WESTINGHOUSE

Engineer

SHIPPINGPORT ISSUE

World Radio History

SHIPPINGPORT STATION

The recent completion of the Shippingport Atomic Power Station—the nation's first large-scale nuclear electric power plant—highlighted an historic achievement.

A primary purpose of this plant, of course, is to further the peacetime use of atomic energy in the field of electric power generation. The technical feasibility of such a plant has long been acknowledged; however, this plant will provide much of the knowledge needed to evaluate nuclear plants from a combined technical, practical, and economic standpoint. The design, manufacture, operation, and maintenance of this full-size plant is providing accurate information that will contribute to improved plants in the future.

Thus the end result—the arrangement of concrete, metals, and other materials that make up the plant—is but a symbol of the achievement itself. And, as frequently happens with technical developments, the magnitude of the task is not fully apparent from the finished product.

The success of any technical development depends largely on the caliber of the people who contribute to it—and with increasing frequency on their ability to function as a smoothly operating team.

In the case of Shippingport, team effort was involved from the very start of the project. Initially, the team consisted of the Atomic Energy Commission, which sponsored the project; Duquesne Light Company, which built the generation portion of the plant, contributed funds toward the nuclear development program and is operating the plant; and Westinghouse, selected to design and develop the nuclear portion of the plant under contract with the Atomic Energy Commission.

The government-industry team grew rapidly and eventually included literally thousands of companies, large and small, all of which made material contributions to the eventual completion of the station.

But industry is made up of people, and within companies men of many different specialties teamed up to achieve the final result. Physicists, chemists, mathematicians, metallurgists, electrical engineers, mechanical engineers, manufacturing engineers, civil engineers, and specialists in almost every other field of science and engineering worked on closely-knit teams. The tireless, unselfish, and dedicated work of these people is impossible to put into adequate words. Therefore, we hope that as you read this special issue of the *Westinghouse* ENGINEER you will try to visualize the magnitude of the problems that arose in the design, development, and construction of this pioneering plant—but more importantly, the people who solved them.

Because this project was a team effort, and because hundreds of people participated, we have purposely not singled out any individuals for mention—far too many deserve mention in our limited space.

Westinghouse is proud of the part we played in the Shippingport Atomic Power Station. The major credit belongs to the people where the project was centered, at the Bettis Atomic Power Division, a facility operated by the Company for the Atomic Energy Commission. Many other divisions of the Company contributed know-how, counsel, and components. The technical competency of all these people, their ability to deal with tough problems, and their plain hard work did much to make Shippingport Atomic Power Station a reality.

Gwilym A. Price

GWILYM A. PRICE
Chairman of the Board

COVER DESIGN: The key component in an atomic power station is the nuclear reactor itself. Thus artist Dick Marsh chose to highlight this element—surrounded by a pattern of fuel elements—on the cover of this special issue. A canned motor-pump, a turbine-generator unit, and a power line round out the story of power from the atom.

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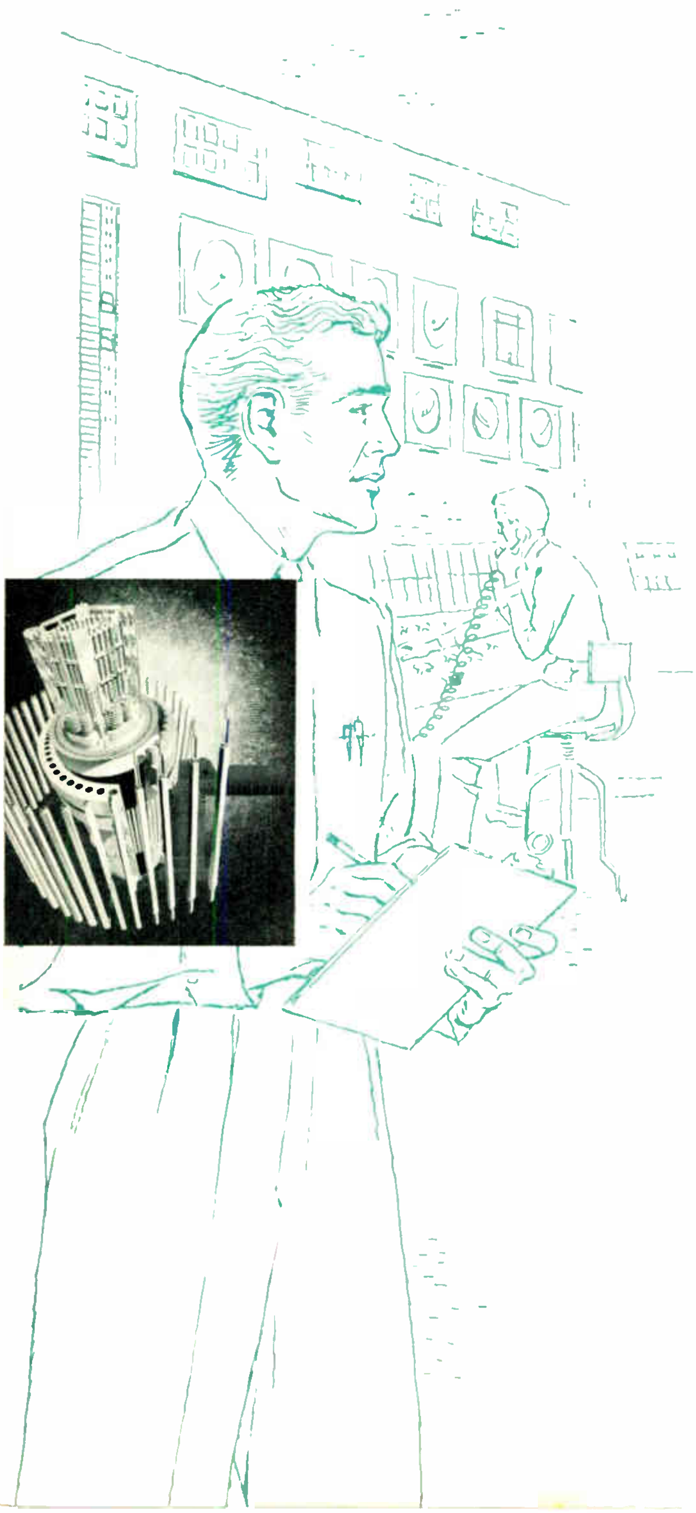
. . . a pioneering project in atomic power



A special report by the staff of the

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BASIC CONCEPTS OF THE PWR PLANT

At this stage of nuclear-power development, power plants of many sizes, shapes, and descriptions have been proposed. Several nuclear electric-utility stations are in advanced stages of design, and a few are under construction. But only one large-scale nuclear generating plant has been completed in this country—the Shippingport Station near Pittsburgh.

This new station is important in many respects. For one, it represents a significant milestone in American progress in nuclear power. But equally important is the experience and technology gained in its development, plus more to be acquired in its operation. The importance of this know-how should not be underestimated. Today, nearly every nuclear reactor is a pioneering venture, but from these plants will come the information needed to develop better ones. While a nuclear system can readily be designed on paper, only in developing and building one can the real problems be discovered and overcome. Thus, the Shippingport Station represents a significant step toward better nuclear plants.

The Shippingport Station has as its heat source a pressurized-water reactor plant. While experimental reactors of many different types have been built on a small scale, the technology of the pressurized-water reactor—the same type used on the submarine *Nautilus*—is by far the most advanced. This was one important reason for its choice for the Shippingport Station.

what is a pressurized-water system?

The source of heat is often said to be the fundamental difference between a nuclear power plant and a conventional utility station. While this comparison is sometimes useful, like most general statements it oversimplifies the situation. In addition to being a radically new heat source, the nuclear portion of the plant also significantly affects the overall plant design and its operation. Consider, then, the basic concept of a pressurized-water system.

As indicated in Fig. 1, a pressurized-water system consists of two main parts—a *primary system* containing the nuclear reactor in which heat is produced, and a *secondary system* to which that heat energy is transferred for use in a steam turbine. Water pumped through the primary system flows over nuclear fuel elements in the reactor, absorbs heat from them, then flows to the heat exchanger, or steam generator. Here it gives up that heat to water flowing in the secondary system.

The whole primary system is under high pressure to prevent the primary water from boiling. The secondary system, however, is a relatively low-pressure system, so that when secondary water absorbs heat in the steam generator it turns to steam, thereby providing energy in the necessary form to drive the turbine.

The reactor, of course, must contain enough fissionable fuel to form a critical mass, i.e., one capable of sustaining a nuclear chain reaction. This chain reaction can be started, stopped, or controlled by neutron-absorbing control rods, which are inserted in the reactor to lower the power level, or withdrawn to increase the power level.

A pressurized-water reactor plant has several significant features. One important fact is that the primary loop is physically isolated from the secondary system, i.e., water from the primary does not enter the secondary. Thus, if any radioactive particles or fission products find their way into the primary system water, they are confined there—they cannot contaminate the secondary system. This means that much of the secondary is available for maintenance during operation.

Another important factor about a pressurized-water system is that fundamentally the plant is designed to contain sources of radioactivity in the reactor itself. Fission products can escape to the primary system only if the cladding that surrounds the fuel becomes defective. Water is maintained at high purity, so that radioactive particles in the system are kept to a minimum. However, even though some fission products may get into the water, as well as some radioactive particles, the primary system—except for the reactor itself—is available for maintenance within a matter of a few minutes after the reactor is shut down.

The primary system in this type of reactor plant is pressurized for several reasons. The chief reason is to prevent boiling in the reactor. While nuclear plants can and have been designed to permit boiling in the reactor, they make use of a different fission process. The pressurized-water system is designed primarily to make use of relatively slow-moving thermal neutrons (see p. 37).

The water in a pressurized-water system serves a dual function as a *coolant* and as a neutron *moderator*. Neutrons given off by the fission process are traveling at high speed. Colli-

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sion with water molecules dissipates some of their energy and slows them to the so-called thermal level, where they are most efficient in causing fission in uranium 235. If the water in a pressurized-water reactor were allowed to boil, the resultant steam would not be as good a moderator and the fission process would slow down because fewer thermal neutrons would be available. Actually, in such a case the reactor would probably oscillate in power output, because of another factor, called negative temperature coefficient. In addition to affecting the nuclear reaction, boiling might also cause local hot spots, or conceivably some melting of the fuel elements, since steam is not as effective a coolant as water.

One of the most important facts about a pressurized-water reactor is that it can be designed to have a *negative temperature coefficient of reactivity.* As a result, the reactor inherently tends to maintain the power level at which it is set. If, for example, the temperature of the water entering the reactor drops for any reason, the reactor automatically produces more heat, and thus a higher outlet water temperature. If the inlet water increases in temperature, the heat output of the reactor automatically drops. Thus, in a properly designed pressurized-water system, the reactor itself automatically maintains the correct power level with no controls being involved. This level-seeking feature can be made to hold true for normal power changes in an electric system, so that control-rod movement is necessary only for large shifts in power output of the plant.

the pwr power plant

While the elementary concept of any reactor is often simple, the actual design of a real plant is a far different matter. Much of the technology of both components and systems must be developed in parallel with the plant's manufacture. Many subsystems must be developed and tested, and each part of the whole plant must be designed to be compatible in every respect with the other parts of the system. The translation of the design into a completed plant is an equally big step. Plans don't have to operate or live up to their specifications—plants do. However, despite the fact that it is a pioneering venture, the PWR system, like any other generating station, is designed to have specific characteristics, and to fit into a specific electrical network.

Several specifications were established by the Atomic Energy Commission to guide Westinghouse in design:

The plant was to have a net electrical output of 60 000 kilowatts at 600 pounds steam pressure.

The reactor was to be cooled by ordinary water at 2000 psia pressure.

The first core was to last 3000 hours at full-power operation.

Refueling was to be accomplished with minimum shut-down time.

The reactor control system was to be as simple as possible.

Commercial equipment was to be used wherever practicable.

The cost of operating the plant was to be the minimum consistent with these requirements.

Like the simple concept, the PWR plant consists of a primary and a secondary system (see Fig. 2). The key element, of course, is the nuclear reactor itself. The reactor pressure vessel, which is some 33 feet high and about 9 feet inside diameter, contains the nuclear core. The core is an assembly of plates and rods arranged in the general shape of a cylinder 6 feet high and 6 feet in diameter. The plates in this core are enriched uranium clad with an alloy of zirconium as protection from the hot water; the rods are hollow zirconium alloy tubes filled with natural uranium oxide (UO_2) pellets. The PWR core is of the *seed-and-blanket* type, the enriched uranium elements constituting the seed, and the natural-uranium elements the blanket.

Another important part of the reactor is the vessel head. This head contains the control-rod drive mechanism. The control rods themselves, of which there are 32, are made of hafnium, and can be inserted or withdrawn from the reactor core to change the power level, or to start or stop the nuclear reaction. The reactor head also contains considerable instrumentation, as well as refueling ports.

