Chapter 46

Steam Generation from Nuclear Energy

Since the early 1950s, nuclear fission technology has been explored on a large scale for electric power generation and has evolved into the modern nuclear power plants. (See frontispiece and Fig. 1.) Many advantages of nuclear energy are not well understood by the general public, but this safe, environmentally benign source of electricity is still likely to play a major role in the future world energy picture. Nuclear electric power generation is ideally suited to provide large amounts of power while minimizing the overall environmental impact.

First generation power plants

The concept of an energy generating plant using nuclear fission was first considered by nuclear physicists in the 1930s. However, peaceful use of the atom was delayed until after World War II. The United States (U.S.) had a head start on nuclear technology because of its work in the atomic weapons program. The U.S. Atomic Energy Commission (AEC) took the lead in research and development for a controlled chain reaction application to energy generation. Many concepts were hypothesized and several promising paths were explored, but the real momentum developed when U.S. Navy Captain Hyman G. Rickover established a division in the AEC to develop a nuclear power plant for a submarine. This program, established in 1949, was to become the forerunner of commercial generating stations in the U.S. and the world. Rickover's design succeeded in 1953. Technology and materials developed by his team became the cornerstone of future U.S. nuclear plants. Concurrently, the AEC established a large testing site in Arco, Idaho where, in 1951, the fast neutron reactor produced the first electricity (100 kW) generated by controlled fission.

The world's first civil nuclear power station became operational in Obninsk in the former Soviet Union (FSU) in mid-1954, with a generating capability of 5 MW. This was about the same energy level produced in the U.S. submarine design.

In 1953, the Navy canceled Captain Rickover's plans to develop a larger nuclear power plant to be used in an aircraft carrier. However, he subsequently transformed this project into a design for the first U.S. civilian power stations. Duquesne Light Company of Pittsburgh, Pennsylvania agreed to build and operate the conventional portion of the plant and to buy steam from the nuclear facility to offset its cost of operation. On December 2, 1957, the Shippingport, Pennsylvania reactor plant was placed in service with a power output of 60 MW. This event marked the beginning of the first generation U.S. commercial nuclear plants.

Several basic concepts were being explored, developed and demonstrated throughout the world during this period. The U.S. submarine and Shippingport plants were pressurized water reactors (PWR) that used subcooled water as the fuel coolant and moderator. The FSU developed enriched uranium, graphite-

Fig. 1 Indian Point Station, New York.

moderated, water-cooled designs. British and French engineers explored natural uranium, graphite-moderated, carbon dioxide-cooled stations. In all these designs, the coolant (gas, liquid metal or pressurized water) transferred heat to a heat exchanger, where secondary water was vaporized to provide steam to drive a turbine-generator. The other major competing approach in the U.S. was the boiling water reactor (BWR). This was similar to the PWR except that the need for a heat exchanger to transfer heat from the coolant to the secondary steam cycle was eliminated by boiling the water in the reactor core and by using this slightly radioactive steam to drive the turbine. (See also Chapter 1.)

Two main classes of reactor fuel evolved: enriched and natural uranium. In the early stage of nuclear power development, only the two major nuclear powers, the U.S. and the FSU, had sufficient fuel production capacity for civilian power generation using enriched uranium. Therefore, natural uranium was chosen as the principal fuel in the United Kingdom (U.K.), France, Canada and Sweden. While the enriched uranium-fueled plants could be smaller and therefore required a lower initial investment, higher costs of the enriched fuel over the life of the plant made its use economically equivalent to that of natural uranium. All six countries launched programs to build civil, commercial nuclear power stations. Other countries collaborated with one of these six for construction technology.

In the U.S., the emerging technology focused on the enriched uranium light water reactor (LWR) that used regular water for the coolant and moderator. This had been the technology selected by the Navy programs and, as a result, the civilian sector gained from the experience of naval applications. The AEC also financed construction of the nuclear-powered merchant ship N.S. *Savannah*, which used this technology in a PWR designed and constructed by The Babcock & Wilcox Company (B&W).

In 1955, the AEC began the Power Reactor Demonstration Program to assist private industry. However, by 1963, only one demonstration plant had started up, and two others had been privately financed. Yankee Rowe in Massachusetts, Indian Point Unit 1 in New York, and Dresden 1 in Illinois operated successfully and demonstrated the viability of this technology for larger plants.

Second generation power plants

From these beginnings, the second generation of nuclear plants began in 1963, when a New Jersey public utility (Jersey Central Power & Light) invited bids for a nuclear power station to be built at Oyster Creek. A comprehensive study showed that it was economically attractive to operate, and the utility was ready to proceed. This bid was won with a turnkey proposal to deliver a plant at a fixed price. The suppliers were expected to absorb initial losses in expectation of profitably building a series of similar plants in the near future. From this study and resulting order came the realization that nuclear power stations could be competitive with other electricity sources. Enthusiasm spread and, in the following years, more new nuclear generating capacity than conventional fossil fuel capacity was ordered. By the mid-1970s, nuclear plants, including new orders, totaled a significant portion of the nation's electric generating capacity. The second generation nuclear plants were comparable in size or larger than contemporary fossil-fueled plants. While nuclear plant capital costs were generally higher than those of the current fossil-fueled plants, the lower fuel cost of the nuclear plant offered a reduction in total generating cost.

The U.S. designs were also attractive for export. U.S. PWR designs were ordered in Japan, South Korea, the Philippines, Spain, Sweden, Switzerland, Yugoslavia, Taiwan, Italy, Brazil, Belgium and Germany, while Italy, Mexico, Spain, India, the Netherlands, Switzerland, Taiwan and Japan ordered BWRs of U.S. design.

In parallel with the rapid U.S. growth in nuclear electric generation capacity, the rest of the economically developed countries also began development and construction. In France, early efforts using natural uranium and gas-cooled reactors could not compete with oil-fired plants and enriched uranium light water reactors. After France developed uranium enrichment capability, the country's national utility, Electricité de France, in conjunction with heavy component designer and manufacturer Framatome, ordered PWR and BWR designs based on U.S. technology. An innovative concept applied by the French was to build a series of nearly identical plants of each design, thereby achieving economies of scale. Subsequently, the PWR design was selected by the French for their standardized series of plants. The basic designs developed were also eventually exported to Belgium, South Korea and South Africa.

In the U.K., initial designs for weapons production reactors were applied to generate commercial power from natural uranium-fueled, graphite-moderated, gas-cooled reactors. In the early 1960s, an advanced gas-cooled reactor (AGR), which used enriched uranium, was designed and subsequently built. Italy and Japan also built versions of these British designs.

The FSU developed a graphite-moderated, boiling water-cooled, enriched uranium reactor from its early Obninsk power station design. A major difference in this reactor from similar Western designs was that it did not include a sturdy reactor containment building. The FSU also developed naval propulsion reactors based upon the PWR concept that were then applied to civilian use. This PWR design was exported to several former Soviet bloc countries and Finland. The FSU PWR was similar to the U.S. designs except it generally provided significantly lower electrical output.

The excellent moderator characteristics of deuterium oxide (heavy water) became the basis for the Canada Deuterium Uranium (CANDU) reactor systems. These natural uranium-fueled reactors were first operated in 1962 (25 MW Nuclear Power Demonstration reactor at Rolphton, Ontario), and subsequent designs increased output to 800 MW.

In Japan, electric utilities experimented with several types of imported nuclear generation plant designs. They ordered gas-cooled reactors from the U.K., and BWR and PWR designs from the U.S. The Japanese government also sponsored a breeder reactor design.

Germany was actively involved in designs primarily associated with high temperature gas reactor concepts. In 1969, the formation of Kraftwerk Union introduced German PWR and BWR designs based on U.S. technology. The intent was to construct a series of these plants to achieve economies of scale. However, unlike the French program, few plants were constructed. Several German PWRs were exported to Brazil, Spain, Switzerland and the Netherlands.

The energy crisis of the mid-1970s caused significant reductions in worldwide electric usage growth rates, and electric utilities began canceling and delaying new nuclear generating capacity construction. Cost increases caused by a number of factors also contributed to this construction decline. The delay of schedules due to slower load growth and/or regulatory hurdles greatly increased financing costs and drove unit costs higher. The high inflation of the period further increased construction costs. In addition, continually changing safety regulations increased costs as changes in design of plants already under construction were required.

PWR installations

By 1978, nine second generation nuclear units designed by B&W were placed into commercial operation, each generating 850 to 900 MW. The first unit was one of three built at Duke Power Company's Oconee Nuclear Station located near Seneca, South Carolina. The chapter frontispiece shows the three units of this station. The three vertical, cylindrical concrete structures are the reactor containment buildings. Although some equipment is shared among units, each is operationally independent. The rectangular building at the right contains the turbine-generators. The control room, auxiliary systems and fuel handling facilities for Units 1 and 2 are located in a structure between reactor buildings 1 and 2. Corresponding equipment for Unit 3 is located separately. The following general description applies to Oconee Unit 1, although it is generally applicable to all U.S. B&W plants; the concepts are applicable to all PWRs.

Containment building

Figs. 2 and 3 show a vertical section and plan of the reactor containment building. The structure is post-tensioned, reinforced concrete with a shallow domed roof and a flat foundation slab. The cylindrical portion is pre-stressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a three way post-tensioning system. The foundation slab is conventionally reinforced with high strength steel. The entire structure is lined with 0.25 in. (6.3 mm) welded steel plate to provide a vapor seal.

The containment building dimensions are: inside diameter 116 ft (35.4 m); inside height 208 ft (63.4 m); wall thickness 3.75 ft (1.143 m); dome thickness 3.25 ft (0.99 m); and foundation slab thickness 8.5 ft (2.59

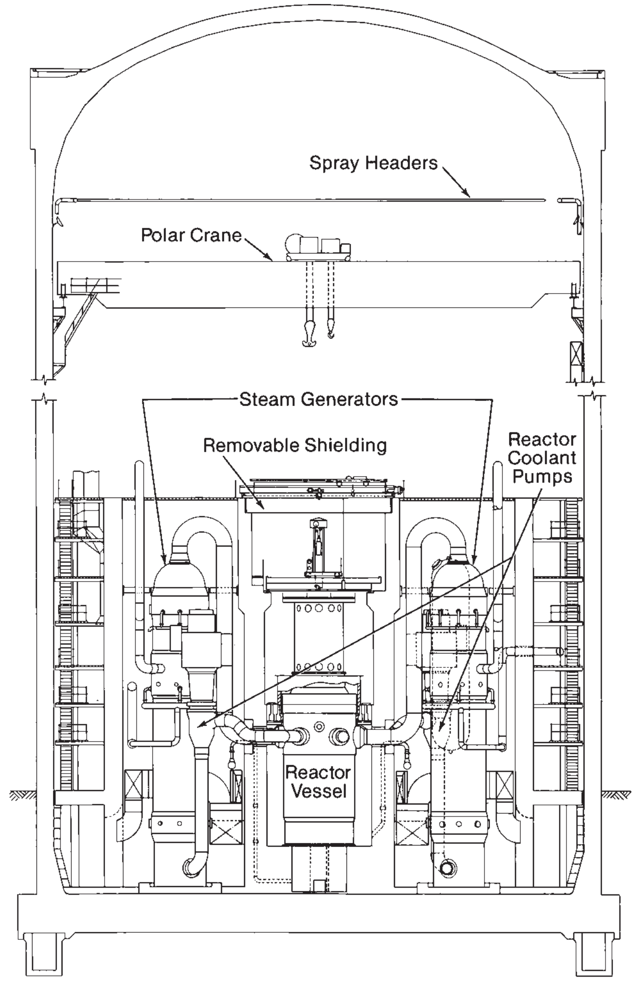

Fig. 2 PWR containment building, sectional view.

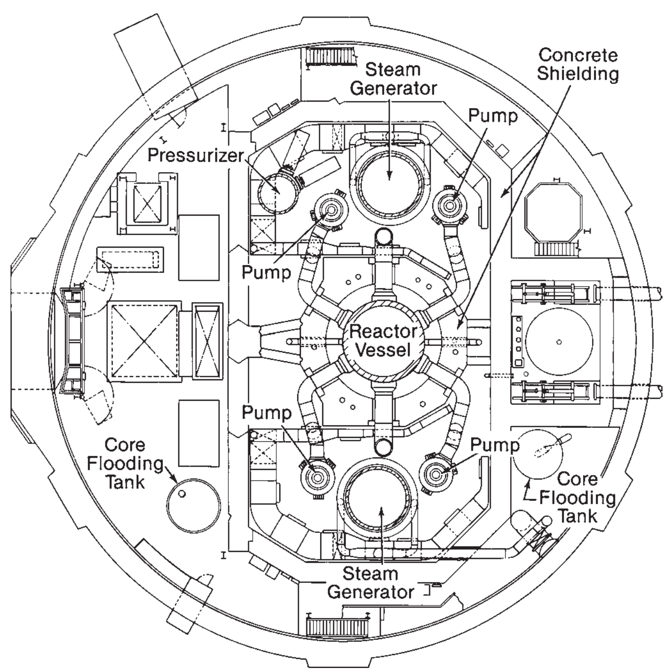

Fig. 3 Reactor containment building, ground floor plan view.

m). The building encloses the nuclear steam supply system and portions of the auxiliary and safeguard systems. The interior arrangement meets the requirements for all anticipated operating conditions and maintenance, including refueling.

The building is designed to sustain all internal and external loading conditions that may occur during its design life. In the event of a major loss-of-coolant accident, the building is designed to sustain the pressure caused by the release of the high-pressure water. To protect against external accidents, extensive tests and analyses have been conducted to show that airplane crashes or objects propelled by tornadoes will not penetrate the wall of the containment.

Nuclear steam supply system (NSSS)

B&W PWR system The major components of the B&W pressurized water reactor NSSS, shown in Fig. 4, include the reactor vessel, two once-through steam generators (OTSGs), pressurizer, primary reactor coolant pumps and piping. As shown in the simplified diagram in Fig. 5 (one of two steam generators shown), the NSSS is comprised of two flow circuits or loops.

The primary flow loop uses pressurized water to transfer heat from the reactor core to the steam generators. Flow is provided by the reactor coolant pump. The pressurizer maintains the primary loop pressure high enough to prevent steam generation in the re-

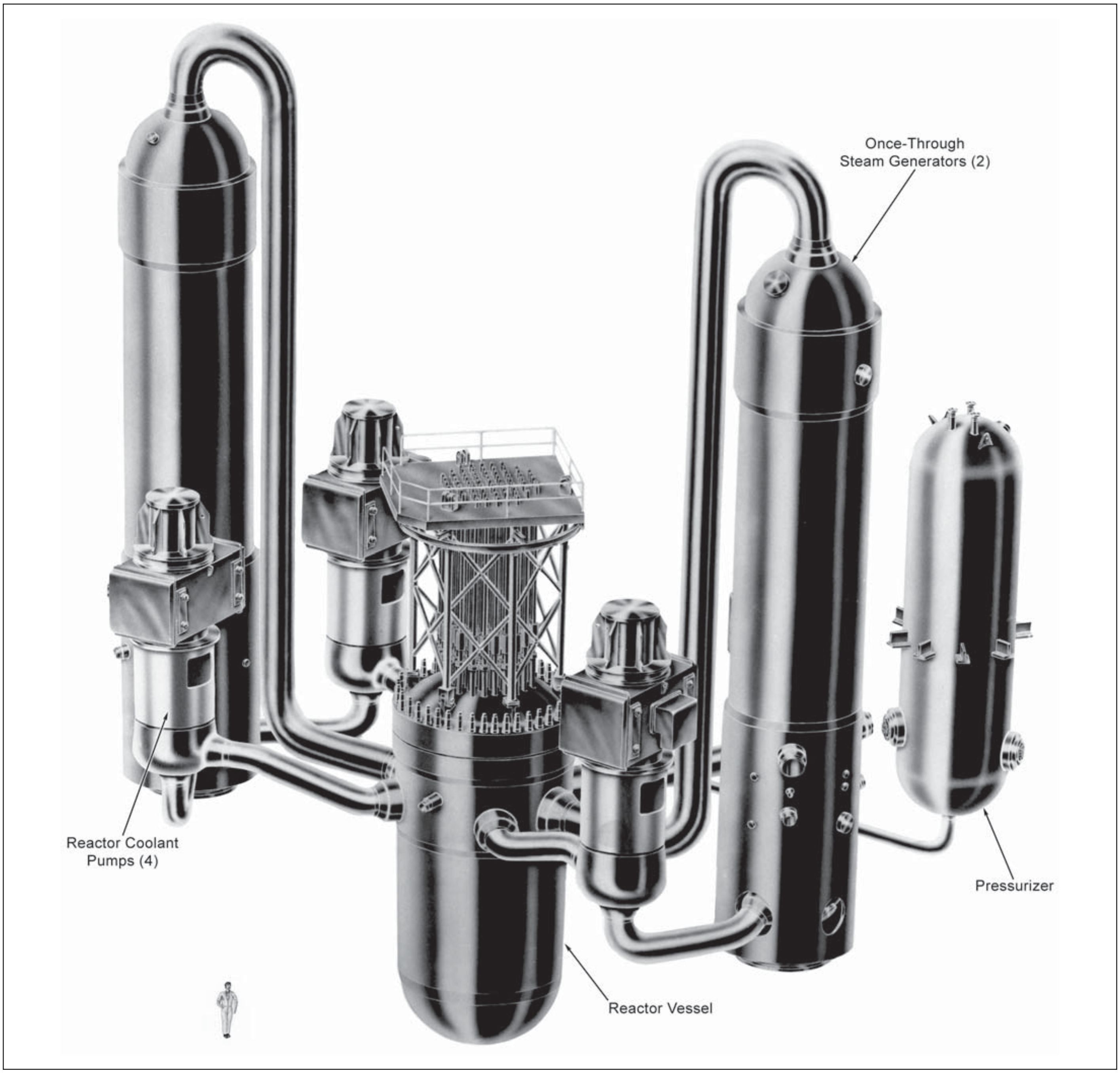

Fig. 4 B&W nuclear steam supply system.

