–80% of the operating costs of fossil-fired plants. Coal dominates and recent changes to the fuel mixes include:

- A mix of eastern high-sulfur coal with low-sulfur, low-cost western coals, often from Powder River Basin (PRB) deposits in Montana and Wyoming. Compared with eastern bituminous coals, PRB coals have LHV, sulfur and ash, but higher moisture content and finer size.

- A mix of 10%–20% gas with coal in a boiler designed for coal firing.
- Orimulsion, a bitumen-in-water emulsion produced only from the Orinoco Basin in Venezuela. This fuel is relatively high in sulfur and vanadium. Power plants that use this fuel need to add scrubbers.
- A mix of coal with petroleum coke, a by-product of refining, whose cost is currently low but whose sulfur content is high.

6.1.2.1 Coal as a Fuel for Electric Power

Coal is the altered remains of prehistoric vegetation that originally accumulated as plant material in swamps and peat bogs. The accumulation of silt and other sediments, together with movements in the earth's crust (tectonic movements) buried these swamps and peat bogs, often to great depth.

With burial, the plant material was subjected to elevated temperatures and pressures, which caused physical and chemical changes in the vegetation, transforming it into coal. Initially the peat, the precursor of coal, was converted into lignite or brown coal—coal-types with low organic maturity. Over time, the continuing effects of temperature and pressure produced additional changes in the lignite, progressively increasing its maturity and transforming it into what is known as subbituminous coals. As this process continued, further chemical and physical changes occurred until these coals became harder and more mature, at which point they are classified as bituminous coals. Under the right conditions, the progressive increase in the organic maturity continued, ultimately to form anthracite.

The degree of metamorphism or coalification undergone by a coal, as it matures from peat to anthracite, has an important bearing on its physical and chemical properties, and is referred to as the "rank" of the coal. Low-rank coals, such as lignite and subbituminous coals, are typically softer, friable materials with a dull, earthy appearance; they are characterized by high-moisture levels and a low-carbon content, and hence a low-energy content. Higher rank coals are typically harder and stronger and often have a black vitreous luster. Increasing rank is accompanied by a rise in the carbon and energy contents and a decrease in the moisture content of the coal. Anthracite is at the top of the rank scale and has a correspondingly higher carbon and energy content and a lower level of moisture.

Large coal deposits only started to be formed after the evolution of land plants in the Devonian period, some 400 million years ago. Significant accumulations of coal occurred during the Carboniferous period (350–280 million years ago) in the Northern Hemisphere, the Carboniferous/Permian period (350–225 million years ago) in the Southern Hemisphere and, more recently, the late Cretaceous period to early Tertiary era (approximately 100–15 million years ago) in areas as diverse as the United States, South America, Indonesia and New Zealand. Of all the fossil fuels, coal is the most plentiful in the world. It is geographically dispersed, being spread over 100 countries and all continents. Coal reserves have been identified that confirm over 200 years of resource availability. The percent of the world reserves categorized by type and use is shown in Figure 6.1 below. Almost one half (48%) of the world's coal reserves is made up of lignite and subbituminous coals, and these coals are used primarily for power generation.

6.1.3 Clean Coal Technology Development

At an increasing rate in the last few years, innovations have been developed and tested aimed at reducing emissions through improved combustion and environmental control in the near term, and in the longer term by fundamental changes in the way coal is preprocessed before converting its chemical energy to electricity. Such technologies are referred to as "Clean Coal Technologies" described by a family of precombustion, combustion/conversion, and post-combustion technologies. They are designed to provide the coal user with added technical capabilities and flexibility, and the world with an opportunity to exploit our most abundant fossil source. They can be categorized as:

- Precombustion, where sulfur and other impurities are removed from the fuel before it is burned.
- Combustion, where techniques to prevent pollutant emissions are applied in the boiler while the coal burns.

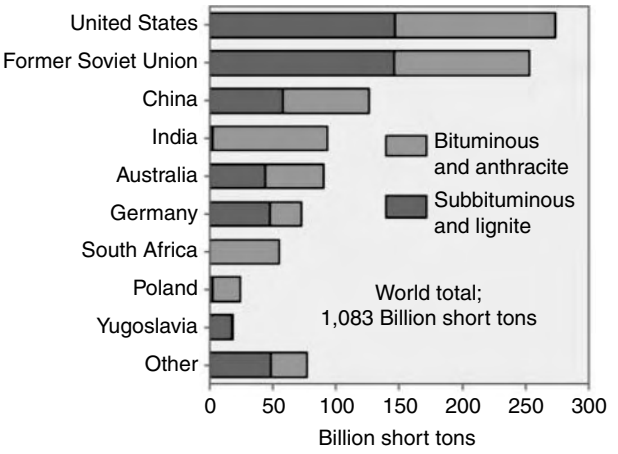

FIGURE 6.1 Worldwide coal reserves. (From Energy Information Administration, International Engergy Annual 2001, DOE/EIA-0219(2001), (Washington, DC, February 2003.) Table 8.2. www.eia.doe.gov/iea/.)

- Post-combustion, where the flue gas released from the boiler is treated to reduce its content of pollutants.
- Conversion, where coal, rather than being burned, is changed into a gas or liquid that can be cleaned and used as a fuel.

6.1.3.1 Coal Cleaning

Cleaning of coal to remove sulfur and ash is well established in the U.S. with more than 400 operating plants, mostly at the mine. Coal cleaning removes primarily pyritic sulfur (up to 70% SO_2 reduction is possible) and in the process increases the heating value of the coal, typically about 10% but occasionally 30% or higher. The removal of organic sulfur, chemically part of the coal matrix, is more difficult, but may be possible using microorganisms or through chemical methods and research is underway [Couch, 1991]. Heavy metal trace elements can be removed also, conventional cleaning removing (typically) 30%–80% of arsenic, mercury, lead, nickel, antimony, selenium and chromium.

6.1.4 New Generation Needs

The exploding global demand for electricity, particularly in the developing world, implies practical electric generating options, and significant increases in efficiency throughout the entire energy chain. A strong portfolio of advanced power generation options would include fossil, renewable and nuclear, all essential to meet these growth requirements, both domestically and globally.

Over the next 20 years, the developed world will meet the need for additional capacity largely with coal and gas-fired plants, while the developing world will continue to rely on indigenous resources, particularly coal in the case of China and India. In the U.S. the likely balance between coal- and gas-fired new generation is a topic of significant debate as discussed above. While gas-fired additions currently predominate, the relative fuel prices (coal to gas), and the timing of additional emissions requirements will ultimately determine the preferred choice between these two fossil fuel options. In the longer term noncarbon sources will begin to supplant both.

6.1.4.1 Central Station Options for New Generation

Coal and gas fuels are expected to continue to dominate U.S. central stations in the next decade, with gas-fired combined cycles supplanting several older fossil steam stations. Timing is still a question though. At the end of 2006, more than 150 new coal-fired plants were being planned in 42 states. Based on the expected mix of coal, oil, gas, nuclear, and renewables through the year 2015 the U.S. central station generation options for fossil fuels may be described as follows:

1. Coal, oil, and gas-fired plants of conventional design mostly Rankine cycles.
   These represent the majority of plants currently in operation. On average they are 35-years old, many (more than 100,000 MW) equipped with SO_2 scrubbers, with NO_x control additions, and with other environmental upgrades. Yet they provide the bulk of our electricity needs, are extremely reliable, and are increasingly in demand as evidenced by an average capacity factor above 70% in the early years of this century.
2. Repowered plants, based on gas-firing and combined cycle operation.
   Many of the gas-fired steam plants that have changed hands are targeted for repowering. That is combustion turbines will be added to provide exhaust heat for producing steam for the existing steam turbines. This combination of gas and steam turbine cycles adds MW, reduces emissions, and improves efficiencies 5% or more.
3. New combined cycles based on gas-firing, and on coal-firing with gasification, utilizing advanced gas and steam turbine technology.
   Gas-fired combined cycles have been the new central plants of choice moving into the new century. Though relatively few are in operation today a considerable number had been planned. However massive deployment of these plants in the future raises questions of gas and gas pipeline availability, and gas prices. Any potential retreat from coal though could have serious future energy consequences.
4. Coal-fired Rankine cycles with advanced steam conditions.
   Advancing steam temperatures and pressures in pulverized steam plants greatly improves overall efficiency. Such ultrasupercritical cycles are already in operation outside the U.S. Advancing steam temperatures to 700°C from current levels of about 590°C enhances efficiency and reduces emissions. When used in coal combined cycles, and with temperatures increased to 750°C or beyond, a coal plant beyond 55% efficiency can be attained. Significant challenges still exist in materials technology.
5. Integrated coal gasification fuel cells perhaps combined with gas turbines with efficiencies of 60% or more.
   The fuel cell is an exciting advance that will change the energy picture in the long term. In the shorter term, and in small sizes, advances are being made in both mobile and stationary applications. If the fuel cell can be used as a combuster for a gas turbine, efficiencies can be raised above 60%. Clearly this is a power source of great promise for the second half of this century.

6.1.5 Pulverized Coal Power Plants

Today more than 50% of total U.S. electricity generation is supplied by about 1000 large coal-fired units, with a concentration in the 500–600 MW unit sizes. These plants are typically aging, 35-plus years old on average, and compete with other regional coal-fired plants at the cost margin. Fuel cost is frequently the determining factor in assuring cost-competitiveness. For this reason—and also to reduce the price of emissions control—many eastern coal-fired units now burn a mixture of eastern coal and coal from the low-sulfur deposits of the Powder River Basin of Wyoming and Montana. Average capacity factor for coal-fired units is about 70%, up from 60% about 10 years ago. Equivalent availabilities average about 85%, close to a 15-year high, and are achieved despite an aging fleet and reduced staffing levels. By the early 2000s sulfur dioxide and nitrogen oxides were down more than 40 and 20%, respectively, since the early 1980s,

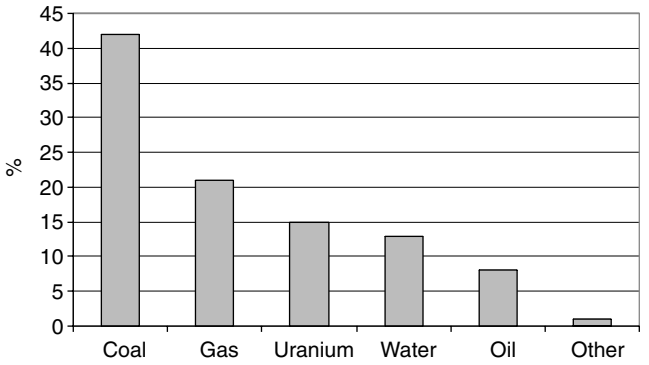

FIGURE 6.2 US installed capacity by fuel percentage.

even though electricity production climbed about 40% in that period. These performance parameters reflect the progressive advances in our understanding of the combustion of coal (Figure 6.2 through Figure 6.4).

6.1.5.1 Cycle Selection

The selection of a supercritical versus a subcritical cycle for a fossil-steam unit is dependent on many site specific factors including fuel cost, emissions regulations, capital cost, load factor, duty, local labor rates, and perceived reliability and availability. In fact the use of subcritical cycles for the limited number of fossil-steam plants that have been build in the U.S. in the last 20 years has been mainly due to relatively low fuel costs which eliminated the cost justification for the somewhat higher capital costs of the higher efficiency cycles. However, in some international markets where fuel cost is a higher fraction of the total cost, the higher efficiency cycles offer a favorable cost-of-electricity comparison and provide lower emissions compared to a subcritical plant. This is true in both Europe (see Figure 6.5) and Japan. Supercritical

FIGURE 6.3 US coal basins.

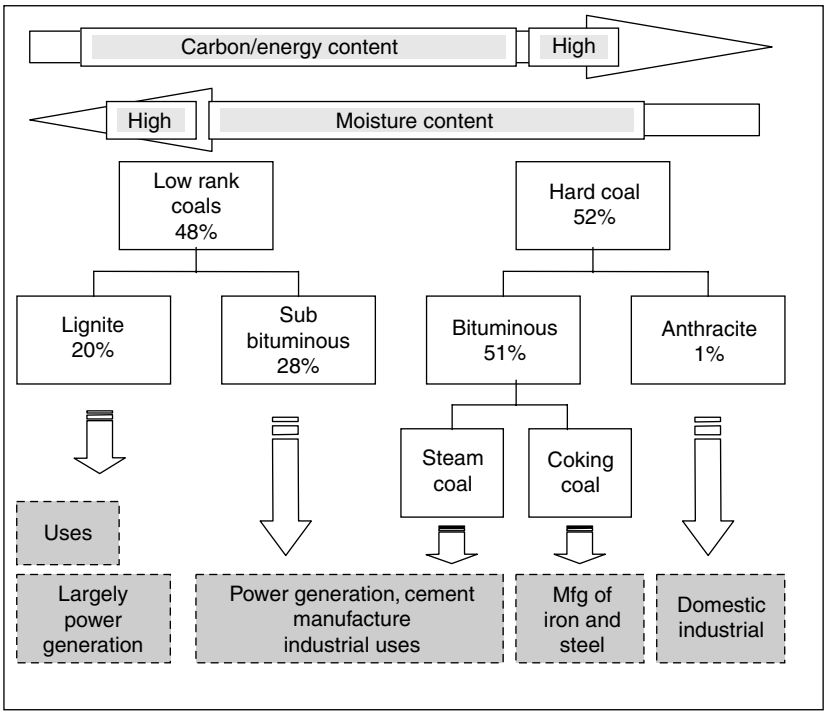

FIGURE 6.4 World use of coal.

cycles have recently been selected for several new fossil plants in the U.S. and the reduction of CO_2 emissions for the supercritical cycle could be a deciding factor as ways are sought to reduce global warming concerns.

6.1.5.2 Environmental Controls for Fossil-Steam Plants

Of all the hurdles facing owners of generating plants, perhaps none is greater than preparing units for meeting environmental limits at minimum cost. In the U.S. by the year 2000, more than 200 SO_2 scrubbers had been installed on more than 100,000 MW of fossil-steam capacity, valuable additions that will permit plants to operate in compliance for many more years. Typically a 450 MW coal-fired plant will emit 75 tn. of SO_2 per day without a scrubber and perhaps 8 tn. per day with a 90% FGD system in place, a difference that can be measured in terms of the market for SO_2 credits. And for NO_x, where most current control activities are focused the same plant might emit 10–35 tn. per day. NO_x control options range from burner optimization to the use of SCR. As for carbon dioxide, the above plant emits about 9000 tn./day at a plant efficiency of 38% which translates to 2452 tn. of carbon. Such emissions are of increasing concern and potential future carbon taxes must be considered. A combined cycle gas plant, for comparison, emits about half of this amount, per MWh, due to the higher plant efficiency and lower carbon content of natural gas. The removal of mercury from the products of fossil fuel combustion, will be an increasingly important task for fossil plant operators. Current plant testing for mercury control is exploring several promising options.

An overall perspective of emissions control technologies from a modern pulverized coal-fired plant is shown in Figure 6.6.

6.1.5.3 NO_x Control

The broad span of options for control may be categorized into combustion and post-combustion methods (Figure 6.7). Low NO_x burners, especially when combined with staging of over-fired air, are currently deployed most often. Combustion optimization techniques may offer a low cost alternative to hardware

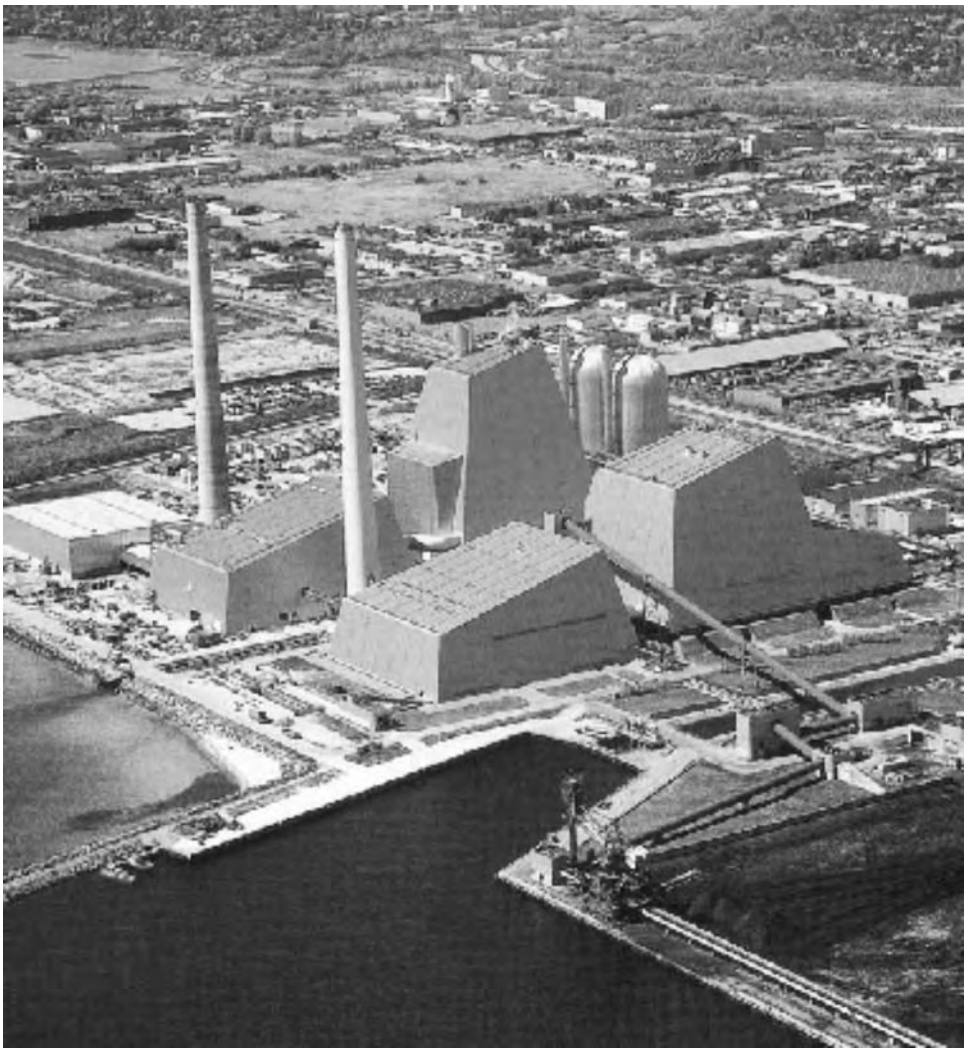

FIGURE 6.5 Modern supercritical coal-fired power plant.

options, particularly where modest reductions up to 30% are needed, and many units are now operating with some form of optimization of air and fuel flows, perhaps utilizing advanced flame diagnostics or software based on neural networks to do the optimizing.

