NUCLEAR POWER FACILITIES, SAFETY

1. Introduction

Nuclear energy is a principal contributor to the production of the world's electricity. As shown in Table , many countries are strongly dependent on nuclear energy. For some countries, more than one-half of the electricity is generated by nuclear means . There are 441 nuclear power plants in operation, generating 16% of the world's electricity and 25 new plants are under construction.. Of these plants, 104 contribute 20% of the electricity in the United States in 2004.

Safety has played a dominant role in achieving the ability to generate electricity by nuclear means. When energy is released from the atom by fission, ie, breaking apart the nucleus of a heavy element, such as uranium or plutonium, two lighter nuclei called fission products are also produced. Most of these fission products are radioactive. Thus, as energy is generated, highly radioactive materials, which can be harmful to living organisms if not kept under strict control, are produced.

Safety provisions establish strict limitations on allowable levels of radiation exposure, assuring that neither the public nor the plant workers are harmed as a result of operation of the nuclear power plant. Control of radioactive materials must be effected by (*1*) careful design and testing of the integrity and reliability of the components and systems that contain and control the radioactive materials in the nuclear power plant; (*2*) fabrication, installation, and construction of these components to meet high quality standards; and (*3*) thorough training of plant operators to assure that the systems and components function as designed and integrity is maintained.

The design of safety systems and components must also provide tolerance for human fallibility, ie, protect against design mistakes, equipment failure, and operational error. Redundancy and diversity are provided for key safety functions. If one component fails, another component, either a duplicate or in different form, is available to carry out that function. In this fail-safe design, a failure of any device should lead to a stable condition. Automatic devices are provided to shut down or reduce power or inject coolant in the event of component failure or operator error. Finally, passive safety features are provided. The design utilizes physical laws to intrinsically bring the system to a stable state when an abnormal condition ensues. An important example of such a passive safety feature is a nuclear fuel system, which is designed to use an intrinsic characteristic that causes the fission reaction rate to reduce if power or temperature increases. With such designs, runaway power excursions are prevented by the laws of physics, not by the operation of equipment or the actions of people. Advanced designs are being developed that utilize intrinsic characteristics to provide emergency cooling in the event of a loss of coolant accident.

Whereas these design measures provide the primary assurance of protection from the harmful effects of radiation, additional protection is provided in the unlikely event that the integrity of the systems or components breaks down. The entire portion of the nuclear plant containing radioactive material is enclosed in a strong containment building. If a release of radioactivity from the plant were to occur, the radioactive material would be captured within the containment building. As a further precaution, these processes are subject to continual cross-checks by people separate from those engaged in the design, fabrication, construction, and operational processes. Cross-checks in the form of design reviews, inspections, operations, and safety audits are carried out. The U.S. nuclear industry has set up the Institute of Nuclear Power Operations (INPO) to establish operations and training standards and to audit nuclear plant operations for compliance with those standards. In addition, a totally independent government body responsible to the public, the U.S. Nuclear Regulatory Commission (NRC), establishes overriding safety regulations, monitors compliance with them, and investigates all abnormal events that threaten safety.

Safety provisions have proven highly effective. The nuclear power industry in the Western world, ie, outside of the former Soviet Union, has made a significant contribution to electricity generation while surpassing the safety record of any other principal industry. In addition, the environmental record has been outstanding. Nuclear power plants produce no combustion products such as sulfuric and nitrous oxides or carbon dioxide (qv), which are significant causes of air pollution and greenhouse gas emissions (see AIR POLLUTION; ATMOSPHERIC MODELING).

The accident at the Three Mile Island (TMI) plant in Pennsylvania in 1979 caused partial melting of the reactor fuel and raised major concerns as to the safety of nuclear power. No harm from radiation resulted to TMI workers, to the public, or to the environment, although the accident caused the loss of a $2 billion investment in the plant. The serious nature of the accident led to recommendations by the NRC for many safety and environmental improvements . The President's Report on the accident recommended some changes to the regulations but the regulations have remained basically the same. The industry made major moves toward more effective risk management. Large scope probabilistic risk assessments (PRAs), building on the pioneering Reactor Safety Study , were applied to identify safety deficiencies. The INPO was formed and initiated safety audits, with independent compliance auditing by NRC. Major improvements were made in training, management practices, the competence of plant operators and management, and the timely reporting to all plants of safety significant operating events. Critical lesssons learned during this period were increased recognition that the reliability of systems supporting normal operations is important to safety and that certain initiating events, such as intersystem loss of coolant and steam generator tube ruptures, can by-pass multiple fission product barriers.

The accident at the Chernobyl plant in the Ukraine in 1986 , on the other hand, caused the immediate deaths of 31 workers from high doses of radiation, led to radioactive contamination of large areas and may have caused thyroid cancer in $\sim$2000 children from low level radiation. Apart from the increase in thyroid cancer, no increases in overall cancer incidence or mortality have been observed that could be attributed to ionizing radiation . This latter accident was unique to Soviet designed Chernobyl-type reactors that did not have the intrinsic protection against a runaway power excursion nor a containment building, as was required on the TMI plant (10–12).

2. Basic Safety Principles

The three fundamental safety objectives advocated by the International Atomic Energy Agency (IAEA) for all nuclear power plants worldwide are (*1*) to protect individuals, society, and the environment by establishing and maintaining in nuclear power plants an effective defense against radiological hazards; (*2*) to ensure in normal operation that radiation exposure within the plant as well as that resulting from any release of radioactive material from the plant is kept as low as reasonably achievable (ALARA) and below prescribed limits, and (*3*) to prevent accidents in nuclear plants with high confidence; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and (*4*) to provide mitigation of the extent of radiation exposures owing to severe accidents; so that the likelihood of serious radiological consequences is extremely small.

IAEA also defines the fundamental responsibilities for nuclear power plant safety as ultimately resting with the operating organization. It is the operators who provide the primary assurance of public safety. Designers, suppliers, constructors, and regulators are also responsible for their separate activities. Responsibility is reinforced by the establishment of a safety culture, ie, "the personal dedication and accountability of all individuals engaged in any activity which has a bearing on the safety of nuclear power plants" . Safety design of nuclear power plants is founded on the defense-in-depth concept, which provides multiple levels of protection to both the public and the workers, in the form of physical barriers and levels of implementation of the associated defenses. Each of the multiple physical barriers prevents the release of radioactive materials, but all envelop a given number of the others so that if an inner barrier fails, the next outer barrier holds back the radioactive material .

Figure shows both the physical barriers and the multiple levels of protection in conceptual form. The first barrier is the nuclear fuel rod that heats up as fission occurs. The fuel rod is made up of corrosion-resistant ceramic and uranium oxide pellets, placed in zorconium alloy tubes called cladding (see NUCLEAR REACTOR TYPES, NUCLEAR FUEL RESERVES), comprising the second barrier that surrounds the nuclear fuel. The cladding is made of a metal alloy, usually Zircaloy, which is highly corrosion resistant. The third barrier is a steel pressure boundary, consisting of the reactor pressure vessel. All the core is placed within this vessel, as is the main coolant piping that contains the cooling water that takes the heat from the fuel and transfers it to provide the electricity. The fourth barrier is the containment building, a massive reinforced concrete or steel structure within which is placed the nuclear portion of the power plant's generation system, called the nuclear steam supply system.

The reliability of the physical barriers is assured by implementation of multiple levels of defense-in-depth, characterized by a sequence of concentric design features and their related operational defenses against the release of radiation from the plant. The first level is the design, fabrication, and construction of the plant to high quality standards together with its reliable operation and maintenance within the prescribed operational bands. The second level of defense is comprised of systems and operating procedures which control abnormal conditions, ie, transients beyond the prescribed bands, so that the basic integrity of the system is maintained. The third level of defense-in-depth is the provision of backup systems and emergency operating procedures that become operative in the event that there is a loss of integrity or a loss of a basic function of the normal nuclear systems, assuring that the radioactive material is not released from the nuclear systems. In each of these first three levels, a separate layer of multiple protection is provided through redundancy and diversity.

The fourth level of defense-in-depth is activated if all of the previous levels fail and radioactivity is released from the power-generating system. This level consists of containment systems and accident management processes that prevent the dissemination of radioactivity to the atmosphere even if it is released from the nuclear systems. The fifth level is the provision for emergency planning outside the plant boundary in the highly unlikely event that all of the first four levels of defense were to fail.

The defense-in-depth process requires that each physical barrier be designed conservatively using substantial margins against failure, on-line monitoring instrumentation, off-line inspections to detect incipient failures, and highly trained operators and maintenance personnel guided by prudent procedures. In particular, the containment building is designed to withstand external assaults from earthquakes, hurricanes, tornadoes, floods, and flying objects, such as crashing airplanes. The safety of the nuclear plant and the integrity of its containment must also be maintained in the event of aggression from terrorists or saboteurs. Stringent security measures are provided at each plant to meet such a challenge. These include large around-the-clock guard forces, modern detection and alarm systems, and vehicle intrusion barriers. Strict personnel checks, emphasis on professional discipline, and the redundant, fail-safe design of the safety systems provide protection against internal sabotage. A prioritization process applies in the design of the barriers and the provisions for defense-in-depth that is based on the principle “prevention first”.

3. Safety Design

3.1. Design Features.

Design safety features are utilized at each of the concentric safety barriers. The most important of these safety features apply at the innermost barriers in what is called the reactor core. A cylindrical arrangement of bundles of nuclear fuel rods are arranged to cause fission and spaced to permit the flow of cooling water through them. Sustained fission is possible because a chain reaction can be established. When a neutron is absorbed by the uranium-235 nucleus causing it to split, 2.44 neutrons on the average are also released. Some of these additional neutrons escape or are absorbed in uranium and other materials in the fuel without causing fission, but if just one of them is absorbed in another uranium-235 nucleus, fission is self-sustained. This is called criticality or a critical mass. If an average of more than one neutron is absorbed, the rate of fission increases, and vice versa.