Uranium in both the seed and the blanket fissions, which liberates heat. Water enters the bottom of the reactor at 508 degrees F, flows up through the core over the rods and plates, absorbing heat. The heated water, now at 538 degrees, flows out piping near the top of the reactor and is pumped directly to the steam generator, where it gives up its heat to water in

Fig. 1—Simplified diagram of a pressurized-water system.

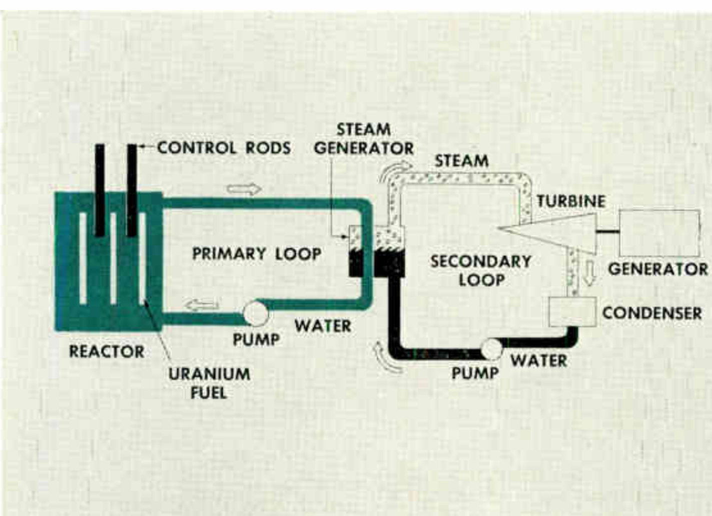
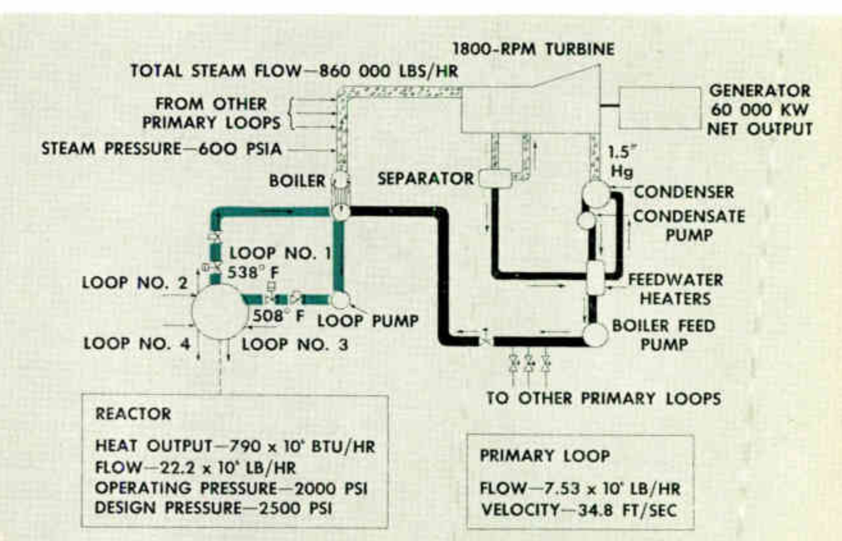
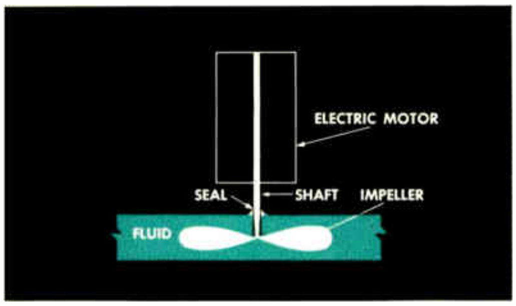
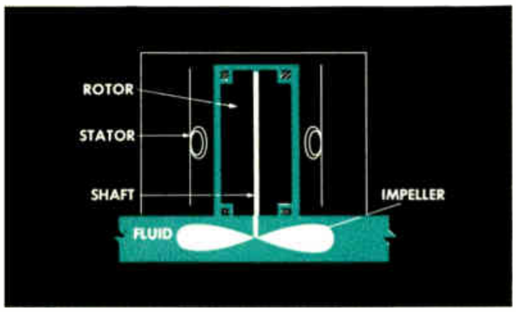


Fig. 2—The PWR primary and secondary systems.

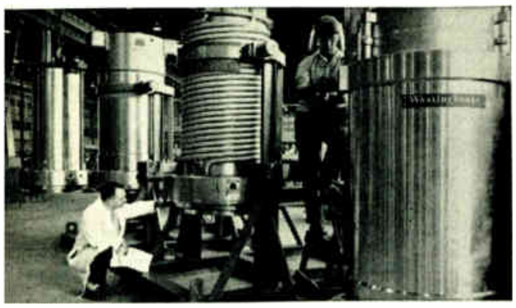




3a Simplified diagram of a conventional pump.



3b Similar diagram of canned motor pump.



3c Photo of PWR canned motor pump.

the secondary system. From the steam generator it flows back through the pump to the inlet at the bottom of the reactor.

The primary system also has a number of subsystems. These include a pressurizer, whose function is to maintain the system at 2000-pounds pressure, systems to sample and continuously purify the water, and many others. These subsystems are discussed in greater detail in another article.

The pump used in the primary system gives a clue to some of the difficulties involved in developing a nuclear plant. While pumps of the size and capacity required in the primary system are not unusual, one factor ruled out the use of conventional pumps. Conventional pumps have an impeller in the liquid to be pumped, connected by a shaft to an electric motor outside the fluid. A seal around the shaft prevents leakage of the fluid from the system (see Fig. 3a); even on the best of pumps, some fluid ultimately leaks around this seal. In a nuclear system, the fact that the fluid being pumped might contain some radioactive particles demands zero leakage; thus a different solution had to be used for the PWR system. This involves a canned motor-pump; in this pump, the impeller, the shaft, and the rotor of the electric driving motor are all in the fluid being pumped, as indicated in Fig. 3b.

Two different types of steam generators are used in the PWR plant, to obtain more experience with this component. One is a straight-through type. Water from the reactor enters the upstream end and flows through hundreds of small tubes. Water from the secondary flows around these tubes and is converted to wet steam and rises to the steam drum, where excess moisture is removed. This steam then flows to the turbine. In the other steam generator, primary water flows in a U path through the device, rather than straight through.

Steam in the secondary system flows to the turbine, then to a condenser, through a pump, and back to the steam generator. The turbine generator has a maximum capability of 100 000 kw, and is a single-cylinder, 1800-rpm unit. The gross turbine heat rate at 100 000 kw is 11 835 Btu per kilowatt-hour. Except for the fact that it operates at somewhat lower pressure than most new turbines, and is therefore somewhat larger, the steam turbine is essentially of conventional design.

Although only one primary system has been described, actually four essentially identical loops—each containing its own pump and steam generator—draw heat from the reactor and provide steam for the turbine.

selection of plant parameters

Selecting the operating conditions for the PWR nuclear plant (see Table 1) was a tough proposition. Cost, development problems, and a number of other considerations entered the picture. The end result is a compromise of many conflicting factors. Although too detailed to present here, several examples may suggest the nature of the problems.

Higher temperatures in the primary system mean higher thermal efficiency—but they also require higher pressure to prevent boiling in the primary system. In general, the cost and size of the reactor plant components increase with pressure, so the design pressure of 2500 psia (for an operating pressure of 2000 pounds) represents a compromise between the two.

Coolant temperature and flow rate are determined by selecting a proper balance between such factors as pumping power costs, steam generator costs, and reactor core costs, as influenced by core surface and allowable heat flux. For a given reactor inlet temperature and reactor heat load, core costs decrease and pumping power costs increase with increasing coolant flow rates. Core costs increase and steam generator costs decrease with higher average coolant temperature. Steam generator surface area is considerably less expensive than reactor core surface, so as much steam generator surface should be used as is feasible. Pressure drops in the reactor coolant system should be kept to a minimum to lessen pumping power requirements; however, reactor core surface area cost usually warrants a considerable cost for coolant pumping power.

Saturated steam pressure in the steam generator should be as high as is consistent with the other parameters. Within the pressure range of the PWR plant, higher steam pressures and temperatures usually mean higher thermal efficiencies and lower steam plant costs per kilowatt. Use of relatively low-

TABLE 1—PWR PARAMETERS

Reactor Power	790 x 10^6 Btu/hr
Gross Electrical Output	68 mw
Net Electrical Output	60 mw
Steam Pressure at Full Load	600 psia
Normal System Operating Pressure	2000 psia
System Design Pressure	2500 psig
Generated Electric Power Range (with reactor in automatic control)	10 mw to full power
Generated Electric Power Range (automatic control of the throttle) (by system load dispatching)	20 mw to full power
Number of Loops	4
Average Reactor Inlet Temperature	508 degrees F
Average Reactor Outlet Temperature	538 degrees F
Weight of Natural Uranium Fuel	14.16 tons
Weight of Enriched Uranium Fuel	75 kg (about 165 pounds)

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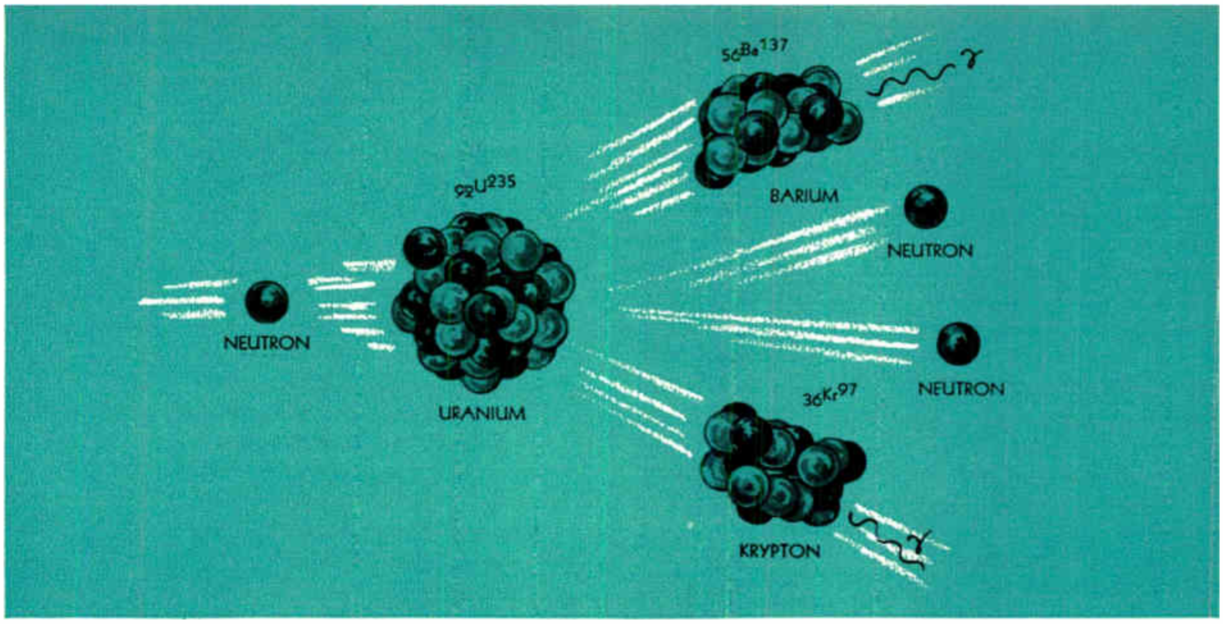


SOME ELEMENTS OF PHYSICS FOR THE PRESSURIZED-WATER REACTOR

When a neutron of the proper energy, or speed, strikes a fissionable nucleus in just the right fashion, the nucleus splits into fragments and releases a proportionately large amount of energy. Such is the source of heat that, multiplied many times, produces sufficient heat energy to drive a steam turbine.

Neutrons are often classed according to their energy or speed. All neutrons released by the fission process are originally *fast* neutrons. Some fast neutrons collide with other atoms in such a way that they neither cause fission nor are captured by the atom. In the collision, however, they lose some of their energy and become *intermediate* speed neutrons. Still other neutrons collide with light elements, thus giving up a larger proportion of their energy, and become slow or *thermal* neutrons.

Obviously, then, not all neutrons that collide with fissionable atoms cause fission. Some are absorbed, others merely collide and lose energy, others cause fission. All three of these processes are of interest in the PWR system.

As far as fissionable material is concerned, two isotopes of uranium—uranium 235 and uranium 238—and plutonium are of primary use in this reactor. Initially, the reactor is designed to operate primarily on the fissioning of uranium 235. Under the right conditions, neutrons of all three speeds can cause fissioning in uranium 235, but thermal neutrons are most efficient.