6.1.5.4 Post-Combustion Options for NO_x Control

Selective catalytic reduction is used widely in Europe (especially Germany where it is installed on more than 30,000 MW of coal-fired boilers) and in Japan, and increasingly in the U.S. In an SCR, ammonia is injected into the boiler exhaust gases ahead of the catalyst bank (at about 550°F–750°F). NO_x and NH_3 then react to produce nitrogen and water, the chemical reactions being:

$$4NO + 4NH_3 + O_2 \rightarrow 4N_2 + 6H_2O$$

$$6NO_2 + 8NH_3 \rightarrow 7N_2 + 12H_2O$$

$$2NO_2 + 4NH_3 + O_2 \rightarrow 3N_2 + 6H_2O$$

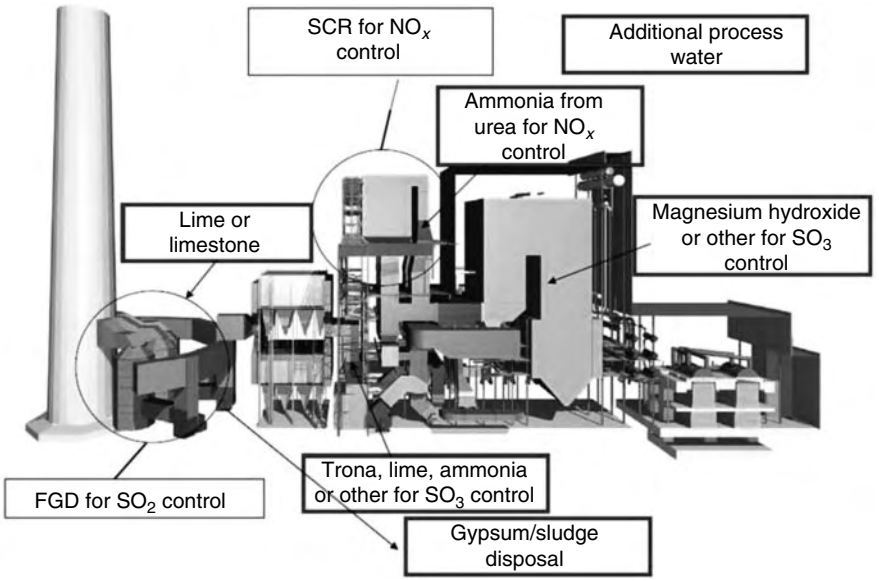

FIGURE 6.6 Emission controls from a modern pulverized coal-fired power plant.

The reaction can result in a potential NO_x removal capability of more than 90%. Retrofit installation of an SCR system can require considerable space, although the reactor can be placed inside the original ductwork if NO_x reduction levels are modest (Figure 6.8).

Selective non-catalytic reduction (SNCR) is a promising lower capital cost alternative to SCR ($10/kW versus more than $50/kW), but with lower performance (20%–35% reduction compared with 50 to as high as 80% for SCR). In SNCR, the injection of a reagent like urea or ammonia into the upper part of the furnace converts NO_x from combustion into nitrogen, this conversion being a direct function of furnace temperature and reagent injection rate.

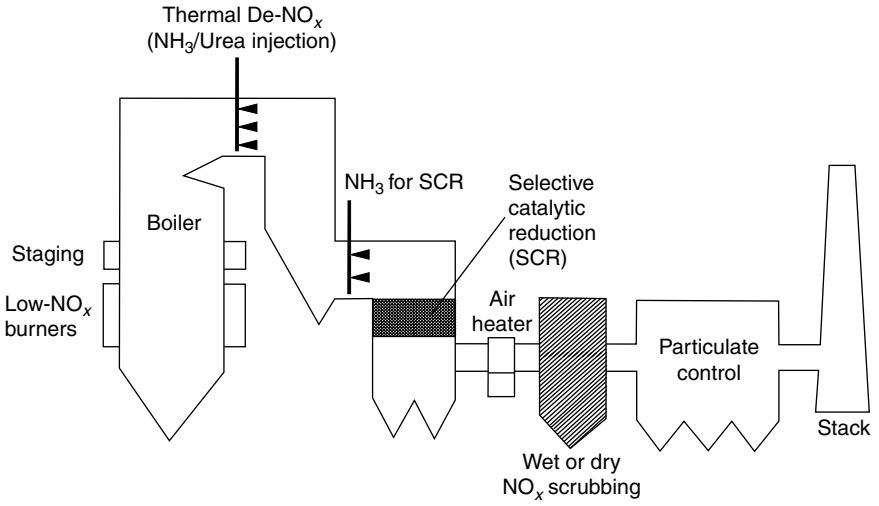

FIGURE 6.7 NO_x control options for fossil boilers.

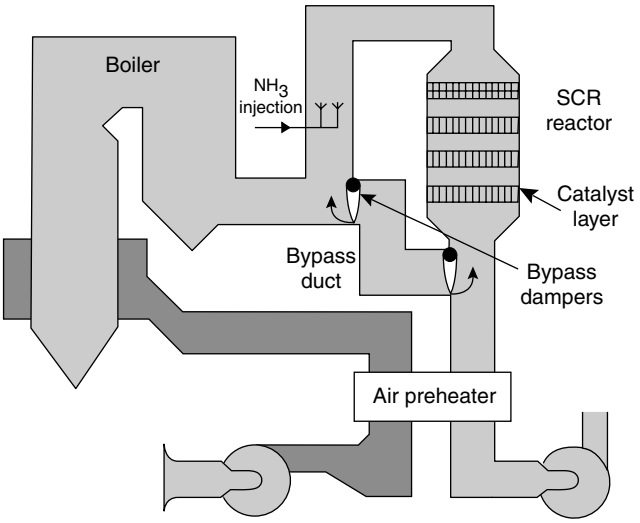

FIGURE 6.8 Selective catalytic reduction system in the boiler flue gas.

6.1.5.5 Sulfur Dioxide Control

The need for removal of sulfur dioxide from flue gases has led to the installation of FGD units in much of the coal fired capacity in the U.S. (Figure 6.9).

More than 200 coal-fired units in the United States use wet or dry scrubbing to remove sulfur. Of these, the majority use wet lime or limestone scrubbers with perhaps between 20 and 30 using dry scrubbing where the sulfur content of the coal is generally lower. Lime may be used alone or in combination with magnesium, carbides, or with alkaline fly ash if the boiler burns subbituminous coals or lignites.

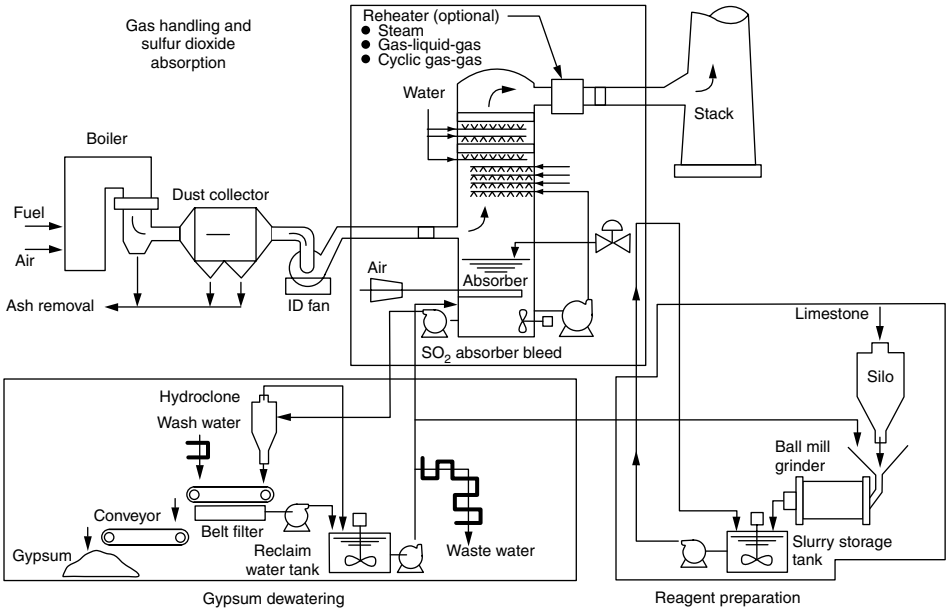

FIGURE 6.9 Wet limestone flue gas desulfurization system.

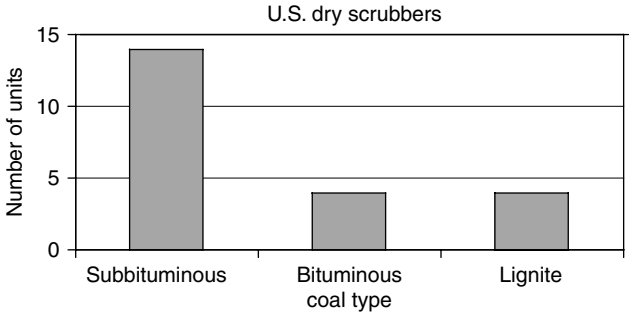

FIGURE 6.10 Use of spray dryers is mainly, though not exclusively, confined to western subbituminous, low sulfur coal-burning units.

The most common wet systems bring lime or limestone slurries into contact with flue gases in a spray tower. SO_2 in the flue gas is absorbed in the slurry and collected in a reaction tank, where it precipitates to produce calcium sulfite or calcium sulfate (gypsum) crystals. A portion of the slurry is then pumped to a thickener where the crystals settle out before going to a filter for final dewatering. Calcium sulfite and/or sulfate are typically disposed of in a landfill. Flue gas desulfurization gypsum can be sold for use in wallboard, cement or agricultural products.

Dry FGD systems employ calcium or sodium reagents that are either injected as dry powders or in spray drying systems as slurries that dry on contact with flue gas. Dry injection systems are most economical for space-constrained sites or applications that require only moderate emissions reductions. Spray drying systems, which can achieve higher SO_2 removal efficiencies, have mainly been applied at units burning low-to-medium-sulfur coals (Figure 6.10).

6.1.5.6 Mercury Control

Various approaches to mercury removal from power plant flue gases are under development though much mercury can be removed through existing air pollution controls for particulate and SO_2 (Figure 6.11).

The removal of mercury from coal-fired units can be accomplished in several ways. Coal cleaning before combustion can remove some mercury and other heavy metals. After combustion, the injection of a sorbent, such as a activated carbon can be very effective. Existing ESPs and SO_2 scubbers can capture from 20 to 60% of mercury. Catalysts and certain chemicals can be injected that oxidize elemental mercury to

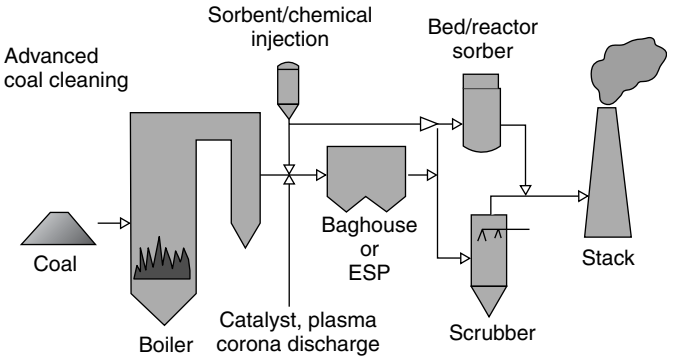

FIGURE 6.11 Options for the removal of mercury.

enhance scrubber capture. Fixed beds, coated with materials such as gold, can form amalgams with mercury.

6.1.6 Fluidized Bed Power Plants

Introduced nearly 30 years ago the atmospheric fluidized bed combustion (AFBC) boiler (Figure 6.12) has found growing application for power generation. From the first FBC boiler, generating 5000 lb/h of steam in 1967, the technology has matured to the 350 MW size units available today.

In the bubbling bed version of the AFBC, the fuel and inert matter, together with limestone or dolomite for SO_2 capture, is suspended through the action of fluidizing air, which flows at a velocity of 3–8 ft./s in essentially a one-pass system. Circulating fluid beds (CFB) differ from bubbling beds in that much of the bed material passes through a cyclone separator before being circulated back to the boiler. In-bed tubes are generally not used for CFB units permitting a much higher fluidizing velocity of 16–26 ft./s. Since the early AFBC designs, attention has been directed towards increasing unit efficiency, and reheat designs are now usual in large units. When SO_2 capture is important a key parameter is the ratio of calcium in the limestone to sulfur in coal. Typical calcium to sulfur ratios for 90% SO_2 reduction are in the range of 3.0–3.5 for bubbling beds and 2.0–2.5 for circulating beds. NO_x levels in AFBCs are inherently low and nominally less than 0.2 lb/MMBtu. It is important to note that for CFBs, boiler efficiencies can be as high as a PC unit. In fact designs now exist for AFBCs with supercritical steam conditions, with prospects for cycles up to 4500 psia, 1100°F with double reheat.

In North America more than 160 units now generate in excess of 9000 MW. Burning coal in a suspended bed with limestone or dolomite permits effective capture of sulfur and fuel flexibility allows a broad range of opportunity fuels. These fuels might include coal wastes (culm from anthracite, gob from bituminous coal), peat, petroleum coke, and a wide range of coals from bituminous to lignite. A low (1500°F) combustion temperature leads to low NO_x formation. Several large size FBC plants are now under construction, including one in Europe with supercritical steam conditions.

Examples of large size generating FBC plants in the Americas are shown in Table 6.1.

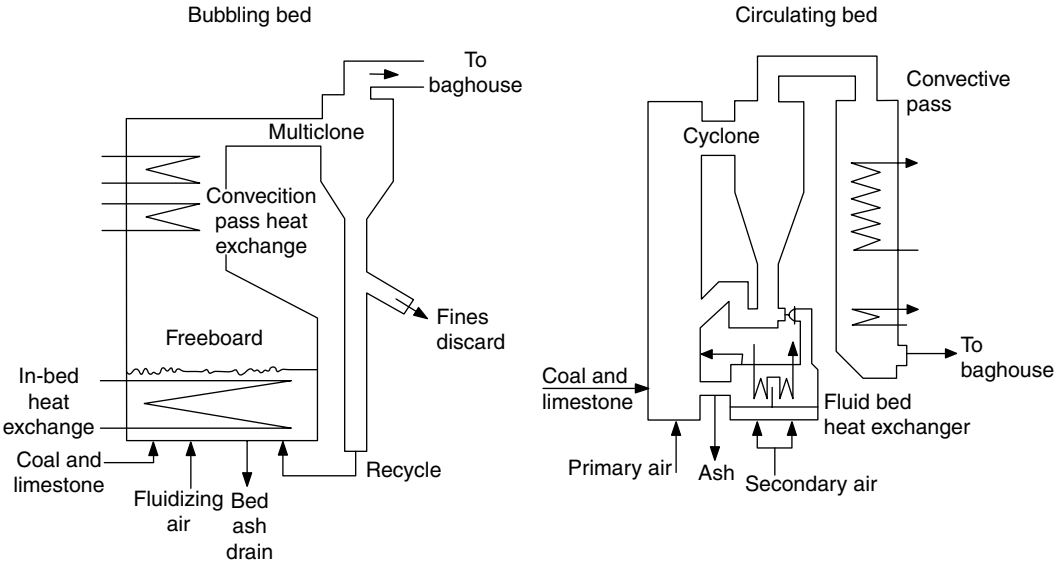

FIGURE 6.12 The primary types of atmospheric fluidized beds are bubbling beds and circulating beds.

TABLE 6.1 U.S., Canadian, and Latin American CFB Units Larger than 75 MW

Plant/Location (Vendor)	Start Up	Capacity, MW (Net)	Fuels
Tri-State Generation and Transmission/Colorado	1987	1×100	Bit. coal
AES Shady Point/Oklahoma	1989	4×75	Bit. coal
AES Thames/Connecticut	1989	2×90	Bit. coal
Schuylkill Energy/Pennsylvania	1989	1×80	Culm
ACE Cogeneration/California	1990	1×97	Low-S bit. Coal
Texas-New Mexico Power/Texas	1990	2×150	Lignite
AES Barbers Point/Hawaii	1992	2×90	Bit. coal
Nelson Industrial Steam Co. (NISCO)/Louisiana	1992	2×110	Coke
Cedar Bay Generating Co./Florida	1993	3×90	Bit. coal
Nova Scotia Power/Nova Scotia	1993	1×165	30% bit. coal and 70% coke
Colver Power/Pennsylvania	1995	1×105	Gob
Northampton Generating Co./Pennsylvania	1995	1×112	Culm
ADM/Illinois	1996/2000	2×132	Bit. coal and up to 5% TDF
ADM/Iowa	2000	1×132	Bit. coal
AES Warrior Run/Maryland	1999	1×180	Bit. coal
Choctaw Generation—the Red Hills project/Mississippi	2001	2×220	Lignite
Bay Shore Power—First Energy/Ohio	2001	1×180	Coke
AES Puerto Rico/Puerto Rico	2002	2×227	Bit. coal
JEA/Florida	2002	2×265	Bit. coal and coke
Southern Illinois Power Cooperative/Illinois	2002	1×113	Waste bit. coal
Termoelectrica del Golfo/Mexico	2002	2×115	Coke
Termoelectrica de Penoles/Mexico	2003	2×115	Coke
Reliant Energy Seward Station/Pennsylvania (ALSTOM)	2004	2×260	Gob and bit. coal
East Kentucky Power Cooperative/Kentucky	2004	1×268	Unwashed high-sulfur bit. coals
Figueira/Brazil	2004	1×128	Bit. coal

6.1.7 Coal Gasification Power Plants

There are currently two coal-based integrated gasification combined cycle (IGCC) commercial-sized demonstration plants operating in the U.S. and two in Europe (Table 6.2). The U.S. projects were supported under the U.S. Department of Energy's (DOE) Clean Coal Technology (CCT) demonstration program.