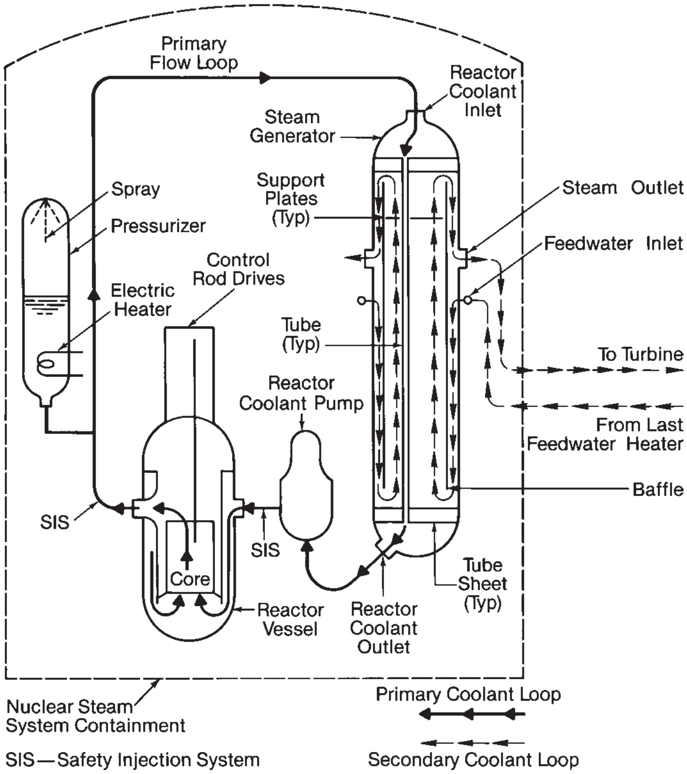

Fig. 5 Simplified schematic of primary and secondary loops.

actor core during normal operation. The steam generators provide the link between the primary coolant loops and the power producing secondary flow loop. In the B&W design, subcooled secondary side water enters the steam generator and emerges as superheated steam, which is sent to the steam turbine to produce power. The primary side water is cooled as it flows vertically downward through the straight tubes and supplies the energy to generate the steam. Table 1 in Chapter 47 of Reference 1 provides a listing of the important design parameters for the 900 MW Oconee type NSSS as well as for a larger 1300 MW system. The first 1300 MW design went into operation in 1987 at Muelheim Kaerlich power station in Germany.

Other PWR systems Fig. 6 shows a PWR system using the recirculating steam generator design. The key components remain the same as the B&W design except that two to four recirculating steam generators replace the OTSGs in providing the link between the primary coolant loop and the power producing secondary side flow loop. In the recirculating steam generator (RSG) design (Fig. 7), the primary coolant passes inside of the U-tubes while the secondary side water passes over the outside of the tubes where it is partially converted to steam. Steam separation equipment then removes residual water which is recirculated back to the bundle for further evaporation. The moisture-free steam is sent to the steam turbine to produce power.

CANDU The CANDU system uses natural uranium fuel and heavy water as the reactor coolant and moderator. The calandria (Fig. 8) is a low-pressure, thin-walled vessel containing the moderator and provides support for the horizontal fuel channel assemblies, shielding, and control mechanisms. In a CANDU 6 system, there are 380 fuel assemblies traversing the entire width of the calandria. Heavy-water coolant flows through the fuel assembly with a full-power flow rate of 1.98 x 10^6 lb/h (249.5 kg/s) at an operating pressure of 1450 psi (10 MPa) and a mean temperature of 553F (289C). From the calandria, reactor coolant is piped to four inverted U-tube recirculating steam generators with design and layout similar to that shown in Fig. 6. In the CANDU system, each of the fuel assemblies can be isolated from the feedwater system, thereby allowing online refueling at full-power operation.

The balance of the following discussion focuses on B&W system components as the key concepts are generally applicable to all PWRs.

PWR reactor and fuel

Reactor vessel and internals

The reactor vessel, which houses the core, is the central component of the reactor coolant system (RCS). The vessel has a cylindrical shell with a spherical bottom head and a ring flange at the top. A closure head is bolted to this flange.

The vessel is constructed of low alloy steel with an internal stainless steel cladding to protect the vessel from corrosion. The vessel with closure head is almost 41 ft (12.5 m) tall and has an inside diameter of 171 in. (4343 mm). The minimum thicknesses of the shell wall and inside cladding are 8.4 and 0.125 in. (213 and 3.18 mm), respectively. Reflective metal (mirror) insulation is installed on the exterior surfaces of the reactor vessel.

The vessel has six major nozzles for reactor coolant flow (two outlet hot leg and four inlet cold leg). The coolant water enters the vessel through the four cold leg nozzles located above the midpoint of the vessel. Water flows downward in the annular space between the reactor vessel and the internal thermal shield, up

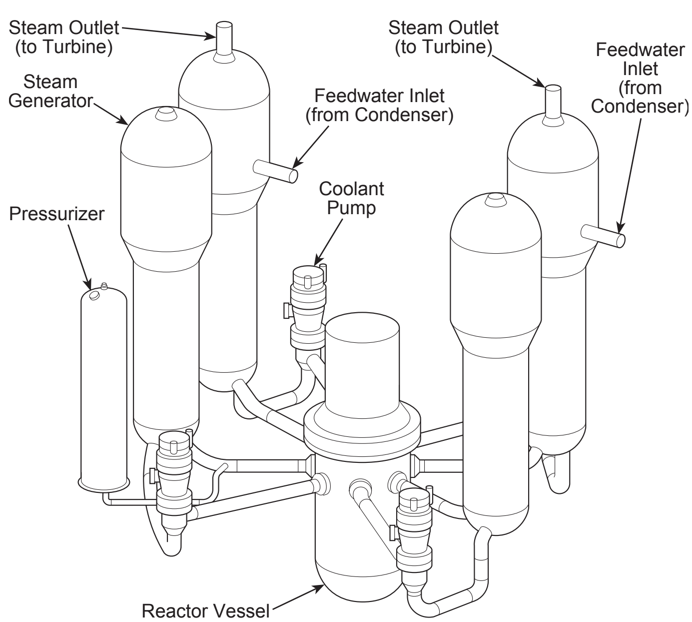

Fig. 6 Four steam generator loop configuration for a pressurized water reactor with recirculating steam generators.

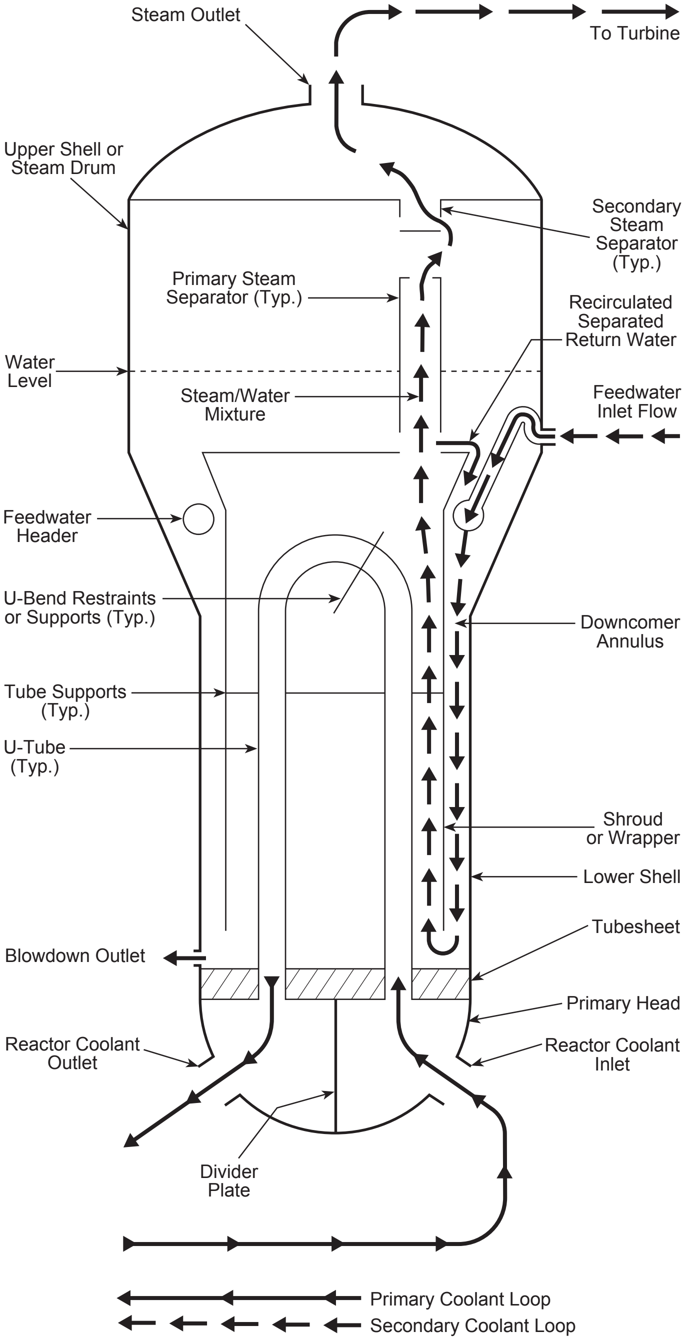

Fig. 7 Simplified recirculating steam generator (RSG) with water preheating configuration.

through the core and upper plenum, down around the inside of the upper portion of the vessel and out through the two hot leg nozzles. Fig. 9 shows a reactor vessel being installed at a nuclear power plant site.

The core, containing the fuel bundle assemblies, is the primary component of the reactor vessel. The remaining major components are the plenum and the core support assemblies (see Figs. 10 and 11).

Plenum assembly The plenum assembly, located directly above the core, includes the plenum cover, upper grid, control rod guide tube assemblies, and flanged plenum cylinder. The plenum cylinder has multiple openings for reactor coolant outlet flow. These flow openings are arranged to form the coolant profile leaving the core. The plenum cover is attached to the top flange of the plenum cylinder. It consists of a square lattice of parallel flat plates, a perforated top plate and a flange. The control rod guide tube assemblies are positioned by the upper grid. These assemblies shield the control rods from coolant crossflow and maintain the alignment of the rods.

Core support assembly The core support assembly consists of the following components:

1. core support shield,
2. core barrel,
3. lower grid assembly,
4. flow distributor,
5. thermal shield,
6. surveillance specimen holder tubes,
7. in-core instrument guide tubes, and
8. internals vent valves.

The core support shield is a flanged cylinder that mates with the reactor vessel opening. The forged top flange rests on the circumferential ledge in the reactor vessel top closure flange, while the core barrel is bolted to the vessel's lower flange. The cylindrical wall of the core support shield has two nozzle openings. These openings form a seal with the reactor vessel out-

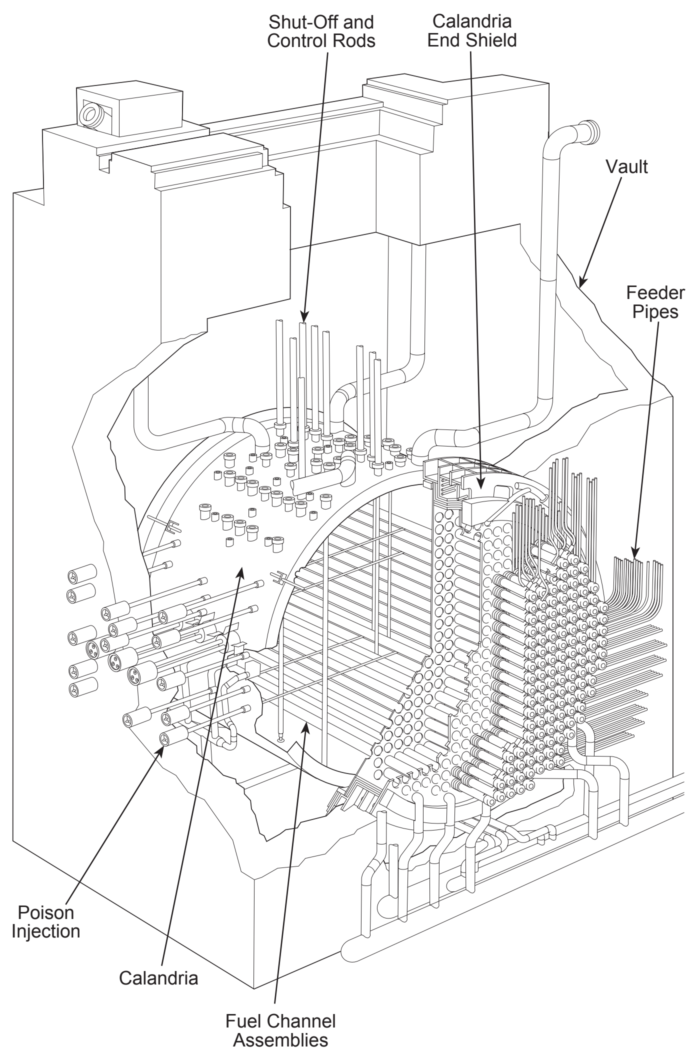

Fig. 8 CANDU 6 reactor assembly.

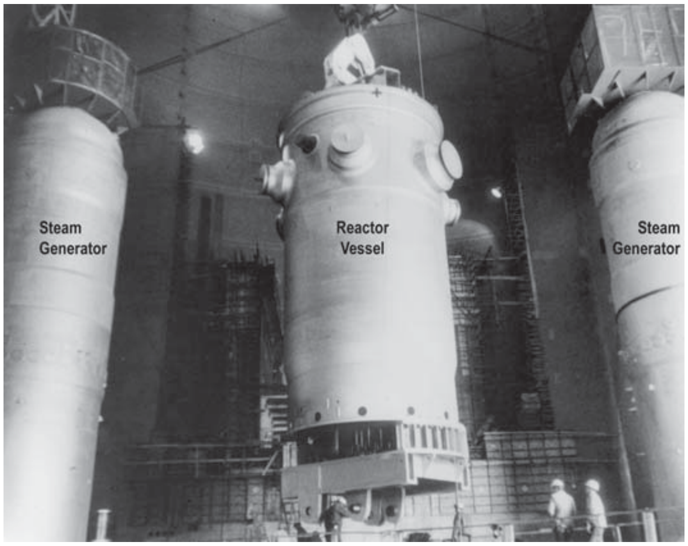

Fig. 9 Reactor vessel placed into position between steam generators.

let nozzles by the differential thermal expansion of stainless and carbon steel. The core support shield also has eight holes in which the internals vent valves are mounted.

The core barrel supports the fuel assemblies, lower grid, flow distributor and in-core instrument guide tubes. This cylinder is flanged at both ends. The upper flange is bolted to the core support shield assembly and the lower flange is bolted to the lower grid assembly. A series of horizontal spacers is bolted to the inside of the cylinder and a series of vertical plates is bolted to the inside surfaces of the horizontal spacers to form walls enclosing the fuel assemblies. Coolant flows downward along the outside of the core barrel cylinder and upward through the fuel assemblies. Approximately 90% of the coolant flow traverses the heat transfer surfaces of the core fuel assemblies. Of the remaining 10% flow, some is directed into the gap between the core barrel and thermal shield primarily to cool the thermal shield. The remainder of the flow bypasses the core through various leak paths in the internals.