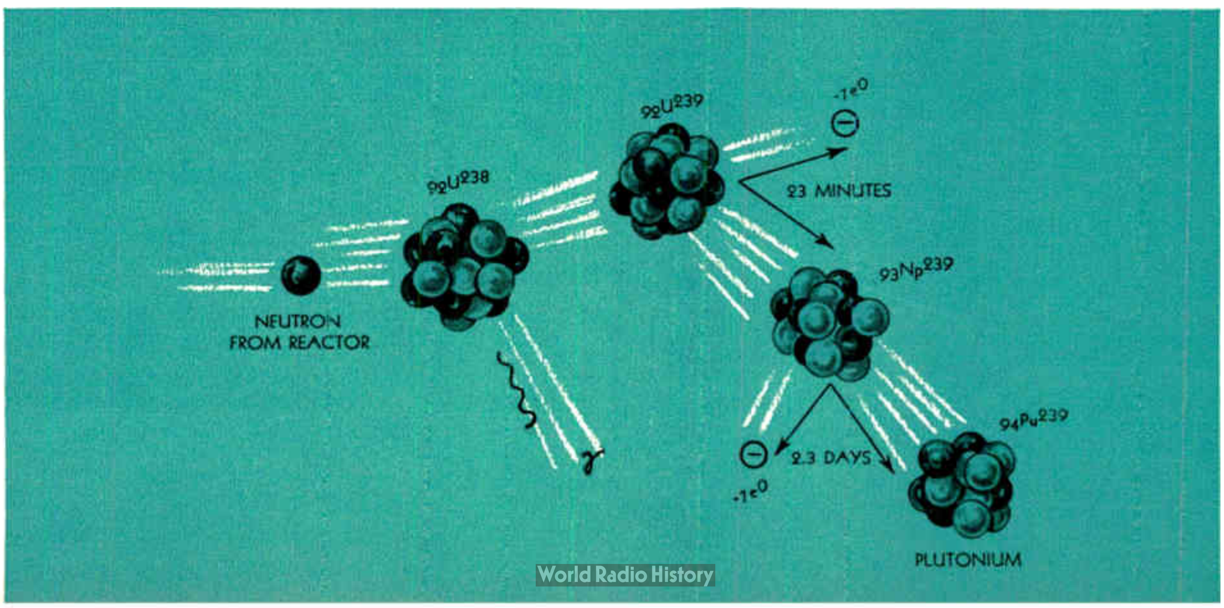
To provide thermal neutrons, fast neutrons must be deliberately slowed to thermal speed in some fashion. This is accomplished by a moderator material, which in the case of the PWR is ordinary water; thus fast neutrons can collide with water atoms and be slowed to thermal speed.

By the laws of probability, not all neutrons collide with water atoms, so in the reactor at any given time is a percentage of both fast and intermediate-speed neutrons. Fast neutrons are effective in causing fission in both uranium 235 and 238, and both these processes occur to some degree in the PWR. Also, intermediate-speed neutrons cause fission in uranium 235, and this process also occurs to some degree in this reactor. Thus fissioning occurs in different ways, but the main one is the fissioning of uranium 235 by thermal neutrons.

Another nuclear reaction is also of key importance in the PWR reactor. Neutrons of particularly low energy are susceptible to capture by uranium 238, thus forming uranium 239. This isotope is extremely unstable, however, and decays to neptunium 239. This element is only slightly more stable, and decays to plutonium 239. Fortunately, plutonium 239 is fissionable byth ermal neutrons.

A nuclear reactor can be designed to produce more than one atom of plutonium for every atom of uranium fissioned. This happens in the so-called *breeder* reactor. The PWR reactor, however, is designed to produce a little less than one plutonium atom for every uranium destroyed, and is thus a *converter*. rather than a breeder.

As mentioned in more detail on page 40, the PWR has fuel elements of two forms. One—called a *seed*—is highly enriched, i.e., it contains a higher proportion of uranium 235 than natural uranium. The other part of the fuel—called the *blanket*—is natural uranium (UO_2). The major part of the plutonium formation occurs in the blanket elements, where uranium 238 is more plentiful. Thus, while the uranium 235 in the seed and blanket elements is being expended, plutonium is being formed in the blanket. The longer the reaction goes on, the more the reactor will depend on plutonium fissioning in the blanket. Actually, as mentioned later, the seed elements will have to be replaced at least once to get full use from the blanket elements.



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pressure saturated steam requires larger steam flow rates and results in a larger turbine for the same kilowatt output than normally found in modern central stations.

the physical arrangement of the pwr plant

The general physical arrangement of the plant is shown below. Note that the entire primary system is underground.

Since this is the first full-scale nuclear power plant, extreme care has been taken to foresee any eventuality, and to build protection against any hazards into the plant. For this reason, the primary system is housed in four containers.

One of these is a 38-foot sphere and contains the reactor itself. The other three containers are roughly cylindrical in shape, with hemispherical ends. One contains two primary loops; the second contains the other two primary loops; and the third contains the pressurizer and auxiliary equipment.

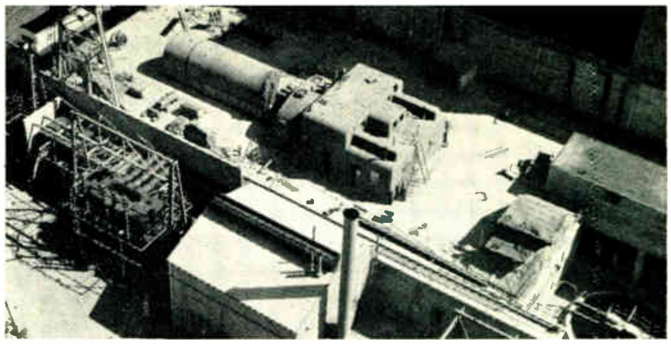
These containers, made of one-inch steel plates, are expensive. Some day they may be eliminated, but for the present they are used as an extra safety feature, to guard against the remote possibility of a hazard arising. In the event of leaks of radioactive material, even radioactive steam, all contamination would be confined to the containers.

Surrounding these containers is a five-foot-thick concrete wall, which serves as a radiological shield as well as a building to contain the plant.

The turbine generator itself is located outdoors on a deck above the three-story turbine-generator building, which houses turbine-plant equipment, water-treating equipment, and similar apparatus for the secondary system.

the pwr plant and the electric utility system

The Duquesne Light Company serves a metropolitan area, which includes the city of Pittsburgh. Before the Shippingport Station was completed, the system capacity was 1 207 000 kw; the system normally operates as part of a large interconnection with 32 000 000 kilowatts of connected capacity.



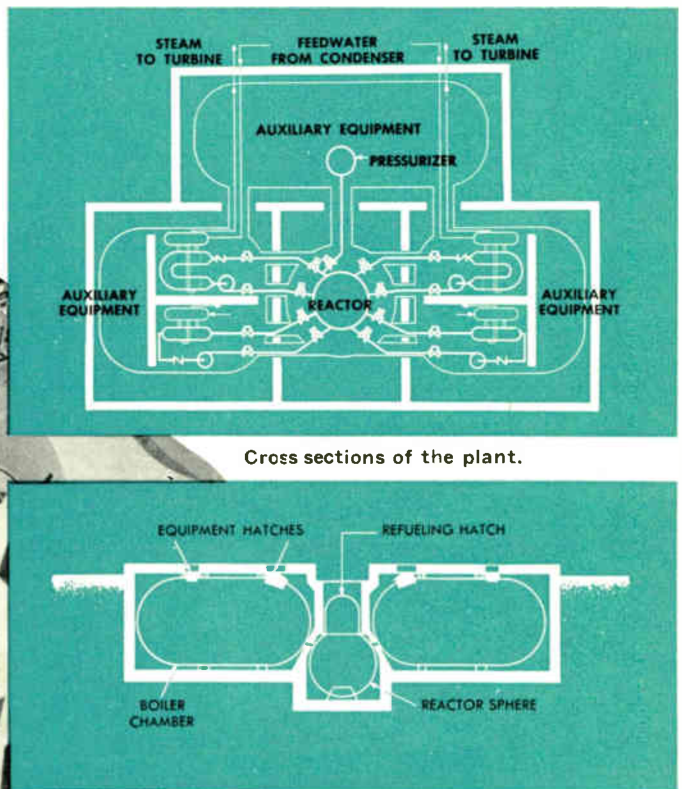
The turbine generator at Shippingport.

Maximum load in the system usually occurs during the day, because of the large industrial load. This industrial load, which accounts for about 55 percent of the system output, includes several continuous steel strip mills and many large electric furnaces. As a result of this type of load, the system load characteristics are erratic and variable, and the output must be changed through a range of 60 000 to 80 000 kw at frequent intervals and at a rapid rate.

The load-changing ability of the PWR station will be equal to or better than that of the average Duquesne Light station using high-pressure steam and conventional coal-fired boilers. Because the reactor will be essentially self-regulating, it has more stability during load changes than the conventional coal-fired equipment. Based on the ultimate capability of 100 mw, the PWR plant must take a ramp change of plus or minus 15 mw at 3 mw per second, or a ramp change of plus or minus 20 kw at 0.4 mw per second; or a step change of plus 15 mw or minus 12 mw. The PWR plant is thus designed to take its place as an integral part of the Duquesne Light Company power system.

This, then, is a simplified picture of the Shippingport plant. However, it says little of the many considerations involved in the design, development, and operation of the various parts of the system, nor the design philosophies involved. The articles that follow delve into systems and components in more detail. ■

A cutaway model of the plant container.



Cross sections of the plant.

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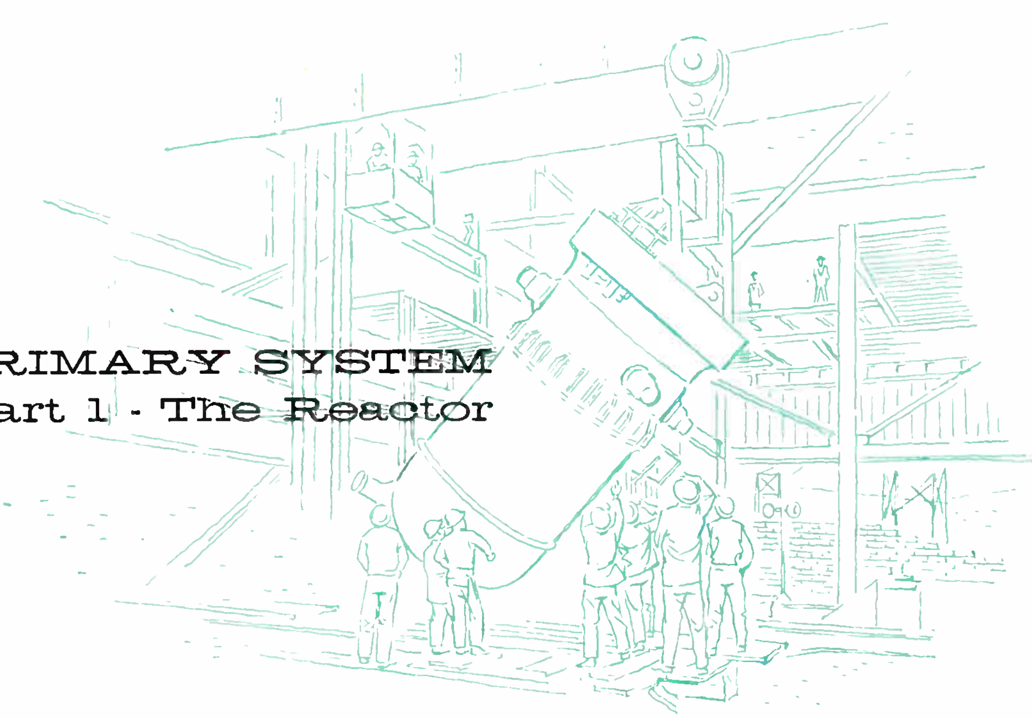

THE PWR PRIMARY SYSTEM
Part 1 - The Reactor

The development of a nuclear reactor has something in common with the development of a vehicle to explore outer space. In both cases the stockpile of available knowledge is inadequate; most problems require pioneering in many different technical fields. And in both cases the solution is not merely to build a device that will work, but to build one that will perform according to exacting standards.

Any nuclear-reactor system represents a deliberate compromise between the many phases of technology involved. The best design from a physics standpoint is not necessarily the best or most practical one from the standpoint of the metallurgist or the mechanical engineer. And, of course, economics must also be factored into the design at every step. For example, in choosing the best operating water temperature for the PWR system, two factors to be considered are core and steam-generator costs. In the range in question, steam-generator costs go down as the temperature increases; however, core costs go up with temperature. The operating temperature of 525 degrees F was ultimately chosen to give the estimated minimum combined cost for the two units.