The 262 MW Wabash River IGCC repowering project in Indiana started up in October 1995 and uses the E-GAS™ (formerly Destec) gasification technology. The 250 MW Tampa Electric Company (TEC) IGCC project in Florida started up in September 1996 and is based on the Texaco (now GE) gasification

TABLE 6.2 Coal-Based, Commercial-Size IGCC Plants

	Gasification Technology	Plant Size, MW	Startup Date
Wabash River, Indiana, U.S.A.	Destec	262	10/95
Tampa Electric, Florida, U.S.A.	Texaco	250	9/96
SEP/Demkolec, Buggenum, The Netherlands	Shell	253	Early 1994
ELCOGAS, Puertollano, Spain	Krupp-Uhde Prenflo	310	12/97 on coal

FIGURE 6.13 The 250-MW coal gasification plant at the polk plant of Tampa Electric.

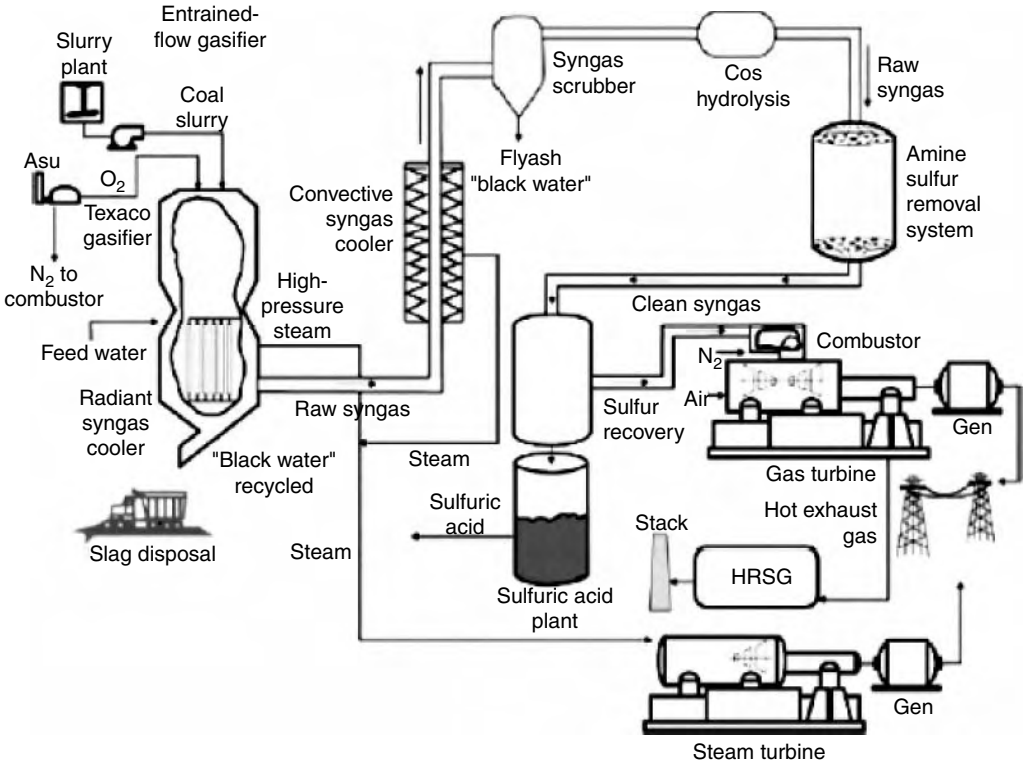

FIGURE 6.14 Tampa electric polk gasification plant.

technology (Figure 6.13). The first of the European IGCC plants the SEP/Demkolec project at Buggenum, the Netherlands, uses the Shell gasification technology and started operations in early 1994. The second European project, the ELCOGAS project in Puertollano, Spain, which uses the Prenflo gasification technology, started coal-based operations in December 1997. New commercial IGCC plants are now in planning in the U.S. and are scheduled for operation within the next 10 years.

6.1.7.1 Tampa Electric Integrated Gasification Combined Cycle Plant

In the DOE Clean Coal project at Tampa Electric coal/water slurry and oxygen are reacted at high temperature and pressure to produce approximately 245 Btu/SCF syngas (LHV) in a Texaco gasifier (Figure 6.14). Molten ash flows out of the bottom of the gasifier into a water-filled sump where it forms a solid slag. The syngas moves from the gasifier to a radiant syngas cooler and a convective syngas cooler (CSC), which cool the syngas while generating high-pressure steam. The cooled gases flow to a water-wash syngas scrubber for particulate removal. Next, a hydrolysis reactor converts carbonyl sulfide (COS) in the raw syngas to hydrogen sulfide (H_2S) that is more easily removed. The raw syngas is then further cooled before entering a conventional amine sulfur removal system and sulfuric acid plant (SAP). The cleaned gases are then reheated and routed to a combined cycle system for power generation. A GE MS 7001FA gas turbine generates 192 MWe. Thermal NO_x is controlled to 0.7 lb/MWh by injecting nitrogen. A steam turbine uses steam produced by cooling the syngas and superheated with the gas turbine exhaust gases in the HRSG to produce an additional 123 MWe. The air separation unit consumes 55 MW and auxiliaries require 10 MW, resulting in 250 MWe net power to the grid. The plant heat rate is 9650 Btu/kWh (HHV).

6.1.7.2 Wabash River Integrated Gasification Combined Cycle Plant

A second U.S. coal gasification plant is at the Cinergy Wabash River plant using a different gasification process from the Tampa Electric approach (Figure 6.15). The Destec, now E-Gas Technology™, process

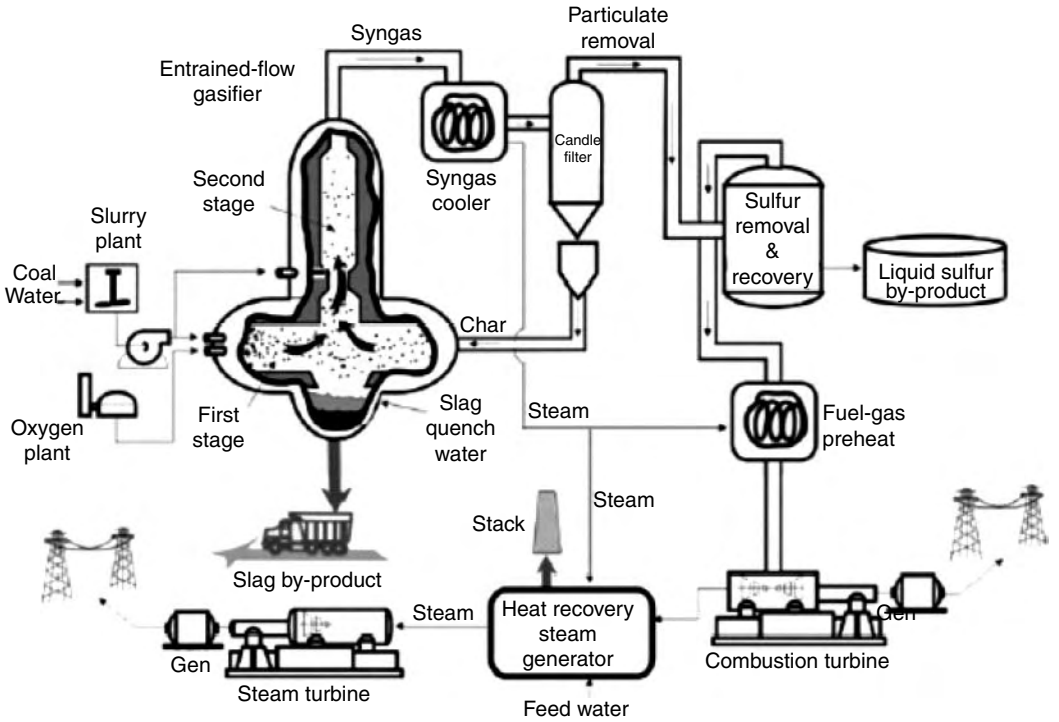

FIGURE 6.15 Sketch of a typical PWR power plant. (From World Nuclear Association, http://www.world-nuclear.org.)

features an oxygen-blown, continuous-slagging, two-stage, entrained flow gasifier. Coal is slurried, combined with 95% pure oxygen, and injected into the first stage of the gasifier, which operates at 2600°F/400 psig. In the first stage, the coal slurry undergoes a partial oxidation reaction at temperatures high enough to bring the coal's ash above its melting point. The fluid ash falls through a tap hole at the bottom of the first stage into a water quench, forming an inert vitreous slag. The syngas flows to the second stage, where additional coal slurry is injected. This coal is pyrolyzed in an endothermic reaction with the hot syngas to enhance syngas heating value and improve efficiency. The syngas then flows to the syngas cooler, essentially a fire tube steam generator, to produce high-pressure saturated steam. After cooling in the syngas cooler, particulates are removed in a hot/dry filter and recycled to the gasifier. The syngas is further cooled in a series of heat exchangers. The syngas is water-scrubbed to remove chlorides and passed through a catalyst that hydrolyzes carbonyl sulfide into hydrogen sulfide. Hydrogen sulfide is removed in the acid gas removal system using absorber/stripper columns. A Claus unit is used to produce elemental sulfur as a salable by-product. The "sweet" gas is then moisturized, preheated, and piped to the power block. The power block consists of a single 192-MWe General Electric MS 7001FA (Frame 7 FA) gas turbine, a Foster Wheeler single-drum heat recovery steam generator with reheat, and a 1952 vintage Westinghouse reheat steam turbine.

TABLE 6.3 Performance Comparisons of Four Clean Coal Technologies

Technology	Units	PC-Sub	PC-SC	AFBC	IGCC
Potential capacity	MW/unit	250–700	350–800	30–300	300–600
Fuel range/diversity		All grades lignite to anthracite	All grades lignite to anthracite	All grades biomass and wastes	All grades bituminous preferred
Build time	Years	3	3	3	3
Manning levels					
Operators	Number/MW	0.16	0.16	0.18	0.18
Total staff	Number/MW	0.31	0.31	0.46	0.33
Expected availability					
Planned outage	%	11.1	11.1	5.7	4.7
Forced outage	%	3.7	3.9	4.1	10.1
Equiv. availability	%	85.7	85.4	90.4	85.7
Expected efficiencies	% (HHV)	34.4–35.7	36.4–37.7	34.6–35.6	39.3–41.1
Expected heat rate	kJ/kWh (HHV)	9570–9930	9050–9380	9600–9870	8310–8680
Emission ranges					
SO_2	Kg/MW-h	0.66–0.68	0.62–0.65	0.66–0.68	0.04–0.22
NO_x	Kg/MW-h	0.66–0.68	0.62–0.65	0.66–0.68	0.23–0.24
CO_2	Kg/MW-h	831–862	786–815	834–857	723–754
Particulates	Kg/MW-h	0.10	0.10	0.10	0.01
Solid waste (Total)	Kg/MW-h	62–113	59–107	55–141	42–44
Ash	Kg/MW-h	45–48	45–48	47–44	42–44
Spent sorbent	Kg/MW-h	12–66	11–62	11–94	0
Cooling water requirements	Cu.M/h/MW	244	236	249	185
Flexibility load range	%	30–100	30–100	30–100 per unit	50–100 per unit 25–100 for 2 trains
Total plant cost	$/kW	1040–1080	1060–1090	1030–1060	1150–1190
O&M cost					
Fixed	$/Kw-yr.	23.9–24.3	23.9–24.4	22.3–22.7	25.6–26.3
Variable	$/MW-h	2–3.1	2–3	2.1–4.4	1.7–1.5
Cost of electricity	$/MW-h	36.2–38.4	36–38	36–39.2	37.1–38.2

6.1.8 Comparison of Clean Coal Options for the Next 10 Years

An overall comparison of the performance and cost of four clean coal technologies discussed above: PC subcritical and supercritical, fluidized bed, and integrated gasification combined cycle, is shown in Table 6.3. It should be recognized that comparisons of this type do not capture the variations in available fuel, site conditions, and operational and business needs that often determine the ultimate choice of technology.

6.1.9 Carbon Dioxide Control

Plant emissions of CO_2 shown in Table 6.3 are mainly a function of the plant heat rate and efficiency. For the combustion technologies (PC, AFBC) there are some minor additional contributions associated with the use of limestone.

It has been shown that if CO_2 removal was ever required that it is much less expensive to remove CO_2 syngas under pressure prior to combustion rather than from the boiler exit ducts of a PC plant where the CO_2 is in more dilute concentrations. Generally, when CO_2 control is deployed, the cost of electricity grows significantly. This is particularly true for PC plants where the CO_2 is at high volume and at atmospheric pressure. In contrast the CO_2 is compressed in gasification plants to less than 1% of the flue gas volume of a PC plant (as are all the gas emissions) and this requires less energy for removal. See Figure 6.16 for rough examples of electricity cost differences. Figure 6.16 assumes plant sizes of about 460 MW. Capacity factor (CF) is 80% except for the second natural gas plant which is 40% CF. Of course these costs also depend on design factors such as coal quality, steam conditions, and coal and gas prices.

Another important question when considering CO_2 control is the impact the additional control equipment will have on the power output and efficiency of the power plant. This is particularly significant when considering the retrofit of control equipment to an existing unit. There are more than 1500 large coal-fired plants in the U.S. that fall into this category. Estimates of the impact of 100% CO_2 control have been made for supercritical PC cycles, natural gas combined cycles, and integrated gasification combined cycles. In this respect the IGCC plant has very significant advantages as shown in Table 6.4. As of the end of 2006, these estimates are still under review as new technology for CO_2 removal suggests that significant reductions in these penalties may be possible.

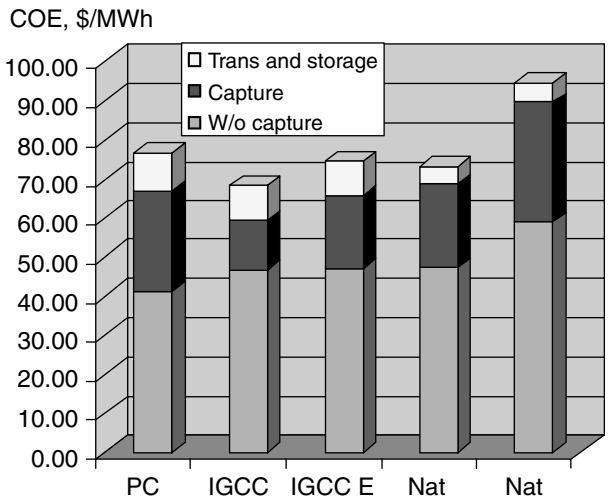

FIGURE 6.16 Cost of electricity with CO_2 removal and storage from fossil-fuel power plants for a PC unit, two gasification units, and two natural gas units. Coal cost is assumed at $1.5/MBtu and natural gas at $5/MBtu.

TABLE 6.4 Estimated Costs for CO_2 Removal

	Output Penalty (%)	Efficiency Penalty	$/$CO_2$ tonne
NGCC	20	−10% pts	60
IGCC	5	−6% pts	18
PC	28	−12% pts	43

6.1.10 Conclusions

Essentially all suggested energy scenarios covering the next 20 years include the continued reliance on large central generating plants, with coal, natural gas, and nuclear fuel as the predominant energy sources. Nearly 800 GW of electric power for the U.S. in 2020 is not going to be possible in this time frame without a substantial number of large unit sizes in the 500 MW+ range. Further, in the light of fuel availability and cost forecasts, most industry observers foresee large coal and nuclear plants essential if the U.S. is to maintain its electrical energy supply and independence in the near term. It is important to note also the question of carbon dioxide control for fossil plants, that use of the nuclear option mitigates.

It is clear though that coal remains a crucial fuel resource for both U.S. and worldwide growth. In Europe and North America, natural gas-fired combustion turbines have been recently favored for new and repowered units, but it appears unlikely that the natural gas supply can meet all the new capacity demand and also be the major replacement fuel for the existing fleet. Substantial recent increases in natural gas prices have thus renewed interest in coal-based generation. Coal's sustained viability beyond the next 20 years will clearly depend on new technologies to improve efficiency, cost, and plant emissions.

It is also clear that, in small sizes, solar, wind, biomass, and other renewable technologies offer very attractive opportunities for cleaner energy sources to reduce mankind's environmental impact. State renewable energy programs are valuable in moving renewables into the market, while increased R&D can be a potent force in driving up the efficiency of these new technologies, simultaneously driving down costs and making them competitive in world markets.

It is clear that regulatory policies, siting-related issues, fuel availability, and risk assessment by financiers and owners will determine which technologies will succeed, and on what timetable.

In a broader perspective, the defining challenge of the coming century will be to balance the "trilemma" of interlocking sustainability issues—the economic aspirations of rapidly expanding populations in the developing world, environmental quality, and natural resource availability. Technology innovation is the best prospect for resolving conflicts between population, prosperity, and pollution. With much advanced technology now mature and ready for application in other parts of the globe, the developing world can leapfrog the slow technology development process and enjoy a vastly improved quality of life while further advances are pursued.

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6.2 Nuclear Power Technologies

Edwin A. Harvego and Kenneth D. Kok

6.2.1 Introduction

Nuclear power is derived from the fission of heavy element nuclei or the fusion of light element nuclei. This chapter will discuss nuclear power derived from the fission process because fusion as a practical power source will not reach the stage of commercial development in the next 20–25 years. In a nuclear reactor, the energy available from the fission process is captured as heat that is transferred to working fluids that are used to generate electricity. Uranium-235 (^{235}U) is the primary fissile fuel currently used in nuclear power plants. It is an isotope of uranium that occurs naturally at about 0.72% of all natural uranium deposits. When ^{235}U is "burned" (fissioned) in a reactor, it provides about one megawatt day of energy for each gram of ^{235}U fissioned (3.71×10^{10} Btu/lb).