First Barrier. The rate of fission must be kept under strict control so as to prevent a runaway power excursion, ie, an excessive increase in fission rate. Control is carried out in two ways: one, intrinsic to the chain reaction, involves a negative coefficient of reactivity; the other is external, through use of control rods. The fission rate is dependent upon the temperature of the fuel and the temperature and density of the coolant. Fuel composition and absorber materials, ratio of fuel to coolant, and geometrical arrangement of the fuel and the fuel rods can be designed so that the fission rate decreases as temperature, coolant density, or power increases. This intrinsic feature can be designed into the fuel system, ie, the core, to cause the fission rate to slow down when temperature, steam content, or power increases. This is called a negative temperature, void, or power coefficient of reactivity. Thus, when an incipient transient in temperature or power occurs in a core having a negative coefficient of reactivity, the physical processes governing the fission rate slow the excursion down to prevent a runaway condition . This intrinsic stability is a requirement in the United States and the rest of the Western world.

The external means of controlling the fissioning rate is through use of control rods. Metal rods composed of strong neutron absorbers can reduce or cut off the chain reaction. These can be inserted into the core to reduce or stop fission or alternatively pulled out of the core to start or increase fission. The control rods are moved by remote control by the operator for normal power control. Redundant and diverse radiation detectors and temperature sensors are installed in the core to signal control rod mechanisms to insert the rods automatically so as to keep the power excursions within the allowable band whenever limiting conditions are reached. Insertion into the core to stop the chain reaction completely is rapid under these circumstances to prevent damage to the fuel. There is a chance that deficiencies in meeting the high quality standards of nuclear fuel manufacture would cause a loss of integrity of the fuel rods. To be on the alert for such an event, radiation monitors check the level of radioactivity in the coolant water to detect incipient fuel rod failure, which would show itself by leakage of fission products from the fuel rods into the coolant stream. For further assurance, on-line monitors detect incipient fuel failure.

Cutting off the chain reaction removes concern that a runaway power excursion involving rapid melting of fuel rods can occur, but does not eliminate the possibility of slow fuel melting. The fission products generated in the fuel rods during power operation continue to emit radioactive particles that are converted to thermal energy in the fuel rods. Although the energy generated is orders of magnitude smaller than that at full power, the fuel rods can slowly heat up to melting temperature unless cooling is maintained. Thus, continued reliability of both the reactor cooling system (second barrier) and back-up emergency cooling provisions (third barrier) is essential during plant shutdown.

Second Barrier. Safety design features at the second barrier involve the primary coolant circuit. These are derived from adherence to rigorous standards in the selection of materials and in conservative design of the coolant system and coolant pressure boundaries. The conservatism is provided by designing for forces, pressures, temperatures, fluid conditions, radiation levels, thermal transients, and fatigue cycles, that are higher than are expected during power operation. This difference between the design levels and the actual levels is called margin.

Margin is provided in the coolant system by designing to keep the peak temperatures in the fuel rods well below the fuel melt temperature, to keep peak coolant temperatures in a range of stable operation, to assure adequate coolant pumping capacity, and to provide reliable component cooling. To assure the integrity of the coolant pressure boundaries, margin is provided by designing these boundaries to withstand pressure surges and steady-state pressures substantially higher than the operating pressure, to prevent fatigue failure of the pressure boundaries after many thermal cycles, to maintain integrity even after substantial loss of ductility occurs for those parts of the system that are exposed to intense radiation, and to withstand the shocks and stresses caused by earthquakes of magnitudes at the upper limit of that which would be expected at the plant site.

Another design feature is the provision for on-line coolant leakage monitors that would signal incipient pressure boundary failure. Radiation monitors are installed in the containment building to detect airborne radioactivity, which would signal incipient loss of integrity of some part of the pressure boundary. Extensive inspection requirements are also stipulated. Sensitive ultrasonic devices are used to check the condition of the piping (see PIPING SYSTEMS). Samples of the reactor pressure vessel material are removed from the reactor zone periodically to be tested for ductility loss. Eddy current detectors are used to inspect the steam (qv) generator tubes for the occurrence of cracking (see NONDESTRUCTIVE EVALUATION).

Third Barrier. At the level of the third barrier, the key design feature is the provision of backup cooling systems that continue to cool the core in the event of a significant loss of integrity that would disable the normal cooling functions of the primary circuit. Separate and redundant coolant injection systems are provided (*1*) the normal coolant recharging systems which replenish the primary coolant circuit; (*2*) gas-pressurized accumulator tanks that force water under high pressure into the primary coolant circuit; and (*3*) safety coolant injection systems that pump water at both high and low pressure into the primary coolant circuit from separate reservoirs.

Fourth Barrier. The design feature of the fourth barrier is the containment building. It is designed to withstand the high temperatures, pressures, and radiation resulting from a severe accident entailing fuel meltdown. Supplementary features are utilized to reduce the consequences of such severe accident conditions: spray systems are installed to reduce the containment temperature; catalytic devices are provided to absorb airborne fission products; igniters are installed to burn off hydrogen gas emitted during an accident before the deflagration temperature is reached; interlocks and alarms are activated to assure that containment hatches are appropriately closed; and periodic testing of the leaktightness of the containment is carried out.

3.2. Operational Safety. Effective human performance is essential to achieving the aims inherent in the safety design as well as assuring reliable and economic operation. Extensive training of operators and maintenance personnel is required that is overseen by industry through the INPO in the United States and its counterpart internationally, the World Association of Nuclear Operators (WANO) . These organizations, in cooperation with the nuclear power plant operators, establish operating standards and procedures, exchange operating experience and best practices, and cooperate in peer reviews of individual plant operational capability. A continuing goal is to achieve excellence in operations and a "safety culture" that gives first priority to safety at all times.

WANO members are nuclear plant owner-operators over the entire world who have pledged to assist each other in the achievement of safe operations. There are four centers from which this international program is administered: one in the United States in Atlanta, Georgia, operated by INPO; one in Paris operated by Electricité de France; one in Moscow operated by the Ministry of Nuclear Power; and one in Tokyo operated by the Central Research Institute for the Electric Power Industry (CRIEPI). Through this mechanism, teams of operators from the United States, Europe, and Asia visit each others' plants to share safety experience and know-how. Similarly, plant personnel from Russian and Eastern European nuclear units visit European, Asian, and U.S. plants. National nuclear power safety regulators, eg, the NRC in the United States, support these same goals through regulations, operational license approvals, and compliance authority (19–22).

3.3. Assessment. It is important to verify that safety is actually being achieved by monitoring the operations of the nuclear plants. The utility is responsible for the monitoring of daily performance. Each operating shift keeps track of the equipment in and out of service and is knowledgeable, through the plant PRA, of the total risk profile with the existing equipment. The Shift Technical Advisor is responsible for monitoroing the overall status of the plant, being aware of what might go wrong and what options are available depending on what goes wrong.

In the United States, NRC and INPO perform key roles in this process. Each organization periodically visits every nuclear power plant in the United States to assess the safety of the operations. The NRC Resident Inspector at each plant monitors its daily performance for NRC. Special inspections occur on the occurrence of a major incident affecting safety. Each plant is given a performance rating, backed up by detailed critiques identifying operational strengths as well as weaknesses. The plant is required to follow up on any corrective actions indicated by these safety audits. The NRC ratings are made public and have significant power in motivating corrective action when that is needed.

Audits by INPO and the U.S. NRC are a culmination of a high degree of self-auditing by the plant operators and the utilities themselves, often assisted by special third-party safety review boards set up to help carry out safety assessments (23–25). Self-auditing reflects the fundamental reactor safety principle that the owner-operator of the plant has the ultimate responsibility for plant safety .

Another element of safety monitoring is the requirement stipulated by the NRC that each utility report any operational event which is out of the ordinary or has safety implications. These licensing event reports (LERs) are placed in the public record. Both INPO and the NRC evaluate the LERs and inform all the utilities of any event that has broad safety significance to the industry. In particular, if one of these field events is judged to be the precursor of a serious accident, all operating nuclear plants are made fully aware of its implications. Thus, steps can be taken to prevent a safety-threatening event from occurring, but if the event does occur, the plant operators would be better prepared to stop the progress of the event before the stage of a severe accident is reached. The methodology used to make these evaluations and communicate them to the nuclear plants was initially developed by the Electric Power Research Institute (EPRI), the collaborative R&D arm of the U.S. electric utilities. NRC reports that there has been substantial reductions in these indicators, eg, the number of significant events has dropped from 0.77/plant in 1988 to 0.01/plant in 2003; and the number of accident precursers/reactor year has reduced from 0.32 in 1988 to 0.01 in 2003.

A further assessment is carried out through the definition and measurement of industry-average performance indexes relating to safety. These indexes have been established by the utilities, working with INPO, EPRI, and the reactor manufacturers. Each index bears on some aspect of safe operation of the nuclear power plant, ie, industrial safety accident rate, unplanned automatic scrams, collective radiation exposure, plant capability factor, and unplanned capability loss factor. Five-year goals are established for average performance of all U.S. plants for each of these performance indexes. A substantial improvement has been made in all of these indexes since the early 1980s. The goals that were set in 1990 have now been achieved. A set of indexes similar to those devised in the United States has been developed through WANO. Measurement of performance against these indeces also shows significant improvement of reactor safety and reliability performance worldwide.

Comparative Risks. All efforts are directed toward reducing to an extremely low level the chance of a severe nuclear accident that would harm the public. The question of how low a level this should be has been addressed in the United States through a safety goal stipulated by the U.S. NRC. The goal is that (*1*) the risk of prompt fatality to an average individual in the vicinity of a nuclear power plant that might result from a reactor accident should not exceed 0.1% of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed, and (*2*) the risk of cancer fatalities to the population in the area near a nuclear power plant in operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes.

Each U.S. plant is evaluated for its ability to meet the NRC safety goal through its PRA, which estimates the probability that the chance of an accident that could initiate melting of the rector fuel, called the core damage frequency (CDF). A particularly important part of that process is the ability to pinpoint detailed weaknesses in design and operational features, in effect pinpointing weak links in the safety defense chain. The PRA assessments of every nuclear power plant in the United States contribute to the judgment of NRC that existing operating plants are adequately safe. The PRAs require a significant amount of detailed knowledge as to the causes of failure and the characteristics a severe accident would have, were one to occur .