This example also illustrates the complexities involved in pioneering a new field. At the time the operating temperature was picked, only rough approximations could be made of either core or steam-generator cost. In fact, fixed-price bids for the steam generators covered a range of about five to one.

components of the nuclear reactor

Before looking at the design considerations involved in the reactor, consider first the elements themselves and their physical arrangement. For the purpose of this discussion, the reactor consists of three main elements—the pressure vessel, the nuclear-fuel configuration, or core, and the control-rod assembly. The reactor also includes instrumentation, but this is discussed in a subsequent article.

Pressure Vessel—The reactor pressure vessel, shown in Fig. 1, is a cylindrical structure, formed of carbon-steel plates and forgings. The inner surfaces of the vessel are clad with stainless steel about a quarter-inch thick to provide corrosion resistance. Internally, the pressure vessel is approximately 31 feet high and 9 feet in diameter. The walls of the vessel are a little over eight inches thick. Thermal shields prevent excess heat generation in the vessel wall from gamma-ray radiation.

Top and bottom sections of the vessel are hemispherical. The top section, or closure head, is bolted onto the main part of the vessel; it has 46 penetrations for control rods, instrumentation, and refueling purposes.

Control-Rod Assembly—The PWR reactor is controlled by moving neutron-absorbing control rods in and out of the fuel core. The rods themselves are cruciform shaped, to fit into similarly shaped openings in the fuel assemblies, and are about $3\frac{3}{8}$ inches wide. The lower section of the rod is made of the little-known metal hafnium, the upper portion of a zirconium alloy. The hafnium portion of the rod, which is the neutron-absorbing end, is about 70 inches long, to correspond to the length of the fuel in the fuel assemblies. When fully inserted into the core, the lowest point of the rod coincides with the lowest point of the fuel alloy.

The movement of the control rods must be positive and accurate. In addition, provision must be made to drop the rods into the core rapidly in a "scram," or emergency condition. Each control rod is moved by a canned-rotor electric motor, which operates a collapsible roller-nut mechanism in contact with a threaded portion of the rod. The mechanisms and motor rotors are located in pressure tubes attached to the pressure-vessel head-mechanism housing. The motor stator and position-indicator coils are installed outside the tubes.

The reluctance motor is driven by a low-frequency current supplied by an inverter. When the inverter is not being rotated, a direct current flows in the motor and holds the control rod in a fixed position.

The roller nuts, normally in contact with the threaded shaft, are pulled away from the shaft by springs if current to the motor is interrupted. This constitutes a scram action, since the nuts disengage from the screw, permitting the shaft and the control rod to drop by gravity into the core.

The Nuclear Core—The fuel assembly, or core, for the PWR plant uses both highly enriched uranium 235 and natural uranium oxide (UO_2). The core is so designed that the assembly of natural-uranium elements is made chain reacting by the help of neutrons leaking from the highly enriched elements.

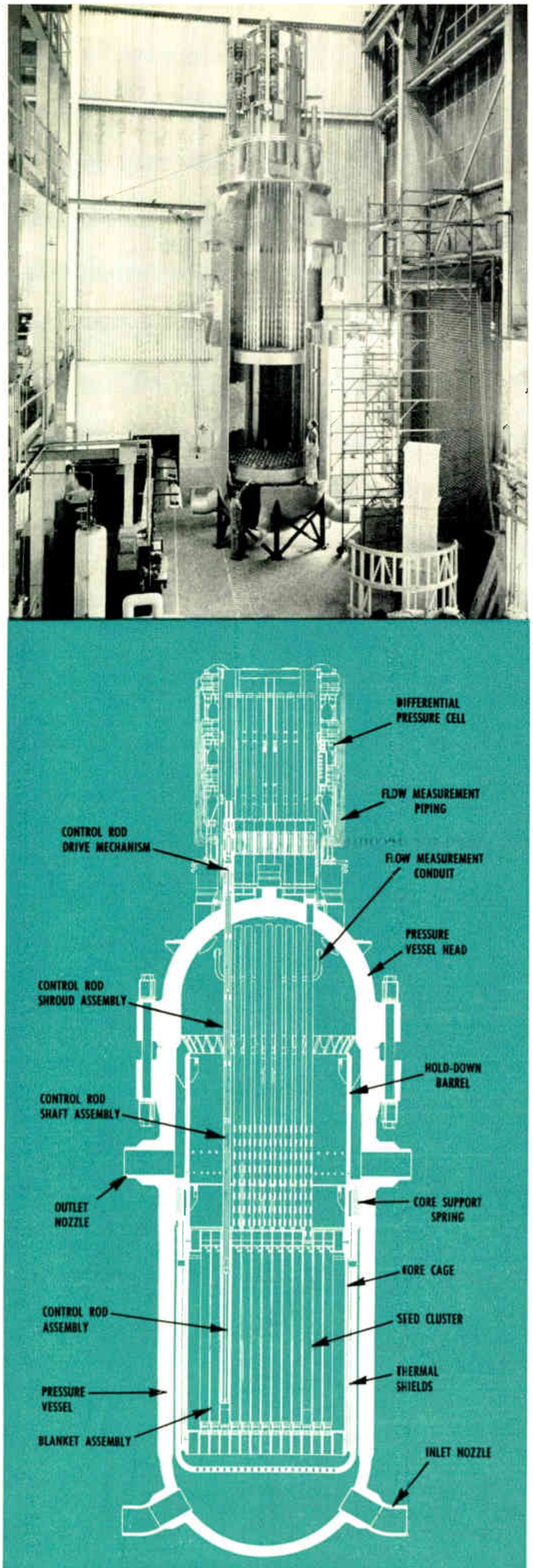

Fig. 1—At top, a full-scale model of the pressure vessel. Below, a cross section of the reactor.

The highly enriched elements are called the *seed*, and the natural-uranium elements, the *blanket*. Initially the seed will provide most of the reactivity, with the blanket acting as a neutron-multiplying reflector.

The entire core is about 70 inches long, and contains 145 fuel assemblies; 32 of these are seed "clusters," the remaining 113 are blanket assemblies. Each fuel assembly has a cross section 5½ inches square, giving a mean core diameter of 6.8 feet. Dimensionally, all seed clusters are designed to be interchangeable with blanket assemblies, and either unit can be removed from the reactor without removing the head.

The basic component of the seed cluster is a fuel plate; this consists of an enriched-uranium fuel plate sandwiched between two zirconium-alloy cover plates and four flange strips. These fuel plates are then built into subassemblies, by stacking 15 fuel plates and two end plates. To make the final cluster, four of these subassemblies are joined, with necessary spacers, to create a cruciform channel between subassemblies for the control rod, as shown in Fig. 2.

The blanket assemblies are of different construction, although their cross-sectional dimensions are identical. The basic element of the blanket assembly is the fuel rod, shown in Fig. 3. These are zirconium-alloy tubes, each filled with 26 uranium-oxide pellets. Each tube is a little under a half inch in outside diameter and 9½ inches long. These rods are assembled into fuel bundles, each containing 120 fuel rods, with a tube sheet on either end with spaced holes to allow coolant flow through the bundle. To make one complete blanket assembly, seven fuel bundles are installed end to end in a shell, giving a nominal fuel length of nearly six feet.

The seed clusters and blanket assemblies are arranged in the core in a definite pattern, for both nuclear and heat distribution reasons. The blanket is divided into four regions, according to the flow requirements of the individual assemblies in each region. The general pattern is shown in Fig. 4.

The four regions are designated according to their position with respect to the seed clusters. The coolant flow is distributed among the seed clusters and blanket assemblies to compensate for heat flux variation from region to region.

design considerations

The Core—Perhaps the most interesting aspect of the nuclear reactor from a physics standpoint is the seed-and-blanket concept of the nuclear fuel. Much basic technology will be gained from this reactor concerning the type of fuel that will go into future nuclear plants.

The seed-and-blanket concept has several important features. In the highly enriched seed, the concentration of uranium 235 not only makes the chain reaction possible, but also makes the seed a copious producer of neutrons.

The natural-uranium blanket elements, on the other hand, contain only one part in 140 of uranium 235, the remainder being uranium 238. This uranium 238 is not fissionable by thermal neutrons; however, it does absorb them without fissioning. As a matter of fact, the uranium 238 absorbs enough neutrons that a chain reaction cannot be maintained in natural uranium. Thus, the seed elements must provide enough neutrons to the blanket elements to keep the reaction going. Were these the only considerations, the seed-and-blanket concept probably would not be practical. However, another nuclear reaction enters the picture. When the uranium 238 absorbs thermal neutrons, it becomes uranium 239, which is unstable and decays to neptunium and ultimately to plutonium 239. Plutonium 239, like uranium 235, is a fuel suitable for use with thermal neutrons. Thus, useful new fuel is being produced in

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the reactor concurrently with burn-up of existing nuclear fuel.

Nuclear reactors can, and have, been designed specifically to produce more than one new fissionable nucleus for each one destroyed. These are the so-called *breeder* reactors. However, in general, for a given core volume, the more effective the reactor is as a breeder the less effective it is as a power producer. On the other side of the ledger, however, the breeding process gives longer life to the core. The PWR core, therefore, represents a compromise between the two extremes. This reactor is a *converter*, which means that some new fuel is formed during its life, but not enough to classify it as a breeder reactor. Actually, in the PWR core, about six plutonium atoms are formed for every ten uranium 235 atoms destroyed, i.e., the conversion ratio is about 0.6.

This ratio is for the core as a whole. In the seed, of course, a relatively small amount of plutonium is formed because of the relatively small proportion of uranium 238 present. However, in the blanket, about 1.1 atoms of plutonium are formed for every atom of blanket uranium fissioned.

As might be expected, this breeding effect in the blanket gives it a longer life than the seed. Actually, to make full use of the blanket, the seed portion of the core will have to be replaced at least once.

At this stage of development, the exact lifetime of any reactor core is impossible to predict. However, the blanket portion of the PWR core is designed to have a minimum equivalent full-power life of 8000 hours, and the seed is designed for a minimum of 3000 hours.

At the start of reactor life, about half the power will be generated in the seed elements and about half in the blanket. As the seed "burns up" its uranium 235, however, its effectiveness as a heat producer diminishes. In the blanket, however, the effect is just the opposite. Since the blanket is producing new fuel faster than it is burning the original fuel it contained, the percentage of power produced in the blanket should increase with time.

However, power distribution between the seed and the blanket is also a function of the reactor temperature and the position of the control rods, so the ultimate distribution of power may well depend upon the use of the reactor.

As far as criticality of the reactor is concerned, in the early life of the reactor this is determined largely by the thickness of the spacing between seed clusters, which determines the neutron leakage from the seed, and by the fuel loading of the seed, which determines the multiplication constant. Initially, the blanket serves primarily as a good reflector of neutrons, i.e., it returns one for every one that enters it.

When the amount of uranium 235 or plutonium left in the reactor decreases beyond a certain point, reactivity ceases. At this point some nuclear fuel is left, but not enough to maintain criticality. This is not wasted fuel, however, since it can be removed by reprocessing.

Another factor in core life is the build-up of nuclear "poisons" in the reactor. Certain products of nuclear fission, notably xenon, are neutron absorbers. During the early life of the reactor they are present in relatively small amounts, and therefore of little consequence. However, as they build up during the operation of the reactor, they ultimately reach the point where they stifle the nuclear reaction, i.e., absorb a sufficient number of neutrons to stop the chain reaction. When this occurs, the reactor must be refueled, regardless of the fact that fissionable fuel remains.

Fortunately, xenon poisoning tends to reach an equilibrium point. The particular isotope of xenon (xenon 135) is formed by a decay process from a direct-fission product, tellurium 135.

Fig. 2—A cross section of a seed cluster.

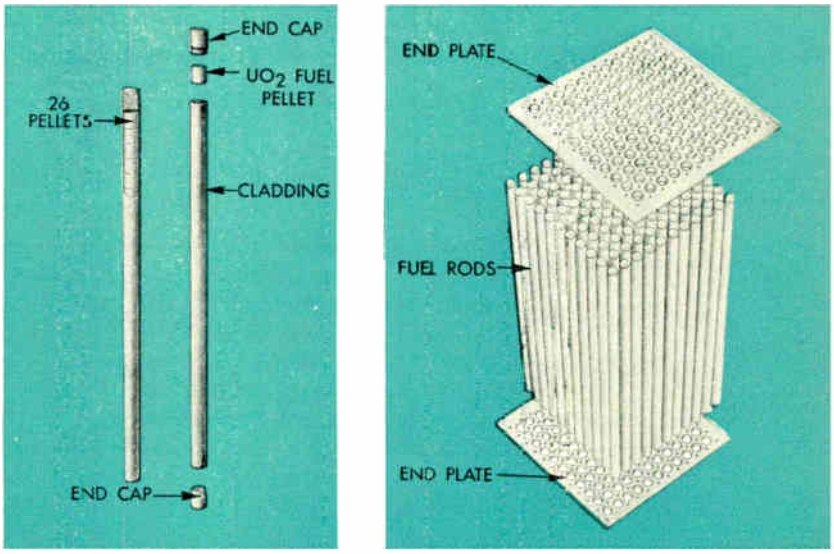

Fig. 3—At left, a blanket fuel element; at right, a blanket assembly.