Core guide lugs are welded to the inside wall of the reactor vessel. In the event of a major internal component failure, these lugs limit the drop of the core barrel to 0.5 in. (12.7 mm) and prevent its rotation about the vertical axis.

The lower grid assembly, consisting of two lattice structures surrounded by a forged flanged cylinder, supports the fuel assemblies, thermal shield and flow distributor. The top flange is bolted to the lower flange of the core barrel. Pads bolted to the top surface of the upper lattice structure provide fuel assembly alignment. A perforated plate midway between the lattice structures aids the uniform distribution of coolant flow to the core.

The flow distributor is a perforated, dished head that is bolted to the bottom flange of the lower grid. The distributor supports the in-core instrument guide tubes and shapes the inlet flow to the core.

The thermal shield is a stainless steel cylinder located in the annulus between the core barrel and the reactor vessel. It is supported and positioned by the

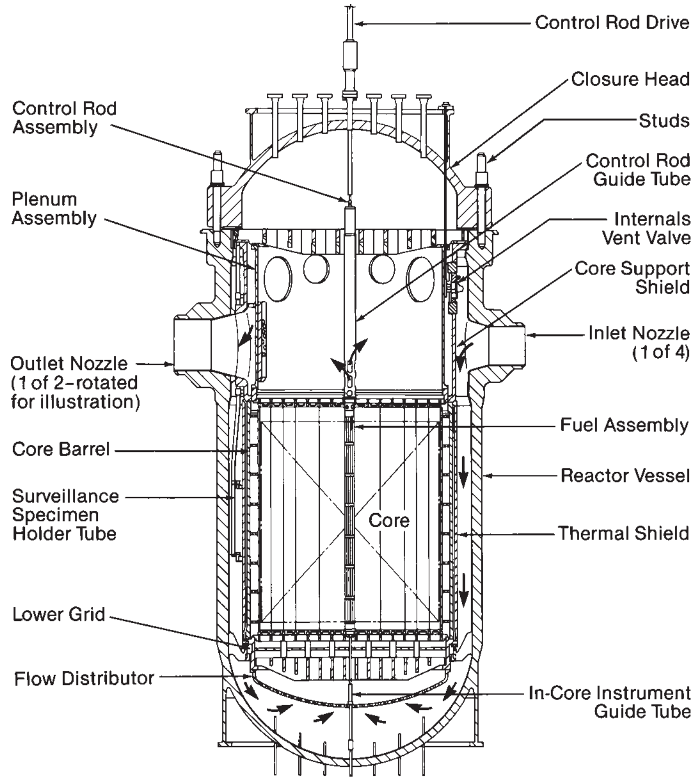

Fig. 10 Cross-sectional view of reactor vessel internals.

top flange of the lower grid. The thermal shield and intervening water annuli reduce the radiation exposure and internal heat generation of the reactor vessel wall by attenuating neutron and gamma radiation.

The surveillance specimen holder tubes are installed on the outer wall of the core support assembly, at approximately mid-height of the core.

The in-core instrument guide tube assemblies guide the in-core assemblies between the penetrations in the bottom head of the reactor vessel and the instrument tubes in the fuel assemblies.

Eight co-planar internals vent valves are installed on the core support shield; they are 42 in. (1067 mm) above the centerlines of the reactor vessel inlet and

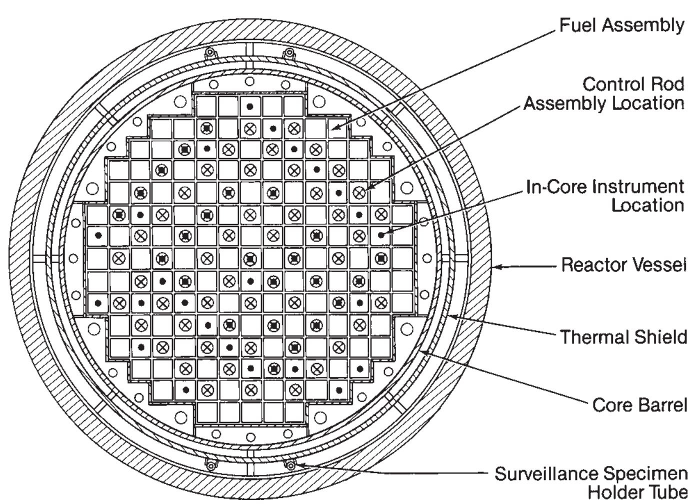

Fig. 11 Plan view of reactor vessel internals.

outlet nozzles. The valve seats are inclined 5 deg (0.087 rad) from vertical and the valve discs naturally hang closed. During normal operation, the valves are forced closed by the differential pressure, approximately 43 psi (296 kPa), between the outer annulus of the reactor vessel and the core outlet region. During a severe accident, such as a large break loss of coolant accident (LBLOCA), however, the differential pressure reverses and the internals vent valves open, permitting steam to be vented directly from the core region to the upper downcomer.

Reactor vessel closure head The closure head is an integral part of the reactor vessel pressure boundary. The head provides access for the replacement of spent fuel, and Alloy 600 penetration nozzles for control rod drive mechanisms and instrumentation. The closure head is typically made of low alloy steel and clad with stainless steel like the rest of the reactor vessel. Recently, many reactor closure heads in existing PWR systems have experienced corrosion damage and are being replaced (see summary in box on the following page.)

Fuel assemblies

In the reactor core, 177 fuel bundle assemblies rest on a lower grid attached to the core barrel. Each fuel bundle assembly contains 208 fuel rods (Fig. 12). Each fuel rod is made up of enriched uranium oxide pellets contained in zircaloy-4 tubing/cladding (Fig. 13). Fuel pellets (Fig. 14) are made up primarily from uranium dioxide (UO_2) powder. The powder is pressed and sintered in a dry hydrogen atmosphere to produce the required pellet size. Final machining of the pellet is done under water. Within the rods, the fuel pellets are spring loaded at both ends to ensure contact between the pellets. The rod is pressurized with dry helium to improve heat transfer across the gap between the pellets and cladding. End caps are laser welded to seal the rods and ensure integrity of the fuel/coolant boundary.

Spacing of the fuel rods in the assembly is maintained with spacer grids at several locations along the length of the bundle (Fig. 15). These grids are formed from strips (Inconel® for the end spacers and zircaloy for the intermediate spacers) that are stamped to form the detailed configurations necessary to ensure adequate contact points with the fuel rods. The strips are welded together to form the square-shaped spacer grid that holds the rods in correct spacing and alignment. Stainless steel end fittings complete the bundle assembly (Fig. 16).

Each fuel assembly is fitted with an instrumentation tube at the center and with 16 guide tubes, each accommodating 16 control rods. (See Fig. 17.) Each group of 16 control rods is coupled together to form a

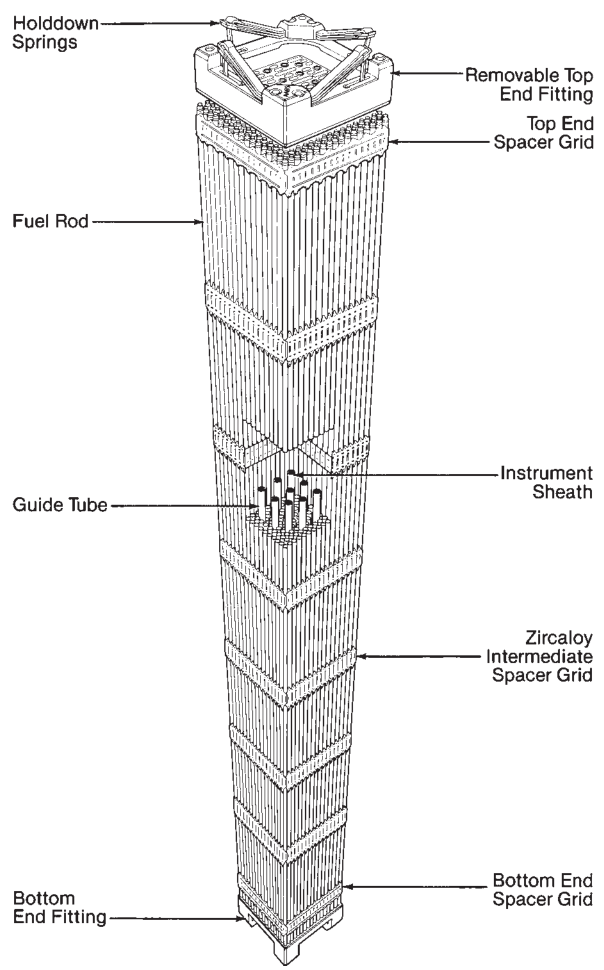

Fig. 12 Nuclear fuel assembly.

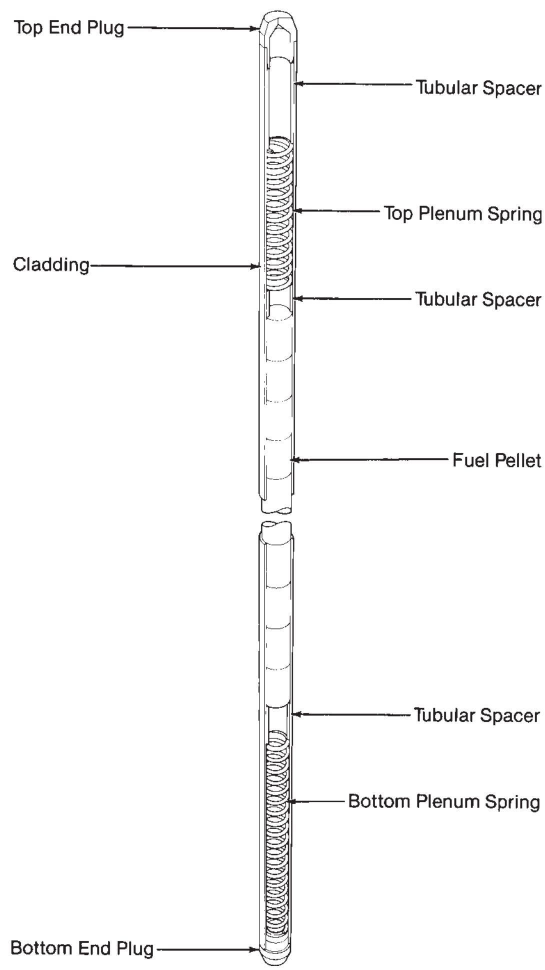

Fig. 13 Typical fuel rod assembly.

Reactor vessel closure heads

In recent years, discovery of cracked and leaking closure head Alloy 600 penetration nozzles, mainly the Control Rod Drive Mechanism (CRDM) nozzles, has raised concerns about the structural integrity of the closure heads. The CRDM nozzle is attached to the reactor vessel closure head and provides access to the core for the CRDMs.

Background

The problem was first discovered at the French Bugey Unit 3 power plant in 1991, when primary water leakage was observed on the vessel closure head at the CRDM nozzle region. Investigation of other French units showed cracking in the CRDM nozzles and the associated welds, and the cause was identified as primary water stress corrosion cracking (PWSCC).

This early French experience prompted examination of some PWR nozzle penetrations in the U.S. Detailed nondestructive examinations showed similar damage. Because the cracks were not through-wall and were within the approved acceptance criteria, the Nuclear Regulatory Commission (NRC) allowed continued plant operation contingent upon increased monitoring. However, in 2001, visual inspections of Oconee Unit 3 CRDM nozzles identified boric acid crystals at 9 of the 69 head penetrations, indicating that some primary water leakage had occurred. Further investigations revealed PWSCC initiated cracks that propagated from the weld radially and axially into the nozzle, allowing primary water leakage.

Evaluation of PWSCC effects has been completed at many of the PWR plants in the U.S. with subsequent decisions to replace some of the heads. These decisions to replace, rather than to follow a monitor and repair strategy, are strongly influenced by the expense of inspection and repair activities, which are challenging to perform in the hostile radioactive environment. It has proven desirable to avoid repairs and replace reactor vessel closure heads in their entirety, as quickly as possible, rather than bear the cost and schedule implications of difficult inspections and time-consuming repairs. In 2001, B&W received the first replacement head order in the U.S. to provide heads for Oconee Units 1, 2 and 3.

Closure head replacement requirements

The principal requirement for replacement closure heads is to provide a *form, fit and function* replacement that is compatible with the original equipment configuration and that provides enhanced reliability suitable for at least a 40 year design life. A form, fit and function replacement design ensures proper and efficient integration of a new closure head into the existing plant, thereby simplifying the safety assessments required to show compliance with NRC 10CFR 50.59 regulations. However, enhanced reliability can only be achieved by having a sound understanding of the cause of the degradation, and by optimizing all aspects of the design that are related to known degradation mechanisms.

Replacement head features

The conditions that induce PWSCC, and that are present in the current CRDM-to-head-attachment region, are:

1. the corrosive environment with primary borated water,
2. the presence of tensile stresses, and
3. the use of materials susceptible to corrosion.

Accepting the presence of the primary water environment, the major focus of the replacement head design has been to provide superior materials and to control the stresses induced at all stages of manufacture.

Materials The material for the CRDM nozzles has been changed from Alloy 600 to thermally treated Alloy 690TT. Similarly, the weld consumables for the CRDM nozzle-to-head weld have been changed from Alloy 182 to Alloy 52 and/or Alloy 152, both having material compositions compatible with the Alloy 690 base material. All of these replacement materials have demonstrated superior resistance to PWSCC if they are specified and procured under a carefully controlled and monitored program.

Fabrication In addition, optimum resistance to PWSCC requires control of mechanical factors such as cold work and residual stresses. Materials for fabrication will start in the annealed condition, essentially free of bulk cold work and residual stress. Some manufacturing steps can reintroduce cold work and stresses which can negatively affect PWSCC resistance (both internal and surface conditions are important). Fabrication sequence and welding methods are important in minimizing internal residual stresses; finishing methods for both individual parts and completed assemblies are important in controlling surface cold work and related residual stresses.

Nozzle-to-head welds The original equipment nozzle-to-head welds were made with a manual shielded metal arc welding (SMAW) process. The replacement units use an automated gas tungsten arc welding (GTAW) process which improves weld quality, and minimizes residual as-welded stresses. To address concerns regarding the effect of as-manufactured surface condition on PWSCC resistance, B&W has employed an electropolishing process for both the CRDM nozzle prior to installation and the finished weld after surface conditioning. This process, unlike abrasive honing or other surface treatments, entirely removes the cold worked surface layer associated with machining and grinding.

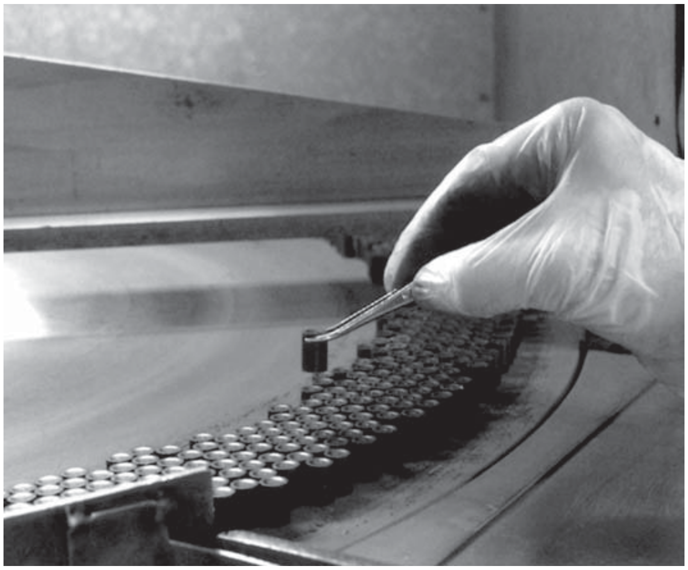

Fig. 14 Nuclear fuel pellets.

control rod assembly as shown in Fig. 18. At any given time, 69 of the 177 fuel assemblies contain a control rod assembly. Because all fuel assemblies are identical and can be used anywhere in the core, the 108 remaining fuel assemblies have their guide tubes partially filled at the top with an orifice rod assembly. This restricts excessive coolant flow through the unused control rod guide tubes and equalizes coolant flow among the fuel assemblies.

The control rod assemblies regulate the reactor reactivity and, therefore, the power output. Each control rod consists of an absorber section made from a neutron absorbing material such as stainless steel with silver-indium-cadmium cladding, and an upper and lower stainless steel end piece. The rods regulate relatively fast reactivity phenomena such as Doppler, xenon buildup and decay, and moderator temperature change effects. The slower reactivity effects, such as fuel burnup, fission product buildup, and hot-to-cold moderator reactivity deficit, are controlled by a soluble

Fig. 15 Spacer grid coordinate measuring machine.

Fig. 16 Final visual inspection of fuel assemblies.

neutron absorbent (usually boron) in the reactor coolant. The concentration of the absorbent is monitored and adjusted by the plant operators.