Nuclear power technology includes not only the nuclear power plants that produce electric power but also the entire nuclear fuel cycle. Nuclear power begins with the mining of uranium. The ore is processed

and converted to a form that can be enriched in the ^{235}U isotope so that it can be used efficiently in today's light-water-moderated reactors. The reactor fuel is then fabricated into appropriate fuel forms for use in nuclear power plants. Spent fuel can then be either reprocessed or stored for future disposition. Radioactive waste materials are generated in all of these operations and must be disposed of. The transportation of these materials is also a critical part of the nuclear fuel cycle.

In this chapter, the development, current use, and future of nuclear power will be discussed. The second section of this chapter is a brief review of the development of nuclear energy as a source for production of electric power, and looks at nuclear power as it is deployed today both in the United States and worldwide. The third section examines the next generation of nuclear power plants that will be built. The fourth section reviews concepts being proposed for a new generation of nuclear power plants. The fifth section describes the nuclear fuel cycle, beginning with the availability of fuel materials and ending with a discussion of fuel reprocessing technologies. The sixth section discusses nuclear waste and the options for its management. The seventh section addresses nuclear power economics. Conclusions are presented in Section 6.2.8.

6.2.2 Development of Current Power-Reactor Technologies

The development of nuclear reactors for power production began following World War II when engineers and scientists involved in the development of the atomic bomb recognized that controlled nuclear chain reactions could provide an excellent source of heat for the production of electricity. Early research on a variety of reactor concepts culminated in President Eisenhower's 1953 address to the United Nations in which he gave his famous "Atoms for Peace" speech, in which he pledged the United States "to find the way by which the miraculous inventiveness of man shall not be dedicated to his death, but consecrated to his life." In 1954, President Eisenhower signed the 1954 Atomic Energy Act that fostered the cooperative development of nuclear energy by the Atomic Energy Commission (AEC) and private industry. This marked the beginning of the commercial nuclear power program in the United States.

The world's first large-scale nuclear power plant was the Shippingport Atomic Power Station in Pennsylvania, which began operation in 1957. This reactor was a pressurized-water reactor (PWR) nuclear power plant designed and built by the Westinghouse Electric Company and operated by the Duquesne Light Company. The plant produced 68 MWe and 231 MWt.

The first commercial-size boiling-water reactor (BWR) was the Dresden Nuclear Power Plant that began operation in 1960. This 200 MWe plant was owned by the Commonwealth Edison Company and was built by the General Electric Company at Dresden, Illinois, about 50 miles southwest of Chicago.

Although other reactor concepts, including heavy-water-moderated, gas-cooled and liquid-metal-cooled reactors, have been successfully operated, the PWR and BWR reactor designs have dominated the commercial nuclear power market, particularly in the U.S. These commercial power plants rapidly increased in size from the tens of MWe generating capacity to over 1000 MWe. Today, nuclear power plants are operating in 33 countries. The following section presents the current status of nuclear power plants operating or under construction around the world.

6.2.2.1 Current Nuclear Power Plants Worldwide

At the end of 2004 there were 439 individual nuclear power reactors operating throughout the world. More than half of these nuclear reactors are PWRs. The distribution of current reactors by type is listed in Table 6.5. As shown in Table 6.5, there are six types of reactors currently used for electricity generation throughout the world. The following sections provide a more detailed description of the different reactor types shown in the table.

6.2.2.2 Pressurized-Water Reactors

Pressurized-water reactors represent the largest number of reactors used to generate electricity throughout the world. They range in size from about 400–1500 MWe. The PWR shown in Figure 6.17 consists of

TABLE 6.5 Nuclear Power Units by Reactor Type, Worldwide

Reactor Type	Main Countries	# Units Operational	GWe	Fuel
Pressurized light-water reactors (PWR)	U.S., France, Japan, Russia	263	237	Enriched UO_2
Boiling light-water reactors (BWR and AWBR)	U.S., Japan, Sweden	92	81	Enriched UO_2
Pressurized heavy-water reactors—CANDU (PHWR)	Canada	38	19	Natural UO_2
Gas-cooled reactors (Magnox & AGR)	U.K.	26	11	Natural U (metal), enriched UO_2
Graphite-moderated light-water reactors (RBMK)	Russia	17	13	Enriched UO_2
Liquid-metal-cooled fast-breeder reactors (LMFBR)	Japan, France, Russia	3	1	PuO_2 and UO_2
		439	362	

Source: Information taken from World Nuclear Association Information Paper "Nuclear Power Reactors".

a reactor core that is contained within a pressure vessel and is cooled by water under high pressure. The nuclear fuel in the core consists of uranium dioxide fuel pellets enclosed in zircaloy rods that are held together in fuel assemblies. There are 200–300 rods in an assembly and 100–200 fuel assemblies in the reactor core. The rods are arranged vertically and contain 80–100 tons of enriched uranium.

The pressurized water at 315°C is circulated to the steam generators. The steam generator is a tube- and shell-type of heat exchanger with the heated high-pressure water circulating through the tubes. The steam generator isolates the radioactive reactor cooling water from the steam that turns the turbine generator. Water enters the steam generator shell side and is boiled to produce steam that is used to turn

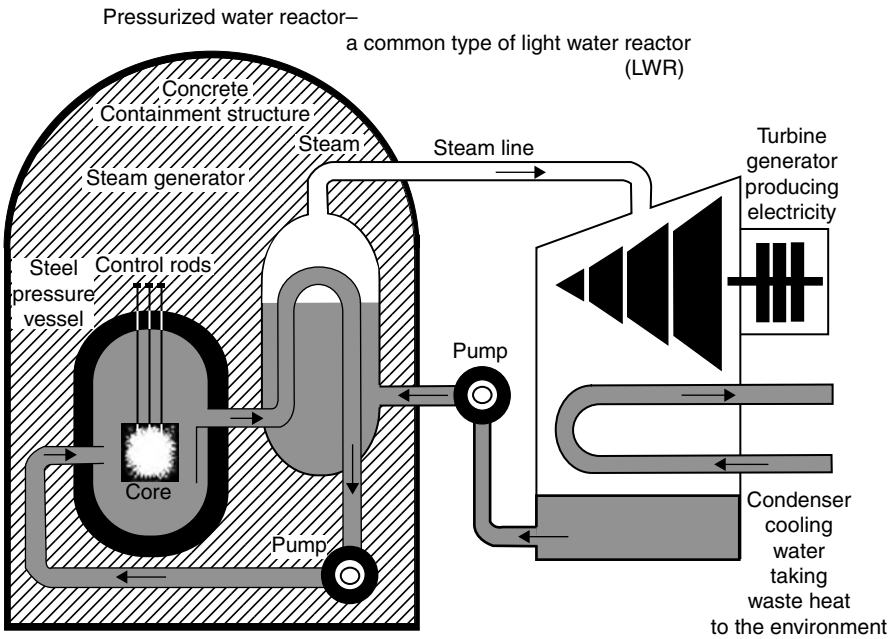

FIGURE 6.17 Sketch of a typical PWR power plant. (From World Nuclear Association, http://www.world-nuclear.org.)

the turbine generator producing electricity. The pressure vessel containing the reactor core and the steam generators are located in the reactor containment structure. The steam leaving the turbine is condensed in a condenser and returned to the steam generator. The condenser cooling water is circulated to cooling towers where it is cooled by evaporation. The cooling towers are often pictured as an identifying feature of a nuclear power plant.

6.2.2.3 Boiling-Water Reactors

The BWR power plants represent the second-largest number of reactors used for generating electricity. The BWRs range in size from 400 to 1200 MWe. The BWR, shown in Figure 6.18, consists of a reactor core located in a reactor vessel that is cooled by circulating water. The cooling water is heated to 285°C in the reactor vessel and the resulting steam is sent directly to the turbine generators. There is no secondary loop as there is in the PWR. The reactor vessel is contained in the reactor building. The steam leaving the turbine is condensed in a condenser and returned to the reactor vessel. The condenser cooling water is circulated to the cooling towers where it is cooled by evaporation.

6.2.2.4 Pressurized Heavy-Water Reactor

The so-called CANDU reactor was developed in Canada beginning in the 1950s. It consists of a large tank called a calandria containing the heavy-water moderator. The tank is penetrated horizontally by pressure tubes that contain the reactor fuel assemblies. Pressurized heavy water is passed over the fuel and heated to 290°C. As in the PWR, this pressurized water is circulated to a steam generator where light water is boiled, thereby forming the steam used to drive the turbine generators.

The pressure-tube design allows the CANDU reactor to be refueled while it is in operation. A single pressure tube can be isolated and the fuel can be removed and replaced while the reactor continues to operate. The heavy water in the calandria is also circulated and heat is recovered from it. The CANDU reactor is shown in Figure 6.19.

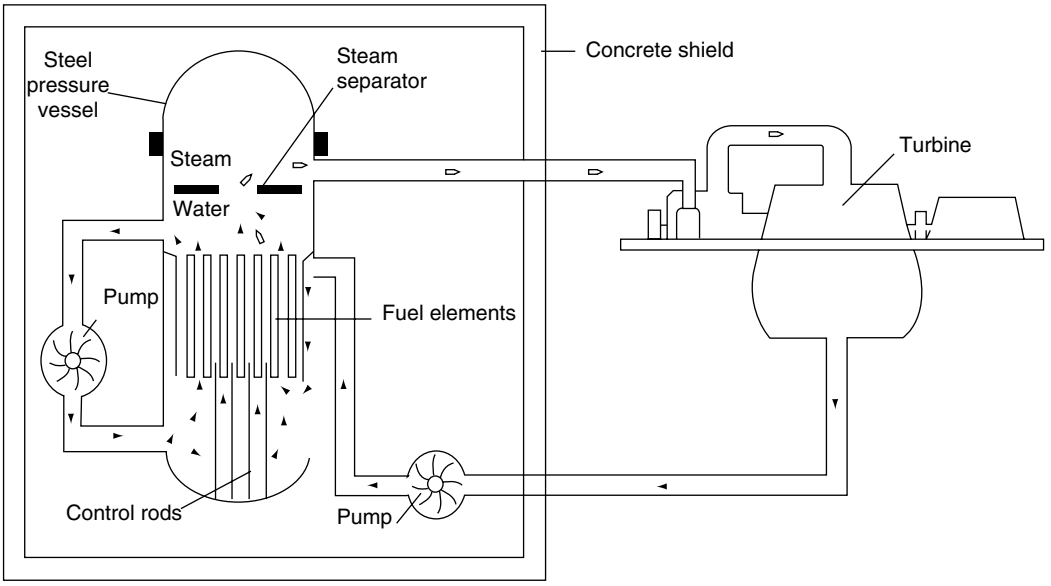

FIGURE 6.18 Sketch of a typical BWR power plant. (From World Nuclear Association, http://www.world-nuclear.org.)

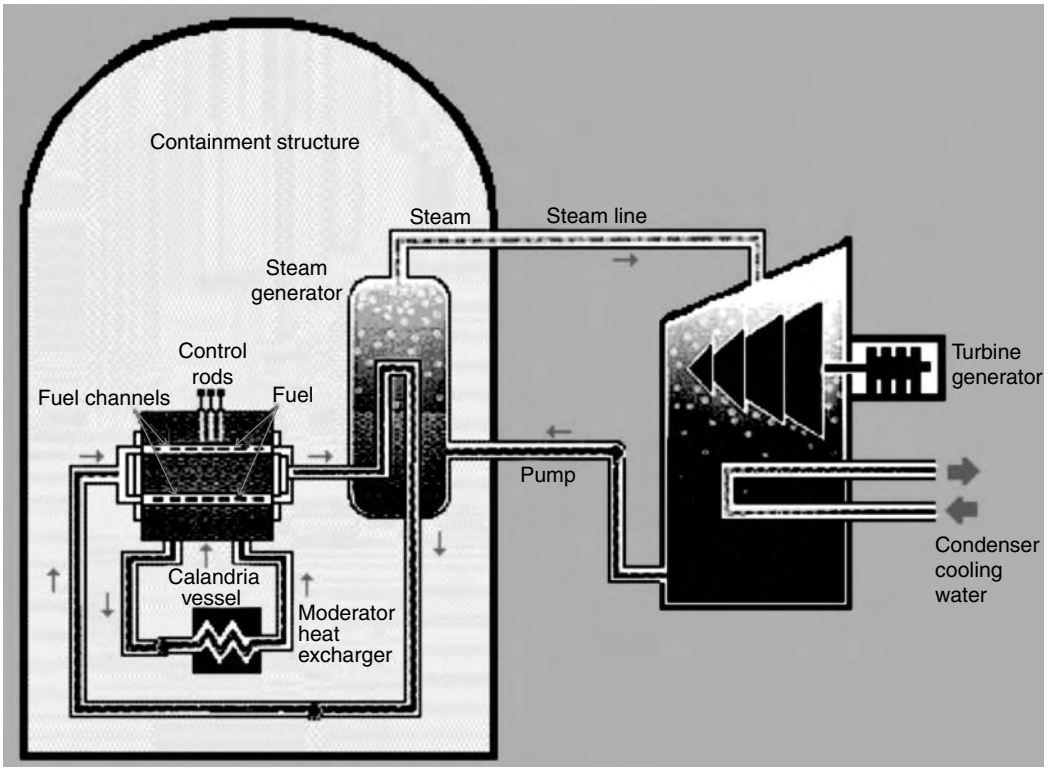

FIGURE 6.19 Sketch of a typical CANDU reactor power station. (From World Nuclear Association, http://www.world-nuclear.org.)

6.2.2.5 Gas-Cooled Reactors

Gas-cooled reactors were developed and implemented in the U.K. The first generation of these reactors was called Magnox, followed by the advanced gas-cooled reactor (AGR). These reactors are graphite moderated and cooled by CO_2. The Magnox reactors are fueled with uranium metal fuel, whereas the AGRs use enriched UO_2 as the fuel material. The CO_2 coolant is circulated through the reactor core and then to a steam generator. The reactor and the steam generators are located in a concrete pressure vessel. As with the other reactor designs, the steam is used to turn the turbine generator to produce electricity. Figure 6.20 shows the configuration for a typical gas-cooled reactor design.

6.2.2.6 Other Power Reactors

The remaining reactors listed in Table 6.5 are the light-water graphite-moderated reactors used in Russia, and the liquid-metal-cooled fast-breeder reactors (LMFBRs) in Japan, France, and Russia. In the light-water graphite-moderated reactors, the fuel is contained in vertical pressure tubes where the cooling water is allowed to boil at 290°C and the resulting steam is circulated to the turbine generator system as it is in a BWR. In the case of the LMFBR, sodium is used as the coolant and a secondary sodium cooling loop is used to provide heat to the steam generator.

6.2.2.7 Growth of Nuclear Power

The growth of nuclear power generation is being influenced by three primary factors. These factors are: (1) current plants are being modified to increase their generating capacity, (2) the life of old plants is being

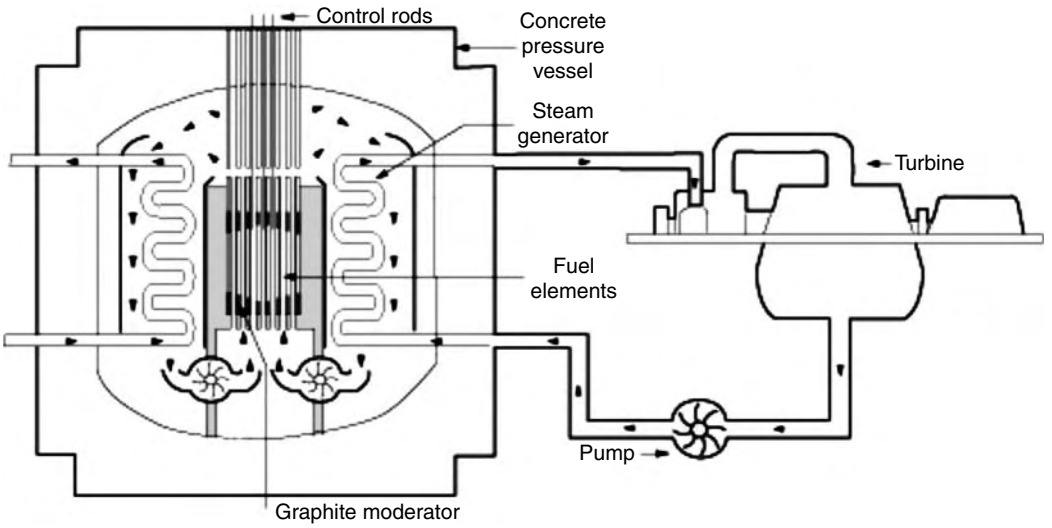

FIGURE 6.20 Sketch of a typical gasd-cooled reactor power station. (From World Nuclear Association, http://www.world-nuclear.org.)

lengthened by life-extension practices that include relicensing, and (3) new construction is adding to the number of plants operating worldwide. According to the IAEA, in May 2005, there were 440 nuclear power plants in operation with a total net installed capacity of 367 GWe. They now anticipate that 60 new plants will be constructed in the next 15 years, increasing the installed capacity to 430 GWe by 2020.

6.2.2.7.1 Increased Capacity

Operating nuclear plants are being modified to increase their generating capacity. Reactors in the U.S., Belgium, Sweden, Germany, Switzerland, Spain, and Finland are being uprated. In the U.S., 96 reactors have been uprated since 1977, with some of them having capacity increased up to 20%. The number of operating reactors in the U.S. peaked in 1991 with a gross electrical generation of over 70,000 MW-years; however, in 2003, the net electrical generation approached 90,000 MW-years from six fewer reactors. The generating capacity increase was due to both power uprating and improvements in operation and maintenance practices to produce higher plant availability. Switzerland increased the capacity of its plants by over 12%, whereas in Spain, uprating has added 11% to that country's nuclear capacity. The uprating process has proven to be a very cost effective way to increase overall power production capacity while avoiding the high capital cost of new construction.