The PRA methodology has been improved greatly over the years, with industry making substantial innovative contributions, eg, the scenario approach (modular event tree models), a basic definition of a risk framework (now adopted by NRC and other agencies), the concept of plant damage states and pinch points in modeling scenarios, comprehensive core and containment response analysis (as part of the risk model), analysis techniques that propogate uncertainty through the risk scenarios, methods of "importance ranking" of risk scenarios, and dose models to account for directional variability and other dynamic changes in atmospheric dispersion models. The PRA was also extended to the off-power and shutdown plant conditions, showing that, although some risk exists in such off-power states, at-power risks should get the emphasis in managing risks. Overall, the PRA has become an intrinsic part of the nuclear power plant operators' decision making process.

Extensive experiments and analyses of severe accident scenarios have been carried out. For example, an experimental reactor (LOFT) at the Idaho National Engineering Laboratory was driven to core melting to measure the rates of release and related release characteristics of the fission products. It has been concluded from these results that the PSA methodology is highly conservative. Not inconsequential among the experimental work has been the evaluation of the TMI accident. Although ~50% of the TMI core melted and some of the molten fuel slumped into the bottom of the reactor vessel causing damage, the molten fuel was held within the vessel.

Analyses and experimental results used to assess the consequences of a postulated severe accident have resulted in substantially reduced estimates of severe accident consequences. Comparing estimates made by the U.S. Atomic Energy Agency in 1975 with those reported by the U.S. NRC in 1990 shows that improved analyses and plant modifications have reduced the core damage frequency by a factor of 3–15, depending on reactor type. Additionally, the fractions of radioactive species that would be released are lower by a factor of 10–100,000, depending on the specific radioactive isotope.

The NRC safety goal has been evaluated by comparison to the risks from accidents incurred from other human activities (Fig.) . The safety goal and the safety record of the nuclear power industry indicate much lower societal risks from commercial nuclear power than from a wide range of other common human activities.

If the comparisons are focused on energy systems, nuclear power safety is estimated to be superior to all other electricity generation methods except for natural gas . Figure is a plot of that comparison in terms of estimated total deaths to workers and the public and includes deaths associated with secondary processes in the entire fuel cycle. The poorer safety record of the alternatives to nuclear power can be attributed to fatalities in transportation, where comparatively enormous amounts of fossil fuel transport are involved. Continuous or daily refueling of fossil fuel plants is required as compared to refueling a nuclear plant from a few truckloads only once over a period of 1–2 years. No death or serious injury has resulted from radiation exposure from commercial nuclear power plants in the United States .

4. Safety Characteristics of the Nuclear Power Plant

4.1. The Reactor. The nuclear power plant reactor types used to produce electricity worldwide are listed in Table (see NUCLEAR REACTOR TYPES). About 80% of the plants worldwide, and all of those in the United States, are light water reactors (LWR) (1). The LWR uses ordinary or light water as distinguishable from heavy or deuterated water to transfer the heat generated from fission in the nuclear fuel assemblies, called the core, to make steam .

The Chernobyl reactor type, designated RBMK, and built only in the former Soviet Union, is also cooled using ordinary water. The water is circulated through fuel tubes inserted in a large graphite block that reduces the energy of the neutrons to increase the interactions to cause fission. The remaining systems utilized worldwide are gas cooled, heavy water cooled, and liquid metal cooled. All of the nuclear reactors designed to meet Western world standards have safety features conceptually similar to those of the LWRs. Since most of the present commercial reactors are LWRs, their design features will be summarized to show the specific ways in which the overall safety concept has been developed.

Pressurized Water Reactor. Figure is a schematic of the PWR power plant. The reactor core made of the nuclear fuel assemblies is installed within the reactor pressure vessel, into which coolant water is pumped. The water passes through the core and is heated. The heated water flows through the tubes of a steam generator and transfers its heat to water on the other side of the tubes. The water turns to steam and is fed to a steam turbine causing the turbine to rotate. The turbine shaft is connected to the shaft of an electrical generator, causing the generator shaft to rotate and generate electricity. A separate tank, called the pressurizer, provides the ability to accommodate volume changes in the primary circuit as well as to maintain a constant pressure. Associated with the steam turbine are conventional systems which enhance the efficiency of heat extraction into work and recirculate the condensed water and unused steam back to the steam generator as feedwater. The reactor pressure vessel containing the core along with the reactor coolant circuit, reactor coolant pumps, pressurizer, steam generators, and emergency cooling systems are installed within the containment building. When the power plant is shut down for refueling or repair, a residual heat-removal system continues to cool the core, which would otherwise heat up from fission-product radioactive decay.

To assure continued cooling of the fuel in the event of loss of normal cooling, the emergency core cooling system is brought into play. Water can be immediately supplied from the accumulator tanks that are constantly under gas (nitrogen) pressure. A high pressure injection system can pump water from the refueling water storage tank into the reactor coolant circuit. Water can also be pumped from the refueling water storage tank to spray nozzles at the top of the containment building to keep the temperature of the containment building under control. These systems are activated automatically and are installed in multiple form to provide redundancy. Emergency electric power is provided from independent diesel generators when power from the reactor plant or off-site power is unavailable.

Control of the core is affected by movable control rods that contain neutron absorbers, soluble neutron absorbers in the coolant (called chemical shim) fixed burnable neutron absorbers in the core, and the intrinsic feature of negative reactivity coefficients. Gross changes in fission reaction rates, as well as start-up and shutdown of the fission reactions, are effected by the control rods. In a typical PWR, $\sim$ 90 control rods are used. These, inserted from the top of the core, contain strong neutron absorbers, such as boron, cadmium, or hafnium.

The most common form is a cadmium–indium–silver alloy, clad in stainless steel. The movement of the control rods is governed remotely by an operator in the control room. Safety circuitry automatically inserts the rods in the event of an abnormal power or reactivity transient.

Chemical shim control is effected by adjusting the concentration of boric acid dissolved in the coolant water to compensate for slowly changing reactivity caused by slow temperature changes and fuel depletion. Fixed burnable poison rods are placed in the core to compensate for fuel depletion. These are made of boron carbide in a matrix of aluminum oxide clad with Zircaloy. As the uranium is depleted, ie, burned up, the boron is also burned up to maintain the chain reaction. This is another intrinsic control feature. The chemical shim and burnable poison controls reduce the number of control rods needed and provide more uniform power distributions.

Boiling Water Reactor (BWR). The BWR differs from the PWR primarily in that steam is generated in the reactor core and sent directly to the steam turbine. The steam generators and secondary coolant circuit can thus be eliminated, as shown in Fig. . There are other differences associated with the core and its controls, as well as in the containment system. There are $>$ 700 fuel assemblies in a typical BWR core, each containing 62 fuel rods. The core diameter is 4.7 m, the length 3.8 m. The movable neutron-absorbing elements are in the form of blades that insert between fuel assemblies, rather than in rods that insert within the fuel assemblies in the PWR. Containment heat removal is aided in the BWR by venting any steam issuing from a pipe break, if such were to occur, to a pool of water located at the bottom of the containment building. The resultant steam condensation would then reduce the pressure and temperature within containment.

4.2. The Nuclear Fuel Cycle. Fuel for a nuclear power plant is provided and dispositioned through the nuclear fuel cycle , shown in Fig. . The disposition process has two options, recycle or once through. In the former, the used fuel is reprocessed in a chemical plant, where residual uranium and plutonium are separated from the fission products. The uranium and plutonium are then recycled to fabricate mixed uranium–plutonium oxide pellets to be used in subsequent reactor refuelings. The fission product waste is then vitrified,, encapsulated in metal casks, and sent to a permanent repository. In the once-through option, the used fuel, with its resisual uranium and plutonium, is sent directly to a permanent repository.

Although recycle is utilized in some countries, once through is currently the most commonly used option worldwide. The recycle option has been completely demonstrated in the United States but its economics have not been favorable. Moreover, concerns have been raised as to the diversion of the plutonium to weapons use. Thus, the once-through option is the only one now in use in the United States.

The safety principles and criteria used in the design and construction of the facilities that implement the nuclear fuel cycle are analogous to those that govern the nuclear power plant. The principles of multiple barriers and defense-in-depth are applied with rigorous self-checking and regulatory overview . However, the operational and regulatory experience is more limited.

One feature of reprocessing plants which poses potential risks of a different nature from those in a power plant is the need to handle highly radioactive and fissionable material in liquid form. This is necessary to carry out the chemical separations process. The liquid materials and the equipment with which it comes in contact need to be surrounded by 1.5–1.8-m thick high density concrete shielding and enclosures to protect the workers both from direct radiation exposure and from inhalation of airborne radioisotopes. Rigid controls must also be provided to assure that an inadvertent criticality does not occur. Shielding protection entails design engineering and installation similar to that provided in a nuclear power plant. Additionally, to protect against exposure to airborne radioactivity, controlled air ventilation and air cleaning is provided. The air flows progressively from clean areas to contaminated ones and is then filtered before being discharged.

The principal methods for preventing criticality are limitations on the mass of the fuel being handled, the equipment size, the concentration of nuclear material in solution, minimization of the presence of water or plastic that would reduce the margin to reaching a critical mass, and the addition of a neutron absorber, eg, cadmium or boron, either in solution or as a solid packing in vessels. Borosilicate glass rings are used as a neutron-absorber packing for tanks that contain many times the critical mass of fuel solutions. At least two and sometimes more of these independent methods usually are employed at fuel-processing facilities to prevent criticality. In addition, control of other parameters individually or in combination permit the safe handling of quantities many times the critical mass .

In plutinium fuel fabrication facilities, protection must be afforded against both inadvertent criticality and inhalation of airborne particles. Plutonium, if inhaled or ingested, is very harmful. Thus, air-flow controls are employed. Fabrication operations are carried out in glove boxes, ie, tightly sealed enclosures maintained at lower pressure than the surroundings, so that any leakage is into the glove box. Ventilation air for the boxes is cleaned through high efficiency particulate air filters. Inadvertent criticality is an even more sensitive safety issue in plutonium fuel fabrication because a criticality accident would emit lethal levels of radiation near the unshielded glove boxes. Strict control of Pu quantities, therefore, is enforced, limiting the amount of Pu handled in a single operation to less than that needed to start a chain reaction.