Refueling

Most U.S. power reactors operate for a planned period of between one and two years before the reactivity of the nuclear fuel is reduced and the unit must be reloaded with fresh fuel. Because only part of the fuel is replaced during the shutdown, this is referred to as *batch refueling*. In some reactors, such as the CANDU design, a continuous refueling mode that does not require reactor shutdown is used.

Most PWRs were designed for annual refueling, replacing approximately one-third of the core at each outage. As operating experience has been gained, the trend has shifted to 18-month refueling cycles. Although fuel costs are higher, the longer refueling cycle is cost-effec-

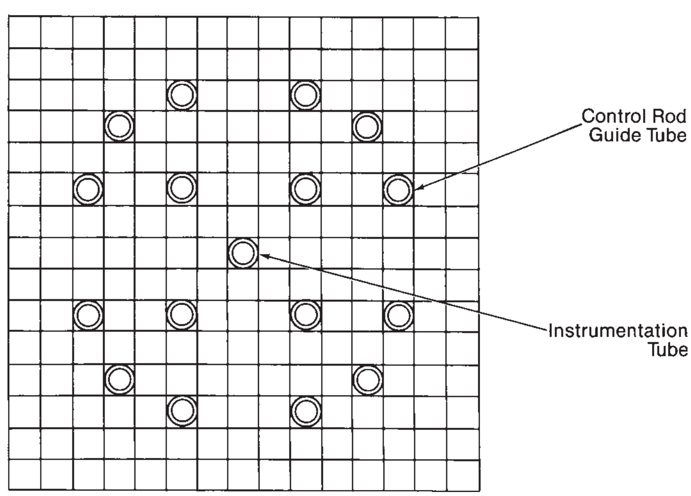

Fig. 17 Diagrammatic cross-section of fuel assemblies showing instrumentation tube and control rod guide tube locations.

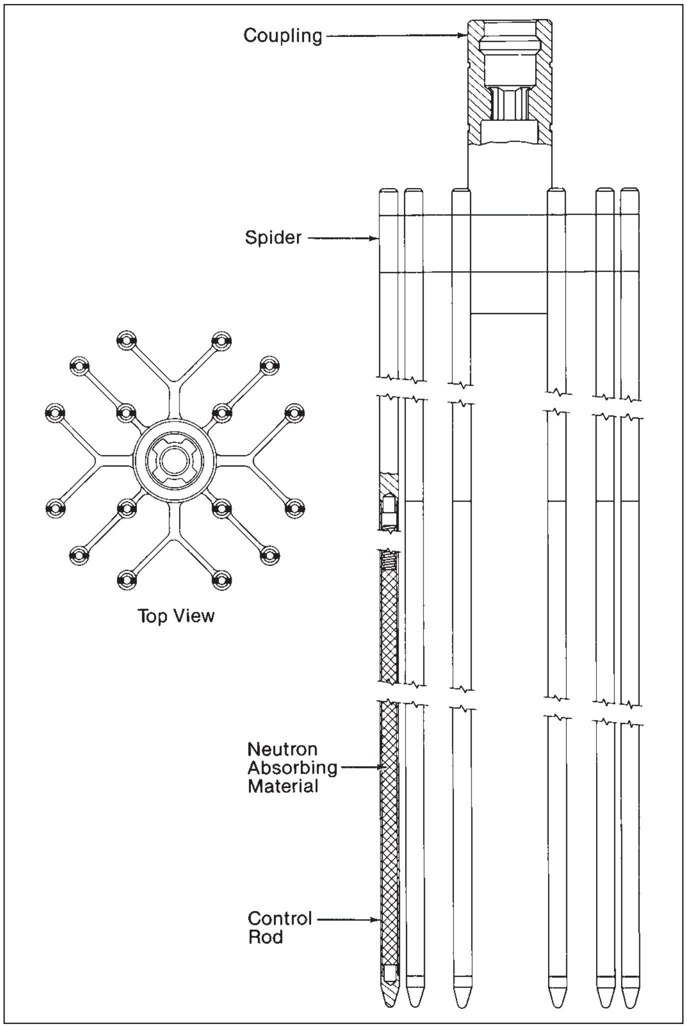

Fig. 18 Control rod assembly.

tive due to reductions in the number of refueling outages, licensing submittals, and replacement power cost. Furthermore, the burnup capability of the fuel has increased, permitting subsequent batch size reductions.

As nuclear fuel is continually consumed in the reactor, the fuel bundle assemblies are subjected to temperature and irradiation effects that change some of their characteristics. Dimensional and structural changes in the assemblies include growth of the fuel rods, spring relaxation, and fuel rod bow. Some corrosion of the cladding takes place on the coolant side. This corrosion effect liberates hydrogen from the coolant, some of which is absorbed by the cladding, and can subsequently decrease cladding mechanical properties. Fission gas release in the fuel rod can cause high internal pressure. The coolant is continually monitored during operation to detect any leakage of these fission gases. While these various effects have been considered and monitored over the years of reactor operation, they have not been found to limit operation of the nuclear systems.

Extreme care and detailed handling procedures are required to remove the spent fuel assemblies for disposition. During their removal from the reactor core and near-term storage in spent fuel pools, water serves as both a shielding and as a cooling medium. The spent fuel assemblies are kept in the spent fuel pool at the reactor site until the radioactive decay and heat generation have declined sufficiently for safe transportation and processing.

Energy transport

Each fission of uranium-235 (U-235) produces approximately 200 MeV, predominantly in the form of kinetic energy of the fission products. This energy dissipates to thermal energy of the fuel. The core heat transfer process involves the fuel, fuel-cladding gap, cladding and coolant. A typical temperature profile is shown in Fig. 19. The mode of heat transfer from cladding to coolant is subcooled forced convection. Film boiling is avoided by maintaining a DNBR above 1.3. DNBR is the ratio of local heat flux at the departure from nucleate boiling (DNB) to actual local heat flux. (See Chapter 5.) The coolant enters the core at approximately 557F (292C) and exits at 607F (319C), thereby establishing the sink temperature for fuel to coolant heat transfer. The cladding to coolant convective and cladding conductive heat transfer rates are relatively high, therefore the fuel temperatures primarily depend on the gap and fuel conductances and the rate of energy deposition. The heat transfer calculation methods are described in Chapters 4 and 5. The melting temperature of unirradiated uranium dioxide (UO_2) is 5080F (2804C). This temperature decreases with burnup at the rate of approximately 58F (32C) per 10,000 MWd/t_m U, due to the accumulation of fission products. A design overpower maximum fuel centerline temperature of 4700F (2593C) has been selected to preclude melting at the fuel centerline.

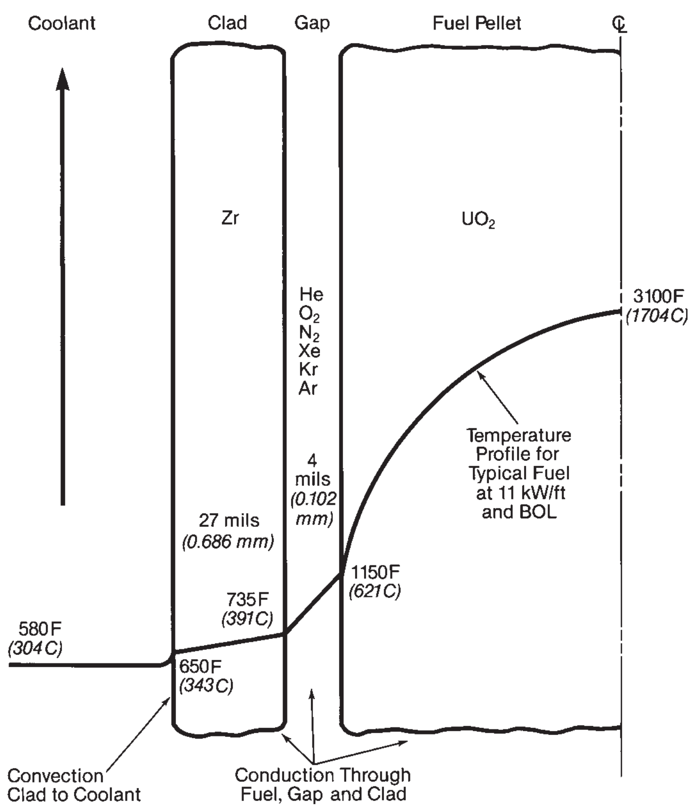

Fig. 19 Radial temperature profile for a fuel rod at 11 kW/ft.

Instrumentation

The reactor vessel instrumentation consists of the nuclear instrumentation and in-core monitoring system.

Nuclear instrumentation Neutron flux levels are monitored using a combination of source, intermediate and power range detectors. Each of these detectors is located at core mid-height, outside the reactor vessel but inside the primary shield. The distribution and types of detectors are:

Range	Detector Type	Number
Source	BF_3 Proportional	2
Intermediate	Compensated ion chamber	2
Power	Uncompensated ion chamber	4

These measurements are supplied to the reactor operator, safety parameter display system, reactor control portion of the integrated control system and reactor protection system. Additionally, the source and intermediate range readings are differentiated to provide startup rate, and the outputs of each pair of power range detectors are differenced to provide top to bottom flux imbalance.

In-core monitoring system Core performance is monitored using 52 in-core detector assemblies. These assemblies are arranged in a spiral fashion outward from the center of the core. Each assembly consists of seven local flux detectors, one background detector and one thermocouple, arranged as shown in Fig. 20. The seven local flux detectors, distributed over the axial core length, indicate the axial flux shape and provide fuel burnup information. The background detector indicates the integrated axial flux. These flux detectors are self-powered rhodium-103 instruments. The thermocouple indicates the exit coolant temperature.

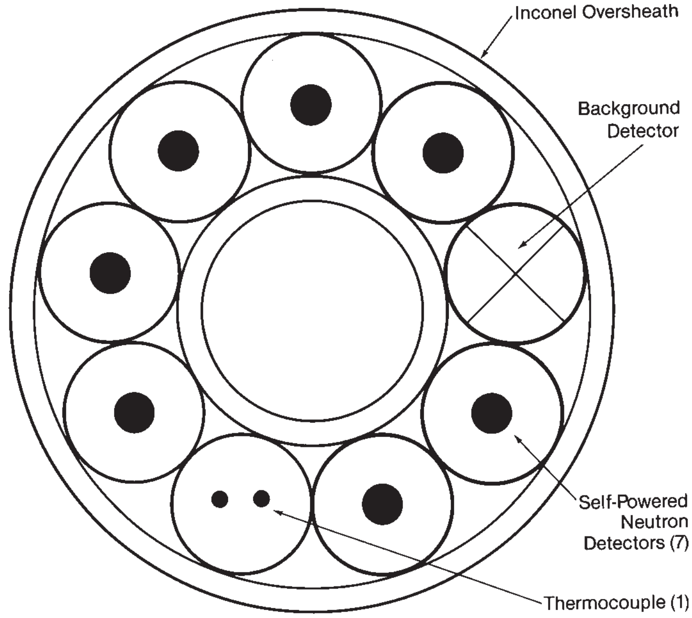

Fig. 20 Instrument tube cross-section.

Steam generators

Nuclear steam generators are particularly significant components in the NSSS and are discussed in more detail in Chapters 48 and 50.

The steam generators in PWR systems transfer heat from the primary coolant loop to the secondary flow loop, effectively functioning as heat sinks for the reactor core. The decrease of primary coolant temperature through the steam generators is virtually the same as the increase of primary coolant temperature through the core. Any difference is attributable to the fluid energy supplied by the reactor coolant pumps, less heat losses to ambient.

A variety of steam generator designs have been used in nuclear reactor systems.[2] However, two have been primarily used on second generation NSSS: the once-through steam generator (OTSG) and the more prevalent recirculating steam generator (RSG).

The OTSG is a straight tube counterflow heat exchanger where the primary coolant flows vertically downward inside the tubes and the secondary side water flows upward changing from slightly subcooled liquid to superheated steam at the outlet.

The RSG uses an inverted vertical U-tube bundle to transfer the energy and generate steam. The primary coolant passes through the U-tubes. The subcooled secondary side water mixes with water from the steam separators and passes up over the outside of the U-tube bundle as it is partially converted to steam. The steam-water mixture passes through multiple levels of steam separation equipment which returns the water to the U-tube bundle for further heating and evaporation and passes the saturated steam to the power-producing system.

Pressurizer

Description

The pressurizer (Fig. 21) is a tall, cylindrical tank connected at the bottom to a reactor coolant loop hot leg through 10 in. (254 mm) diameter surge line piping. Spray is introduced near the top of the pressurizer through a nozzle and 4 in. (102 mm) diameter line from a cold leg. Three replaceable heater bundles are installed over the lower portion of the pressurizer. Pressure relief devices are mounted at the top of the unit; these include two code safety valves and the power operated relief valve (PORV).

Function

The pressurizer controls primary system pressure. During normal operation, the pressurizer volume of 1500 ft^3 (42.5 m^3) contains equal portions of saturated liquid and saturated steam. Should the primary system liquid expand, such as through an increase of coolant temperature due to a load reduction, the excess volume displaces liquid into the surge line, raising the pressurizer level. This evolution is termed an *insurge*. The pressurizer steam is compressed, raising the primary system pressure and causing some of the steam to be condensed. The specific volume of liquid is much less than that of steam, therefore the steam condensation counteracts the ongoing pressure rise. The opposite occurs during an *outsurge*, the displacement of pressurizer liquid toward the hot leg through the surge line. In this case, the primary system pressure decreases, saturated pressurizer liquid flashes to steam and the net increase of specific volume through the change of phase suppresses the depressurization.

The pressurizer liquid volume is sized to maintain liquid above the pressurizer heaters and to maintain pressure above the HPI actuation point during the outsurge due to a reactor trip. The pressurizer steam volume is sized to retain a steam bubble during the insurge due to a turbine trip. Operation without a pressurizer steam bubble is termed solid plant operation. The pressurizer is sized to avoid this mode in which pressure control is encumbered.

Heaters

The pressurizer heaters maintain the pressurizer liquid at the saturation temperature, replacing the heat losses to ambient. They also raise and/or restore primary system pressure by elevating the saturation temperature of the pressurizer fluid. The three bundles of heaters are divided into five banks. There are two essential banks and three nonessential banks. The essential banks are energized automatically, based on primary system pressure. Essential bank no. 1 is energized below 2150 psig (14.82 MPa) and is subsequently de-energized above 2160 psig (14.89 MPa); the corresponding pressures for essential bank no. 2 are 2145 and 2155 psig (14.79 and 14.86 MPa). Nonessential bank no. 3 is sized to maintain the pressurizer fluid temperature during normal, steady-state operation. Banks no. 4 and no. 5 provide additional heater capacity that can be used during system startup or load changes.

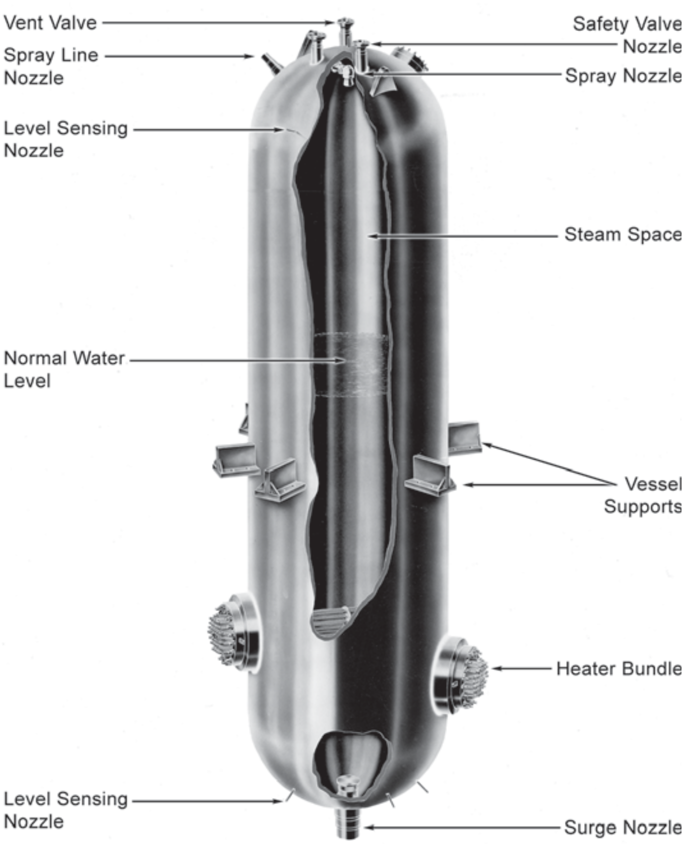

Fig. 21 Nuclear steam system pressurizer.