6.2.2.7.2 Plant-Life Extension

Life extension is the process by which the life of operating reactors is increased beyond the original planned and licensed life. Most reactors were originally designed and licensed for an operational life of 40 years. Without life extension, many of the reactors that were built in the 1970 s and 1980 s would reach the end of their operational lives during the years 2010–2030. If they were not replaced with new plant construction, there would be a significant decrease in nuclear-based electricity generation as these plants reached the end of their useful life.

Engineering assessments of current nuclear plants have shown that they are able to operate for longer than their original planned and licensed lifetime. Fifteen plants in the U.S. have been granted 20-year extensions to their operating licenses by the U.S. Nuclear Regulatory Commission (NRC). The operators of most of the remaining plants are also expected to apply for license extensions. This will give the plants an operating life of 60 years. In Japan, operating lifetimes of 70 years are envisaged.

The oldest nuclear power stations in the world were operated in Great Britain. Chalder Hall and Chaplecross were built in the 1950 s and were expected to operate for 20–25 years. They were authorized to

operate for 50 years, but were shut down in 2003 and 2004 for economic reasons. In 2000, the Russian government extended the lives of their 12 oldest reactors by 15 years for a total of 45 years.

Although life extension has become the norm throughout the world, many reactors have been shut down due to economic, regulatory, and political reasons. Many of these reactors were built early in the development of nuclear power. They tended to be smaller in size and were originally built for demonstration purposes. However, the political and regulatory process in some countries has led to the termination of nuclear power programs and the shutdown of viable reactor plants.

6.2.2.7.3 New Nuclear Plant Construction

New nuclear power plants are currently being constructed in several countries. The majority of the new construction is in Asia. Plants currently under construction are listed in Table 6.6.

6.2.3 Next-Generation Technologies

The next generation, generation-III nuclear power reactors, are being developed to meet power production needs throughout the world. These reactors incorporate the lessons that have been learned by operation of

TABLE 6.6 Power Reactors under Construction

Start Operation	Country, Organization	Reactor	Type	MWe (net)
2005	Japan, Tohoku	Higashidori 1	BWR	1067
2005	India, NPCIL	Tarapur 4	PHWR	490
2005	China, CNNC	Tianwan 1	PWR	950
2005	Ukraine, Energoatom	Khmelnitski	PWR	950
2005	Russia, Rosenergoatom	Kalinin 3	PWR	950
2006	Iran, AEOI	Bushehr 1	PWR	950
2006	Japan, Hokuriku	Shika 2	ABWR	1315
2006	India, NPCIL	Tarapur 3	PHWR	490
2006	China, CNNC	Tianwan 2	PWR	950
2006	China, Taipower	Lungmen 1	ABWR	1300
2007	India, NPCIL	Rawatbhata 5	PHWR	202
2007	Romania, SNN	Cernavoda 2	PHWR	650
2007	India, NPCIL	Kudankulam 1	PWR	950
2007	India, NPCIL	Kaiga 3	PHWR	202
2007	India, NPCIL	Kaiga 4	PHWR	202
2007	USA, TVA	Browns Ferry 1	BWR	1065
2007	China, Taipower	Lungmen 2	ABWR	1300
2008	India, NPCIL	Kudankulam 2	PWR	950
2008	India, NPCIL	Rawatbhata 6	PHWR	202
2008	Russia, Rosenergoatom	Volgodonsk-2	PWR	950
2008	Korea, KHNP	Shin Kori 1	PWR	950
2009	Finland, TVO	Oikiluoto 3	PWR	1600
2009	Japan, Hokkaido	Tomari 3	PWR	912
2009	Korea, KHNP	Shin Kori 2	PWR	950
2009	Korea, KHNP	Shin Wolsong 1	PWR	950
2010	Russia, Rosenergoatom	Balakovo 5	PWR	950
2010	Russia, Rosenergoatom	Kalinin 4	PWR	950
2010	India, NPCIL	Kalpakkam	FBR	440
2010	Pakistan, PAEC	Chashma 2	PWR	300
2010	Korea, KHNP	Shin Wolsong 2	PWR	950
2010	North Korea, KEDO	Sinpo 1	PWR(KSNP)	950
2010	China, Guangdong	Lingao 3	PWR	950
2010	Russia, Rosenergoatom	Beloyarsk 4	FBR	750
2011	China, Guangdong	Lingao 4	PWR	950
2011	China, CNNC	Sanmen 1 and 2	PWR	?
2011	China, CNNC	Yangiang 1 and 2	PWR	?

Source: From World Nuclear Association, Plans for New Reactors Worldwide, http://www.world-nuclear.org, 2005.

nuclear power systems since the 1950 s. The reactors are designed to be safer, more economical, and more fuel efficient. The first of these reactors were built in Japan and began operation in 1996.

The biggest change in the generation-III reactors is the addition of passive safety systems. Earlier reactors relied heavily on operator actions to deal with a variety of operational upset conditions or abnormal events. The advanced reactors incorporate passive or inherent safety systems that do not require operator intervention in the case of a malfunction. These systems rely on such things as gravity, natural convection, or resistance to high temperatures.

Generation-III reactors also have:

- Standardized designs with many modules of the reactor being factory constructed and delivered to the construction site leading to expedited licensing, reduction of capital cost and reduced construction time
- Simpler designs with fewer components that are more rugged, easier to operate, and less vulnerable to operational upsets
- Longer operating lives of 60 years and designed for higher availability
- Reduced probability of accidents leading to core damage
- Higher fuel burnup reducing refueling outages and increasing fuel utilization with less waste produced

The following sections describe the different types of generation-III reactors being developed worldwide.

6.2.3.1 Light-Water Reactors

Generation-III advanced light-water reactors are being developed in several countries. These will be described below on a country by country basis.

6.2.3.1.1 United States

Even though no new reactors are being built in the U.S., U.S. companies have continued to design advanced systems in anticipation of sales both in the U.S. and other parts of the world. In the U.S., the commercial nuclear industry in conjunction with the U.S. Department of Energy (DOE) has developed four advanced light-water reactor designs.

Two of these are based on experience obtained from operating reactors in the U.S., Japan, and western Europe. These reactors will operate in the 1300-MW range. One of the designs is the advanced boiling-water reactor (ABWR). This reactor was designed in the U.S. and is already being constructed and operated in Asia. The NRC gave final design certification to the ABWR in 1997. It was noted that the design exceeded NRC "safety goals by several orders of magnitude." The other type, designated System 80+, is an advanced PWR. This reactor system was ready for commercialization, but the sale of this design is not being pursued.

The AP-600 (AP=advanced passive), designed by Westinghouse, was the second reactor system to receive NRC certification. The certification came in 1999. The reactor is designed with passive safety features that result in projected core damage frequencies nearly 1000 times less than current NRC licensing requirements.

The Westinghouse AP-1000 (a scaled up version of the AP-600) received final design approval from the NRC and is scheduled for full design certification in 2005. The passive safety systems in this reactor design lead to a large reduction in components including 50% fewer valves, 35% fewer pumps, 80% less pipe, 45% less seismic building volume, and 70% less cable.

Another aspect of the AP-1000 is the construction process. After the plant is ordered, the plant will be constructed in a modular fashion, with modules being fabricated in a factory setting and then transported to the reactor site. The anticipated design construction time for the plant is 36 months. The construction cost of an AP-1000 is expected to be $1200/kW and the generating costs are postulated to be less then 3.5 cents/kWh. The plant is designed to have a 60-year operating life. China, Europe, and the U.S. are considering purchases of the AP-1000.

General Electric has created a modification of the ABWR for the European market. The European simplified BWR is a 1300 MWe reactor with passive safety systems. It is now called the economic and simplified boiling-water reactor (ESBWR). General Electric has a 1500-MWe version of this reactor in the preapplication stage for design certification by the NRC.

An international project being led by Westinghouse is designing a modular 335-MWe reactor known as the international reactor innovative & secure (IRIS). This PWR is being designed with integral steam generators and a primary cooling system that are all contained in the reactor pressure vessel. The goal of this system is to reach an eight-year refueling cycle using 10% enriched fuel with an 80,000-MWd/t burn-up. U.S. Nuclear Regulatory Commission design certification of this plant is anticipated by 2010.

6.2.3.1.2 Japan

Japan has three operating ABWRs. The first two, Kashiwazaki Kariwa-6 and Kashiwazaki Kariwa-7, began operation in 1996, and the third, Hamaoka-5, started up in 2004. These plants are expected to have a 60-year life and produce power at about $0.07/kWh. Several of these plants are under construction in Japan and Taiwan.

Hitachi has completed systems design of three additional ABWRs. These are rated at 600, 900, and 1700 MWe and are based on the design of the 1350-MWe plants. The smaller versions are designed with standardized components that will allow construction times on the order of 34 months.

Westinghouse and Mitsubishi, in conjunction with four utilities, are developing a large, 1500-MWe advanced PWR. This design will have both active and passive cooling systems and will have a higher fuel burn-up of 55 GWd/t of fuel. Mitsubishi is also participating with Westinghouse in the design of the AP-1000.

6.2.3.1.3 South Korea

The South Koreans have the APR-1400 system that evolved from the U.S. System 80+ and is known as the Korean next generation reactor. The first of these will be Shin-Kori-3 and Shin-Kori-4. Capital cost for the first systems is estimated to be $1400/kW with future plants coming in at $1200/kW with a 48-month construction time.

6.2.3.1.4 Europe

Four designs are being developed in Europe to meet the European utility requirements that were derived from French and German requirements. These systems have stringent safety requirements.

Framatome ANP has designed a large (1600–1750 MWe) European pressurized-water reactor (EPR). This reactor is the new standard design in France and it received design approval in 2004. The first of these units is scheduled to be built at Olkiluoto in Finland and the second at Flamanville in France. It is capable of operating in a load following manner and will have a fuel burn-up of 65 GWd/t. It has the highest thermal efficiency of any light-water reactor at 36%.

Framatome ANP, in conjunction with German utilities and safety authorities, is developing the supercritical-water-cooled reactor (SWR), a 1000–1290-MWe BWR. This design was completed in 1999 and is ready for commercial deployment. Framatome ANP is seeking U.S. design certification for this system.

General Electric and Westinghouse are also developing designs for the European market. The General Electric system, known as the ESBWR, is 1390 MWe and is based on the ABWR. They are in the preapplication stage for a 1500-MWe version of this reactor for design certification by the U.S. NRC. Westinghouse is working with European and Scandinavian authorities on the 90+ PWR to be built in Sweden. These reactors all have passive safety systems.

6.2.3.1.5 Russia

Russia has also developed several advanced PWR designs with passive safety systems. The Gidropress 1000 MWe V-392 is being built in India with another planned for Novovoronezh. They are also building two VVER-91 reactors in China at Jiangsu Tainwan. The VVER-91 is designed with western control systems.

OKBM is developing the VVER-1500 for replacement of two plants each in Leningrad and Kursk. The design is planned to be complete in 2007 and the first units will be commissioned in 2012–2013.

Gidropress is developing a 640-MWe PWR with Siemans control systems which will be designated the VVER-640. OKBM is designing the VVER-600 with integral steam generators. Both of these designs will have enhanced safety systems.

6.2.3.2 Heavy-Water Reactors

Heavy-water reactors continue to be developed in Canada by AECL. They have two designs under development. The first, designated CANDU-9, is a 925–1300-MWe extension of the current CANDU-6. The CANDU-9 completed a two-year license review in 1997. The interesting design feature of this system is the flexible fuel requirements. Fuel materials include natural uranium, slightly enriched uranium, uranium recovered from the reprocessing of PWR fuel, mixed oxide (MOX) fuels, direct use of spent PWR fuel, and also thorium. The second design is the advanced CANDU Reactor (ACR). It uses pressurized light water as a coolant and maintains the heavy water in the calandria. The reactor is run at higher temperature and pressure, which gives it a higher thermal efficiency than earlier CANDU reactors.

The ACR-700 is smaller, simpler, cheaper, and more efficient than the CANDU-6. It is designed to be assembled from prefabricated modules that will cut the construction time to a projected 36 months. Heavy-water reactors have been plagued with a positive-void reactivity coefficient, which led some to question their safety. The ACR-700 will have a negative-void reactivity coefficient that enhances the safety of the system, as do the built-in passive safety features. AECL is seeking certification of this design in Canada, China, the U.S., and the U.K.

A follow-up to the ACR-700 is the ACR-1000, which will contain additional modules and operate in the range of 1100–1200 MWe. Each module of this design contains a single fuel channel and is expected to produce 2.5 MWe. The first of these systems is planned for operation in Ontario by 2014.

The long-range plan of AECL is to develop the CANDU-X, which will operate at a much higher temperature and pressure, yielding a projected thermal efficiency of 40%. The plan is to commercialize this plant after 2020 with a range of sizes from 350 to 1150 MWe.

India is also developing an advanced heavy-water reactor (AHWR). This reactor is part of the Indian program to utilize thorium as a fuel material. The AHWR is a 300-MWe heavy-water-moderated reactor. The fuel channels are arranged vertically in the calandria and are cooled by boiling light water. The fuel cycle will breed ^{233}U from ^{232}Th.

6.2.3.3 High-Temperature Gas-Cooled Reactors

The third generation of HTGRs is being designed to directly drive a gas turbine generating system using the circulating helium that cools the reactor core. The fuel material is a uranium oxycarbide in the form of small particles coated with multiple layers of carbon and silicon carbide. The coatings will contain the fission products and are stable up to 1600°C. The coated particles can be arranged in fixed graphite fuel elements or contained in "pebbles" for use in a pebble-bed-type reactor.

In South Africa, a consortium lead by the utility Eskom is developing the pebble-bed-modular reactor (PBMR). This reactor will have modules with power outputs of 165 MWe. It will utilize the direct gas turbine technology and is projected to have a thermal efficiency of 42%. The goal is to obtain a fuel burn-up of 90 GWd/t at the outset and eventually reach 200 GWd/t. The intent is to build a demonstration plant for operation in 2006 and obtain commercial operation in 2010.

In the U.S., a larger system is being design by General Atomics in conjunction with Minatom of Russia and Fuji of Japan. This reactor, designated the gas turbine-modular helium reactor (GT–MHR), utilizes hexagonal fuel elements of the kind that were used in the Fort St. Vrain reactor. The initial use of this reactor is expected to be to burn the weapons-grade plutonium at Tomsk in Russia.

6.2.3.4 Fast-Neutron Reactors

Several nations are working on developing improved fast-breeder reactors (FBRs). Fast-breeder reactors are fast-neutron reactors and about 20 of these reactors have operated since the 1950 s. They are able to use

both ^{238}U and ^{235}U as reactor fuel, thus making use of all the uranium. These reactors use liquid metal as a coolant. In Europe, research work on the 1450-MWe European FBR has been halted.

In India, at the Indira Gandhi Centre for Atomic Research, a 40-MWt fast-breeder reactor has been operating since 1985. This reactor is used to research the use of thorium as reactor fuel by breeding ^{233}U. India has used this experience and began the construction of a 500-MWe prototype fast-breeder reactor in 2004. This unit at Kalpakkam is expected to be operating in 2010.

In Japan, the Joyo experimental reactor has been operating since 1977 and its power is now being raised to 140 MWt.

In Russia, the BN-600 FBR has been supplying electricity since 1981. It is considered to be the best operating reactor in Russia. The BN-350 FBR operated in Kazakhstan for 27 years and was used for water desalinization as well as electricity production. The BN-600 is being reconfigured to burn plutonium from the military stockpiles.

Russia has also begun construction of the BN-800 (880 MWe), which has enhanced safety features and improved fuel economy. This reactor will also be used to burn stockpiled plutonium. Russia has also experimented with lead-cooled reactor designs. A new Russian design is the BREST fast-neutron reactor. It will operate at 300 MWe or more and is an inherently safe reactor design. A pilot unit is being built at Beloyarsk. The reactor is fueled with plutonium nitride fuel and it has no blanket so no new plutonium is produced.

In the U.S., General Electric is involved in the design of a 150-MWe modular liquid-metal-cooled inherently safe reactor called PRISM. This design, along with a larger 1400-MWe design being developed jointly by GE and Argonne, has been withdrawn from NRC review.

6.2.3.5 Summary of Generation-III Reactors

As can be seen from the discussion above, there are many reactor systems of many types under development. The key feature of all of these reactors is the enhancement of safety systems. Some of these reactors have already been built and are in operation, whereas others are under construction. This activity indicates that there will be a growth of nuclear-reactor-generated electricity during the next 20 years. Table 6.7, taken from World Nuclear Association information on advanced nuclear power reactors, shows the advanced thermal reactors that are being marketed around the world.

6.2.4 Generation-IV Technologies

As discussed earlier, the development of nuclear power occurred in three general phases. The initial development of prototype reactor designs occurred in the 1950s and 1960s, development and deployment of large commercial plants occurred in the 1970s and 1980s, and development of advanced light-water reactors occurred in the 1990s.

Although the earlier generations of reactors have effectively demonstrated the viability of nuclear power, the nuclear industry still faces a number of challenges that need to be overcome for nuclear power to achieve its full potential. Among these challenges are (1) public concern about the safety of nuclear power in the wake of the Three Mile Island accident in 1979 and the Chernobyl accident in 1986, (2) high capital costs and licensing uncertainties associated with the construction of new nuclear power plants, (3) public concern over potential vulnerabilities of nuclear power plants to terrorist attacks, and (4) issues associated with the accumulation of nuclear waste and the potential for nuclear material proliferation in an environment of expanding nuclear power production.