The sum total of risks of the nuclear fuel cycle, most of which are associated with conventional industrial safety, are greater than those associated with nuclear power plant operation (34,36–42). However, only 1% of the radiological risk is associated with the nuclear fuel cycle so that nuclear power plant operations are the dominant risk. Public perception, however, is that the disposition of nuclear waste poses the dominant risk.

4.3. Used Fuel and Radioactive Waste. The basic safety objective governing radioactive waste management is to protect the public and the workers from radiation, at a minimum meeting federal regulatory standards for maximum allowable radiation dosage. This protection is provided for both used fuel and plant radioactive wastes during the transfer and treatment processes at the plants, temporary storage of wastes at the plants, and transportation from the plants to storage sites and repositories. Plant radioactive wastes arise from the coolant systems of the nuclear power plants; from auxiliary process wastes; from the enrichment, reprocessing, and fuel fabrication facilities; and from the decommissioning of these facilities. These wastes, called low level radwastes (LLW), are of lower radiation intensity than those arising from the nuclear fuel, called high level radwastes (HLW).

The safety objective for the final storage facilities for the low level wastes and permanent repository for the used fuel or the separated high level wastes is to isolate the radioactive materials from the biosphere, ie, to package the waste in rugged containers and bury these containers in stable geologic formations far from ground water so that they do not come in contact with humans directly or indirectly. The specific means and difficulties of meeting this objective can vary in regard to whether the once through or recycling option is being utilized.

For the once-through option in the United States, the used fuel is initially stored in water pools at the reactor site, where the relatively high decay heat is removed by natural circulation of the water in the pool. After $\sim$10 years, the used fuel is moved to a dry storage facility at the reactor site where natural circulation of air provides sufficient cooling, all the while awaiting transfer to either an interim centralized storage facility or a permanent repository . Once removed from the water pool, the used fuel assemblies are placed in a stainless steel or titanium container, called a multipurpose canister, which provides an inner shell from which the fuel assembly need never be removed again. This shell is inserted into various other overpacks of concrete or steel, depending on whether the fuel is being stored on site, is being transported, or is being placed in the permanent repository.

In keeping with the overall safety principles, the used fuel repository is designed using concentric barriers . The first barriers are the same as those for the nuclear power plant, ie, the solid, corrosion-resistant ceramic fuel pellets and the Zircaloy cladding that surrounds the pellets. The next barrier is the canister or inner shell, which becomes a permanent element and within which the fuel assemblies are placed. The third barrier is the overpack of concrete or steel. This set of barriers makes up the engineered package. The last barrier is the geologic surroundings within which the engineered package is buried. Even after 1000 years or more, when the integrity of the engineered package may become reduced, the dry, impermeable ground formation will contain the radioactive material with high probability for indefinitely long periods of time.

Scientific studies of the impact of geologic change have been accompanied by probabilistic performance assessment, ie, PRA methodology adapted to radioactive waste disposition. Analyses have been performed to develop estimates of the probabilities that the radioactive material might enter the biosphere. These analyses are being used to assure compliance with the emerging regulatory requirements. The half-lives of the radioactive species (the time it takes for radioactive species to diminish by a factor of 2 through radioactive decay) are important characteristics of the evaluations. However, half-life in itself is not the dominant characteristic of concern. Otherwise, stable nonradioactive toxic wastes that have infinite half-lives would be risk dominant. Rather, it is the toxic and chemical characteristics in combination with the radioactivity which determine the radioisotopes (qv) of dominant risk.

The primary assurance of repository public safety is the prevention of groundwater radioactive contamination. The long-lived radioactive species pertinent characteristic is their solubility in water. The long-lived actinides, such as plutonium, are metallic and insoluble even if water were to penetrate the repository. Some fission product isotopes are soluble and therefore represent the principal hazard. The dominant radionuclides with potential for causing a dose outside the repository are very few in number and are for the most part just two, Tc^{99} ($<$ 2000 years) and Np^{237} (for long times beyond).

A major effort is underway in the United States to provide a deep underground disposal facility for spent fuel. Yucca Mountain, Nevada was chosen by Congress as a promising location. Extensive scientific studies for determining whether Yucca Mountain is a suitable site have been completed . In 2002, based on these scientific studies, Congress and the President approved Yucca Mountain for a repository. The primary effort now is to obtain a license from the U.S Nuclear Regulatory Commission to construct the repository. The safety criteria include an individual protection standard after permanent closure, a human intrusion scenario, and separate standards for protection of groundwater. The licensing effort is moving slowly. The safety criteria for the repository have not been fully determined because of differences of opinion between the governing regulatory agencies (EPA and NRC) and court challenges to the proposed criteria. The PRA methodology, modified for the waste disposition application and called probabilistic performance assessment, is being used to assess the radiation risk to the public. Results to date indicate that the risk is negligible.

In all of the transportation and storage steps, sensitive radiation monitors are located at and around the used fuel to detect incipient leakage. If such leakage were to be detected, steps would be taken to repair the defect. Even for the permanent repository, radiation monitoring would be kept up indefinitely and provision made for retrieving the spent fuel for a period of at least 50–100 years to effect repairs of any defects in the engineered package. Based on the experience gleaned in that initial period, a decision would then be made as to whether the repository were fully suitable as permanent. The first 50–100 years could be considered as interim storage, as the first phase of the permanent facility. When the repository is licensed in the next decade, it is planned to provide an adjacent above ground interim storage facility to permit orderly removal of used fuel from the reactor site and preparation for repository emplacement. Such a facility has been constructed in Sweden. Storage of up to 5000 metric tons of used fuel has been initiated within the facility, called CLAB. It is in an underground manmade rock cavern $\sim$ 40 m deep. Sweden treats CLAB as a separate interim storage facility. A permanent repository is under development .

The high level waste of the recycle (qv) option is made up primarily of fission products having only residual amounts of plutonium and other actinides following the reprocessing . The fission-product wastes come from the chemical reprocessing plant in liquid form and have to be converted to a solid. Vitrification of the waste is planned, so that the first barrier in radioactive containment design is a highly corrosion-resistant glass. The vitrified form is in pellets or logs stored in stainless steel or titanium canisters, which in turn are installed in an overpack to make up the engineered package. This package would then be disposed of geologically. Repository safety advantages exist in this option because the bulk of the long-lived plutonium has been removed.

In addition, processes are under development to separate the other long-lived, and largely non-fissile, actinides from the fission products and recycle these materials into the reactor along with the residual fuel. A promising pyrometallurgical reprocessing method for such actinide separations is under development. This recycling is most effective in a liquid metal cooled reactor because in its high energy neutron spectrum, neutrons are not absorbed appreciably by the actinides, and thus the efficiency of the chain reaction is maintained. By contrast, efficiency would be poor in a lightwater-cooled reactor, which has a low energy neutron spectrum, and actinides become strong neutron absorbers. Other improvements could be made to the waste by converting the soluble fission products into insoluble forms. If the economics of recycling were improved, that option would become preferable for used fuel because the permanent repository issues of the residual fission products would be easier to handle. The economic value of the energy generated from the recycled plutonium and uranium would substantially allay the costs of the repository as compared to the once-through option.

Another safety issue to be considered that might be exacerbated in the recycle option is that the plutonium generated in power reactors, called reactor-grade plutonium, can be used to make a nuclear explosive. Since reactor-grade plutonium contains plutonium-241, which is subject to spontaneous fission , it is extremely difficult to make an effective nuclear weapon from it. However, an explosive device could be built using this isotopic mixture if control of detonation is sacrificed .

When reactor-grade plutonium is left in spent fuel, the large size of the fuel assemblies and the lethal radiation fields make it extremely difficult to divert the material covertly. Once the reactor-grade plutonium is separated in the commercial reprocessing option, however, the radiation barrier is almost eliminated, and in certain steps of the process the plutonium is in powder or liquid form, which is much more easily diverted than large, bulky fuel assemblies. This issue is under study and strict standards of control of separated reactor-grade plutonium have been instituted . In the United States, the Nuclear Nonproliferation Act of 1978 was passed to strengthen control over export trade of plutonium-bearing components by U.S. industry. In addition, under the Nonproliferation Treaty, which most larger nations have signed, the IAEA monitors plutonium from power reactors so as to detect covert diversion. Nevertheless, clandestine weapons development has occurred, often under cover of commercial nuclear power or nuclear research programs, in countries such as India, Israel, Pakistan, North Korea, and Iran. In some of these cases, highly enriched uranium as well as plutonium has been produced .

Although none of the above proliferating actions has involved separated plutonium from used commercial nuclear fuel, concern about the potential diversion of separated reactor-grade plutonium has led to a reduction in U.S. governmental support of development of both plutonium recycle and the liquid metal reactor. This latter ultimately depends on plutonium recycle to achieve its long-range purpose of fuel sustainability: extending nuclear fuel resources for centuries.

5. Radiation Exposure and Health Standards

In the United States, each person receives an average of ~ 0.0036 Sv/year of radiation exposure. About 0.003 Sv comes from natural background sources such as radon gas, cosmic rays, and radioactive elements present in the air, soil, and rocks. Another 0.0006 Sv come from other sources, primarily medical treatments and consumer products. Nuclear utility workers may be exposed to an additional occupational radiation exposure. In any given year, ~ 50% of nuclear industry workers receive no measurable radiation. The remaining 50% of the workers are exposed to an additional average 0.003 Sv/year, for a total of 0.0066 Sv annually . To assure that this additional radiation exposure is not harmful, several measures are taken (*1*) standards are set for the maximum allowable radiation dosage by national and international commissions of radiation health experts (53–55) and incorporated into federal regulations ; (*2*) dosimeters, ie, radiation monitors, are worn by all workers potentially exposed to occupational radiation and accurate records are kept of the accumulated exposure to each worker to assure that maximum allowable levels are not exceeded; and (*3*) the concept of keeping occupational radiation exposure to as low a level as reasonably achievable (ALARA) is practiced so that a relatively small number of workers come close to the maximum allowable levels.