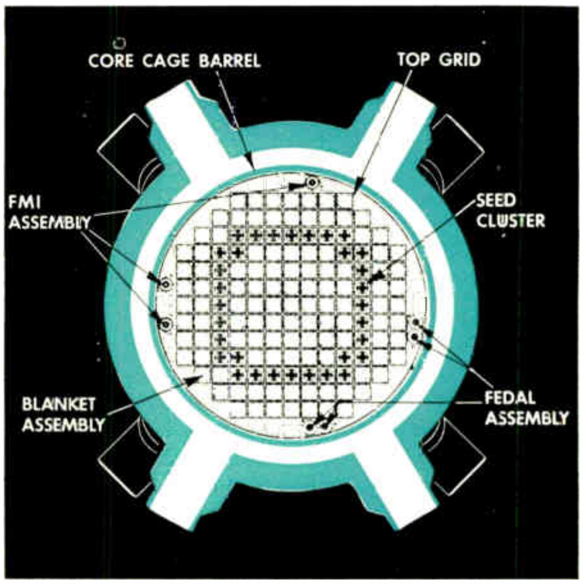

Fig. 4—A cross section of the reactor, showing arrangement of core.

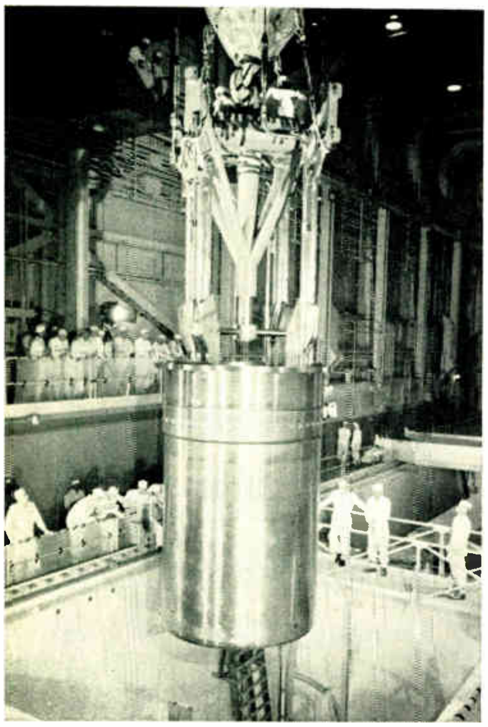

Fig. 5—The 58-ton core being lowered into the reactor.

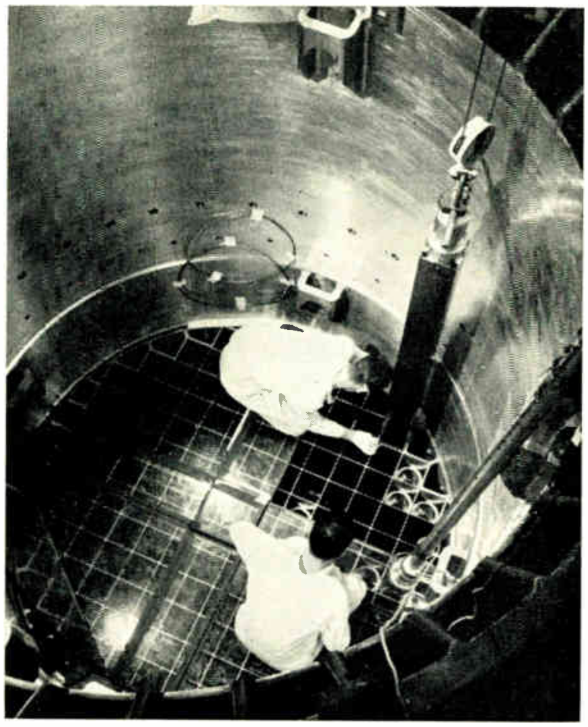

Fig. 6 A fuel element being placed in the reactor core.

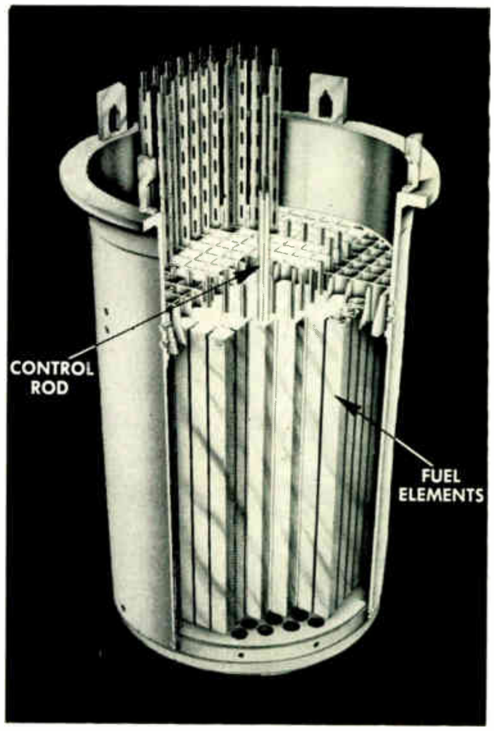

Fig. 7 A cutaway sketch of the reactor core.

However, xenon 135 is not stable and ultimately decays to cesium. Thus, during the lifetime of the reactor, at some point xenon is decaying at about the same rate it is being formed and the concentration reaches equilibrium. By adjustment of control rods, the effect of xenon can be overridden. Other fission products that absorb neutrons, however, ultimately build up to the point where the chain reaction is choked off.

Control—Control of the reactor is provided by hafnium control rods. However, the negative temperature coefficient (see facing page) designed into the reactor simplifies the whole control problem.

Other than under emergency conditions, the chief function of the control rods is to establish and maintain the power level of the reactor. At the start of reactor life, criticality with full power will probably be obtained with 12 control rods fully inserted. When equilibrium xenon has been built up, four of these rods will have to be withdrawn; as fuel burn-up and poisoning continue, rods will be withdrawn in groups of four.

At any given stage of reactor life, control rods will have a specific operating position. At this operating position, whatever output that has been established for the reactor will be maintained. The chain reaction will be just sustained, with the right number of neutrons being produced for the heat level desired. If higher output is desired, the control rods will be moved out of the core until the new heat rate is attained, and then moved back to the operating position. A larger number of neutrons is then put in circulation, more fissions occur, and more heat is liberated. To lower the heat output, the rods are inserted further into the core until sufficient neutrons are taken out of circulation, i.e., until the heat production drops, and then moved back to the operating point.

However, control rods will not be needed for most power-level changes once the plant is in normal operation. This is where the *negative temperature coefficient* enters the picture. For example, suppose the demand on the electric generator increases. This, in turn, demands more power from the turbine, and thus greater steam flow. The turbine adjusts to this condition by admitting more steam. This lowers the steam pressure and the temperature in the secondary system. As a result, more heat is absorbed from the primary, in the heat exchanger, and the temperature of the primary coolant leaving the steam generator decreases below its previous value. When this lower-temperature coolant reaches the reactor, it causes the reactor to automatically step up its level of heat production, thus giving a higher temperature to coolant leaving the reactor. By proper reactor design, the amount of heat added to the coolant is made sufficient to provide for the higher electrical demand.

Actually, the plant is designed to have a constant average coolant temperature. Therefore, if the temperature of the water at the reactor inlet goes down for any reason, the reactor provides enough heat to raise the coolant temperature at the reactor outlet in direct proportion. In operation, the negative temperature coefficient is expected to take care of load changes on the order of 10 percent of full power rating without control-rod movement.

Materials—Among the most important considerations in design of a reactor is the materials problem. Materials exposed to the primary coolant must have unusual properties. Not only must they be resistant to the highly corrosive action of hot water, but also they must have good nuclear properties.

The basic requirements for structural materials used in the reactor are that they do not interfere with the fission process, that they be unaffected by radiation in the lifetime of the component in which they are used, and that they be corrosion resistant in hot water. Unfortunately, few existing materials satisfy these requirements.

To date, zirconium has proved the best material for a core structural material; it has good corrosion resistance in water at 600 to 700 degrees F, and low interference with the fission process, i.e., it is a poor neutron absorber. An alloy of zir-

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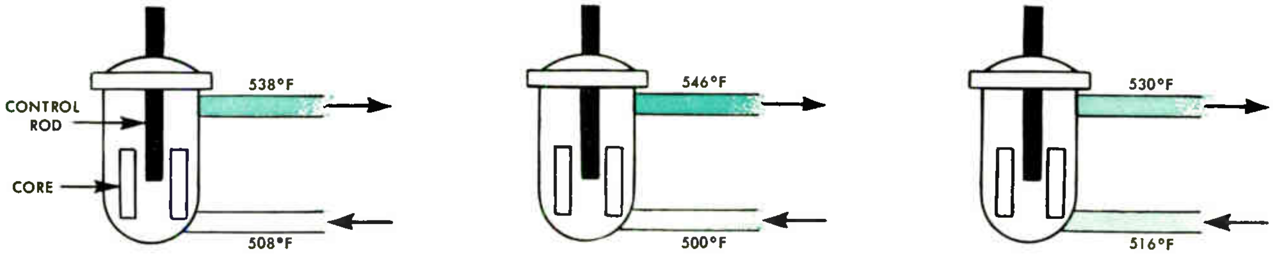
EFFECT OF NEGATIVE TEMPERATURE COEFFICIENT



Under normal operating conditions, with control rod in operating position, temperature of water at inlet is 508 degrees F, and at outlet is 538 degrees.

If temperature of inlet water drops, i.e., if load on the generator increases, the reactor adjusts itself to produce more heat, thus raising outlet temperature of water. Note that *average* coolant temperature remains same. No rod motion is required for normal load changes.

If temperature of inlet water increases, i.e., if load on the generator decreases, reactor adjusts itself to produce less heat, thus lowering temperature of outlet water. Again, note that *average* coolant temperature remains constant and no rod motion is required.

conium, called Zircaloy 2, is used extensively in the PWR.

Corrosion resistance is extremely important, if high radioactivity levels are to be confined to the reactor. Especially important is protection of the fuel itself from the circulating hot water, since any fission products in the coolant would contaminate the entire primary system.

The blanket fuel offers other materials problems, and illustrates some of the conflicting factors that must be resolved. From a nuclear standpoint, pure uranium would be more effective as a blanket fuel, since no oxides or alloying elements would be present to interfere with the fission process. However, pure uranium distorts under continued irradiation, and would ultimately block the cooling channels. Also, of course, pure uranium is more expensive than natural uranium because of the additional refining necessary. The blanket fuel itself should also have good corrosion resistance, since the possibility of a flaw in the zirconium rod always exists. For these reasons, uranium oxide was chosen for the blanket fuel. Ultimately a uranium alloy may provide the best answer, but this has not yet been achieved.

The hafnium control rods of the reactor are another example of materials problems. A control-rod material must first of all be a good neutron absorber. But also, it must have good corrosion resistance, and maintain constant dimensions under operating conditions, including high irradiation. Hafnium is a good control-rod material, but, like zirconium, is expensive. As a matter of fact, it is produced as part of the zirconium process, but not enough is obtained in this manner to satisfy industry demands and the cost is high.

The safety and mechanical perfection that must be designed into this first full-scale plant also introduce materials problems. Ordinary safety precautions are not enough. The different types of mechanical failure must be weighed carefully so that certain failures are not reasonably possible. In other cases, where failure is a possibility, no matter how remote, the worst possible sequence of events must be calculated to assure that the emergency can be handled safely. All this returns to the various materials, how they fail, and why they fail.

One other fact in connection with materials should be mentioned. The PWR plant was designed using the best information that could be obtained at the time the design had to be finalized. Some of the materials questions had been partially resolved by the experience with the *Nautilus;* other materials problems could be solved in time to meet PWR schedules. There is no doubt that better nuclear materials can be found, that many of the safeguards included on this first plant because of materials can be omitted; but this all requires extensive and time-consuming research and development. For example, one estimate states that if a material such as an aluminum alloy could be used in place of zirconium, 80 percent of the present cost of zirconium elements might be saved. However, so far no aluminum alloy has been found with resistance to water anywhere near comparable with zirconium.