Spray

The pressurizer spray system lowers primary system pressure and counters an increasing pressure transient. The actuation of pressurizer spray introduces 550F (288C) cold leg fluid into the pressurizer steam space. The resulting condensation decreases the pressurizer fluid volume, thereby decreasing pressure.

The spray enters the pressurizer through the spray nozzle. This nozzle imparts rotational motion to the fluid, generating a downward directed, hollow spray cone. This pattern enhances steam condensation efficiency while minimizing contact of the cold spray with the pressurizer vessel walls. Between the pressurizer wall and the spray nozzle, the spray line makes a bend in the vertical plane. This loop seal line configuration is designed to maintain liquid in the spray line, thereby alleviating the temperature changes experienced by the spray nozzle.

The spray line is attached to the cold leg piping just downstream of the reactor coolant pump. The spray is driven by the RCS fluid pressure drop across the reactor vessel, and the full spray flow rate at normal operating conditions is approximately 170 GPM (10.7 l/s). The spray flow control valve is normally operated in a modulated rather than on/off mode. When in automatic mode, the spray flow control valve opens 40% when the primary system pressure increases to 2200 psig (15.17 MPa) and closes when pressure decreases to 2150 psig (14.82 MPa). The 40% open setting permits a spray flow rate of approximately 90 GPM (5.68 l/s).

A continuous flow rate of approximately 1 GPM (0.063 l/s) is maintained to minimize the thermal transients of the spray nozzle; this is accomplished by a bypass line around the spray flow control valve. An auxiliary system provides pressurizer spray when the reactor coolant pumps are inactive, such as during decay heat removal cooling of the RCS.

Pressure control devices

Pressure increases are countered by pressurizer steam condensation through compression and by pressurizer spray actuation. Should pressure continue to rise, the PORV opens at 2450 psig (16.89 MPa). A pressure actuated solenoid operates the pilot valve, which actuates the PORV. The PORV discharges into the pressurizer quench tank and can be isolated by closing the PORV block valve. A hypothetical continuous rod withdrawal accident from low power would cause core power generation to greatly exceed steam generator heat removal. The two code safety valves are sized to prevent RCS pressure from exceeding the 2750 psig (18.96 MPa) limit. These valves open at 2500 psig (17.24 MPa) and also discharge to the pressurizer quench tank.

Pumps and piping

Description and function

Coolant is transported from the reactor vessel to the two steam generators through hot leg piping. The coolant is returned to the reactor vessel from the generators through four cold legs, two per generator. Fluid circulation is provided by four reactor coolant pumps, one per cold leg. The arrangement of the piping and pumps is shown in Figs. 2, 3 and 4.

The hot leg and cold leg piping is fabricated from carbon steel and is clad on the inside with type 304 or 316 stainless steel. The inside diameters of the hot leg and cold leg piping are 36 and 28 in. (914 and 711 mm) and their nominal wall thicknesses are 2.75 and 2.5 in. (69.9 and 63.5 mm), respectively. The other piping attached to the RCS is either fabricated from stainless steel or clad with stainless steel.

Thermal sleeves are used to minimize the thermal stresses generated by rapid fluid temperature changes; they are installed on the four high pressure injection (HPI) nozzles and two core flood nozzles.

Pump characteristics

The reactor coolant pumps are vertical suction, horizontal discharge, centrifugal units. They are further characterized as diffused flow, single stage, single suction, constant speed, vertical centrifugal, controlled leakage pumps having five-vane impellers. The pumps are driven by constant speed, vertical squirrel cage induction motors. Pump sealing is accomplished using three-stage mechanical seals. The pump and motor are just over 27 ft (8.2 m) tall (see Fig. 22). The motor develops approximately 9000 hp (6714 kW) at cold conditions and 1185 rpm (124 rad/s) using a 6600 volt, three-phase power supply. The locked rotor starting current is approximately 3800 amperes. At normal operating conditions, the motor develops approximately 8300 hp (6192 kW), drawing a current of 685 amperes. Each unit's rotational moment of inertia is in excess of 70,000 lbf ft^2 (28,928 Nm^2). This rotational inertia is sufficient to sustain rotation for approximately 45 seconds following a power interruption. Each reactor coolant pump has a nominal capacity of 92,400 GPM (5829 l/s) with a discharge head of 403.5 ft (122.99 m) at operating RCS conditions.

Integrated control system

Function

The integrated control system (ICS) simultaneously controls the reactor, steam generators and turbine to obtain a smooth, rapid response to load changes. The ICS operates the control rods to regulate core power, adjusts the feedwater flow to control the rate of steam production, and operates the turbine throttle valves to control electrical power output. This control method combines the advantages of two alternative approaches.

The NSSS responds inherently to load changes without an ICS, but this response causes undesirable system variations. Consider the response to an increased load demand without integrated control. The increased electrical load causes the turbine throttle valves to open to maintain turbine speed. The increased steam demand lowers steam pressure at the turbine and within the steam lines and steam generators. Feed flow rate is increased to match the increased steam flow rate. The increased steam generator heat transfer lowers the temperature of the reactor coolant returning to the core. Core power gradually increases due to the reactivity gained from the negative temperature coefficient. This means that the increased reactor coolant density enhances the moderation of neutrons, thereby increasing the thermal neutrons available to cause thermal fissions. Also, the control rods are withdrawn to maintain the average primary fluid temperature. These primary system interactions would be reversed as the core and steam generator stabilized at the new, higher power level. This method of feedback control has the advantage of immediately supplying the required electrical load.

An extremely stable but slow method of NSSS control is termed the turbine-following method. The turbine output is varied only as the steam pressure is adjusted for the revised conditions. Again consider the response to an increased load demand. The control rods are withdrawn to increase reactor power and the feed flow rate is increased to support a higher rate of steam generation. As the turbine throttle pressure begins to increase, the throttle valves are gradually opened, allowing the turbine output to respond to the increased load demand.

The ICS operates all three systems: turbine, feedwater flow and control rods, to combine the rapid response of the unregulated system with the stability of the turbine-following system. The key to ICS operation is the turbine header pressure setpoint. The turbine governor valves maintain turbine speed and turbine header steam pressure. Because of the thermal inertia of the NSSS, the steam production rate can not be changed as rapidly as the steam demand varies in response to a turbine load change. This time delay is handled by temporarily changing the turbine

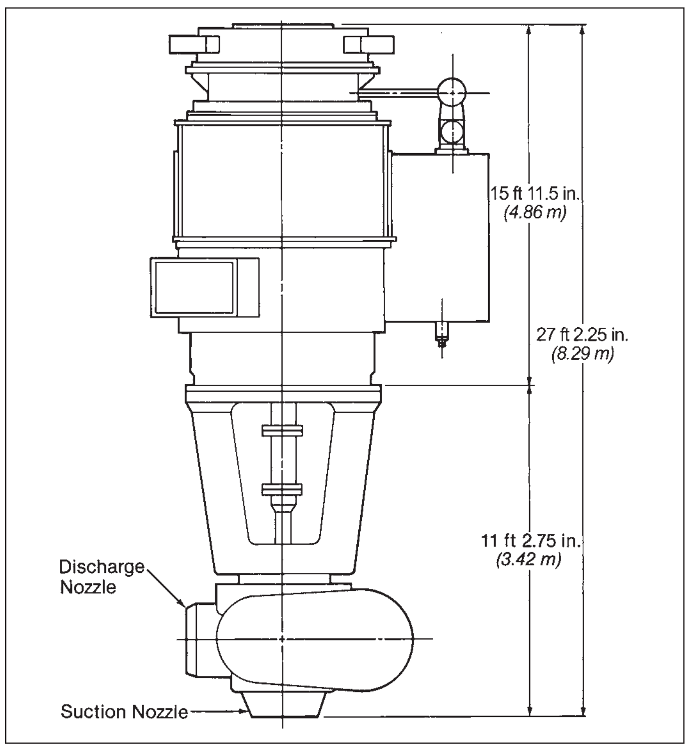

Fig. 22 Reactor coolant pump arrangement.

header pressure setpoint. For example, again consider a load increase. The ICS causes the control rods to be withdrawn and the feedwater flow rate to be increased. Simultaneously, the ICS temporarily reduces the turbine header pressure setpoint. The turbine governor valves open to maintain turbine speed and to reduce the header steam pressure to the new setpoint. As core power, primary to secondary heat transfer and steam flow rate gradually increase, the turbine header pressure recovers and its set point is returned to the steady-state value. In this fashion, the ICS obtains a rapid response and a smooth transition between turbine loads.

Limits and controls

The ICS recognizes a variety of limiting conditions, such as a reactor coolant pump trip, a feed pump trip or an asymmetric control rod fault. It imposes the appropriate load and load change limits corresponding to these conditions.

Special ICS signals are imposed below 15% of full power. During system startup, the reactor and steam generator power levels are increased to approximately 10% power by steaming through the turbine bypass valves. The turbine-generator is rolled, synchronized with the electrical distribution system and gradually loaded. The turbine control station is placed in automatic mode at approximately 15% turbine load. The reactor is then controlled to maintain a constant average coolant temperature of 582F (306C) above 15% power, using the RC subsystem. From 0 to 15% power, manual control is used to increase the average coolant temperature from 532F to 582F (278 to 306C).

Reactor protection system

Function

The reactor protection system (RPS) trips the reactor when limiting conditions are approached. These limits prevent boiling of the reactor coolant, limit the local core power generation rate (the linear heat rate) and minimize challenges to the PORV.

Multiple independent RPS channels monitor critical reactor trip functions. If any two of these channels detect single or multiple trip functions outside normal parameters, the control rod drive breakers are opened, tripping the reactor. Depending upon the NSSS design, the trip functions can include:

1. ARTS (anticipatory reactor trip system) main feedwater pump trip,
2. ARTS turbine trip,
3. overpower trip,
4. high outlet temperature trip,
5. high pressure trip,
6. reactor building high pressure trip,
7. pressure/temperature trip,
8. low pressure trip,
9. power/imbalance/flow trip, and
10. power/reactor coolant pumps trip.

The two ARTS trip functions trip the reactor before the RCS pressure rises to the reactor set point, thereby minimizing the activation of the PORV (power operated relief valve). The reactor can also be manually tripped from the control room.

Safety features actuation system

The safety features actuation system (SFAS) engages the emergency core cooling system (ECCS) in the event of a breach of the primary system boundary. The two SFAS signals are low RCS pressure and high reactor building pressure. A schematic of the various systems is shown in Fig. 23. The SFAS operates the following systems:

1. high pressure injection,
2. low pressure injection,
3. reactor building cooling, and
4. reactor building spray.

In addition, the SFAS activates the two emergency diesel generators.

Each of the three independent measurements of RCS pressure is fed to a trip bistable. This input is combined with the output of a reactor building pressure bistable; the trip of either pressure bistable trips the output. This output signal, combined with those of the other two pressure logic signals, engages the SFAS should two out of three trip. Reactor building spray is also initiated using two out of three logic, but is based only on reactor building pressure. In addition, the reactor building spray is delayed five minutes from the time of actuation.

Certain features are bypassed when the system is normally depressurized. These include high pressure injection, low pressure injection, reactor building isolation and reactor building cooling. Bypass must be initiated at an RCS pressure of less than 1850 psig (12.76 MPa); the system is automatically reinstated when the pressure exceeds 1850 psig (12.76 MPa).

High pressure injection system

The high pressure injection (HPI) system provides water to the RCS during a loss of coolant accident (LOCA). It is designed to prevent core uncovery in the event of a small RCS leak, during which the RCS pressure remains elevated, and to delay core uncovery in the event of an intermediate size break. The HPI system is activated by the SFAS signal when RCS pressure decreases to 1600 psig (11.03 MPa) or when reactor building pressure increases to 4 psig (27.6 kPa).

The HPI system consists of three pumps, a common discharge header and four lines that are equipped with motor operated isolation valves leading to the four cold legs. One HPI pump operates continuously to provide water for the reactor coolant pump seals. Two pumps are always available for automatic safety injection. The SFAS signals activate both pumps and their auxiliary equipment. The system also opens the four isolation valves to preset throttled positions to produce 125 GPM (7.89 l/s) per line using either of the two HPI pumps.

The HPI system is designed redundantly. Either pump is sufficient to meet the system requirements. The pumps are located in separate rooms and draw

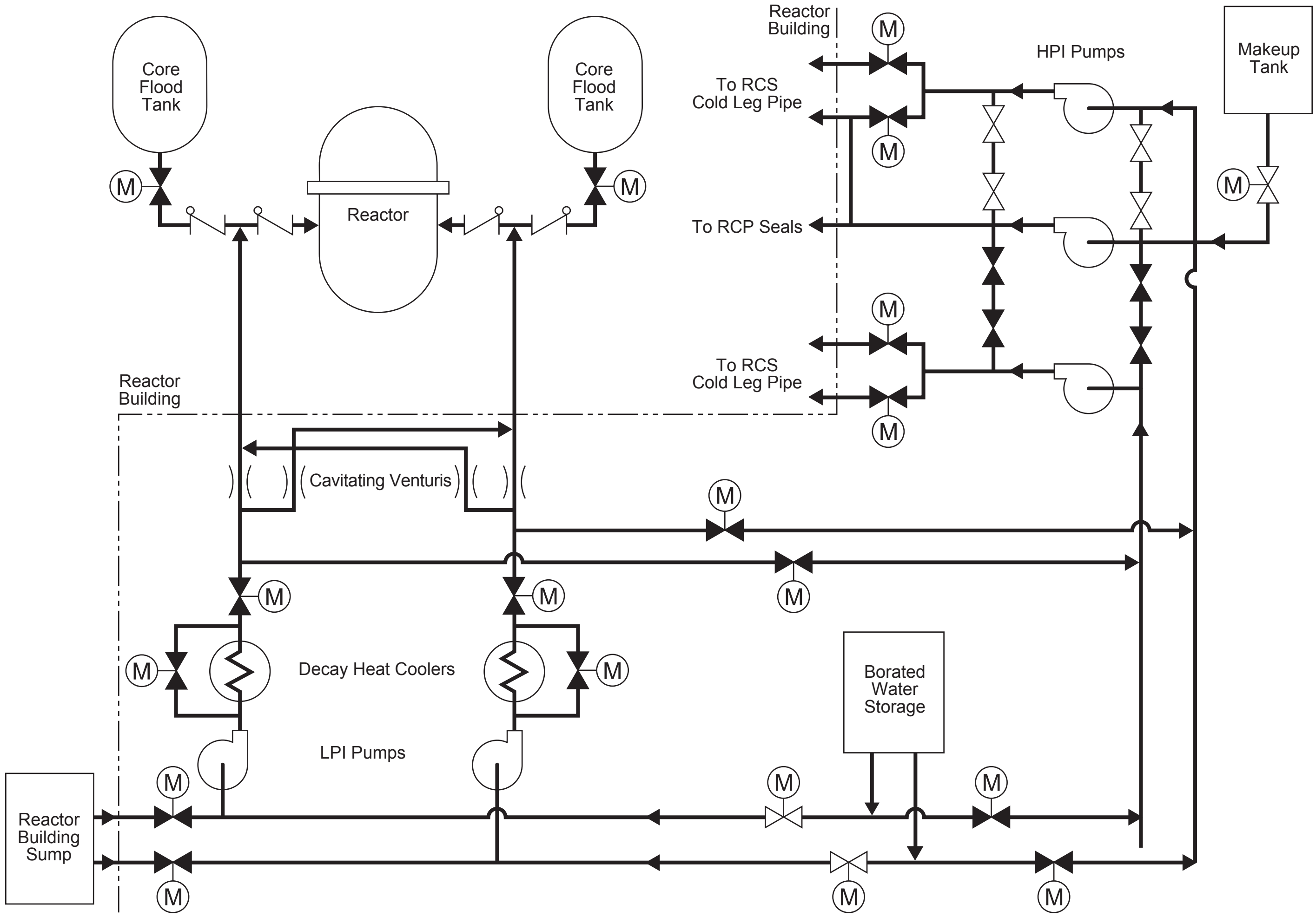

Fig. 23 Emergency core cooling/injection system.

water from the borated water storage tank (BWST) through separate lines. The SFAS channels are physically separated. The HPI pump and valve power supply lines are independent, physically separated and energized from independent power sources. The SFAS also activates the emergency diesel generators that provide backup power to the HPI system.

HPI fluid enters each RCS cold leg midway in the down-sloping piping run at the reactor coolant pump discharge. The fluid enters horizontally at the side of the cold leg pipe. The HPI nozzles are equipped with sleeves to minimize the thermal stress caused by the cold water injection.

The HPI system can be used during normal operation as part of the makeup and purification system. Either HPI pump can supply RCS makeup and seal injection flow to the reactor coolant pumps; these functions are normally performed by the makeup pump.