High levels of radiation exposure received over a short period of time (minutes to hours) can cause both near- and long-term effects. Near-term effects include radiation sickness or death. Long-term effects predominantly involve the incidence of cancer . Studies of radiation exposure to researchers in the early twentieth century, Japanese atomic bomb survivors, patients undergoing medical radiation exposures, and radiation accidents have led to standards of maximum allowable radiation exposure . Short-term exposures of several sieverts are required to cause severe radiation sickness or death; exposures of tenths of sieverts may induce cancer in humans. For lower exposures, the risk of cancer or genetic effects is difficult to assess. The number of these effects seen in exposed individuals is about the same as the number occurring in people who receive only normal background radiation exposure.

Most of the data on radiation health effects have come from medical monitoring of Japanese atomic bomb survivors. For survivors who received radiation exposures up to 0.10 Sv, the incidence of cancer is no greater than in the general population of Japanese citizens. For the ~ 1000 survivors who received the highest radiation doses, ie, >2 Sv, there have been 162 cases of cancer. About 70 cases would have been expected in that population from natural causes. Of the ~ 76,000 survivors, as of 1995 there have been a total of ~ 6000 cases of cancer, only ~ 340 more cases than would be expected in a group of 76,000 Japanese citizens who received only background radiation exposure . As discussed above, the Chernobyl accident is providing additional data. Apart from an increase in thyroid cancer, no increases in overall cancer incidence or mortality have been observed to date that is attributable to ionizing radiation. Thousands of laboratory and epedimeological studies of radiation and its risks have been conducted. Yet, there is no conclusive evidence that low levels of radiation exposure cause either cancer or birth defects. The nuclear industry operates on the conservative ALARA approval, assuming that any exposure involves some risk.

For radiation doses <0.5 Sv, there is no clinically observable increase in the number of cancers above those that occur naturally . There are two risk hypotheses: the linear and the nonlinear. The former implies that as the radiation dose decreases, the risk of cancer goes down at roughly the same rate. The latter suggests that risk of cancer actually falls much faster as radiation exposure declines. Because risk of cancer and other health effects is quite low at low radiation doses, the incidence of cancer cannot clearly be ascribed to occupational radiation exposure. Thus, the regulations have adopted the more conservative or restrictive approach, ie, the linear hypothesis. Whereas nuclear industry workers are allowed to receive up to 0.05 Sv/year, the ALARA practices result in much lower actual radiation exposure.

Reduction in occupational radiation exposure is portrayed in Fig. . In the decade between 1983 and 1993, the annual total radiation dosage received by U.S. nuclear plant workers dropped by 54% whereas the annual MW/year of electricity generated increased by 51%. Thus, the annual ratio of total occupational radiation exposure to total electricity generated dropped by almost a factor of 5. This achievement can be credited in part to improved management practices, but a series of technological innovations have also made a significant contribution .

The dominant sources of residual radiation in the primary circuit outside the reactor core in nuclear plants are cobalt isotopes: ^{60}Co and ^{58}Co form by neutron absorption in ^{59}Co and ^{58}Ni. These last two species are naturally occurring isotopes in commonly used plant construction materials. The processes of transport, activation, and deposition of cobalt-containing corrosion products in the PWR primary system is shown in Fig. . Similar processes apply to the BWR primary circuit. Technological approaches to reduce this residual radioactive cobalt are as follows. (*1*) Minimize the cobalt impurities in the structural materials, replacing the high cobalt hardfacing alloys where practicable. Development of a cobalt-free hardfacing alloy and preparation of cost-effective materials procurement specifications that minimize cobalt content both contribute to significant reductions in ^{60}Co in the primary circuit . (*2*) Precondition out-of-core primary circuit surfaces to minimize the release of corrosion products and the resuspension of radioactive species. Protective surface films can be provided by electropolishing and preoxidizing as well as by electroplating (qv) a thin film of chromium. (*3*) Specify and control primary water chemistry to minimize corrosion and the transport of corrosion products into the core, the disposition and subsequent activation of these products, and resuspension in the coolant. Coolant chemistry guidelines have been developed that specify the allowable levels of impurities, the addition of lithium in the PWR coolant to maintain the proper pH in the presence of boric acid, and the injection of hydrogen and addition of zinc in the BWR systems. (*4*) Remove the residual radiation in the out-of-core primary circuit by decontamination. Several decontamination processes, such as CITROX, CANDECON, and LOMI, have been developed. The last, LOMI has been the most widely used.

6. Safety of Future Reactors

Substantial research and development is ongoing to define the characteristics of improved lightwater-cooled nuclear power plants (62–71). The safety area is no exception. The development of computer capabilities in hardware and software, related instrumentation and control, and telecommunication technology has provided an opportunity for improvement in safety (see COMPUTER TECHNOLOGY). Plant operators now use a variety of user-friendly diagnostic aids to assist in plant operations and incipient failure detection. Communications can be more rapid and dependable within the plant and between plants. Safety control systems can be made even more reliable and maintenance-free. Moreover, passive safety features to provide emergency cooling for both the reactor system and the containment building have been developed. In such passive designs, the emergency cooling is provided through gravity flow from elevated tanks or pressurized containers. In a loss of coolant accident, no operator action is required for 72 h to maintan cooling. The designs are simpler and even more reliable since electric or steam driven pumps and their associated valves have been eliminated. These passive designs have been developed through the Electric Power Research Institute (EPRI) advanced lightwater reactor (ALWR) program .

The ALWR program, supported by electric utilities in the United States, Europe and Asia, the United States suppliers, and the U.S. Department of Energy, has focused on advancing LWR technology because LWRs are expected to continue to be used in the near term deployment of new nuclear plants . The full scope of the program includes the development of both PWR and BWR reactors of passive and evolutionary design, ranging in power outputs from 600 to 1350 MW.

Three evolutionary designs are under development: the PWR System 80+ designed by ABB-CE, the advanced BWR (ABWR) designed by GE, and the Advanced PWR (APWR) designed by Westinghouse. ABWRs have been built and are operating in Japan and System 80 + is under construction in South Korea. Other evolutionary plants are under development by international firms, although not under the sponsorship of the ALWR Program: (*1*) Electricité de France has developed and built a 1400-MW PWR plant called the N-4, (*2*) Nuclear Electric in the United Kingdom is sponsoring a 1350-MW PWR plant developed jointly with Westinghouse/BNFL,and (*3*) Siemans of Germany and Framatome of France are jointly developing a 1350-MW PWR, called the EPR, a 1600 MW version of which has been ordered by Finland. The evolutionary reactors are based on the same design concept as is used in the lightwater reactors of the mid-1990s. Many significant improvements have been made, such as selection of alloys having more corrosion resistance, eg, Inconel 690, for steam generator tubes; a high pressure system for the removal of decay heat; and the reactor vessel materials and weldments chosen to reduce radiation embrittlement and shielded to reduce the fast neutron fluency.

Passive reactors are under development in the United States: 600 and 1000 MW PWRs called AP600 and AP1000 designed by Westinghouse, and a 1350 MW BWR called ESBWR , designed by GE. These designs combine the experience-fed improvements of the larger reactors with passive emergency cooling features. A schematic of the nuclear steam supply system and the containment system for AP600 is shown in Fig. . The power train of the AP600 and AP1000 use proven technology: a UO_2-fueled core and plant components with extensive worldwide operating experience. The burden on the equipment and systems has been reduced by increasing design margins through reductions in coolant temperature, flow rate, and core power density and by selecting higher quality materials and more robust components.

Passive cooling in the AP designs is provided by a passive emergency core cooling system (ECCS) and a passive containment cooling system. The passive ECCS consists of a combination of cooling water sources: gravity drain of water (from two core make-up tanks and a large refueling water storage tank suspended above the level of the core) and water ejected from two accumulator tanks under nitrogen pressure. If a feedwater supply interruption renders the steam generators inoperable, core decay heat is removed through a passive residual heat exchanger located in the refueling water storage tank. This transfers core decay heat to the refueling water by natural circulation. Containment integrity is ensured by cooling the containment shell through evaporation of water that is gravity-fed from a large tank located above the containment. The heat is ultimately removed to the atmosphere by a natural circulation air system. Only active automatic valve operations, ie, no operator action and no pump, diesel, or fan operations, are required to provide emergency core cooling and containment cooling after a significant energy release into containment from the maximum loss-of-coolant accident. The ESBWR has basically the same passive emergency core cooling features and in addition is capable of operating at full power on natural selection, eliminating main coolant pumps as well as emergency cooling pumps, leading to great simplification.

Safety objectives have been established to make all the ALWRs even safer than the plants of the early 1990s and safer than required by the safety goals established by the U.S. NRC. The ALWR safety objectives are that there would be only one chance in 10×10^4 per reactor-year that a severe accident would be initiated, a factor of 10 better than the U.S. NRC safety goal. Mitigation of the accident through the containment systems would reduce the risk by another factor of 10, so that the chance that the radiation dose at the boundary of the plant would be as high as 0.25 Sv, the level below which there is no clinically observable effect, would be one in 1×10^6. An additional objective has been set to limit the level of occupational radiation exposure. No $>$ 1.00-Sv/year occupational exposure should be received by all the workers in each plant, an average of about 0.001 Sv/year. Improved performance objectives have also been set to provide an additional power margin. This places less burden on both the equipment and operators in running the plant, resulting in increased reliability and lower operating and maintenance costs.

Another overall objective of the ALWR Program is to achieve standardization of families of plants in design, construction, and operation. Two fundamental bases for that standardization are common owner operator (utility) requirements and common regulatory requirements. The common utility requirements are contained in substantial detail in utility requirements documents (URD) , which have been developed with experienced utility personnel and the reactor designers. The URD has been accepted by the U.S. NRC. As an appropriate basis for design of new plants. The URD applies the operational experience of existing plants to define optimum plant characteristics from the operators' standpoint, which in turn govern the entire design, operating procedures, and configuration control during plant life. A parallel set of utility requirements, the European Requirement Document (EUR), have been developed by the France and Germany.