To address these concerns and to fully realize the potential contributions of nuclear power to future energy needs in the United States and worldwide, the development of a new generation of reactors, termed generation IV, was initiated in 2001. The intent or objective of this effort is to develop multiple generation-IV nuclear power systems that would be available for international deployment before the year 2030. The development of the generation-IV reactor systems is an international effort, initiated by the U.S. DOE with participation from 10 countries. These countries established a formal organization referred to as the Generation IV International Forum (GIF). The GIF countries included Argentina, Brazil, Canada, France,

TABLE 6.7 Advanced Thermal Reactors Being Marketed

Country and Developer	Reactor	Size (MWe)	Design Progress	Main Features
U.S.-Japan (GE-Hitachi-Toshiba)	ABWR	1300	Commercial operation in Japan since 1996–1997, in U.S.: NRC certified 1997, first-of-a-kind engineering	Evolutionary design More efficient, less waste Simplified construction (48 months) and operation
South Korea (derived from Westinghouse)	APR-1400 (PWR)	1400	NRC certified 1997, Further developed for new S. Korean Shin Kori 3 and 4, expected to be operating in 2010	Evolutionary design Increased reliability Simplified construction and operation
U.S.A (Westinghouse)	AP-600	600	AP-600: NRC certified 1999, FOAKE	Passive safety features Simplified construction and operation 3 years to build 60-year plant life
	AP-1000 (PWR)	1100	AP-1000 NRC design approval 2004	
Japan (Utilities, Westinghouse, Mitsubishi)	APWR	1500	Basic design in progress, planned at Tsuruga	Hybrid safety features Simpliified construction and operation
France–Germany (Framatome ANP)	EPR (PWR)	1600	Confirmed as future French standard, French design approval, to be built in Finland	Evolutionary design Improved safety features High fuel efficiency Low-cost electricity
U.S.A (GE)	ESBWR	1390	Developed from the ABWR, precertification in U.S.A	Evolutionary design Short construction time Enhanced safety features
Germany (Framatome ANP)	SWR-1000 (BWR)	1200	Under development, precertification in U.S.A	Innovative design High-fuel efficiency Passive safety features
Russia (OKBM)	V-448 (PWR)	1500	Replacement for Leningrad and Kursk plants	High-fuel efficiency Enhanced safety

Russia (Gidropress)	V-392 (PWR)	950	Two being build in India, likely bid for China	Evolutionary design 60-year plant life Enhanced safety features
Canada (AECL)	CANDU-9	925-1300	Licensing approval 1997	Evolutionary design Single stand-alone unit Flexible fuel requirements Passive safety features
Canada (AECL)	ACR	700	ACR-700: precertification in U.S.A	Evolutionary design Light-water cooling Low-enriched fuel Passive safe features
		1000	ACR-1000 proposed for U.K.	
South Africa (Eskom, BNFL)	PBMR	165 (module)	Prototype due to start building, precertification in U.S.A	Modular plant, low cost Direct cycle gas turbine High-fuel efficiency Passive safety features
U.S.A-Russia et al. (General Atomics, Minatom)	GT-MHR	285 (module)	Under development in the U.S.A. and Russia by multinational joint venture	Modular plant, low cost Direct-cycle gas turbine High-fuel efficiency Passive safety features

Source: From World Nuclear Association, Plans for New Reactors Worldwide, http://www.world-nuclear.org, 2005.

Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the United Kingdom, and the United States. The intent of the GIF is "…to develop future-generation nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public perception concerns."

The process used by the GIF to identify the most promising reactor concepts for development (referred to as the Generation IV Technology Roadmap) consisted of three steps. These steps were (1) to develop a set of goals for new reactor systems, (2) solicit proposals from the worldwide nuclear community for new reactor systems to meet these goals, and (3) using experts from around the world, evaluate the different concepts to select the most promising candidates for further development.

The eight goals developed by the GIF for generation-IV nuclear systems were:

- Sustainability 1: Generation-IV nuclear energy systems will provide sustainable energy generation that meets clean air objective and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.
- Sustainability 2: Generation-IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden in the future, thereby improving protection for the public health and the environment.
- Economics 1: Generation-IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
- Economics 2: Generation-IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
- Safety and reliability 1: Generation-IV nuclear energy systems operations will excel in safety and reliability.
- Safety and reliability 2: Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
- Safety and reliability 3: Generation-IV nuclear energy systems will eliminate the need for offsite emergency response.
- Proliferation resistance and physical protection: Generation-IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

Over 100 generation-IV candidates were evaluated by experts from the GIF countries and six reactor systems were selected for further evaluation and potential development. The six reactor systems selected were:

6.2.4.1 Gas-Cooled Fast-Reactor System

The gas-cooled fast-reactor system (GFR) is a fast-neutron spectrum reactor that uses helium as the primary coolant. It is designed to operate at relatively high helium outlet temperatures, making it a good candidate for the high-efficiency production of electricity or hydrogen. As shown in Figure 6.21 below, a direct Brayton cycle is used for the production of electricity with the helium gas delivered from the reactor outlet to a high-temperature gas turbine connected to a generator that produces electricity. In alternative designs, the high-temperature helium can also be used to produce hydrogen using either a thermochemical process or high-temperature electrolysis, or for other high-temperature process heat applications.

The reference plant is designed to produce 288 MWe using the direct Brayton cycle with a reactor outlet temperature of 850°C. The fuel forms being considered for high-temperature operation include composite ceramic fuel, advanced fuel particles, or ceramic clad elements of actinide compounds. Alternative core configurations include prismatic blocks and pin- or plate-based assemblies. The GFR's fast-neutron spectrum also makes it possible to efficiently use available fissile and fertile materials in a once-through fuel cycle.

6.2.4.2 Very-High-Temperature Reactor

The very-high-temperature reactor (VHTR) is a helium-cooled reactor designed to provide heat at very high temperatures, in the range of 1000°C for high-temperature process heat applications. In particular, the

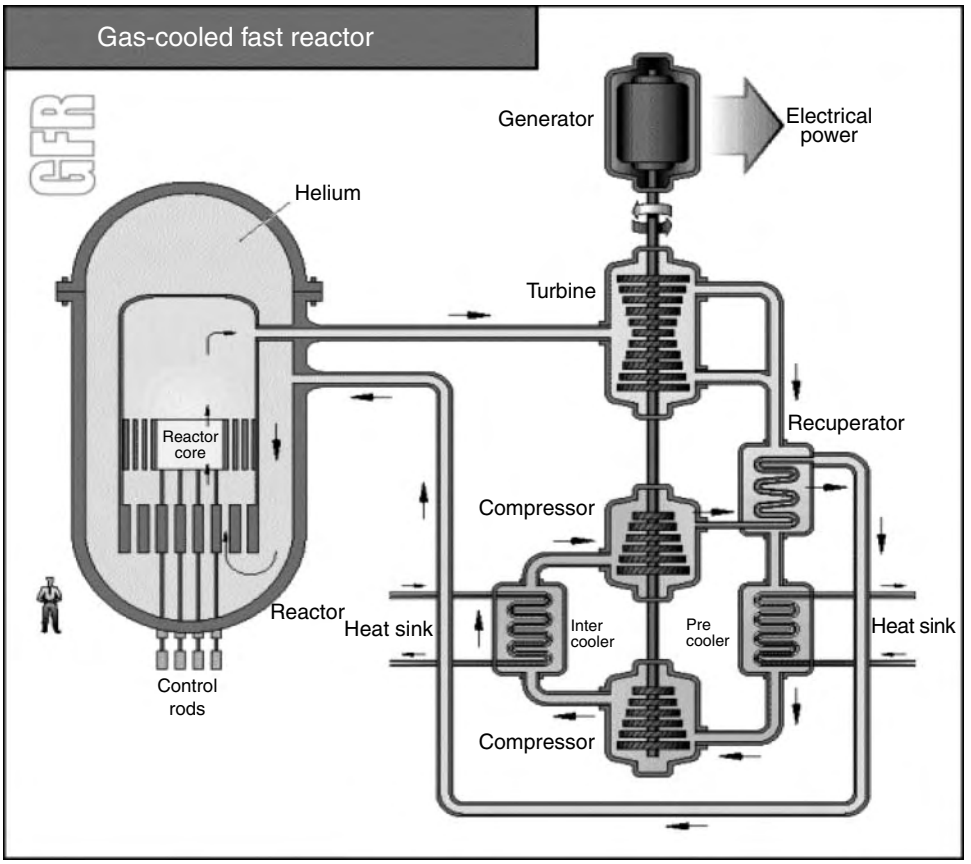

FIGURE 6.21 Gas-cooled fast reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum. 2002. *A Technology Roadmap for the Generation IV Nuclear Energy Systems*.)

1000°C reactor outlet temperature makes it a good candidate for the production of hydrogen using either thermochemical or high-temperature electrolysis processes. As shown in Figure 6.22 below, heat for the production of hydrogen is delivered through an intermediate heat exchanger that serves to isolate the reactor system from the hydrogen production process.

The reference design for the VHTR is a 600-MWt reactor with an outlet temperature of 1000°C. The reactor core uses graphite as a moderator to produce the thermal neutrons for the fission process. The core configuration can be either graphite blocks or pebbles about the size of billiard balls in which fuel particles are dispersed. For electricity production, either a direct Brayton cycle gas turbine using the primary helium coolant as the working fluid, or an indirect Rankine cycle using a secondary working fluid can be used. The high-temperature characteristics of this reactor concept also make it an ideal candidate for cogeneration applications to meet both electricity and hydrogen production or other high-temperature process heat needs.

6.2.4.3 Supercritical-Water-Cooled Reactor

The supercritical-water-cooled reactor (SWR) is a relatively high-temperature, high-pressure reactor designed to operate above the thermodynamic critical point of water, which is 374°C and 22.1 MPa. Because there is no phase change in the supercritical coolant water, the balance of plant design, shown in Figure 6.23, utilizes a relatively simple direct-cycle power-conversion system. The reference design for this concept is a 1700-MWe reactor operating at a pressure of 25 MPa with a reactor outlet temperature ranging

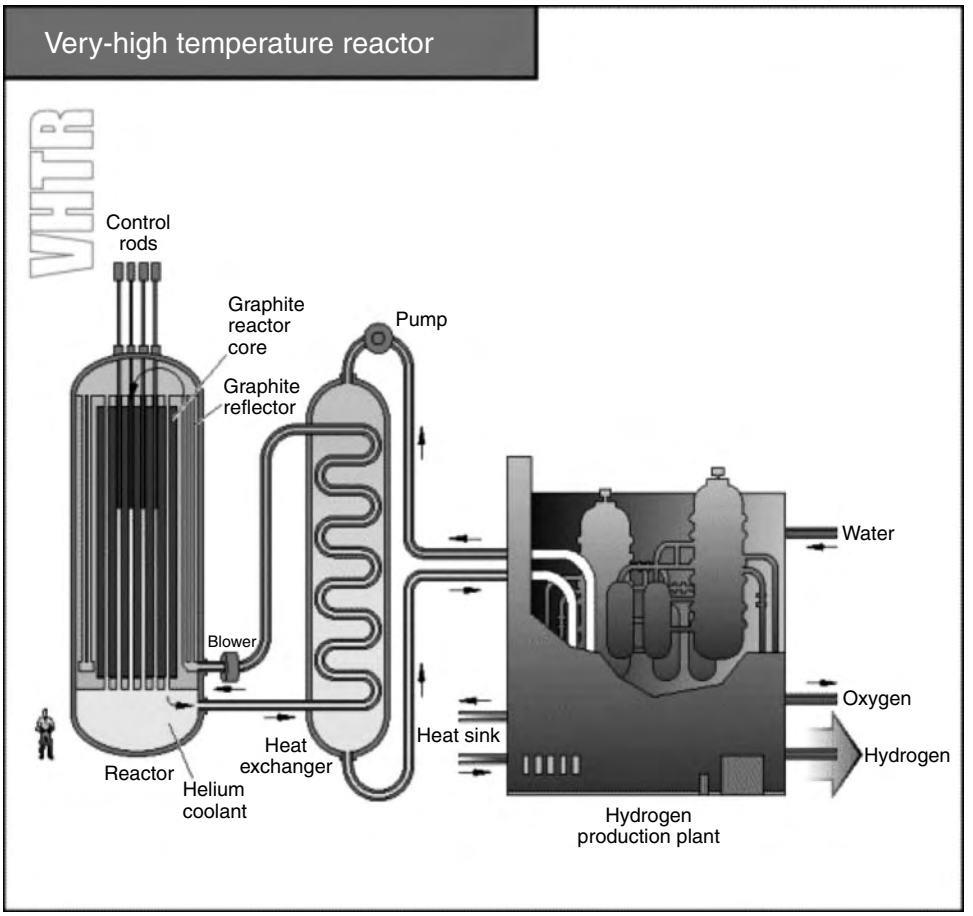

FIGURE 6.22 Very-high temperature reactor. (From US DOE Nuclear Energy Research Advisory committee and the Generation IV International Forum, *A Technology Roadmap for the Generation IV Nuclear Energy Systems*.)

between 510 and 550°C. This reactor can be designed as either a fast-neutron-spectrum or thermal-neutron-spectrum reactor. The relatively simple design also allows for the incorporation of passive safety features similar to those of the simplified boiling-water reactor discussed earlier. However, unlike the previously discussed concepts, the lower reactor outlet temperature is not well suited for the efficient production of hydrogen, which requires minimum temperatures in the range of 850°C–900°C. Therefore, this reactor concept is primarily intended for the efficient, low-cost production of electricity.

6.2.4.4 Sodium-Cooled Fast Reactor

The sodium-cooled fast reactor (SFR), shown in Figure 6.24, is a sodium-cooled fast-neutron-spectrum reactor designed primarily for the efficient management of actinides and conversion of fertile uranium in a closed fuel cycle. Two reference designs to support different fuel reprocessing options have been defined for this concept. The first is a medium-sized sodium-cooled reactor with a power output between 150 and 500 MWe that utilized uranium-plutonium-minor-actinide-zirconium metal alloy fuel. This reactor concept is supported by a fuel cycle based on pyrometallurgical processing in which the processing facilities are an integral part of the reactor plant design.

The second reactor reference design is a large sodium-cooled reactor with a power output capability between 500 and 1500 MWe that utilizes uranium-plutonium oxide fuel. This reactor design is supported

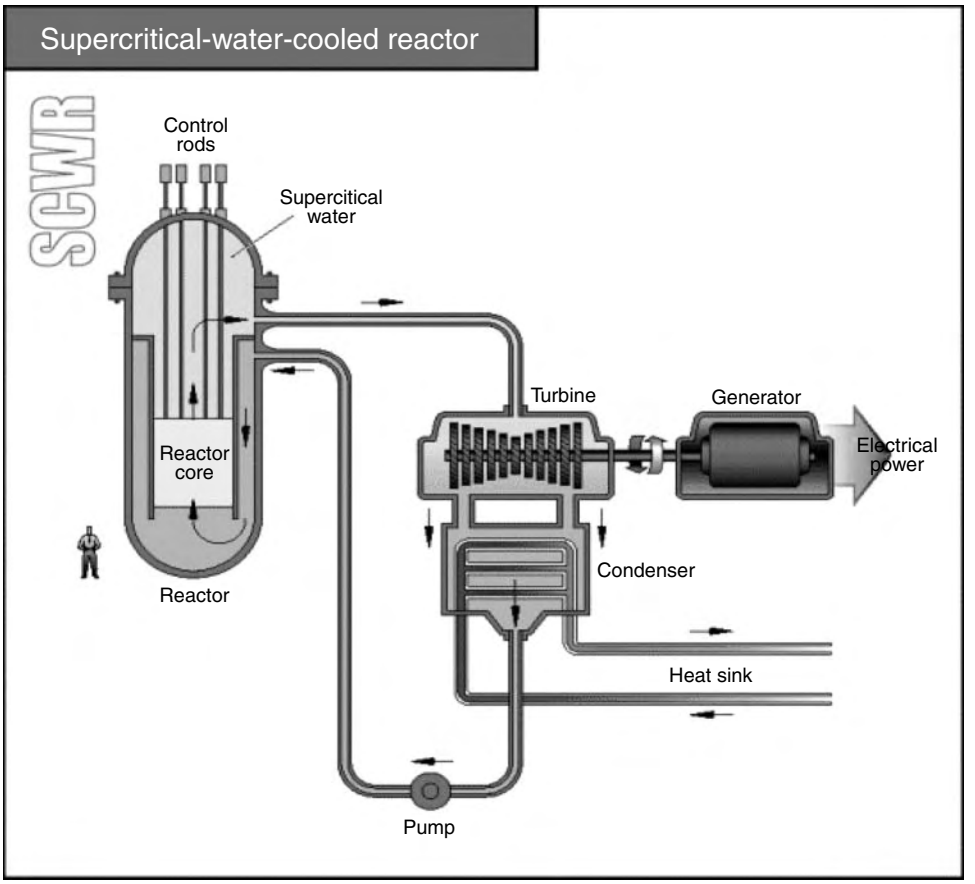

FIGURE 6.23 Supercritical-water-cooled reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, 2002. *A Technology Roadmap for the Generation IV Nuclear Energy Systems*.)

by a fuel cycle based on an advanced aqueous process that would include a centrally located processing facility supporting a number of reactors.

Both versions of this reactor concept would operate at coolant outlet temperatures in the range of 550°C, and are intended primarily for the management of high-level waste and the production of electricity. In addition to design innovations to reduce capital costs, these reactors incorporate a number of enhanced safety features that include:

- Long thermal response time
- Large margin to coolant boiling
- Primary system that operates near atmospheric pressure
- Intermediate sodium system between the radioactive sodium in the primary system and the water and steam in the power plant.