At present, much of the reactor and associated components must be made of stainless steel, because it appears to be the best and most dependable material. In conventional plants, less expensive steel is used, because chemicals can be added to the water to prevent it from attacking the metal. In a nuclear reactor, these chemicals are damaged by radiation as they pass through the reactor. An active program is now being pursued to find out how to build such a plant of carbon steel. For the PWR plant, however, stainless steel was the only known answer; the cost penalty is at least 10 percent of the reactor-plant cost.

As mentioned, the nuclear reactor for the PWR plant is a compromise of many different technical and economic facts; it makes full use of all the knowledge of pressurized-water plants accumulated with previous reactors. Future reactors will be better, and more economic—but this will be possible because of the technology gained in this development. ■

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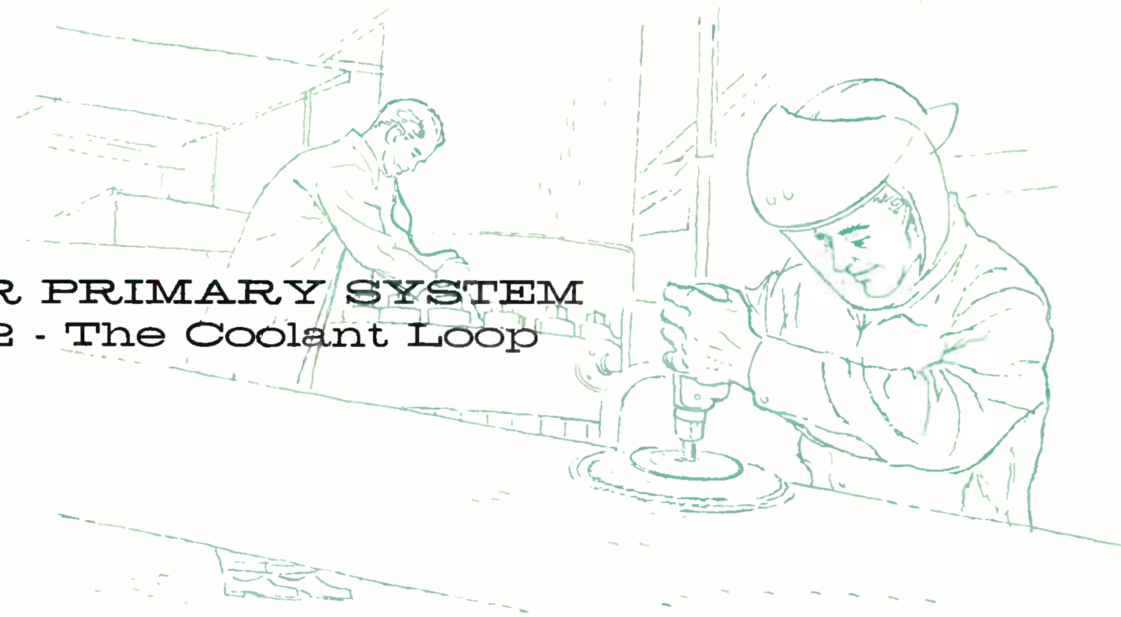

THE PWR PRIMARY SYSTEM
Part 2 - The Coolant Loop

Although the reactor is obviously the key component in a nuclear power plant, it is actually but the first step in the process of gaining useful power from the fission process. Components in the remaining portion of the primary loop must pump and purify the coolant, maintain it at constant pressure, and deliver it to the steam generator where its heat content can be transferred to the secondary loop.

The principal elements of the primary system, aside from the reactor itself, are the steam generator, a canned motor-pump, and valves.

However, the primary also includes a whole host of auxiliary systems, such as the pressurizing system, valve-operating systems, and coolant-purification system.

steam generators

Each of the four primary loops has its own steam generator. Two have a straight-through-type heat exchanger, and the other two use a U-type (Fig. 2).

Both kinds consist of three basic elements—a heat exchanger, in which primary coolant flows through tubes and gives up its heat to secondary water surrounding the tubes; the steam drum, which dries the steam for use in the turbine; and piping (i.e., risers and downcomers) to carry steam to the drum and return water. Each is rated to provide 263 x 10^6 Btu per hour at 600-psia steam pressure at the design power.

Straight-through type—The heat exchanger in the straight-through generator is entirely stainless steel. Coolant entering

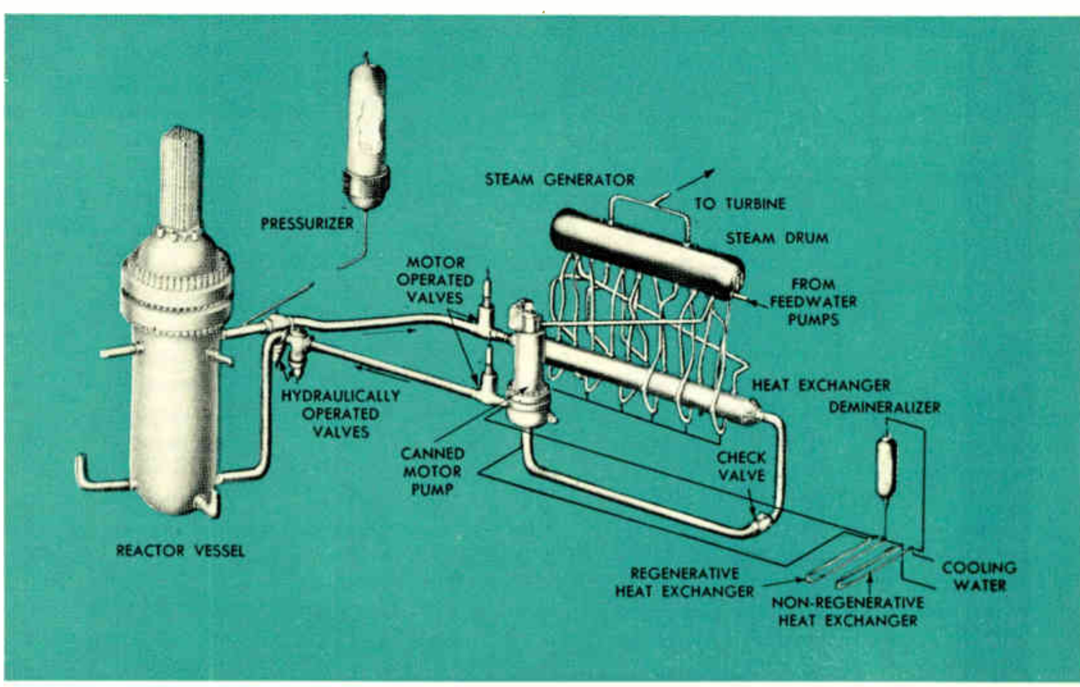

Fig. 1 Perspective sketch of reactor plant.

Fig. 2— Sketch of U-type steam generator.

Fig. 3 —Photo of a main coolant pump.

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the heat exchanger flows through 2096 stainless steel tubes, each of which is a half-inch in diameter and 36 feet long. These tubes are rolled and welded into tube sheets and enclosed in the exchanger shell, which is 43 inches in diameter. The steam drum is made of carbon steel, since only secondary water and steam pass through it. Twelve 8-inch risers and six 8-inch downcomers connect the heat exchanger to the steam drum.

U-type—Water flowing into the U-bend heat exchanger enters 921 three-quarter-inch stainless-steel tubes. These tubes average 50 feet long, and are bent in a U shape. The tubes are contained in a U-shaped shell 38 inches in diameter. This heat exchanger differs from the straight-through type in that carbon steel clad with stainless steel is used for the tube sheets, and for the hemispherical ends of the exchanger. The steam drum itself is of conventional design. Fourteen 4-inch downcomers and eighteen 5-inch risers connect the heat exchanger and steam drum.

canned motor-pumps

Water is pumped through the primary by canned motor-pumps, one to each loop. As mentioned previously, these pumps are designed for zero leakage. A single-winding, two-speed motor of approximately 1400-kw input drives a single stage 18 300 gpm centrifugal pump. Power supply for both speeds is 2300 volts, 60 cycle, three phase.

Both rotor and stator of these pumps are encased in stainless-steel cans designed to withstand full system pressure. Primary water flows around the rotor between the rotor and stator. Because the motor is an integral part of the pump, no seals are required between motor and impeller, as in conventional pumps—thus eliminating the major source of leakage.

As in other parts of the primary loop, the hot water must serve as a lubricant for the rotor of the pump. The exterior of the stator is cooled by coils to prevent overheating.

primary loop valves

As indicated in Fig. 1, each primary loop has three different types of valves. *Main stop valves* are located within the reactor container, one in the inlet and one in the outlet of the reactor. Since access to these valves during reactor operation is not feasible, the valves must be remotely operated, as well as have a high degree of reliability.

Since leakage must be essentially zero and the operating pressure is high, valves sealed with bellows or stem packing were avoided. Normal maintenance of such valves would be impractical. Therefore, hydraulic cylinder valves are used in the PWR system. These hydraulic valves make it possible to quickly isolate any loop from the system by remote control at the main control console.

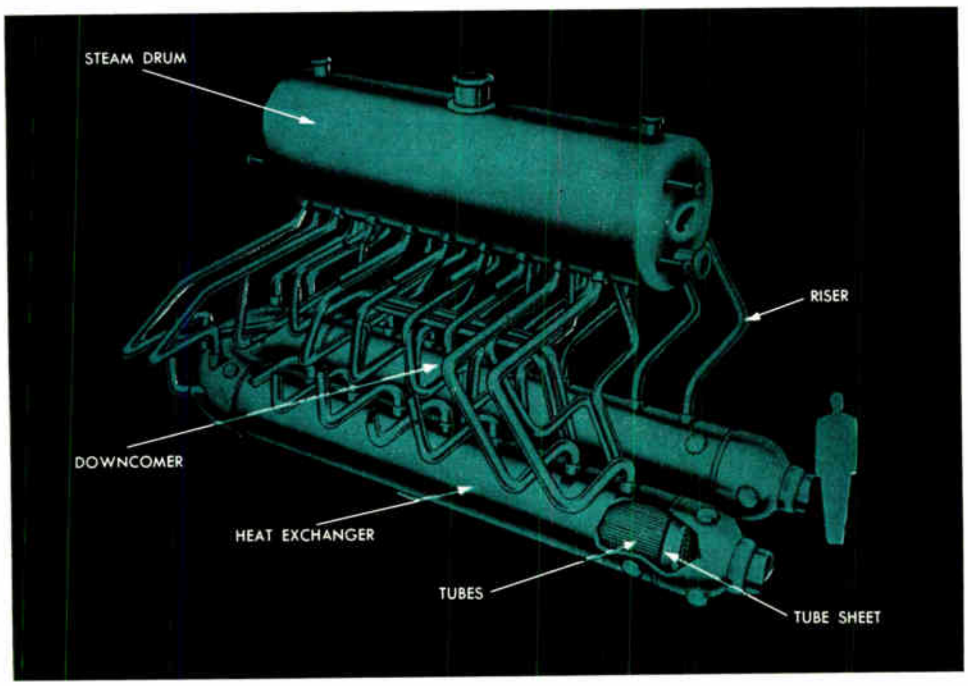
For safety reasons, each main valve is backed up by a *motor-operated gate valve* located inside the boiler chamber. Normally, these valves will be used only when some portion of the loop is being repaired. Before starting the repairs, the valves are closed, by installing a motor-operator and closing the valve locally; thus loop repairmen are protected from the possibility of accidental opening of the remotely controlled main isolation valves.

In addition to these stop valves, each loop has one check valve. The function of this valve is to prevent too much back flow of coolant in the event a pump is stopped for any reason. These valves are designed with low resistance to normal flow, even at very low flow rates.

auxiliary systems

The simple concept of a primary loop with a few key elements ignores many functions that are important and necessary to the reactor-plant operation. The functions performed by these so-called auxiliary systems are many and varied. On the surface, some functions of these auxiliary systems—such as pressure relief and water purification—seem like problems common to other types of systems. However, the fact that this is a nuclear system alters the functions sufficiently that the problems are essentially brand-new ones.