In the U.S. common regulatory requirements are specified through a standardization process defined by the U.S. NRC , which provides for certification of a design from a safety standpoint. This can be used in replicate on many sites and also provides for early site approval so that an approved site can be matched with a certified design and a combined construction and operating license obtained. Successful implementation of standardization is a significant contributor to safety because the regulator, the operator, and the supporting industry are then focusing resources and sharing experience on a small number of plant designs and operational processes.

Extensive testing of both the AP and ESBWR passive safety features has been completed. Tests were carried out in industrial and government laboratories in the U.S., Europe, and Asia. They have successfully established the performance and validated the designs. The U.S. NRC independently sponsored similar testing in Japan that confirmed these findings. The evolutionary designs, ABWR and System 80+, and the passive design AP600 have been granted design certification by the U.S. NRC. The AP1000 has received final design approval by NRC and the ESBWR has been submitted to the NRC for licensing review.

Another advanced system offers promise for near term deployment is the gas-cooled reactor , that can operate at significantly higher temperatures than existing designs and therefore can be used to provide the energy for process heat applications in industry as well as the production of hydrogen. This reactor is an advanced version of the gas-cooled reactors listed in Table , employing the direct cycle (heat from the reactor goes directly to the gas turbine rather than through a secondary cooling system).

A long-term international collaboration on advanced reactors is underway to develop more advanced nuclear power systems, called the Generation IV International Forum (GIF), the U.S. counterpart of which is the Generation IV Program. Extensive RD&D is planned for advanced reactor systems that have the potential for achieving fuel sustainability as well as offering improvements in safety, reliability, proliferation resistance, and economy . The following advanced systems and their variants are being considered: gas-cooled fast reactor systems, lead alloy land sodium liquid metal-cooled reactor systems, molten salt reactor systems, supercritical water-cooled reactor systems, and very high temperature gas reactor systems. A related U.S. RD&D program, the Advanced Fuel Cycle Initiative (AFCI) is pursuing new reprocessing and refabrication systems, such as pyrometallurgical reprocessing, that have the potential to be more proliferation resistant and economical than the present operational systems . A primary long-range goal for such an RD&D effort is to achieve a dramatic expansion of fuel resources. Recycling systems can provide a level of total nuclear fuel resources some ten times that of the total of all fossil fuel resources. But without recycle, total nuclear fuel resources are only a fraction of fossil resources .

Nuclear power plants for the future are being developed with the objective of better fulfilling their role as safe, reliable, and economic bulk power producers that will be more broadly accepted and implemented. Use of these will help stem the tide of environmental damage caused by air pollution, or possibly by greenhouse gas emissions, from fossil-fuel combustion products .

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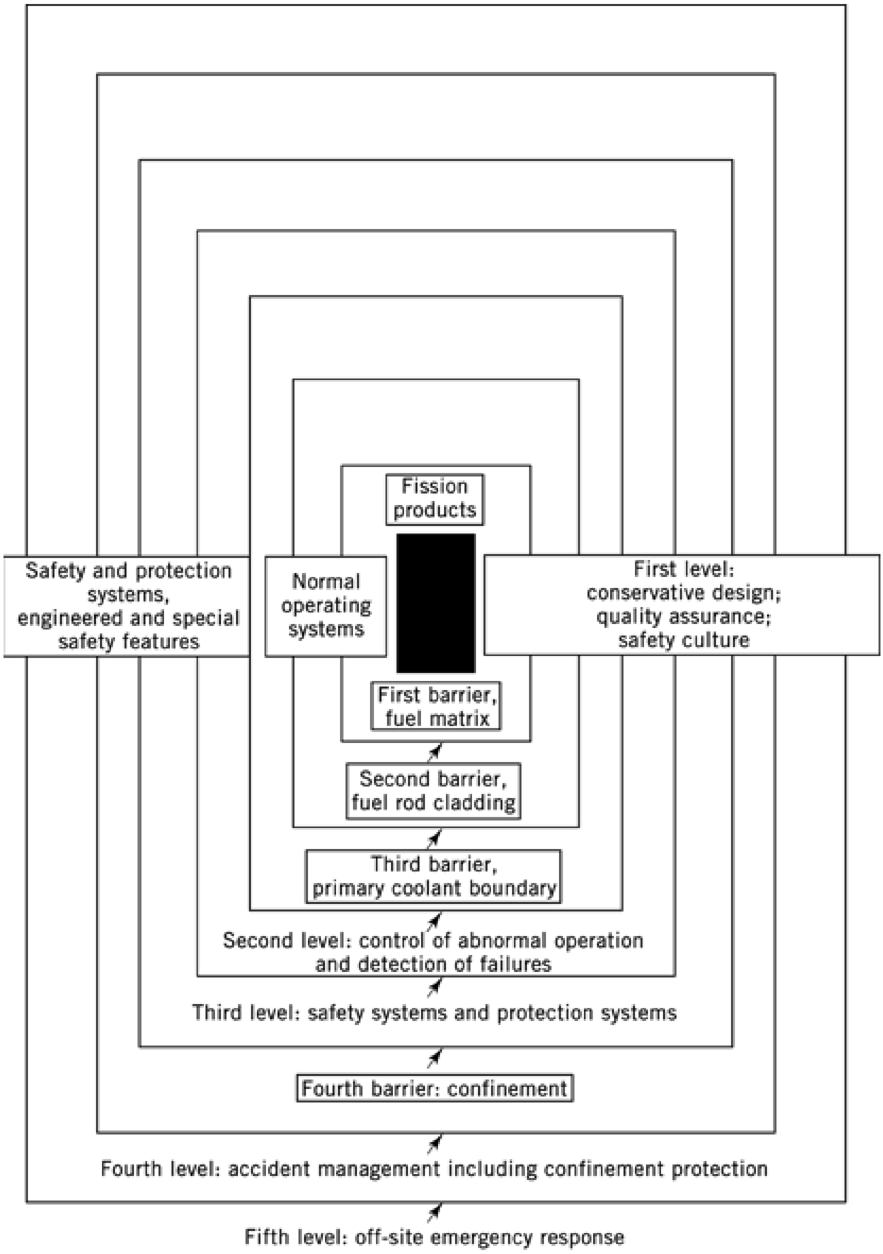

Fig. 1. The relation between physical barriers and levels of protection in defense-in-depth design of a nuclear facility 13. (Courtesy of IAEA.)

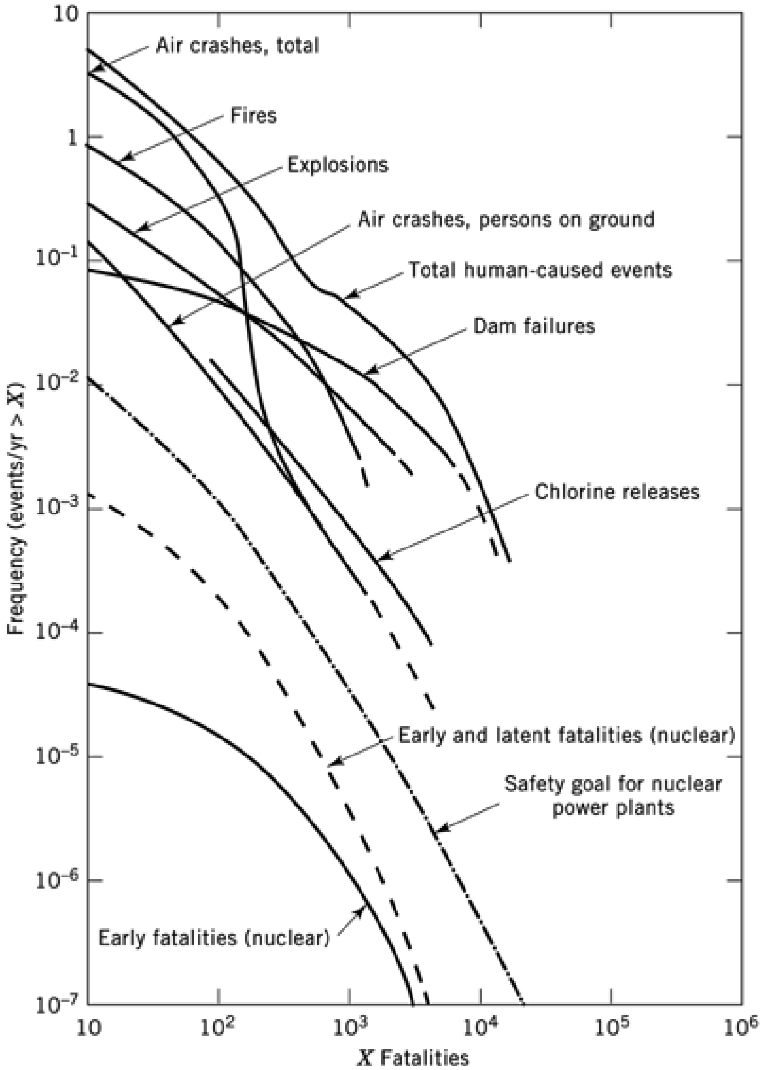

Fig. 2. Frequency of fatalities owing to human-caused events (———) and those caused by nuclear reactor accidents (- - -) together with proposed nuclear power plant safety goals (· –).