6.2.4.5 Lead-Cooled Fast Reactor

The lead-cooled fast reactor (LFR) is a fast-neutron-spectrum reactor cooled by either molten lead or a lead-bismuth eutectic liquid metal. It is designed for the efficient conversion of fertile uranium and the management of actinides in a closed fuel cycle. The reactor core for this design, shown in Figure 6.25,

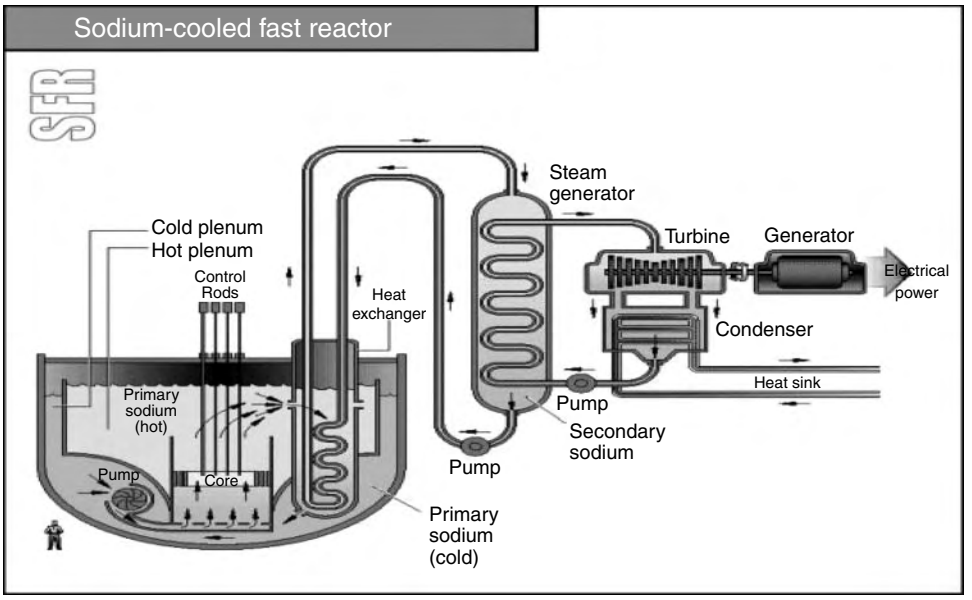

FIGURE 6.24 Sodium-cooled fast reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, 2002. *A Technology Roadmap for the Generation IV Nuclear Energy Systems*.)

utilizes a metal or nitride-based fuel containing fertile uranium and transuranics. As shown in Figure 6.25, the LFR relies on natural convection to cool the reactor core. The outlet temperature for the current reactor concept is about 550°C, but with advanced materials, reactor outlet temperatures of 800°C may be possible. An indirect-gas Brayton cycle is used to produce electrical power.

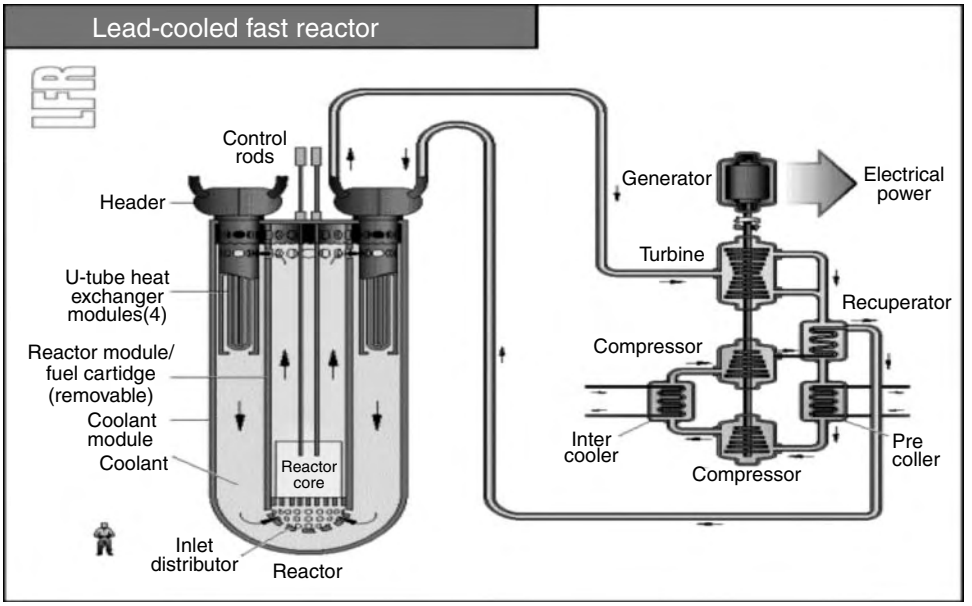

FIGURE 6.25 Lead-cooled fast reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, 2002. *A Technology Roadmap for the Generation IV Nuclear Energy Systems*.)

There are currently three versions of the reference design for this concept. The smallest design, rated at 50–150 MWe is intended for distributed power applications or electricity production on small grids. This reactor design, referred to as a battery, features modular design with a factory fabrication "cassette" core. The reactor is designed for very long refueling intervals (15–20 years), with refueling accomplished by replacement of the cassette core or reactor module.

The other two versions of this design are a modular system rated at 300–400 MWe, and a large plant rated at 1200 MWe. The different power options for this design are intended to fill different needs or opportunities in the power market, and be economically competitive with comparable alternative power sources.

6.2.4.6 Molten-Salt Reactor

The molten-salt reactor (MSR), shown in Figure 6.26, produces power by circulating a molten salt and fuel mixture through graphite-core flow channels. The slowing down of neutrons by the graphite moderator in the core region provides the epithermal neutrons necessary to produce the fission power for sustained operation of the reactor. The heat from the reactor core is then transferred to a secondary system through an intermediate heat exchanger and then through a tertiary heat exchanger to the power conversion system that produces the electric power. The circulating coolant flow for this design is a mixture of sodium, uranium, and zirconium fluorides. In a closed fuel cycle, actinides such as plutonium can be efficiently burned by adding these constituents to the liquid fuel without the need for special fuel fabrication.

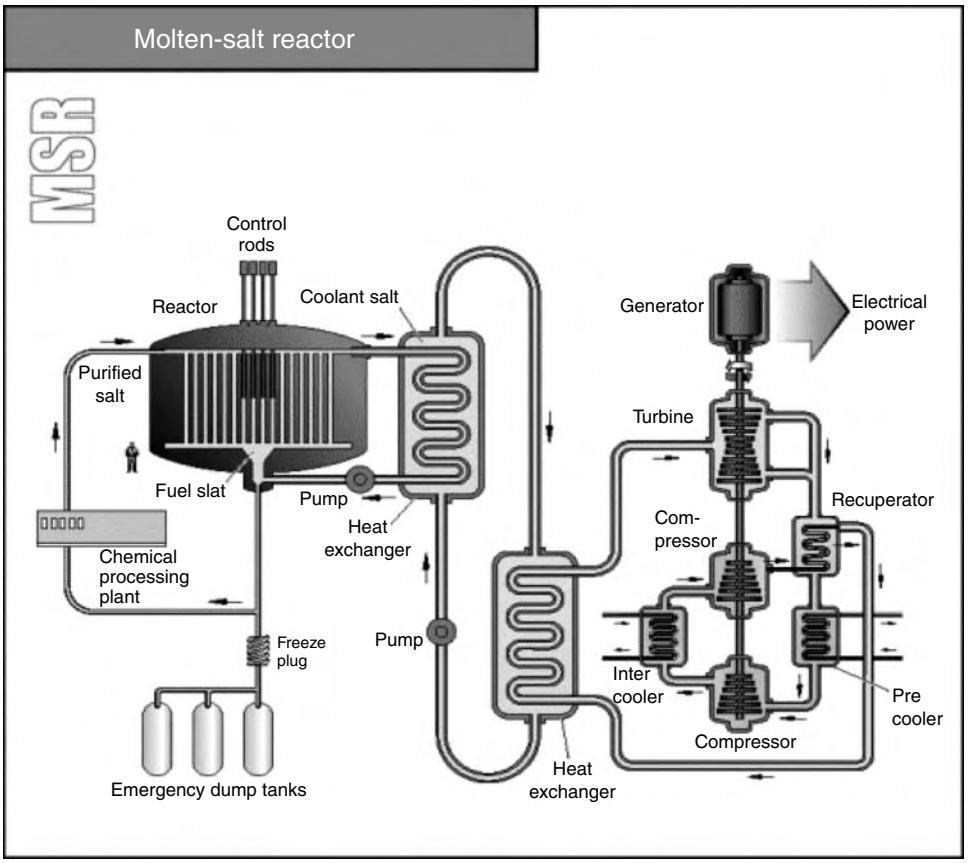

FIGURE 6.26 Molten-salt reactor. (From US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, 2002. *A Technology Roadmap for the Generation IV Nuclear Energy Systems*.)

The reference design for this concept is a 1000 MWe power plant with a coolant outlet temperature of 700°C. To achieve higher thermal efficiencies for this concept, coolant outlet temperatures as high as 800°C may also be possible.

6.2.5 Fuel Cycle

The process of following the fuel material from the uranium or thorium mine through processing and reactor operation until it becomes waste is called the fuel cycle for nuclear systems. After a discussion of the fuel cycle in general, the fuel cycle will be examined by looking at uranium and thorium resources, mining and milling, enrichment, reactor fuel use, spent fuel storage, nuclear materials transportation, and reprocessing. Nuclear waste will be addressed in a separate section.

General discussion of the fuel cycle will often include the terms "open" or "closed." The open fuel cycle is also called the *once-through* cycle. In the once-through fuel cycle, the uranium fuel is fabricated and run through the reactor once and then disposed of as waste. There is no reprocessing of the fuel. In the closed cycle, the fuel is reprocessed after leaving the reactor so that it can be reused to improve overall fuel utilization.

In the open cycle, the fuel is introduced into the reactor for one to two years. It is then removed and placed into long-term storage for eventual disposal. The impact of this cycle is the waste of about 95% of the energy contained in the fuel. The U.S. adopted the open cycle in 1977 when President Carter issued an executive order to stop reprocessing as a part of the fuel cycle. Canada has also adopted the open cycle.

The closed cycle was envisioned when the development of nuclear power began. The uranium and plutonium removed from reactors would be reprocessed and returned to reactors as fuel. Breeder reactors would be used to breed additional plutonium for use in thermal reactors. Thorium could also be used as a breeding material to generate ^{233}U as a reactor fuel. The intent of the closed fuel cycle was to maximize the use of available reactor fuel resources while minimizing waste generated by operating reactors.

Currently, reprocessing is used in Europe and Japan, but the benefits of the closed cycle have not been fully realized because there has only been limited use of the separated plutonium. As discussed above, the U.S. and Canada, for reasons described later, have not pursued closed cycle reprocessing of spent fuel. As a result, only a small fraction of the available fuel resources are utilized, and disposal of large quantities of potentially usable spent fuels has become a major issue for the U.S. nuclear industry.

6.2.5.1 Uranium and Thorium Resources

Uranium is a common material in the earth's crust. It is also present in sea water. Thorium is about three times more plentiful then uranium. Typical concentrations of uranium measured in parts per million (ppm) are shown in Table 6.8.

The amount of recoverable uranium is dependent upon the price. As the price increases, more material is economically recoverable. Also, more exploration will occur and it is likely that additional orebodies will be discovered. An orebody is defined as an occurrence of mineralization from which the metal, in this case

TABLE 6.8 Typical Concentrations of Uranium

Source	Uranium Concentration (ppm)
High-grade ore: 2% U	20,000
Low-grade ore: 0.1% U	1000
Granite	4
Sedimentary rock	2
Earth's continental crust (avg)	2.8
Seawater	0.003

Source: From World Nuclear Association, Supply of Uranium, http://www.world-nuclear.org, 2004.

TABLE 6.9 Known Recoverable Resources of Uranium

Country	Tons of Uranium	Percentage of Total
Australia	989,000	28
Kazakhstan	622,000	18
Canada	439,000	12
South Africa	298,000	8
Namibia	213,000	6
Brazil	143,000	4
Russian Federation	158,000	4
U.S.A	102,000	3
Uzbekistan	93,000	3
World total	3,537,000	—

Source: From World Nuclear Association, Supply of Uranium, http://www.world-nuclear.org, 2004.

uranium, can be recovered economically. Because of the uncertainties of price and its impact on exploration, any statement of recoverable amounts of uranium is simply a picture at an instant in time and is likely to change many times in the future. There is also a store of highly enriched uranium that is being recovered as nuclear weapons are dismantled. In addition, there are millions of tons of ^{238}U that are the results of previous enrichment activities around the world. The ^{238}U can be blended with highly enriched uranium or plutonium to make fuel for nuclear power plants. The ^{238}U can also be used to breed plutonium in FBR fuel cycles.

Table 6.9 presents a list of recoverable resources of uranium. The table is taken from information gathered by the World Nuclear Association from other sources and was generated in 2004.

The 3.5 Mt is enough to fuel the world's current reactors for 50 years assuming the same fuel cycles currently in use. IAEA estimates the world supply at over 14 Mt, which provides a supply exceeding 200 years at the current rate of use. This estimate does not include the uranium in phosphate deposits estimated at 22 Mt or the uranium available in seawater estimated at 1400 Mt. In addition, the ability of nuclear reactors to achieve higher burn-ups (utilize more of the uranium in the fuel) has also increased. This increases the efficiency of uranium use. Because thorium is not included in these fuel supply numbers, and as noted above is about three times as plentiful as uranium, there does not appear to be a fuel supply limitation for nuclear power in the foreseeable future.

6.2.5.2 Mining and Milling

Uranium is being mined using traditional underground and open-pit excavation technologies, and also using in situ leaching or solution-mining techniques.

Underground mining is used when the orebody is deep underground, usually greater than 120 m deep. In underground mines, only the orebody material is extracted. Underground mining is hazardous and made more so by high concentrations of radon from the radioactive decay of the uranium. Once mined, the extracted ore is sent to a mill where the uranium in the ore is concentrated.

Open-pit technology is used when the orebody is near the surface. This leads to the excavation of large amounts of material that does not contain the ore itself. The ore that is recovered is also sent to a mill for further processing.

Solution mining involves the introduction of an aqueous solution into the orebody. The solution, oxygenated ground water, is pumped into the porous orebody and the uranium is dissolved. The uranium-rich solution is then extracted and sent to the mill for further processing.

The milling process for the solid ore material involves crushing the ore and then subjecting it to a highly acidic or alkaline solution to dissolve the uranium. Mills are normally located close to the mining activity and a single mill will often support several mines. The solution containing the uranium goes through a precipitation process that yields a material called *yellow cake*. The yellow cake contains about

80% uranium oxide. The yellow cake is packaged and sent to a conversion and enrichment facility for further processing.

6.2.5.3 Conversion and Enrichment

Prior to entering the enrichment process, the impure U_3O_8 is converted through a series of chemical processing steps to UF_6. During these processes, the uranium is purified. Conversion facilities are operating commercially in the U.S., Canada, France, the U.K., and Russia. UF_6 is a solid at room temperature but converts to its gaseous form at moderate temperature levels, making the compound suitable for use in the enrichment process. UF_6 is very corrosive and reacts readily with water. It is transported in large cylinders in the solid state.

Conversion of the U_3O_8 to UO_2 is also done at conversion facilities. The natural UO_2 is used in reactors such as the CANDU that do not require enriched uranium as fuel.

The first enrichment facilities were operated during the 1940 s. The electromagnetic isotope-separation process was used to separate the ^{235}U used in the first atomic bomb. The process used a magnetic field to separate the ^{235}U from the ^{238}U. As the ions were accelerated and turned, they moved differently because of the difference in their masses. Multiple stages were required and the process was very difficult to run efficiently; it was therefore soon abandoned.

Today, only two processes—gaseous diffusion and gas centrifugation—are used commercially. The capacity of enrichment plants is measured in separative work units (SWU). The SWU is a complex term that is dependent on the amount of uranium that is processed, and the concentration of ^{235}U in the product and in the tails. It is a measure of the amount of energy used in the process.

The first commercial enrichment was carried out in large gaseous diffusion plants in the U.S. It has also been used in Russia, the U.K., France, China, and Argentina. Today, operating plants remain in the U.S., France, and China, with a total nominal capacity of 30 million SWU.

In the gaseous diffusion process, UF_6 is pumped through a series of porous walls or membranes that allow more of the light ^{235}U to pass through. Because the lighter ^{235}U particles travel faster then the heavier ^{238}U particles, more of them penetrate the membrane. This process continues through a series of membranes with the concentration of ^{235}U increasing each time. For commercial reactor fuel, the process continues until the ^{235}U concentration is 3%–5%. The slower ^{238}U particles are left behind and collect as a product referred to as *tails*. The tails have a reduced concentration of ^{235}U and are commonly referred to as *depleted uranium*. This process uses a very large amount of energy and thus is very expensive to operate.

In the centrifuge enrichment process, the gaseous UF_6 is placed in a high-speed centrifuge. The spinning action forces the heavier ^{238}U particles to the outside while the lighter ^{235}U particles remain closer to the center. To obtain the enrichment required for power reactor fuel, many stages of separation are required. The arrangement is know as a *cascade*. Again, the process is continued until the ^{235}U concentration is 3%–5%. The centrifuge process uses only about 2% of the energy required by gaseous diffusion.

Table 6.10 shows the location and size of enrichment facilities around the world.

6.2.5.4 Fuel Fabrication and Use

Following enrichment, the UF_6 is shipped to a fuel fabrication facility. Here, the UF_6 is converted to UO_2 and pressed into cylindrical ceramic pellets. The pellets are sintered, heated to high temperature, and inserted in the fuel cladding tubes. The tubular material is zircaloy, an alloy of zirconium. The tubes are sealed forming fuel rods that are assembled into fuel assemblies and shipped to a reactor for use. All of the dimensions of the pellets and fuel rods are very carefully controlled to assure uniformity throughout the fuel assemblies.

The primary hazard in the fabrication facility is the potential for an accidental criticality because they are working with enriched uranium. Therefore, all of the processing quantities and the dimensions of the processing vessels must be controlled. This must be done even with low-enriched uranium.