Low pressure injection system

The low pressure injection (LPI) system provides water to the core in the event of a large LOCA which depressurizes the RCS. The two LPI pumps discharge 3000 GPM (189.2 l/s) each at a rated head of 350 ft (106.7 m). [Shutoff head is 435 ft (132.6 m).] The borated water storage tank (BWST) supplies the LPI pumps. The pumps are actuated independently and automatically by the SFAS signal, as are the four motor operated isolation valves. They discharge to the reactor vessel upper downcomer through two core flood nozzles.

The two BWSTs have a capacity of 450,000 gal (1,703,430 l) each. The boron concentration is maintained above 1800 ppm at 80F (27C). The storage tank inventory provides at least 30 minutes of operation of all ECCS pumps. When the BWST inventory is depleted, the LPI pumps are manually transferred to the reactor building emergency sump, which collects water lost from the RCS. In the event that the HPI pumps are still required, such as with RCS pressure higher than the LPI pump shutoff head, the HPI pumps can be realigned to draw from the discharge of the LPI pumps. This is referred to as piggy-back operation.

Core flood system

The core flood system provides a passive water supply to the core in the event of a large break loss of coolant accident. The system consists of two tanks, associated piping and valves. Each tank has a volume of 1410 ft^3 (39.9 m^3). Approximately two-thirds of this volume, or 7500 gal (28,391 l), is borated water; the remainder is nitrogen gas pressurized to 600 psig (4.14 MPa). Should the RCS depressurize to less than

600 psig (4.14 MPa) in the event of an LBLOCA, the nitrogen cover gas forces the core flood tank (CFT) water into the RCS. The boron concentration of the CFT water is maintained between 2270 and 3490 ppm.

Each CFT discharges through 14 in. (356 mm) lines, dual check valves and a core flood nozzle into the upper downcomer of the reactor vessel. The two CFTs use separate piping and nozzles. Each tank is equipped with a motor operated isolation valve to prevent tank discharge when the RCS is normally depressurized. These valves are open when the RCS is pressurized.

Reactor building cooling and spray systems

The reactor building emergency cooling and spray systems are actuated in the event of a loss of coolant accident. The spray system also reduces the post accident level of fission products in the reactor building atmosphere through chemical reaction. The systems are designed such that either one can offset the heat released by the escaping reactor coolant.

The systems are activated by the SFAS signals. The emergency cooling system is actuated within 35 seconds after the RCS pressure decreases to 1600 psig (10.03 MPa) or the reactor building pressure increases to 4 psig (27.6 kPa). The emergency spray system is actuated 5 minutes after the reactor building pressure reaches 30 psig (206.8 kPa). Both systems can also be actuated manually.

The reactor building emergency cooling system includes four units, each consisting of an air circulator and a cooling coil. The 40,000 ft^3/min (18.9 m^3/s) circulators draw air from a point high within the reactor building dome and discharge downward toward the cooling units. Two of the four units are equipped with activated charcoal filters for removing fission products. The cooling units reject heat to the nuclear service cooling water system.

The reactor building emergency spray system includes two spray trains, each consisting of a pump, spray header, spray additive tank and eductor and associated isolation valves, piping and controls. The pumps are supplied from the BWST and, later in an accident, from the reactor building emergency sump.

The 300 hp (224 kW), 1500 GPM (94.6 l/s) spray pumps discharge through the spray additive eductors, drawing in sodium hydroxide. The spray additive concentration is sufficient to bring the entire post-accident inventory of reactor building water to a pH of 9.3. The spray discharge is distributed through 100 spray nozzles per header that are arranged to provide a uniform spray throughout the reactor building, above the operating floor.

Decay heat removal system

The RCS is periodically cooled and depressurized for maintenance and refueling. The first portion of the cooldown is accomplished by circulating the primary coolant using reactor coolant pumps and by removing heat using the steam generators. This method becomes impractical as the RCS pressure is reduced towards the net positive suction head (NPSH) of the reactor coolant pumps and as the primary to secondary system temperature difference diminishes.

The decay heat removal system (DHRS) is used to complete the RCS cooldown below 225 psig (1.55 MPa) and 290F (143C). The DHRS also performs the following functions:

1. purifies the reactor coolant during cold shutdown,
2. refills the RCS following maintenance,
3. cools and adds boron to the spent fuel pool, and
4. transfers water between the borated water storage tank and the fuel transfer canal during refueling.

The DHRS also functions as the low pressure injection system during a loss of coolant accident.

The DHRS is activated approximately six hours after reactor shutdown. It can reduce the RCS temperature from 280F to 140F (138 to 60C) within 14 hours. The DHRS pumps draw reactor coolant from the hot leg piping, just beyond the reactor vessel, through a 12 in. (305 mm) decay heat drop line. The DHRS fluid is discharged through coolers to the reactor vessel upper downcomer through the core flood nozzles. The injected DHRS fluid then follows the usual flow path down the reactor vessel downcomer, up through the core and out the hot leg to the DHRS inlet, thereby completing the flow circuit. The DHRS flow rate and rate of RCS cooling are controlled by bypassing a portion of the DHRS flow around coolers. The flow rate per cooler is limited to approximately 3000 GPM (189.2 l/s). The heat transferred in the coolers is removed by the nuclear service cooling water (NSCW) system, and the temperature difference between the RCS and the DHRS cooler outlet is maintained at approximately 30F (17C). The decay heat removal suction header temperature is measured to monitor RCS cooldown and the return header flow rates are also measured and remotely indicated. The DHRS suction block valve is interlocked to prevent inadvertent actuation at RCS pressures above 225 psig (1.55 MPa).

A portion of the DHRS flow rate can be diverted to the makeup and purification system for purifying the reactor coolant. This flow path is from the discharge of a DHRS pump; through the letdown filter, purification demineralizers and makeup filters; and back to a DHRS pump inlet header. The purification demineralizers are limited to a maximum fluid temperature of 135F (57C).

Makeup and purification system

The functions of the makeup and purification system are as follows:

1. control RCS inventory,
2. purify and degasify the reactor coolant,
3. maintain coolant boron concentration,
4. add chemicals to the reactor coolant for pH control,
5. supply seal injection flow to the reactor coolant pumps and handle seal return flow, and
6. add borated water to the core flood tanks.

The makeup and purification system can also function as part of the high pressure injection system.

The makeup and purification system consists of the letdown, purification and makeup portions. The letdown portion draws reactor coolant from the cold leg suction through a 2.5 in. (63.5 mm) line. The effluent is cooled using one of the three letdown coolers. Radioactive nitrogen-16 is allowed to decay in the letdown delay line, a 15 ft (4.6 m) length of 12 in. (305 mm) piping. Depressurization and flow control are provided by the letdown orifice; the nominal flow rate of 45 GPM (2.83 l/s) may be increased to 140 GPM (8.83 l/s) by bypassing the orifice.

The flow rate of 45 GPM (2.83 l/s) processes one RCS volume daily. Letdown flow rate, temperature and pressure are measured. Multiple motor operated valves are used to isolate the system in the event of SFAS operation.

The purification portion of the system consists primarily of the letdown filters, mixed bed demineralizers and makeup filters. The demineralizers are provided temperature protection by temperature actuated isolation valves. The purified effluent may be directed to the makeup tank or diverted to the bleed tanks when reducing the RCS fluid inventory.

The 4400 gal (16,657 l) makeup tank is central to the makeup and purification system. It receives purified letdown flow and seal return flow. It also serves as the point of boron and hydrogen addition to the RCS. Boron is used for reactivity control, lithium hydroxide provides pH control and hydrogen or hydrazine is used for oxygen control. The makeup tank acts as a surge tank to accommodate temporary changes of RCS inventory and provides water for the makeup pump. Finally, it can be used to degasify the reactor coolant.

A 4 in. (101.6 mm) line connects the makeup tank to its pump, with cross connects to the HPI pumps. The pump discharges are recombined and a 2.5 in. (63.5 mm) diameter flow line is routed back to the seal return coolers to cool the operating pumps. The makeup flow rate is measured and remotely indicated and is controlled by the pressurizer level control valve. Makeup is injected into the RCS through the cold leg HPI nozzle at the reactor coolant pump discharge, thereby completing the flow circuit.

During power operation, the makeup and purification system operates continuously to regulate RCS inventory. The makeup flow control valve is adjusted automatically to maintain pressurizer level. The flow rate of the makeup pump is the sum of the makeup flow rate (to the RCS), the seal injection flow rate to the reactor coolant pumps and the makeup pump recirculation flow rate. The net makeup system flow rate to the RCS is the sum of the makeup and the seal injection flow rates, less the seal return flow rate.

Abnormal transient operating guidelines

The abnormal transient operating guidelines (ATOG) represent a symptom-oriented response to plant transients. These guidelines provide the operator with a clear and effective method of correcting abnormal conditions. ATOG involves the identification and correction of key upset conditions, regardless of their cause. These actions alone are sufficient to ensure core covery and cooling and to ensure plant safety.

The three key symptoms of ATOG are loss of subcooling margin, inadequate heat transfer and excessive heat transfer.

The subcooling margin refers to the difference between the saturation temperature at RCS pressure and an RCS temperature. A positive subcooling margin ensures that the core is covered and therefore cooled. A loss of subcooling margin, on the other hand, indicates steam generation within the RCS and may indicate core uncovery. The operator must take action to restore the subcooling margin, such as by activating HPI pumps or a primary system heat removal process.

Subcooling margin is the key ATOG indicator. However, inadequate or excessive heat transfer indications must also be remedied. Inadequate heat transfer is countered by restoring primary to secondary system cooling and/or by initiating HPI-PORV cooling. In the latter method, core decay heat is removed by actuating full HPI and the PORV. The HPI fluid flows into the cold leg discharge piping leading to the reactor vessel. It flows through the core, out the hot leg, into the pressurizer through the surge line and out the PORV. Excessive heat transfer is remedied by reducing the rate of primary to secondary system heat transfer. This can be done by throttling the flow of auxiliary feedwater to the steam generators, restoring steam generator secondary pressure and reducing the primary to secondary temperature difference.

Exclusive among the events that can give rise to abnormal ATOG symptoms is the steam generator tube rupture (SGTR) accident. This event is readily identified using radiation indications and alarms and indications of steam generator conditions: pressure, temperature and level. Event oriented operator actions are taken to minimize radiation release and to ensure core cooling.

NSSS design: today and in the future

There are currently 103 operating, fully licensed nuclear power reactors in the United States representing 97.5 GW of capacity and generating 780 billion kWh in 2002, or 20% of the total U.S. generation.[2,3] These 103 NSSS units fall into one of three categories:

	No.	MW
Pressurized Water Reactor (PWR)		
Recirculating Steam Generator (RSG)	62	59,793
Once-Through Steam Generator (OTSG)	7	5,915
Boiling Water Reactor (BWR)	34	31,792
Total	103	97,500

As of May 2004 there were 440 operating nuclear power systems worldwide with a capacity of 362 GW producing approximately 16% of global electricity generation.[4,5] More than 30 new plants are under construction.[6] As discussed in the introduction to this chapter, commercial NSSS designs outside of the U.S. fall into one of six general categories:

1. PWR – RSG
2. PWR – OTSG
3. BWR
4. PHWR (pressurized heavy water reactor, including the Canadian CANDU design)
5. GCR (gas-cooled reactor)
6. FBR (fast breeder reactor)

As with the PWRs, the PHWR, GCR and FBR also include steam generators to provide a heat sink for the primary side reactor coolant system and to generate steam for the secondary side power cycle. The PHWR system steam generators are very similar in design to the PWR recirculating steam generators. (See Chapter 48.) In particular, 33 Canadian CANDU PHWR systems with 21 GW of capacity are in operation in Canada, India, Pakistan, China, Argentina, Romania and South Korea. An additional 18 units (48 GW) in India are based upon the CANDU technology.

The current (second) generation of nuclear power plants in the U.S. has demonstrated decades of safe and reliable performance. Continual improvements have been made in maintenance and operation. In the past 20 years, the average capacity factor has increased from about 60% to more than 90%. This significant increase translates into an additional 23,000 MW of power to the grid – the equivalent of 23 new plants. Production costs (fuel, operations and maintenance) of most plants are less than US$0.02/kWh, and in the best plants about $0.01/kWh. The proven reliability has resulted in 25 of the current operating plants being granted licenses to continue operations for another 20 years beyond their original license period.

Despite this excellent performance, no new nuclear plants have been ordered in the U.S. for the past 26 years, although design of a third generation of nuclear systems, commonly designated Generation III, has proceeded. The key attributes of these new designs are:

1. simple and more rugged making these more resistant to upsets and easier to operate,
2. standardized to reduce cost and schedule,
3. increased availability,
4. extended design life (approximately 60 years),
5. reduced possibility of core meltdown accidents,
6. minimized environmental impact,
7. enhanced fuel burn-up to reduce waste, and
8. extended fuel life.

The most significant change has been the incorporation of passive or inherent safety features which do not require operator or control system intervention to avoid accidents if equipment malfunctions. These systems may include gravity assisted flow, natural circulation within and between components, and resistance to deterioration at elevated temperatures. Most of these NSSS designs (including PWRs, BWRs, PHWR/PWR hybrid, high temperature GCR) are evolutionary from the existing second generation systems. These systems are being introduced in other countries with two in operation by 2003 and others being built or ordered.

Beyond the Generation III designs, a number of countries including the U.S. have formulated a general agreement to explore the potential of several more revolutionary designs, designated Generation IV. Introduction of these new designs may be feasible starting around 2020. In addition to further improving the key attributes of the Generation III designs, the Generation IV designs will try to offer some radical new approaches to solve future energy needs and resolve environmental concerns. The new designs will employ fast cores and coolants that allow a marked increase in operating temperatures. The high temperatures can be used for chemical processes, including production of low cost hydrogen. The advanced fuel cycles can be arranged to extract more energy from the spent fuel and also to drastically reduce the toxicity of high level waste.

Nuclear ship propulsion

Nuclear power for propulsion has been applied to both commercial and naval vessels beginning with the U.S.S. *Nautilus*, the world's first nuclear ship (see Fig. 24). The development of nuclear ship propulsion also formed the basis for commercial pressurized water reactors (PWR) for land-based electric power generation. For submarines, nuclear propulsion has proven to be the greatest single advancement in post-World War II technology. This dramatic achievement virtually abolished range limitations and enabled the submarine to become a true submersible, freed from the need to make regular forays to the surface to recharge batteries.

Nuclear power has also proven extremely valuable for naval surface ships such as aircraft carriers and escort vessels. While the advantages are not as readily apparent as for submarines, the unlimited, sustained power available with nuclear propulsion allows the ship commander to devote his full attention toward effectively executing his mission without the continuous logistical concern for fuel oil supplies.

While any type of reactor system may be used for ship applications, only the PWR and the sodium-cooled reactor have been used. Of these, the PWR is the predominant system. A typical PWR schematic arrangement for shipboard application is shown on Fig. 25.

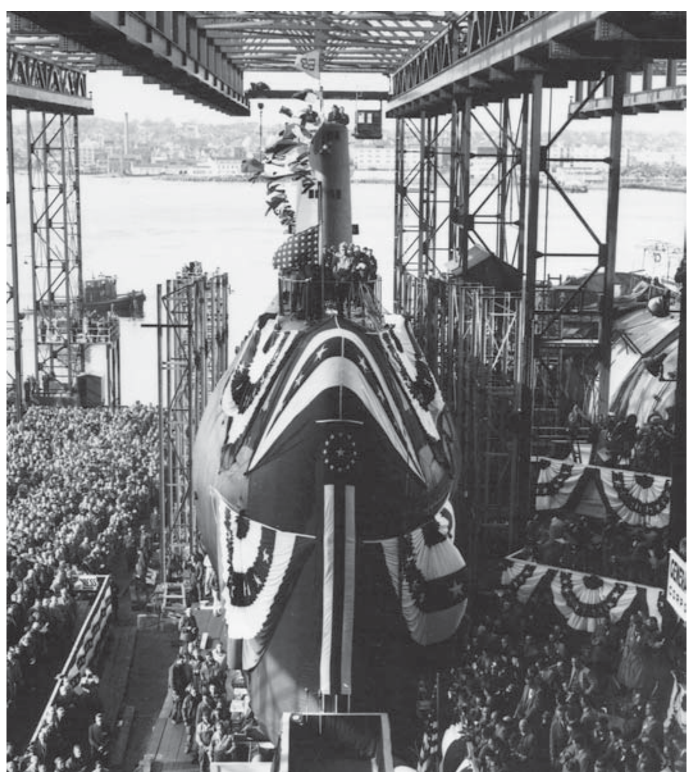

Fig. 24 Launching of the U.S.S. *Nautilus*, the world's first nuclear powered ship.