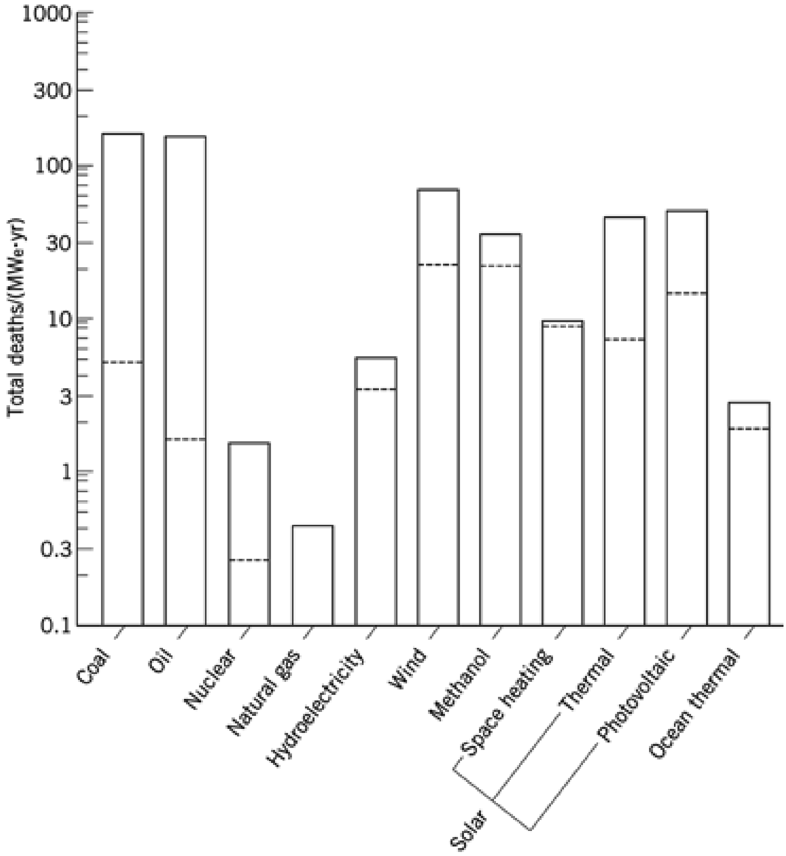

Fig. 3. Total deaths/100 MW year as a function of energy system. The space above the dashed line in each bar represents the range of uncertainty in each estimate. (Courtesy of the Atomic Energy Control Board, Canada.)

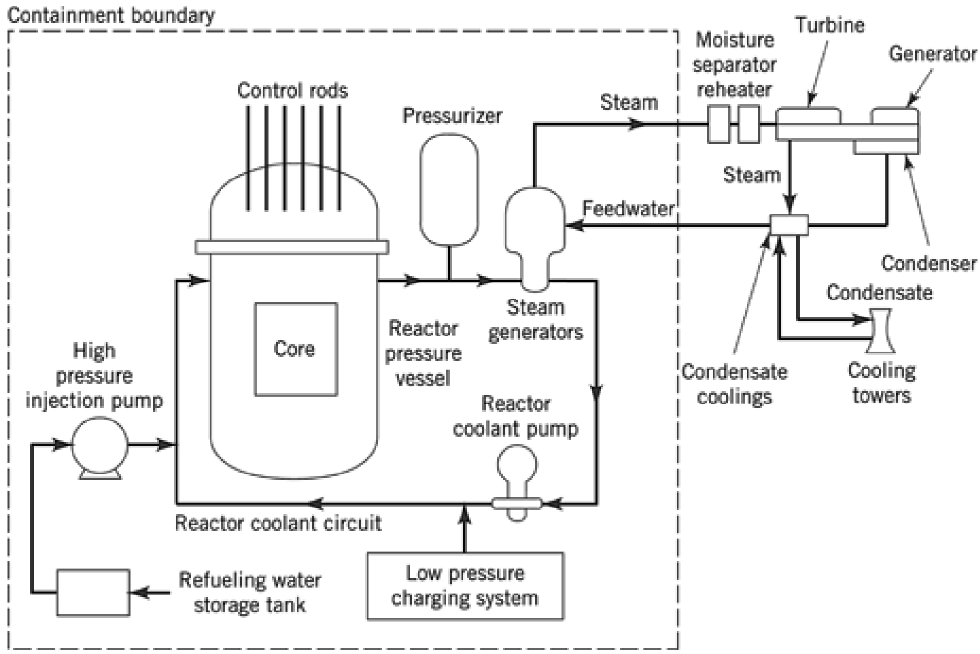

Fig. 4. Pressurized water reactor power plant schematic. (Courtesy of Westinghouse Electric Corp.)

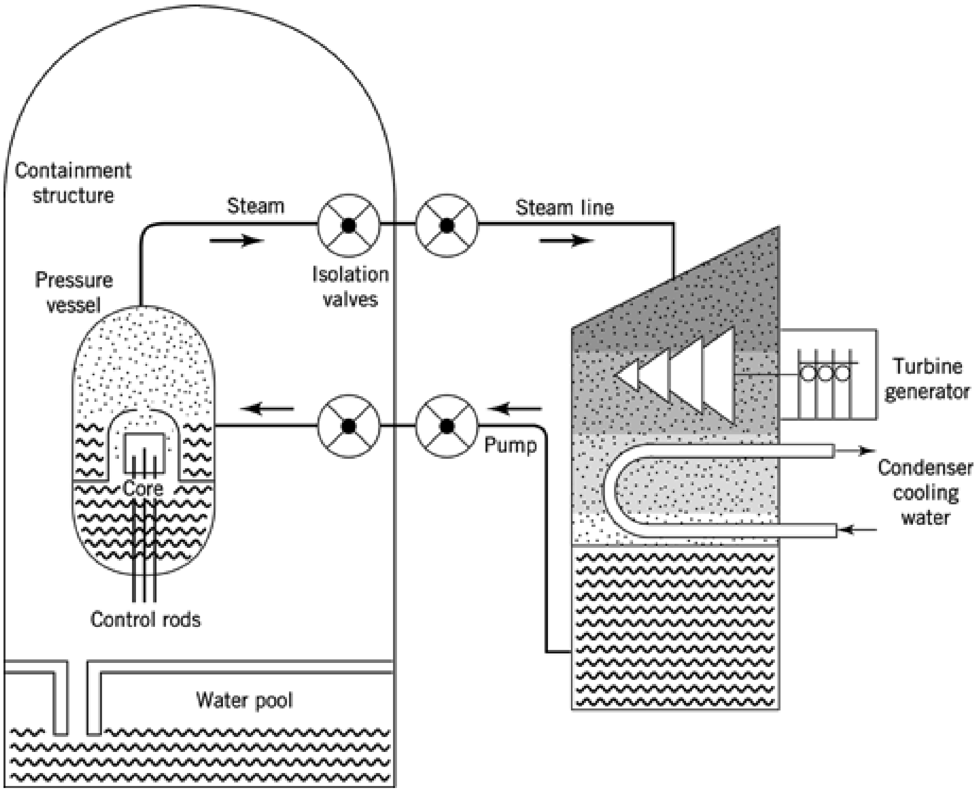

Fig. 5. Schematic of a BWR plant. (Courtesy of Atomic Industrial Forum, Inc.)

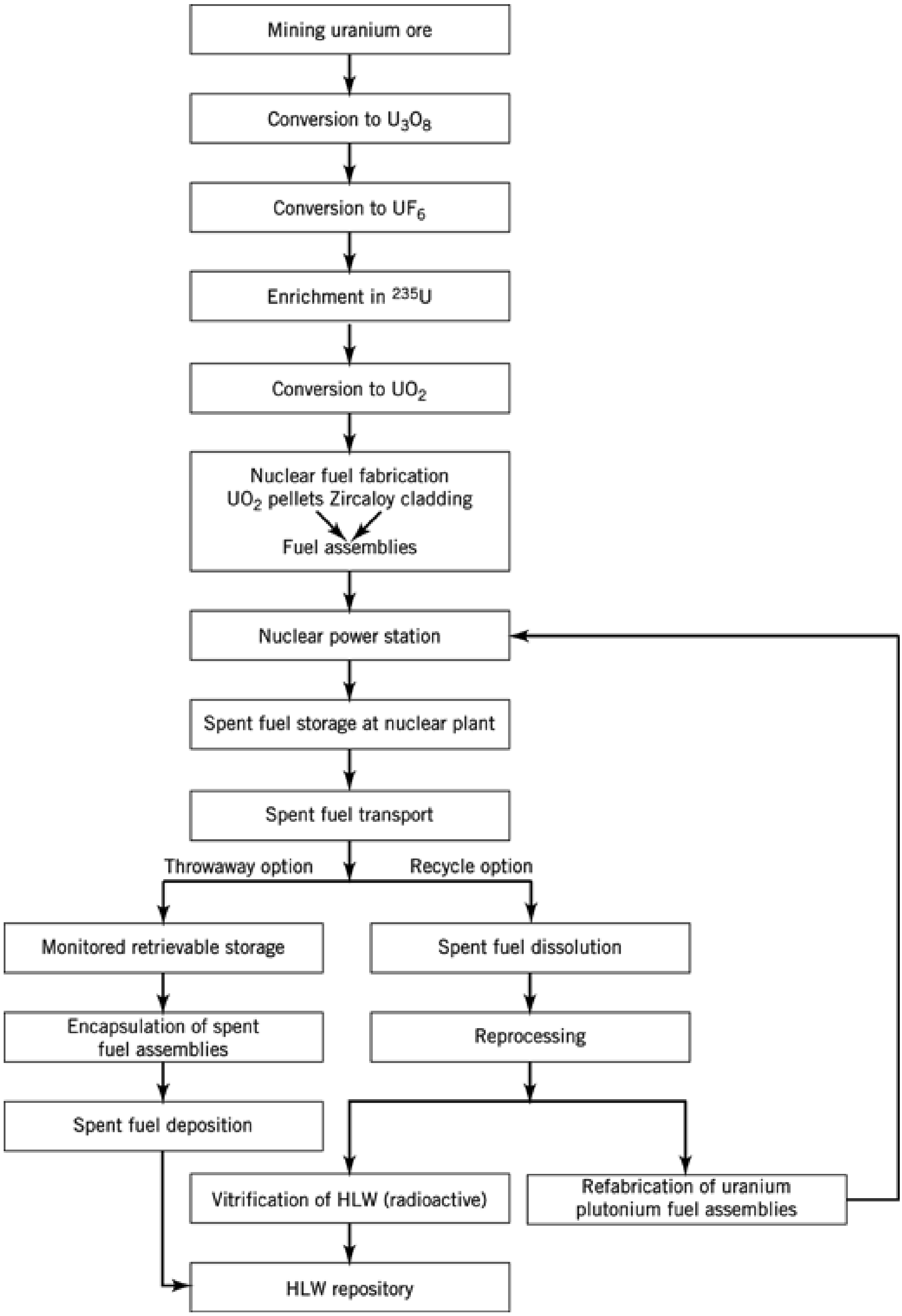

Fig. 6. The nuclear fuel cycle. HLW = high level waste.