Pressurizing and Pressure-Relief Systems—Because it is important that no boiling occur in the primary system, the pressure in the system under normal operating conditions must be maintained within fairly narrow limits despite sudden demands on the system. This is the function of the pressurizing



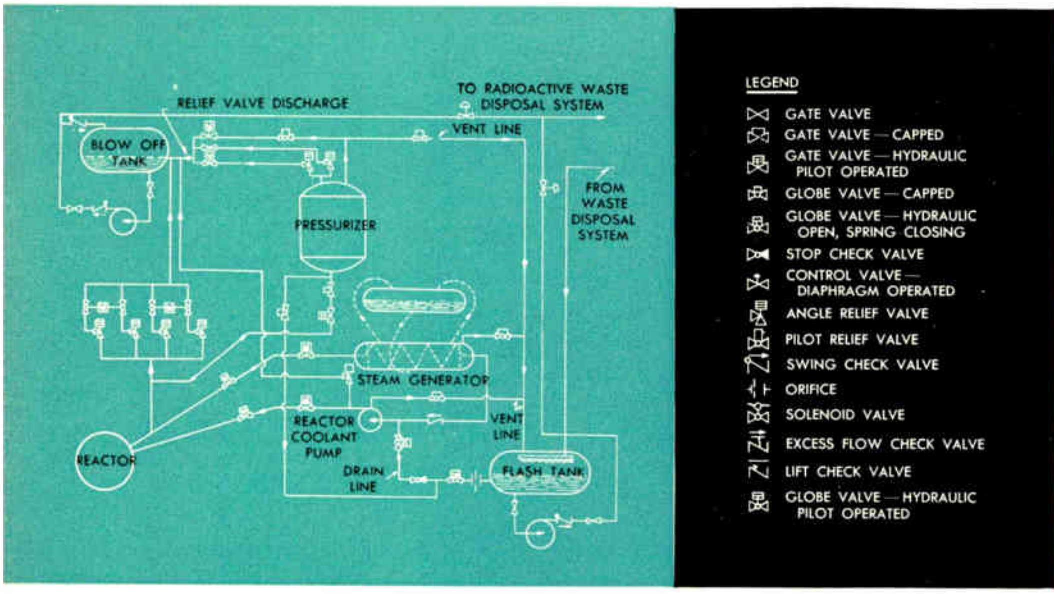


Fig. 4—Diagram of pressure-relief system.

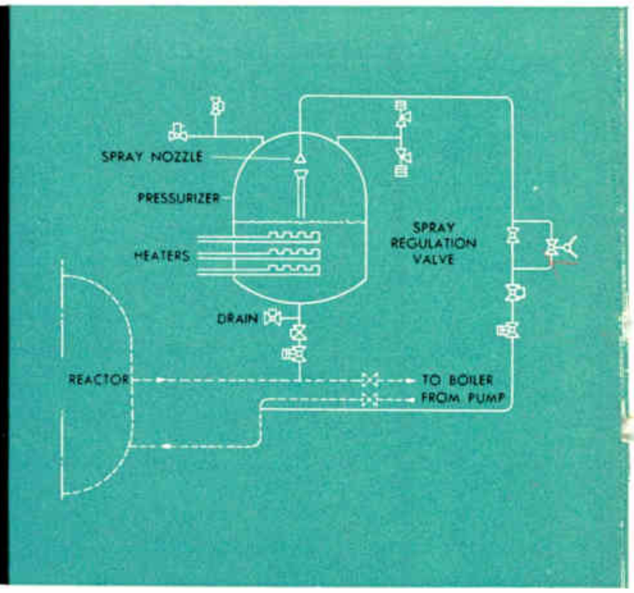


Fig. 5—Diagram of pressurizing system.

system (see Fig. 5). The pressure-relief system is designed to prevent excessive system pressures during especially large power transients.

The main component of the *pressurizing system* is the pressurizing tank, which is tied into the primary coolant system by piping connected to the main loop between the reactor outlet and the steam generator. During normal operation, a little over one-third (100 cubic feet) of the volume of the 260 cubic-foot tank is filled with water. Electric immersion heaters in the water supply enough heat to maintain a 2000-psia steam head in the tank.

The pressurizer is designed to hold the pressure reasonably constant in the primary, despite normal sudden power demands. Suppose, for example, the coolant pressure decreases for any reason. This lowers the pressure in the pressurizer, thus allowing some of the water in the pressurizer to flash to steam. This compresses the steam in the dome of the pressurizer, thereby raising the pressure in the primary.

Should the pressure rise above the normal range, a spray nozzle in the top of the pressurizer goes into action. This sprays reactor coolant into the top of the dome, condensing some of the steam, and thus lowering the system pressure.

The maximum power excursions to which the Shippingport Station will normally be subjected are: plus 15 mw or minus 12 mw as a step change; plus or minus 15 mw at 3 mw per second; and plus or minus 20 mw at 25 mw per minute. Under these conditions, the pressurizing system will maintain pressure within a range of 1850 to 2180 psia.

The pressurizer is not designed to maintain pressure by itself during startup, since it cannot accommodate the volume change involved in going from room temperature to operating temperature. Thus, during this period, about 400 cubic feet of water must be drained from the primary system.

As mentioned, the pressurizer tank takes care of all normal surges due to normal power demands. Suppose, however, the demand should drop to zero suddenly, or pumping power should suddenly be lost. Either of these incidents would result in excess heat in the primary, and thus excess temperature—and it would be a rapid and large increase. For this possible eventuality, a *pressure-relief system* is incorporated (Fig. 4). Pressure relief is provided for each portion of the primary that can be isolated by stop valves—the pressure vessel, the pressurizer, and each of the four main coolant loops.

Pressure settings on all the relief valves are established so that the valve with the greatest reliability of reseating tightly has the lowest setting. Two valves on the pressurizer, which would pass steam, have the lowest settings; a pilot valve is set to relieve at 2175 psig, and a self-actuated valve at 2260 psig. A self-actuated valve at the pressure vessel relieves at 2450 psig, and the loop valves relieve at 2830 psig. All water released by these valves goes to a blow-off tank so that the plant container will not be contaminated by radioactive water.

Coolant-Purification System—Despite the care taken in the primary system to prevent fission or corrosion products from being released to the coolant, perfection is not likely.

Impurities must be kept to a minimum for several reasons. First of all, to maintain safe radiation limits, radioactivity in the primary loop must be restricted to specific values. Second, impurities depositing on heat-transfer surfaces would change the heat-flow characteristics. And third, particle build-up in some areas would plug small clearance passages in the core and other components.

The purification system consists of two identical loops. Each loop serves two of the primary-coolant loops in such a way that it can admit coolant from either reactor loop and discharge it to either reactor loop. However, purification for the entire reactor plant can be handled by one purification loop alone, if necessary.

The purification system consists essentially of a demineralizer, a regenerative heat exchanger, and a nonregenerative heat exchanger. The function of the heat exchangers is to lower the temperature of the reactor coolant from about 508 degrees to 120 degrees F to prevent damage to the resins in the demineralizer, and then to heat it again to about 435 degrees for reentry into the reactor loop. This reheating reduces thermal losses and lessens thermal shock to the reactor loop.

Part of the reactor coolant is continuously bypassed through the purification system. The flow of bypassed water from the reactor loop is through the tubes of both the regenerative and nonregenerative heat exchangers, through the demineralizer, through the shell side of the regenerative heat exchanger, and back to the primary loop.

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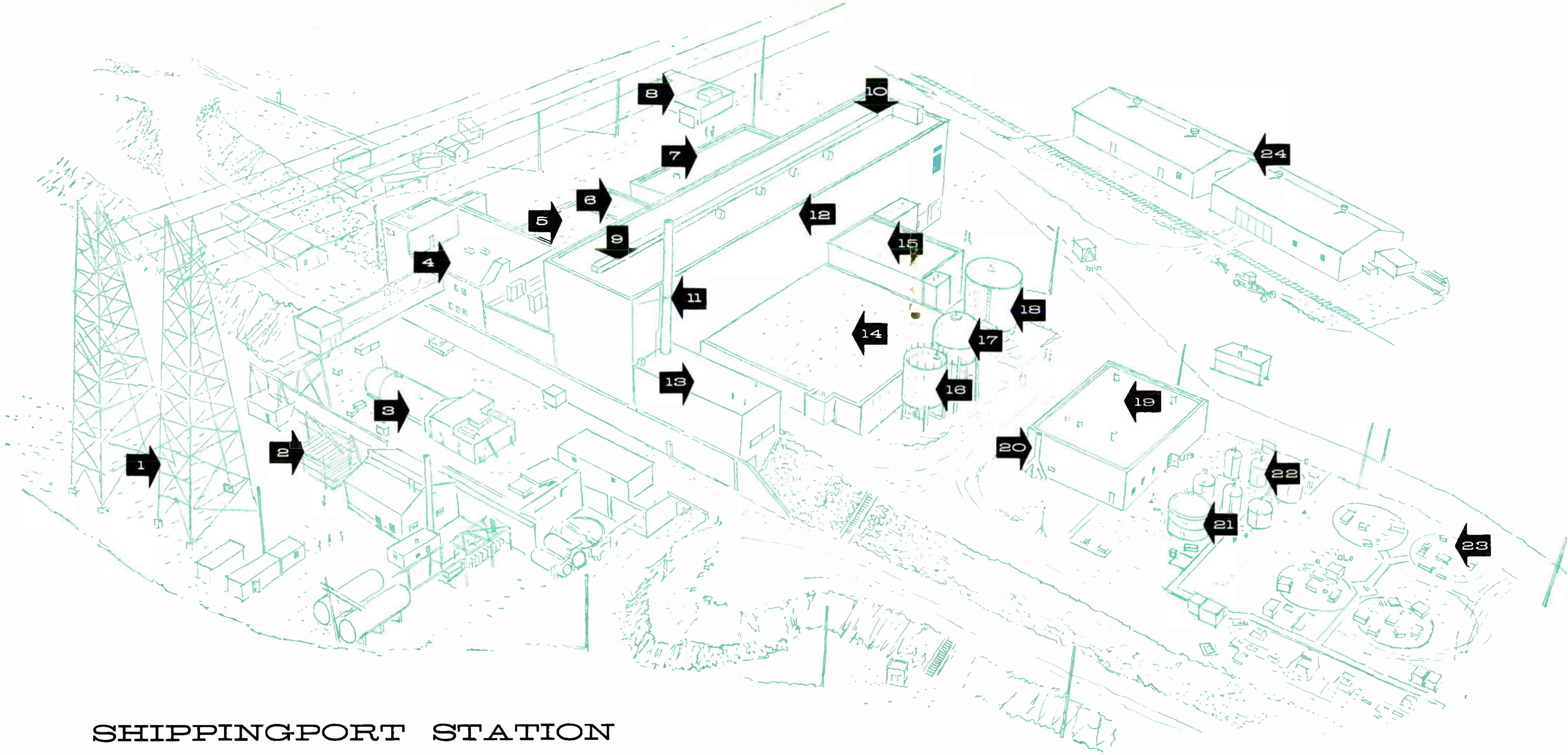
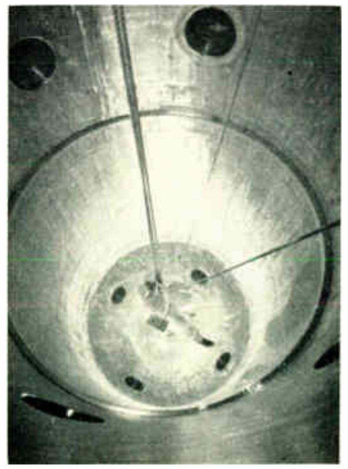


SHIPPINGPORT STATION

Stainless steel is much in evidence in the PWR plant, as witness these parts of the nuclear reactor core.



An inspector views the interior of the pressure vessel. Holes at bottom are inlets for primary water, those at top are outlets.

Workmen vacuum test welds on the reactor vessel thermal insulation.

1 *Transmission Towers.*

2 *Transformers.*

3 *Turbine generator.*

4 *Control room, power distribution panels, and administrative offices.*

5 *East boiler container under this building.*

6 *Auxiliary equipment room.*

7 *Laboratories.*

8 *Guardhouse.*

9 *Approximate location of reactor (underground).*

10 *Decontamination and core assembly rooms.*

11 *Air treatment stack for plant containers.*

12 *Fuel handling canal building.*

13 *Air conditioning room.*

14 *Auxiliary equipment room and west boiler container (under building).*

15 *Fuel storage vault.*

16 *Condensate tank for turbine-generator plant.*

17 *Primary system water tank.*

18 *Canal water storage tank.*

19 *Waste-disposal building.*

20 *Incinerator and gas discharge stack for waste-disposal building.*

21 *Spray recycle tank.*

22 *Test and special monitoring tanks.*

23 *Decay tanks.*

24 *Temporary buildings.*

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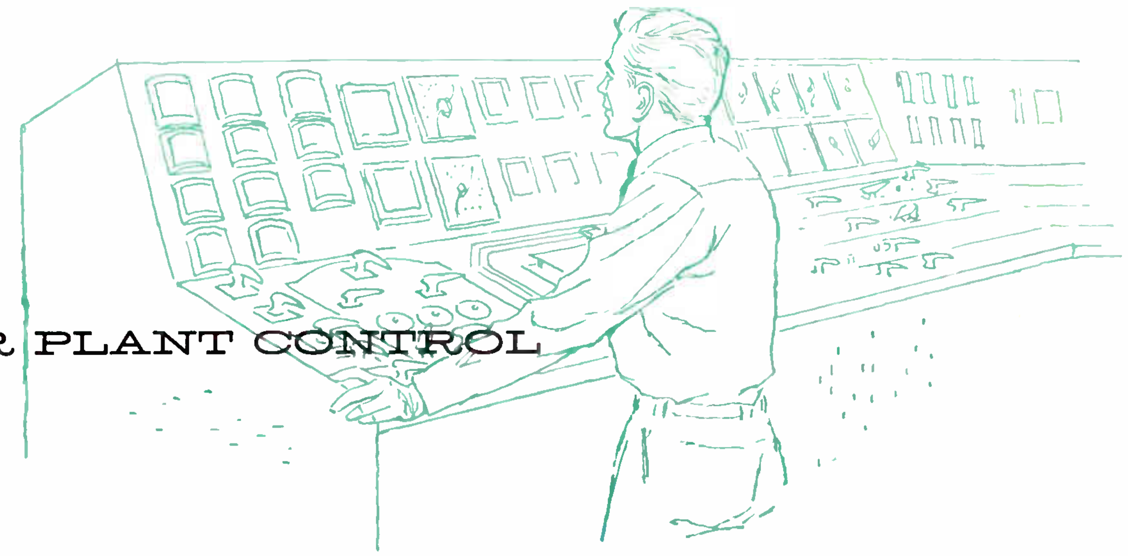

REACTOR PLANT CONTROL

Like most other control systems, that for the PWR primary plant consists of two basic parts—sensing elements and instrumentation that indicate existing conditions, and controls that react to these signals and maintain or restore proper conditions.