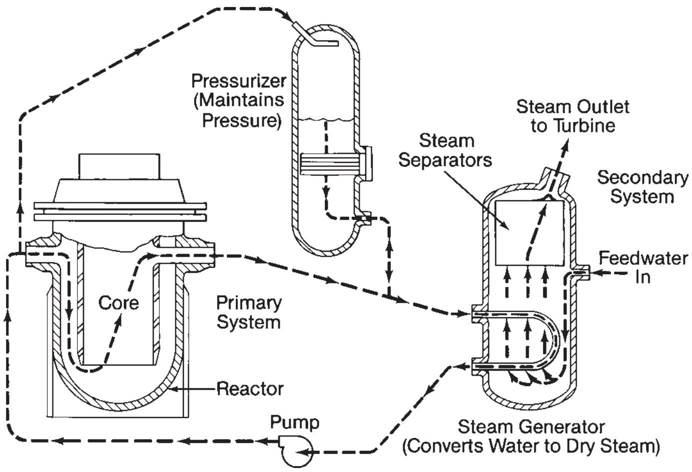

Fig. 25 Pressurized water reactor system – naval type.

There are important differences between shipboard reactors and land-based installations. These differences involve weight and space limitations, plant reliability and on-board maintenance, plant safety, and problems inherent with a moving platform. While the weight and size of the nuclear reactor itself does not present a problem, the weight and size of the radiation shielding around the NSSS is significant. These considerations usually dictate a more compact arrangement of the major components, as compared to land installations. However, the need to account for maintenance requirements forces the designer to balance compactness against access needs. Because access to ship units is more limited, reliability of shipboard components assumes even greater importance.

Safety is, of course, a major consideration. Shipboard nuclear plants are more subject to external hazards than land-based plants, particularly for naval units that may be exposed to extreme battle shock conditions. In the early stages of development, commercial ships showed promise. However, the public fear of accidents, especially with ships in harbor, essentially has prevented any further exploration of commercial ship installations. On the other hand, the U.S. Navy has maintained an excellent performance and safety record over the years, and U.S. naval ships are welcomed in major ports throughout the world.

Commercial nuclear ships

In the 1950s, several countries became interested in applying nuclear power to commercial shipping. These efforts resulted in the launching of the U.S. ship N. S. *Savannah* and the soviet ice-breaker *Lenin*. Later in the 1970s, Germany launched the N. S. *Otto Hahn* and Japan launched the cargo ship N. S. *Mutsu*. Fig. 26 is a schematic diagram of a typical marine propulsion plant.

Nuclear merchant ship *Savannah*

A nuclear merchant ship was first proposed by President Dwight D. Eisenhower in 1955, as evidence of the U.S.'s interest in promoting the peaceful use of atomic energy. After Congress approved the funding in 1956, the President directed the AEC and the Maritime Administration (MARAD) to design and construct the vessel that was subsequently named the *Savannah*.

The program's major objectives were to demonstrate the peaceful use of nuclear energy and resolve the problems of commercial marine reactor operation. Plant requirements included a conservative design with a long core life, use of commercially available materials and equipment wherever practical, and safety of operation.

The *Savannah* (Fig. 27) was a single-screw, geared turbine vessel, 595 ft (181.4 m) long, with a beam of 78 ft (23.8 m), draft of 29.5 ft (9 m), and a displacement of 21,900 t (19,867 t_m). The ship had a design speed of 22 knots (40.7 km/h) at 22,000 shaft hp (16,412 kW), accommodations for 60 passengers and 652,000 ft^3 (18,463 m^3) of cargo space.

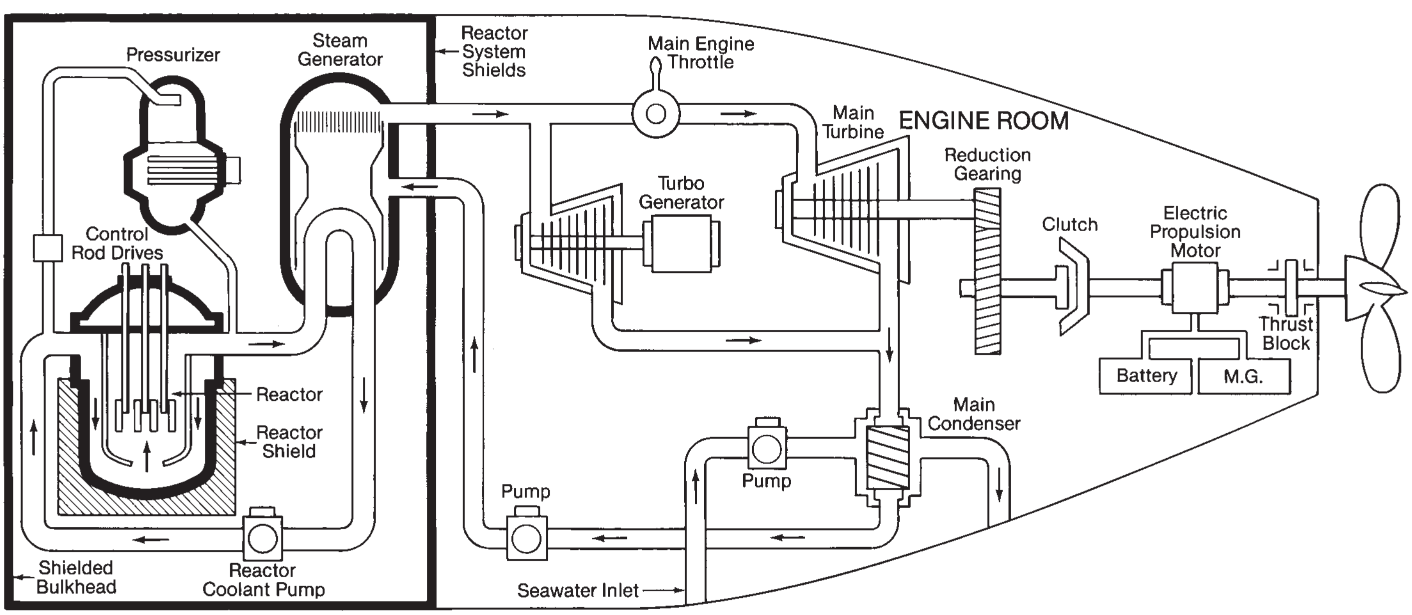

Fig. 26 Shipboard nuclear propulsion system.

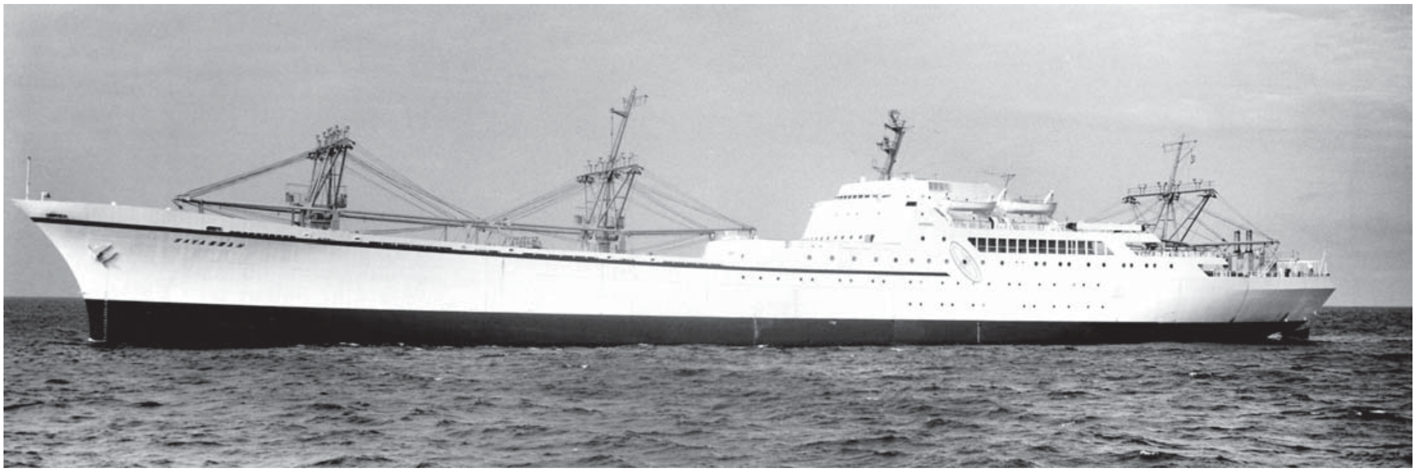

Fig. 27 N.S. *Savannah* – first nuclear merchant ship.

B&W supplied the PWR nuclear propulsion plant and auxiliaries (see Fig. 28) for the *Savannah* and trained the operating crew. Pertinent design data for the power plant are given in Table 1.

The keel of the *Savannah* was laid in 1958, and the ship was launched 14 months later. In 1962, the ship was delivered to the operating agent, and port visitations began. The *Savannah's* reactor was fueled by 15,653 lb (7100 kg) of uranium enriched to an average of 4.4% uranium-235. On her first fuel core, the *Savannah* traveled approximately 330,000 mi (531,089 km) and developed 15,000 full power hours with no shuffling of fuel. By late 1970, the *Savannah* had traveled more than 450,000 mi (724,205 km), visited 32 different U.S. ports in 20 states and 45 different foreign ports in 27 countries, and had been visited by more than 1,500,000 people.

The operation of the *Savannah* provided technology for future development of nuclear ships, and established standards for the design of the ship and reactor, operating practices and safety. After completing her mission, the *Savannah* was retired and is now a museum ship located in Charleston, South Carolina.

Nuclear merchant ship *Otto Hahn*

After building the N.S. *Savannah* reactor, B&W developed an improved nuclear marine plant known as the Consolidated Nuclear Steam Generator (CNSG). The CNSG was designed to achieve more economic nuclear propulsion systems for merchant ships, and has potential application for small- to medium-sized land-based plants.

The CNSG design incorporates the reactor and once-through steam generators within a single pressure vessel, achieving a compact arrangement and eliminating some of the auxiliary equipment.

The CNSG design was used successfully for the nuclear plant of the German N.S. *Otto Hahn* (Fig. 29). This ship was a single-screw geared turbine ore carrier 565 ft (172.2 m) long, with a beam of 77 ft (23.5 m), a draft of 30 ft (9.1 m), dead weight of about 15,000 t (13,608 t_m) and a design speed of 16 knots (30 km/h) at 10,000 shaft hp (7460 kW).

The nuclear plant for this ship was designed and constructed by the German-Babcock Interatom consortium with the assistance of B&W. A cross-section of the pressure vessel is shown in Fig. 30. Pertinent design data for the power plant at normal load are given in Table 2. The *Otto Hahn* began commercial service in 1970, and was decommissioned in 1978.

Naval nuclear ships

U.S. naval nuclear propulsion program

Background In 1946, the U.S. Navy's Bureau of Ships recognized that atomic energy might ultimately be developed for ship propulsion. Under the direction of Captain (later Admiral) Hyman G. Rickover, a team was established to transform existing theory and concepts into practical engineering designs.

The basic requirements for applying nuclear power to shipboard propulsion were clear. The reactor would have to produce sufficient power so that the ship would have military usefulness, and it would have to produce that power safely and reliably. The reactor plant would have to be rugged enough to meet the stringent requirements of a combatant ship, and be

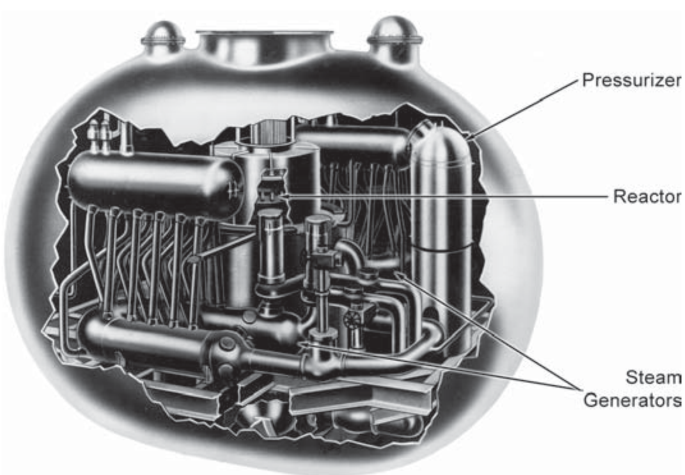

Fig. 28 N.S. *Savannah* nuclear steam supply system arrangement.

Table 1
N.S. *Savannah*

Power plant design data:	
Maximum shaft power	22,000 hp (16,412 kW)
Reactor power	70 MW
Total steam flow	226,000 lb/h (28.5 kg/s)
Turbine inlet pressure	430 psi (2.96 MPa)
Feedwater temperature	340 F (171C)
Reactor coolant data:	
Pressure in reactor	1735 psi (11.96 MPa)
Temperature in core	508 F (264C)
Flow	9,400,000 lb/h (1184.4 kg/s)

designed for operation by a Navy crew. A suitable reactor would require new corrosion resistant metals that could sustain prolonged periods of intense radiation, effective shielding to protect personnel from radiation, and the development of new components that would operate safely and reliably for prolonged time periods.

These problems were even more difficult for submarine applications because the reactor and associated steam plant had to fit within the confines of a small hull, and be able to withstand extreme battle shock. Although the application of nuclear power to submarines was a major challenge, success would revolutionize submarine warfare. No longer limited to submerged operation on battery power, a true submarine was possible, one that could travel submerged at high speed for long periods. Because of the immense challenge posed by this revolutionary approach, the Navy established extremely rigorous guidelines covering design, fabrication, quality, testing, and training for the suppliers and Navy personnel. These new guidelines demanded significant improvements in performance from all involved parties and established the design and engineering philosophy which underlies the basis of the commercial and Naval nuclear industry today.

Three reactor concepts were initially considered for naval nuclear propulsion. A study of a gas-cooled reactor showed that this concept was not then suitable. The pressurized water reactor and liquid metal-cooled

Fig. 29 N.S. *Otto Hahn (courtesy of Gesellschaft fur Kernenergieverwertung in Schiffbau und Schiffahrt mbH).*

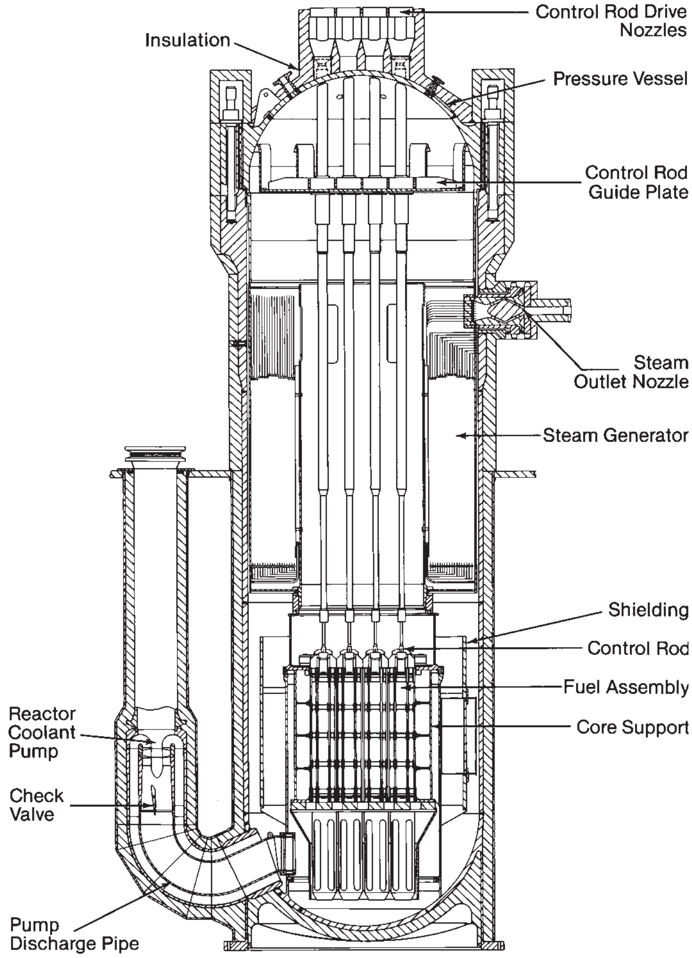

Fig. 30 Reactor vessel cutaway (N.S. *Otto Hahn*) (*courtesy of Gesellschaft fur Kernenergieverwertung in Schiffbau und Schiffahrt mbH*).

reactor approaches were found promising and carried through to full scale prototype plants, and thereafter to shipboard applications. B&W became actively involved in both designs.