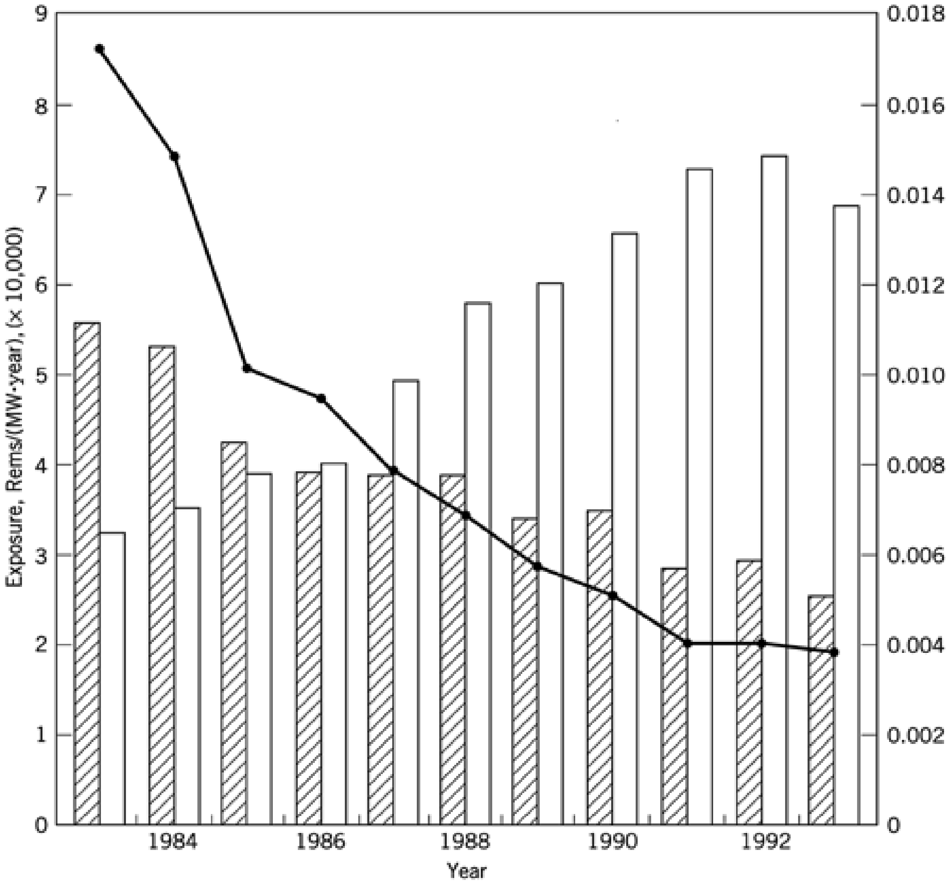

Fig. 7. U.S. nuclear power plant occupational radiation exposure, where (▨) corresponds to total radiation exposure, (Λ) to the electricity generated, and (—●—) to the radiation exposure per unit of electricity 5[Sv/(MW year)] . (Courtesy of the Electric Power Research Institute.)

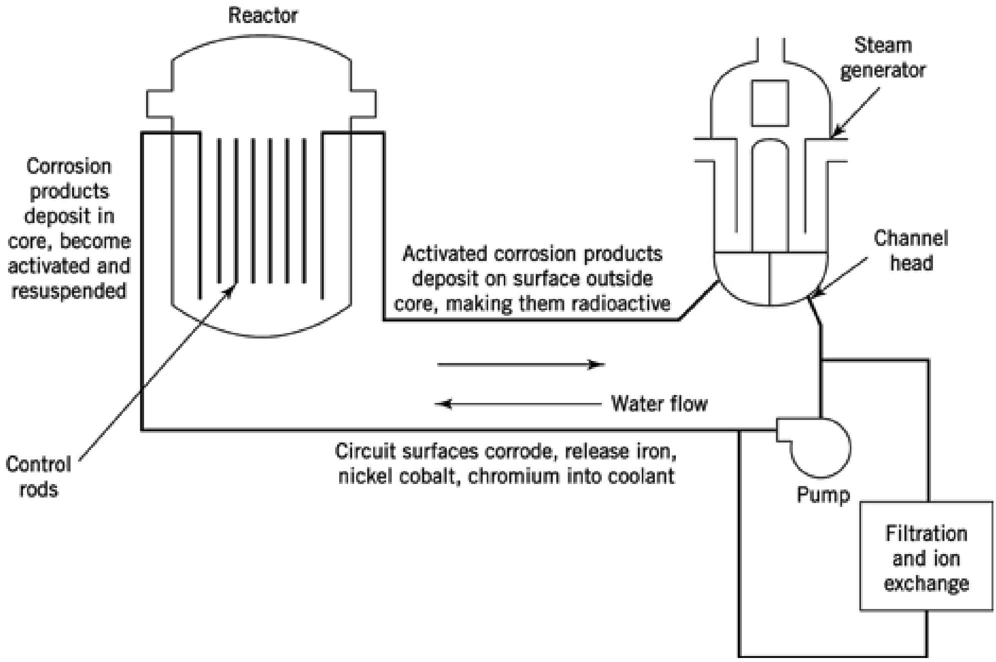

Fig. 8. The activation of cobalt-containing corrosion products in a PWR primary circuit. See text. (Courtesy of the Electric Power Research Institute.)

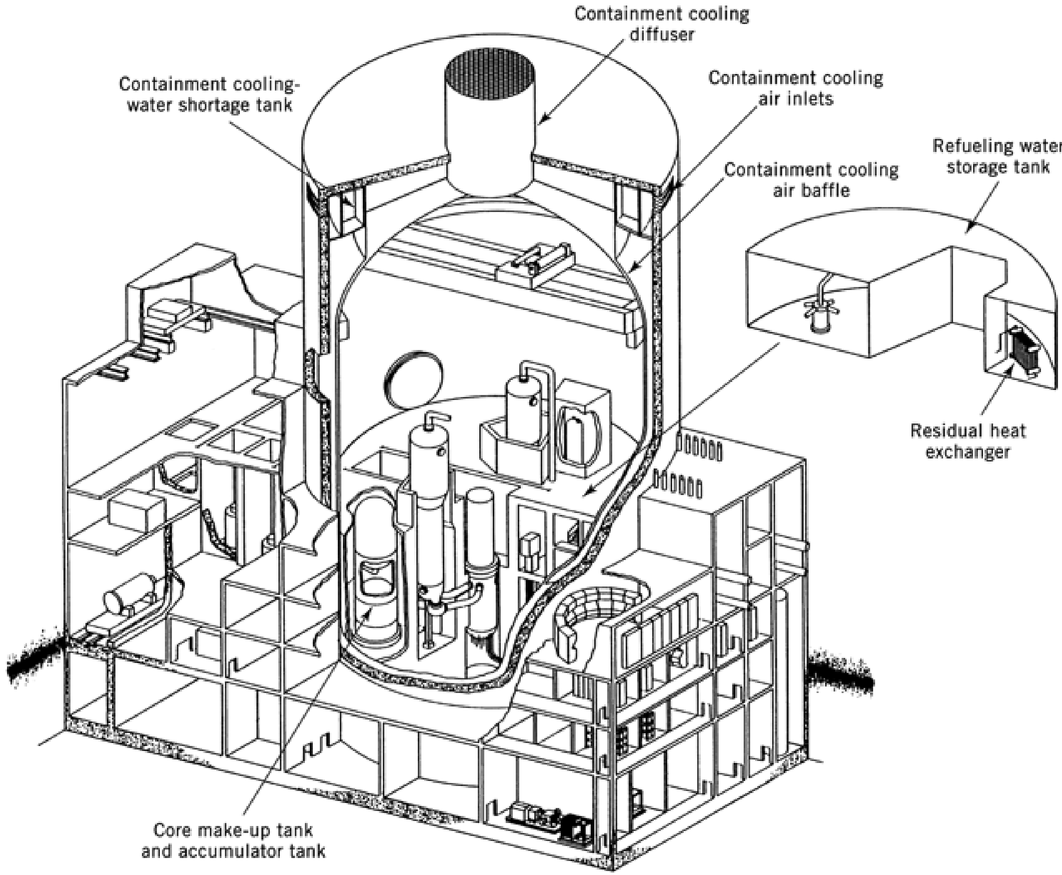

Fig. 9. Sketch showing AP600 passive plant configuration (66). [Courtesy of the American Association for the Advancement of Science (64).]

Table 1. **Nuclear Power Units in Operation Worldwide**[a]

Nation	Number of units[b]	Net power, MW[b]	% of total electricity generated[c,d]
Belgium	7	5,801	9
Brazil	2	1,901	4
Bulgaria	4	2,722	38
Canada	17	12,143	13
China	9	6,587	2
Czech Republic	6	3,548	31
Finland	4	2,656	27
France	59	63,363	78
Germany	18	20,663	28
Hungary	4	1,755	33
India	14	2,550	3
Japan	49	45,444	25
South Korea	19	15,850	40
Lithuania	2	2,270	80
Mexico	2	1,310	5
Russia	31	20,973	17
Slovakia	6	2,442	57
South Africa	6	1,800	6
Spain	9	7,584	24
Sweden	11	9,451	50
Switzerland	5	3,220	40
Taiwan	6	4,884	38
Ukraine	15	13,107	46
United Kingdom	23	11,852	24
United States	104	97,966	20
other	9	6,379	6
totals	*441*	*367.422*	*16*

[a]Ref. 1.
[b]
[c]
[d]

Table 2. **Nuclear Power Units by Reactor Type Worldwide**[a]

Nuclear reactor	Number of units	Net power, MWe $\times 10^3$
lightwater reactor		
pressurized water reactor (PWR)	245	215.7
boiling water (BWR)	92	75.9
gas-cooled reactor	35	11.7
heavy-water reactor	34	18.5
graphite-moderated lightwater reactor	15	14.8
liquid metal-cooled fast breeder reactor	3	0.9

[a]Ref. 1.