TABLE 6.10 Location, Size, and Type of Enrichment Facilities Around the World

Country	Owner/Controller	Plant Name/Location	Capacity (1000 SWU)
		Gaseous Diffusion Plants	
China	CNNC	Lanzhou	900
France	EURODIF	Tricastin	10,800
		Paducah, KY	11,300
United States	U.S. Enrichment Corporation	Portsmouth, OH (Closed since May, 2001)	7400
Subtotal			30,400
		Centrifuge Plants	
		Hanzhong	500
China	CNNC	Lanzhou	500
Germany	Urenco	Gronau	1462.5
	JNC	Ningyo Toge	200
Japan	Japan Nuclear Fuel Limited (JNFL)	Rokkasho-mura	1050
The Netherlands	Urenco	Almelo	1950
Pakistan	Pakistan Atomic Energy Commission (PAEC)	Kahuta	5
		Ural Electrochemical Integrated Enterprise (UEIE), Novouralsk	7000
		Siberian Chemical Combine (SKhK), Seversk	4000
Russia	Minatom	Electrochemical Plant (ECP), Zelenogorsk	3000
		Angarsk Electrolytic Chemical Combine (AEKhK), Angarsk	1000
United Kingdom	Urenco	Capenhurst	2437.5
Subtotal			23,105
Total			53,505

Source: From WISE Uranium Project, World Nuclear Fuel Facilities, http://www.wise-uranium.org, 2005.

A typical 1000-MWe reactor will use about 27 tons of UO_2 each year. Typical burn-up in current reactors is 33 GWd/t of uranium fed to the reactor. The energy available from the fission of uranium is 1 MW/g of uranium or 1000 GW/t. Using these numbers, the actual amount of uranium burned is only 3%–5%. This means that the unused energy available from the spent fuel, if it could be completely burned, is over 95%. During the operation of the reactor, some of the ^{238}U is converted to plutonium, which also contributes to the thermal energy of the reactor.

Advanced fuel use in reactors is estimated to be up to 200 GWd/t. In this case, about 80% of the energy available from the uranium remains in the spent fuel. These facts are the driving force behind the questions regarding reprocessing. In the once-through fuel cycle, the spent fuel will be disposed of as waste. In the closed cycle, the spent fuel is reprocessed and the remaining uranium and also the plutonium are recovered.

6.2.5.5 Reprocessing

In the 1940 s, reactors were operated solely for the production of plutonium for use in weapons. The fuels from the production reactors were reprocessed to recover the plutonium. The chemical processes were developed to separate the fission products and the uranium from the plutonium. The most common process was the PUREX process. This is the process that is used today by countries that reprocess power reactor fuels.

The purpose of reprocessing is to recover the uranium and plutonium in the spent fuel. As discussed above, these materials contain a large amount of potential energy if they are reused as reactor fuel. Plutonium separated in the PUREX process can be mixed with uranium to form a MOX fuel. Plutonium from the dismantlement of weapons can be used in the same way.

The potential availability of separated plutonium is seen by some as a potential mechanism for the proliferation of nuclear weapons. This was the basis of the U.S. decision to halt reprocessing. In the 1970 s, research began into methods for modifying the chemical process so that the plutonium and uranium would remain together at the end of the process. In this method, called coprocessing, the short-lived fission products would be separated and the remaining uranium, plutonium, and other actinide elements would remain together. This remaining mixture would be highly radioactive, but could be remotely processed into new reactor fuel. A blend of fast neutron and thermal reactors could be used to maximize the use of this material.

The current worldwide reprocessing capability is shown in Table 6.11. These facilities all use the PUREX technology. More then 80,000 tons of commercial fuel have been reprocessed in these facilities.

Three processes are considered to be mature options for reprocessing fuel: PUREX, UREX+, and pyroprocessing. Each of these processes has certain advantages and disadvantages.

TABLE 6.11 World Commercial Reprocessing Capacity

Type of Fuel	Location	Tons/year
LWR fuel	France, La Hague	1700
	U.K., Sellafield (THORP)	900
	Russia, Ozersk (Mayak)	400
	Japan	14
	Subtotal	3000
Other nuclear fuels	U.K., Sellafield	1500
	India	275
	Subtotal	1750
Civilian capacity	Total	4750

Source: From Uranium Information Centre, Nuclear Issues Briefing Paper 72, Processing of Used Nuclear Fuel, http://www.uic.com.au, 2005.

6.2.5.5.1 PUREX

The PUREX process is the oldest and most common reprocessing option. It uses liquid–liquid extraction to process light-water reactor spent fuel. The spent fuel is dissolved in nitric acid, and then the acid solution is mixed with an organic solvent consisting of tributyl phosphate in kerosene. The uranium and plutonium are extracted in the organic phase and the fission products remain in the aqueous phase. Further processing allows the separation of the uranium and plutonium. The advantage of this process is the long-term experience with the process. The disadvantage is that it cannot separate fission products such as technetium, cesium, and strontium, nor can it separate actinides such as neptunium, americium, and curium.

6.2.5.5.2 UREX+

The UREX+ process is a liquid–liquid extraction process like PUREX. It can be used for light-water reactor fuels and it includes additional extraction steps that allow separation of neptunium/plutonium, technetium, uranium, cesium/strontium, americium, and curium. The advantage of this process is that it meets the requirements for continuous recycle in light-water reactors and it builds on current technology. The disadvantage is that it cannot be used to process short-cooled fuels and it cannot be used for some specialty fuels being developed for advanced reactors.

6.2.5.5.3 Pyroprocessing

This process was developed and tested at Experimental Breeder Reactor-2 (EBR-2) by Argonne National Laboratory in the U.S. It is an electrochemical process rather then a liquid–liquid extraction process. Oxide fuels are first converted to metals to be processed. The metallic fuel is then treated to separate uranium and the transuranic elements from the fission products. The advantage of this process is the ability to process short-cooled and specialty fuels designed for advanced reactors. The disadvantage is that it does not meet the requirements for continuous recycle from thermal reactors; however, it is ideal for fuel from fast-neutron reactors.

6.2.5.6 Spent-Fuel Storage

Spent fuel is routinely discharged from operating reactors. As it is discharged, it is moved to the spent-fuel storage pool that is an integral part of the reactor facility. Reactors are built with storage pools that will hold fuel from many years of operation. The pools are actively cooled by circulating cooling water. The fuel stored at many of the older reactors is reaching the capacity of the on-site storage pools. At this point, the fuel is being transferred to dry storage. Dry storage takes place in large metal or concrete storage facilities. These dry facilities are passively cooled by the air circulating around them.

6.2.5.7 Spent-Fuel Transportation

Spent fuel is transported in large engineered containers designated as type-B containers (casks). The casks provide shielding for the highly radioactive fuel so that they can be safely handled. They are constructed of cast iron or steel. Many of them use lead as the shielding material. They are also designed to protect the environment by maintaining their integrity in the case of an accident. They are designed to withstand severe accidents, including fires, impacts, immersion, pressure, heat and cold, and are tested as part of the design certification process.

Casks have been used to transport radioactive materials for over 50 years. The IAEA has published advisory regulations for safe transportation of radioactive materials since 1961. Casks are built to standards designed to meet the IAEA advisory regulations specified by licensing authorities such as the NRC in the U.S.

Spent fuel is shipped from reactor sites by road, rail, or water. The large casks can weigh up to 110 tons and hold about 6 tons of spent fuel. Since 1971, about 7000 shipments of spent fuel (over 35,000 tons) have been transported over 30 million km with no property damage or personal injury, no breach of containment, and a very low dose rate to the personnel involved.

6.2.6 Nuclear Waste

Radioactive wastes are produced throughout the reactor fuel cycle. The costs of managing these wastes are included in the costs of the nuclear fuel cycle and thus are part of the electricity cost. Because these materials are radioactive, they decay with time. Each radioactive isotope has a half life, which is the time it takes for half of the material to decay away. Eventually, these materials decay to a stable nonradioactive form.

The process of managing radioactive waste involves the protection of people from the effects of radiation. The longer lived materials tend to emit alpha and beta particles. It is relatively easy to shield people from this radiation but if these materials are ingested the alpha and beta radiation can be harmful. The shorter lived materials usually emit gamma rays. These materials require greater amounts of shielding.

6.2.6.1 Types of Radioactive Wastes

The strict definitions of types of radioactive waste may vary from country to country. In the following discussion, the more generally accepted terminology will be used.

6.2.6.1.1 Mine Tailings

Mining and milling of uranium produces a sandy type of waste that contains the naturally occurring radioactive elements that are present in uranium ore. The decay of these materials produces radon gas that must be contained. This is often accomplished by covering the tailings piles with clay to contain the radon gas. Technically, tailings are not classified as radioactive waste.

6.2.6.1.2 Low-Level Wastes

Low-level wastes (LLW) is generated from medical and industrial uses of radioactive materials as well as from the nuclear fuel cycle. In general, these wastes include materials such as paper, clothing, rags, tools, filters, soils, etc., that contain small amounts of radioactivity. The radioactivity tends to be short-lived. These materials generally do not have to be shielded during transport and they are suitable for shallow land burial. The volume of these materials may be reduced by compacting or incinerating prior to disposal. They make up about 90% of the volume of radioactive waste but contain only about 1% of the radioactivity of all the radioactive waste.

6.2.6.1.3 Intermediate-Level Wastes

Intermediate-level wastes (ILW) are generated during the operation of nuclear reactors, in the reprocessing of spent fuel, and from the decommissioning of nuclear facilities. These materials contain higher amounts of radioactivity and generally require some shielding during storage and transportation. Intermediate-level wastes is generally made up of resins, chemical sludges, fuel cladding, and contaminated materials from decommissioned nuclear facilities. Some of these materials are processed before disposal by solidifying them in concrete or bitumen. They make up about 7% of the volume and have about 4% of the radioactivity of all the radioactive waste.

6.2.6.1.4 High-Level Wastes

High-level wastes (HLW) is generated in the operation of a nuclear reactor. This waste consists of fission products and transuranic elements generated during the fission process. This material is highly radioactive and it is also thermally hot so that it must be both shielded and cooled. It accounts for 95% of the radioactivity produced by nuclear power reactors.

6.2.6.1.5 Managing HLW from Spent Fuel

The form of HLW from spent fuel is either the spent fuel itself or the waste products from reprocessing. The level of radioactivity from spent fuel falls to about one thousandth of the level it was when removed from the reactor in 40–50 years. This means the heat generated is also greatly reduced.

Currently, 270,000 tons of spent fuel are in storage at reactor sites around the world. An additional 12,000 tons are generated each year and about 3,000 tons of this are sent for reprocessing.

When spent fuel reprocessing is used, the uranium and plutonium are first removed during reprocessing, and then the much smaller volume of remaining HLW is solidified using a vitrification process. In this process, the fission products are mixed in a glass material, vitrified in stainless steel canisters and stored in shielded facilities for later disposal.

High-level waste will eventually be disposed of in deep geologic facilities. Several countries have selected sites for these facilities and they are expected to be commissioned for use after 2010.

6.2.6.1.6 Managing Other Radioactive Wastes

Generally, ILW and LLW are disposed of by burial. Intermediate-level wastes generated from fuel reprocessing will be disposed of in deep geological facilities. Some low-level liquid wastes from reprocessing plants are discharged to the sea. These liquids include some distinctive materials such as ^{99}Tc that can be discerned hundreds of kilometers away. Such discharges are tightly controlled and regulated so that the maximum dose any individual receives is a small fraction of natural background radiation.

Nuclear power stations and reprocessing facilities release small quantities of radioactive gases to the atmosphere. Gases such as ^{85}Kr and ^{133}Xe are chemically inert, and gases such as ^{131}I have short half-lives. The net effect of these gases is too small to warrant further consideration.

Table 6.12 provides a summary of waste management adopted by countries throughout the world.

6.2.7 Nuclear Power Economics

Any discussion of the economics of nuclear power involves a comparison with other competitive electric generation technologies. The competing technologies are usually coal and natural gas.

Nuclear power costs include capital costs, fuel cycle costs, waste management costs and the cost of decommissioning after operation. The costs vary widely depending on the location of the generating plant. In countries such as China, Australia and the U.S. coal remains economically attractive because of large accessible coal resources. This advantage could be changed if a charge is made on carbon emissions. In other areas nuclear energy is competitive with fossil fuels even though nuclear costs include the cost of all waste disposal and decommissioning.

As previously stated, nuclear power costs include spent fuel management, plant decommissioning, and final waste disposal. These costs are not generally included in the costs of other power generation technologies.

Decommissioning costs are estimated to be 9%–15% of the initial cost of a nuclear plant. Because these costs are discounted over the life of the plant, they contribute only a few percent to the investment cost of the plant and have an even lower impact on the electricity generation cost. This impact in the U.S. is about 0.1–0.2 cent/kWh or about 5% of the cost of electricity produced.

Spent-fuel interim storage and ultimate disposal in a waste repository contribute another 10% to the cost of electricity produced. This cost is reduced if the spent fuel is disposed of directly. This does not account for the energy that could be extracted from the fuel if it was reprocessed.

Costs for nuclear-based electricity generation have been dropping over the last decade. This reduction in the cost of nuclear-generated electricity is a result of reductions in nuclear plant fuel, operating costs, and maintenance costs. However, the capital construction costs for nuclear plants are significantly higher than coal- and gas-fired plants. Because the capital cost of nuclear plants contribute more to the cost of electricity than coal- or gas-fired generation, the impact of changes in fuel, operation costs, and maintenance costs on the cost of electricity generation is less than those for coal- or gas-fired generation.

One of the primary contributors to the capital cost of nuclear plants has been the cost of money used to finance nuclear plant construction. The financing costs increase when the time required to license and construct a plant increases. Two factors are leading to the reduction in this portion of the cost. First, especially in the U.S., the licensing process is changing so that a plant receives both the construction permit and the operating license prior to the start of construction. Under this process, there is no large investment in plant hardware prior to completion of a significant portion of the licensing process, leading to a reduction in time required for the plant to begin producing revenue. Second, the new generation of nuclear

TABLE 6.12 Waste Management Policies for Spent Fuel for Countries Throughout the World

Country	Policy	Facilities and Progress Toward Final Disposition
Belgium	Reprocessing	Central waste storage and underground laboratory established Construction of repository to begin about 2035
Canada	Direct disposal	Underground repository laboratory established Repository planned for use 2025
China	Reprocessing	Central spent fuel storage in LanZhou
Finland	Direct disposal	Spent fuel storages in operation Low and intermediate-level repositories in operation since 1992 Site near Olkiluoto selected for deep repository for spent fuel, from 2020
France	Reprocessing	Two facilities for storage of short-lived wastes Site selection studies underway for deep repository for commissioning 2020
Germany	Reprocessing but moving to direct disposal	Low-level waste sites in use since 1975 Intermediate-level wastes stored at Ahaus Spent fuel storage at Ahaus and Gorleben High-level repository to be operational after 2010
India	Reprocessing	Research on deep geological disposal for HLW
Japan	Reprocessing	Low-level waste repository in operation High-level waste storage facility at Rokkasho-mura since 1995 Investigations for deep geological repository begun, operation from 2035
Russia	Reprocessing	Sites for final disposal under investigation Central repository for low and intermediate-level wastes planned from 2008 Central interim HLW store planned for 2016
South Korea	Direct disposal	Central low- and ILW repository planned from 2008 Investigating deep HLW repository sites Low and intermediate-level waste repository in operation
Spain	Direct disposal	Final HLW repository site selection program for commissioning 2020 Central spent fuel storage facility in operation since 1985
Sweden	Direct disposal	Final repository for low to intermediate waste in operation since 1988 Underground research laboratory for HLW repository Site selection for repository in two volunteered locations Central interim storage for high-level wastes at Zwilag since 2001 Central low and intermediate-level storages operating since 1993
Switzerland	Reprocessing	Underground research laboratory for high-level waste repository with deep repository to be finished by 2020 Low-level waste repository in operation since 1959
United Kingdom	Reprocessing	High-level waste is vitrified and stored at Sellafield Underground HLW repository planned
U.S.A	Direct disposal	Three low-level waste sites in operation 2002 decision to proceed with geological repository at Yucca Mountain

Source: From World Nuclear Association, Waste Management in the Nuclear Fuel Cycle, http://www.world-nuclear.org, 2004.

plants will be highly standardized and modularized. This will allow a significant reduction in the time required to construct a new plant. It is estimated that the time from the start of construction to the start of operation will be reduced from nearly 10 years to 4–5 years. This will have a significant impact on capital costs.

The reduced capital costs associated with the licensing and construction of new nuclear power plants, and the fact that nuclear power is inherently less susceptible to large fluctuations in fuel costs, have made nuclear power an attractive energy option for many countries seeking to diversify their energy mix in the face of rising fossil fuel costs.

6.2.8 Conclusions

The development of nuclear power began after World War II and continues today. The first power-generating plants were constructed in the late 1950 s. During the 1960 s and 1970 s, there was a large commitment to nuclear power until the accidents occurred at Three Mile Island in 1979 and then at Chernobyl in 1986. The new safety requirements and delays caused by these accidents drove up the costs and at the same time caused a loss of public acceptance. In the U.S., many plant orders were canceled; in other countries, entire nuclear programs were canceled.

The ability of nuclear reactors to produce electricity economically and safely without the generation of greenhouse gasses has revitalized the interest in nuclear power as an alternative energy source. Many lessons have been learned from the operation of current power plants that have allowed the safety of newly designed plants to be improved. This, coupled with the desire of many nations to develop secure energy sources and a diversity of energy options, have resulted in the continuing development of a whole new generation of nuclear plants to meet future energy needs.

Nuclear power is also not as susceptible to fluctuation in fuel costs as petroleum and natural gas. As shown, the supply of uranium is very large, and if it is supplemented with thorium, the fuel supply is seemingly unlimited. This drives many other aspects of the fuel cycle, such as the choice between closed and open fuel cycles discussed earlier. For example, because of the large uranium resource and the fears of nuclear proliferation, the once-through (open) fuel cycle is favored by many. This will require large deep geologic waste repositories for the disposal of large quantities of spent fuel. However, when reprocessing is included in the closed fuel cycle, the amount of needed repository space is greatly reduced, but the expense of operation is increased. Finally, it may be possible to essentially eliminate the need for repositories by utilizing advanced fuel cycles that utilize almost all of the energy available in the uranium and the other transuranic products of reactor operation.

The need for energy and the use of electricity as the primary energy source for the end user will drive the increase in electricity generation around the world. The drive to reduce the production of greenhouse gases will contribute to a wider use of nuclear power for electricity generation. The recognition that nuclear power can safely provide large base-load generating capacity at a reasonable cost using known technologies will also be a major factor in its future development.

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