U.S.S. *Nautilus*

In 1949, the U.S. Navy's Chief of Naval Operations issued a formal requirement for the development of a nuclear powered submarine. The following year, the U.S. Congress authorized funds for a land-based prototype of the pressurized water reactor that would power the world's first nuclear powered ship. B&W was

Table 2
German N.S. *Otto Hahn*

Power plant design data:	
Shaft power	10,000 hp (7460 kW)
Reactor power	38 MW
Total steam flow	141,000 lb/h (17.8 kg/s)
Superheater outlet press.	440 psi (3.03 MPa)
Superheater outlet temp.	523 F (273C)
Feedwater temp.	365 F (185C)
Reactor coolant data:	
Pressure in reactor	918 psi (6.33 MPa)
Temp. in reactor core	523 F (273C)
Flow	5,280,000 lb/h (665.3 kg/s)

selected to provide major reactor system components. Just three years later, the prototype began operation and for the first time, a reactor produced sufficient energy to drive power machinery.

On January 17, 1955, the *Nautilus* (Fig. 31) put to sea for the first time and radioed her historic message: "underway on nuclear power." On her shakedown cruise, the *Nautilus* steamed submerged more than 1300 mi (2092 km) in 84 hours – a distance that was 10 times greater than had been traveled continuously by a submerged submarine.

Within months, the *Nautilus* would break virtually every submarine speed and endurance record. Her global odyssey carried her under the seven seas, and on the first voyage in history across the top of the world, passing submerged beneath the North Pole.

On her first fuel core the *Nautilus* steamed more than 62,500 mi (100,584 km), more than half of which were totally submerged. During 25 years of service, she traveled a total of 600,000 mi (965,606 km) on nuclear energy, making the dream of Jules Verne come alive. The *Nautilus* is now a historic museum ship and is located in New London, Connecticut.

The *Nautilus* was the first application of a reactor power plant using pressurized water both as the primary coolant and as the heating fluid for converting the secondary water into steam. The recirculating steam generator (Fig. 32) was comprised of a straight tube and shell heat exchanger with riser and downcomer pipes connected to a separate steam drum. B&W designed and fabricated the prototype test steam generator, and the reactor vessel and pressurizer for the *Nautilus*.

U.S.S. *Seawolf*

The U.S.S. *Seawolf*, the second U.S. nuclear submarine, was launched in 1955, and her liquid metal-cooled reactor attained initial criticality in 1956. To help ensure tube integrity, the *Seawolf's* sodium-heated steam generators (Fig. 33) utilized sodium-potassium as a third or *monitoring* fluid in the annulus of the double-tube design. Although the ship operated satisfactorily for almost two years on its sodium-cooled reactor, overriding technical and safety considerations (mainly the potential for violent reaction between sodium and water) led to the abandonment of this type of reactor for propelling U.S. naval ships.

Fig. 31 U.S.S. *Nautilus*.

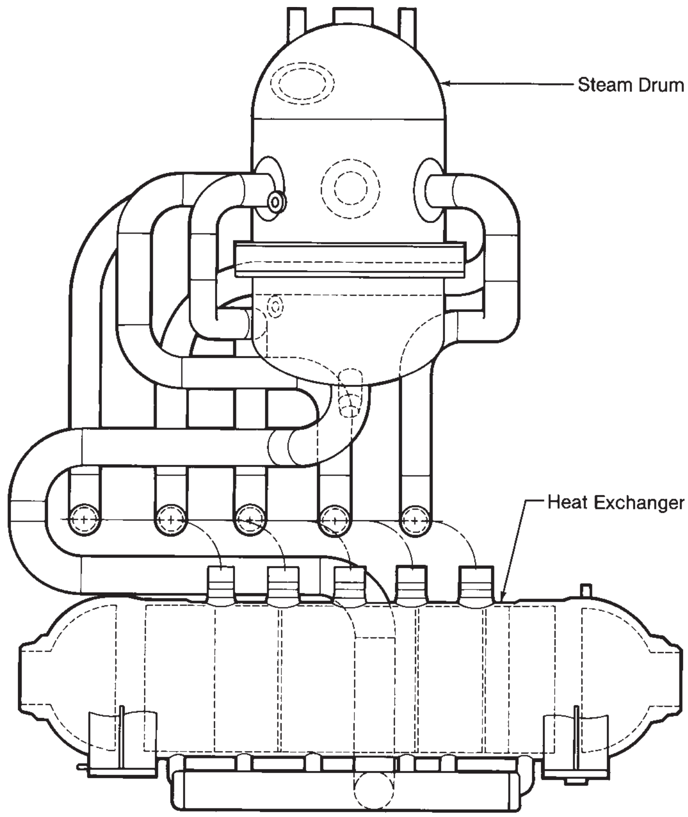

Fig. 32 U.S.S. *Nautilus* steam generator.

While liquid sodium is a much more efficient heat transfer medium than water, it can be very troublesome in service. Two problems are particularly noteworthy: the sodium has to be kept molten at all times or it will solidify and can damage primary system piping, and the sodium must be kept isolated from water to prevent a violent reaction.

In 1958, the *Seawolf* entered a shipyard where her sodium-cooled plant was replaced with a pressurized

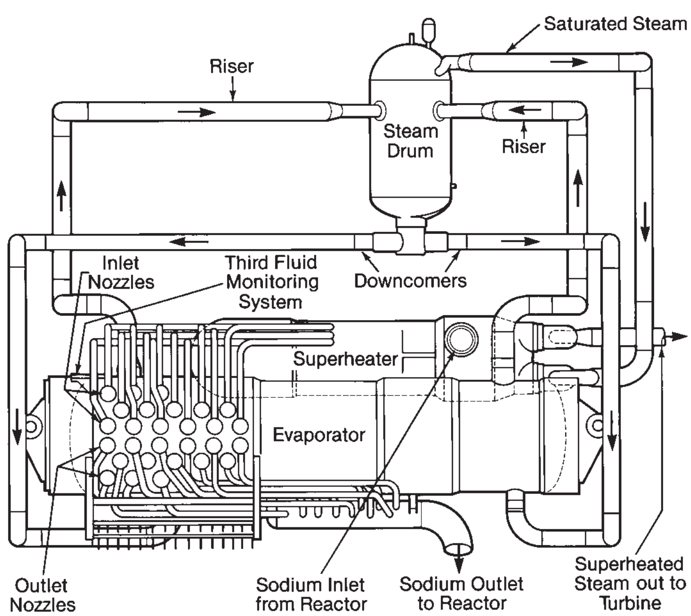

Fig. 33 Sodium-heated steam generator for U.S.S. *Seawolf*.

water reactor similar to that in the *Nautilus*. When her sodium plant was shut down for the last time, the *Seawolf* had steamed a total of 71,611 mi (115,247 km) of which 57,118 mi (91,923 km) were fully submerged.

U.S.S. *Skipjack*

The U.S.S. *Skipjack* attack submarine class combined the PWR plant with a streamlined *Albacore* hull shape to provide increased speed and reduced flow noise. In 1956, B&W received a contract for the design and fabrication of steam generators for the *Skipjack*. These were the first vertical recirculating PWR steam generators with integral steam separators and were the forerunner to the recirculating steam generator designs in current commercial PWR (non-B&W) plants and in CANDU reactor plants.

Technological advancements

Each new naval propulsion plant is a balance between the desire to make technological advances and a commitment to deliver a reliable fighting ship on schedule. Each new design can incorporate only a portion of all the potential improvements in technology. The ultimate success of any power plant design will depend on how well that balance is achieved.

In developing power plant components for modern naval vessels, improvements have been made in thermal hydraulic design, materials, structural design, and fabrication. Significant improvements in the art of steam-water separation have contributed greatly to the performance and compactness of naval steam generators.

Significant advances have also been made in reactor fuel technology. Reactor fuel lifetime has been extended from 2 years for the first Nautilus core, to more than 15 years for cores delivered in the 1980s. Efforts have been underway to develop reactor fuel that will last the life of the ship, making expensive and time consuming refueling unnecessary.

While much research has been done in the U.S. and in other countries on alternate forms of submarine power production, such as magnetohydrodynamics and fuel cells, nuclear power has proven to be highly reliable, safe and cost effective, and continues to be the system of choice for the U.S. Navy and other international fleets for the foreseeable future.

International naval nuclear programs

All of the principal maritime nations have studied the application of nuclear power to naval ship propulsion. The U.S., Great Britain, France, the People's Republic of China, and the FSU have built nuclear vessels. These nations all have naval nuclear fleets that rely primarily on pressurized water reactor technology. However, the Soviet Navy reportedly utilizes liquid metal cooled reactors in at least one attack submarine class for higher power output and greater operational speeds.

The Soviet *Typhoon*-class ballistic missile submarine (Fig. 34) is the largest submarine type ever built with a length of 563 ft (171.6 m) and a submerged displacement of 26,500 t (24,041 t_m). The Soviets' titanium-hulled *Alfa*-class attack submarines are reportedly the world's fastest and deepest diving with a speed of 45 knots (83 km/h) and an operational depth of 2500 ft (762 m).

Nuclear ship development is proceeding in other countries. The FSU is continuing development of new attack and missile submarines. The overall picture is somewhat unclear due to the economic conditions in the FSU and also because of continued safety issues, particularly in the aftermath of the sinking of the Kursk submarine in 2000. Both issues have led to a marked downturn in Soviet ship deployment. China has been developing new attack and ballistic missile nuclear submarines based on early Soviet designs. India is trying to develop a nuclear ballistic missile submarine.

Future U.S. Navy

The U.S. Navy has the largest nuclear fleet in the world, comprising 72 submarines and 10 aircraft carriers. In line with the ever changing situation in the world, the naval strategies and ship requirements are constantly evolving to meet the perceived global issues. Ship designs and power propulsion systems are constantly upgraded to meet the navy needs and to incorporate the newest technological advances.

Submarines

With the demise of the Cold War, the need for a large contingent of ballistic missile submarines has greatly diminished and strategy is now more focused on providing multi-purpose submarines with a broad range of missions. The U.S.S. *Virginia* (SSN774) is the first of a new class of attack submarines intended as a more cost-effective follow-on to the current Seawolf class. The ship is capable of both deep ocean

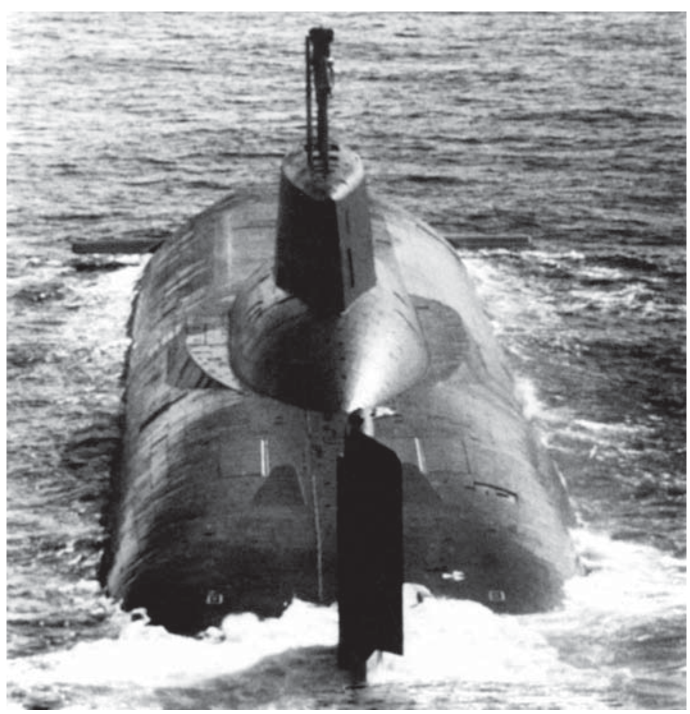

Fig. 34 Soviet *Typhoon*-class ballistic missile submarine.

warfare and shallow water operations of all types. Missions of the new ship include anti-submarine warfare, covert operations, personnel delivery, intelligence gathering, convert mine warfare, and Battle Group Support. The Virginia displaces 7800 tons and is 370 feet long. Fig. 35 shows a Virginia-class submarine under construction.

Aircraft carriers

In 2006, the U.S.S. *George W. Bush* will be the last of the Nimitz-class carriers to be built (see Fig. 36). Construction of a new class of carrier, the CVN 21 with 100,000 ton displacement, will begin at Newport News Shipbuilding in 2007 for delivery in 2014. The new ship will incorporate a new design of nuclear power plant, expanded flight deck, and a new electrical power distribution system. The new power plant will greatly increase the electrical power supply enabling the deployment of an electromagnetic aircraft launch system and offering scope for advancements in new electromagnetic weapons systems. The power plant and related equipment are designed to reduce maintenance and provide substantial reduction of personnel required to operate and staff the ship.

Fig. 35 Virginia-class submarine nears completion.

Fig. 36 U.S.S. *Nimitz* and U.S.S. *Ronald Reagan* are part of the current Nimitz-class aircraft carrier group.

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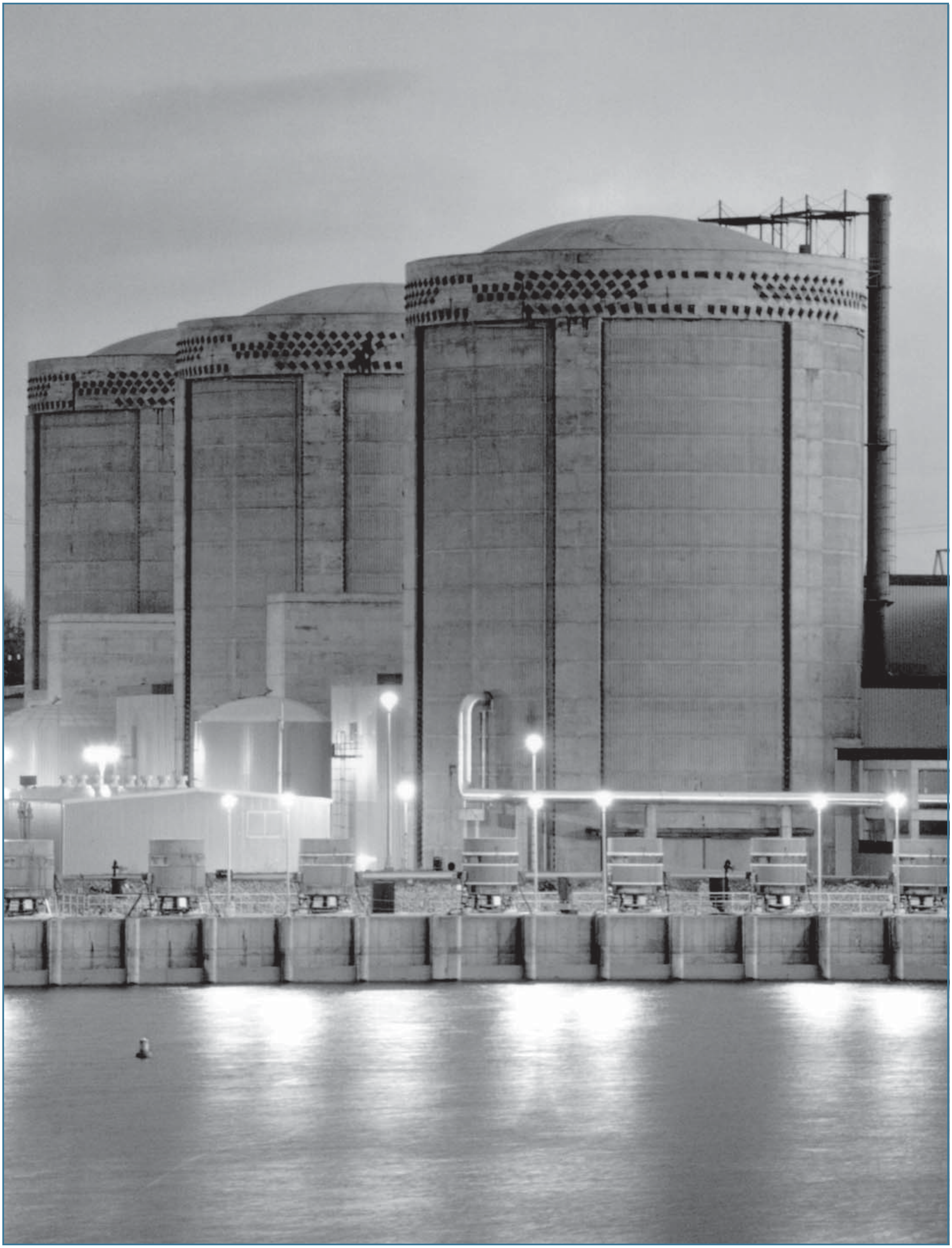

Oconee Nuclear Power Station, Duke Power Company.