As might be expected, instrumentation in this pioneer plant is extensive. Much of it is included exclusively to provide information to aid future designs; the remainder is used directly for system control.

With a negative temperature coefficient, the reactor inherently adjusts to meet power demand without external controls. This does not eliminate the need for reactor controls but does simplify the problem considerably. In normal operation, the control rods are used primarily to correct coolant temperature drift caused by fission-product poison variations. The control rods may also be used to assist the negative temperature coefficient in providing a more rapid restoral of normal coolant pressure and temperature following unusually large load increases and decreases. When so called upon, they return to the position occupied previous to the load increase or decrease, as the coolant conditions (temperature) return to normal.

functions of control system

The operational requirements of the Duquesne Light Company system were an important consideration in the design of the PWR plant. The more important design requirements were the ability to start up and shut down in a reasonably short time; to have the ability to respond to both normal and abnormal power-system transients without exceeding plant limitations. Also, the plant must be able to protect itself against accidents; these might include complete loss of load, loss of auxiliary power, sudden changes in reactor output caused by equipment failure or operator error, or failures in the main-coolant circulating system resulting in reductions in flow.

More specifically, the reactor-plant control is designed to accomplish several different functions. It must provide for manual start-up of the reactor in either a hot or cold condition. Manual regulation of reactor power must also be possible, since this type of operation is used at power levels below 10 percent of normal power, and whenever temperature or pressure is below normal. The control system must also provide means for automatic or manual control above 10 percent of normal power, and at normal temperatures and pressures.

In addition, the control system must be capable of shutdown under two different circumstances. It must be able to shut down the system under normal conditions, and protect all equipment from damage by decay heat. Also, the control must be able to shut down the reactor automatically under conditions that could lead to equipment damage if allowed to continue. Last, the control must operate all auxiliary fluid systems in the main coolant loops.

To accomplish these functions, information about conditions in the system must be provided quickly, accurately, and for the most part continuously. The instrumentation system must report such things as reactivity level in the reactor, and coolant pressure, temperature, and flow.

The control system as a whole must be capable of monitoring all important actions in the reactor system, must give indication of conditions to operators or act automatically, and in some circumstances give an alarm to warn of potential trouble.

plant control subsystems

Control of the PWR plant is accomplished by several different subsystems. Reactivity conditions in the reactor itself are sensed by a *nuclear-instrumentation system*. Basically, this consists of neutron detectors, which measure neutron flux, and a conversion system that changes this information into indications of power level and its rate of change.

The neutron detectors are located in the water that serves as a neutron shield for the reactor. Three channels of neutron detectors are provided, placed around the core to view three different segments of the core.

Information from this system is provided to the main control console as well as to the automatic control and protection systems. In addition to this nuclear instrumentation, conditions within the core are also measured, but this is for information purposes only and is not used in control. Core temperature is measured by thermocouples located in seed elements, and flow is measured in seed-and-blanket elements by differential pressure cells. These elements are so arranged to provide useful information as to temperature distribution and flow within the core. This information is recorded to provide data for core design calculations.

Reactor plant instrumentation provides all the necessary information about conditions in the primary system. As indicated in Fig. 1, temperature, pressure, pressure drop, and flow are all measured at various parts of the system, and the information relayed to the necessary indicators, recorders, or controls.

The *reactor-control system* consists of two basic parts—a *rod-control system* and a *reactor power-and-temperature control*.

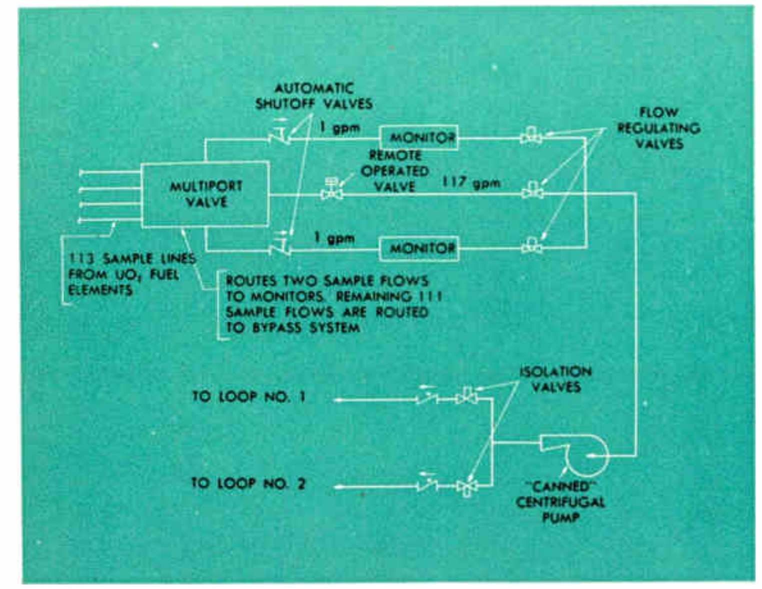

Fig. 6—Diagram of failed element detection and location system.

Coolant Discharge and Vent System—Any radioactive liquids or gases released in any part of the primary system must be carefully collected and conveyed to the waste-disposal system. This is the function of the coolant discharge and vent system. The function necessitates connection with each part of the system that may release radioactive liquids and gases; it also requires that this system must have equipment to cool and depressurize the material; and have the means of safely discharging these materials to the waste-disposal facilities.

Essentially, the system consists of a blow-off tank and a flash tank. The flash tank cools and depressurizes reactor-system drainage; the blow-off tank is designed to contain all the steam or reactor coolant discharged through various pressure-relief devices.

Additional functions of this system are to bleed off excess reactor coolant during startup operations, to maintain proper fluid level in the pressurizer, and to drain an isolated loop.

Decay-Heat Removal System—Some auxiliary systems are designed to handle specific but abnormal eventualities. For example, suppose a-c power to the reactor coolant pumps should suddenly be lost. In this event, the reactor would be shut down immediately but, with little or no circulation in the primary system, sufficient residual heat to damage the core might exist. This eventuality is taken care of by the decay-heat removal system, which actually consists of a steam-relief valve in the secondary system set to operate at 707 psia and sized to dissipate 7000 kw of heat. The valve is normally isolated, but is placed in operation automatically if a-c power to all reactor coolant pumps is lost.

Natural convection heat flow occurs in the primary loop, the heat is transferred via the steam generators, and resultant steam is released through the relief valve. The valve is so designed that it will act soon enough to prevent other relief valves in either the primary or secondary from operating.

Failed Element Detection and Location System—A prime purpose of this PWR plant is to collect data that will further reactor technology. Many subsystems and much instrumentation are included in the plant largely for that purpose. One of these systems is designed to detect failed blanket fuel elements.

Considerable experience has been amassed with plate-type fuel elements containing enriched uranium. However, relatively little is available with the type of element used in the blanket, i.e., a hollow zirconium-alloy rod containing uranium-oxide pellets. For this reason only the blanket fuel elements are monitored.

A failed element can be detected by the presence of delayed neutrons in the coolant. Individual sample tubes draw off coolant from the region of each of the 113 blanket assemblies. These pass through a multiport valve mounted in the reactor pressure-vessel head. The valve routes samples (see Fig. 6) from two of the tubes to separate monitoring systems, where presence of decayed neutrons can be detected. The rest of the samples are bypassed. Two samples are checked at a time, in sequence, until all tubes have been monitored, and then the operation repeats.

When the level of decayed neutron emission in any sample indicates failure of an element, a signal is sent to the control room indicating the location of the assembly.

Originally, on the basis of tests, the system seemed capable of locating a defect as small as a 5-mil hole. However, later, it was discovered that the zirconium used in the PWR contained a tiny amount of uranium; the background from this uranium decreases the sensitivity of the system, so that only major defects or many small defects in a single channel can be located. This, however, is adequate sensitivity. The development of the system added one more small bit of information to reactor technology. Efforts are now being made to control the uranium content of zirconium.

Other Auxiliary Systems—A double check on the coolant purification system is furnished by a separate *coolant-sampling system*. This draws samples upstream and downstream of the purification system's demineralizers. Another function is to provide the necessary chemical information for the *chemical-addition system*. This addition system maintains oxygen level in the coolant below certain levels. Hydrazine is added during initial precritical startup to reduce the oxygen content of the coolant to acceptable limits (0.14 ppm); during reactor operation, gaseous hydrogen is added to consume oxygen formed by decomposition of water in the reactor. Lithium hydroxide is added by this system to maintain the pH level of the water at the nominal value of 10.

In addition to these functions, the chemical-addition system is designed to add boric acid to the water to shut down the reactor chemically. If any control rods should become stuck out of the core, a normal operating condition could be maintained, but it might not be possible to shut down the reactor to a cold, subcritical condition. In this event, boric acid—a good neutron absorber—would be inserted in the coolant. This method, of course, would be used only if normal means for effecting rod insertion have failed.

The *coolant-charging system*, as its name implies, is used to fill all parts of the primary system before operation, and can also be used to help flush out the system after shutdowns. Another important auxiliary is the *safety-injection system*, which is an emergency system designed to help cool the core if primary coolant is lost at a greater rate than it can be made up by the charging pumps.

As can be seen, the reactor plant, including its auxiliaries, is designed to provide maximum safety and information. Some auxiliary systems, in all likelihood, will not be necessary in future plants. In reviewing the various components and functions of this plant, it is relatively simple to spot portions of the system that might be omitted or simplified in future plants. But only through the construction and operation of the Shippingport Station can the feasibility and safety of design changes be proved conclusively. ■

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The rod-control system converts reactivity demand signals into control-rod motion, or maintains the rods stationary when there are no signals. The rods can be controlled manually from the console, and operated automatically by the power-and-temperature-control and the reactor-protection system.

As mentioned in a previous article, 12 of the 32 control rods will remain fully inserted in the core. In effect, these are "compensating" rods, in that they will be withdrawn only when it becomes necessary to compensate for xenon build-up and fuel depletion.

The rod program calls for motion of four rods at any one time. By using different gear ratios, they can be moved in one-inch or three-inch increments. Each rod, on an average, has a value of about one percent in reactivity in full travel. Normal speed of the rods is at the rate of 11 inches per minute; in any emergency condition (i.e., a "scram") the control rods can, of course, be dropped their full length into the core.

The *power-and-temperature control* receives information as to primary coolant temperature and nuclear power level, and reacts to maintain the average coolant temperature constant. A schematic diagram of this system is shown in Fig. 2. While this control constantly monitors the primary system, the negative temperature coefficient maintains constant average coolant temperature in the steady state under most conditions except that of changes in xenon 135 concentration. Should large changes in power level take place, the power-and-temperature control provides the necessary signals to initiate rod motion. It also initiates rod motion when changes in xenon 135 concentration cause the temperature at which the reactor is self-regulating to change.

Reactor-plant control includes both manual and automatic operations of the primary and auxiliary fluid systems. This includes control of valve and pump operation, and of pressurizer spray and heater action.

The *reactor-protection system* protects the core from damage. It can cut back power when conditions become potentially dangerous, or shut down the reactor quickly by release of control rods when core damage is more imminent. The protection system also provides alarms for those conditions, and in addition includes interlocks to prevent cold water from an inactive loop from being suddenly inserted into the core.

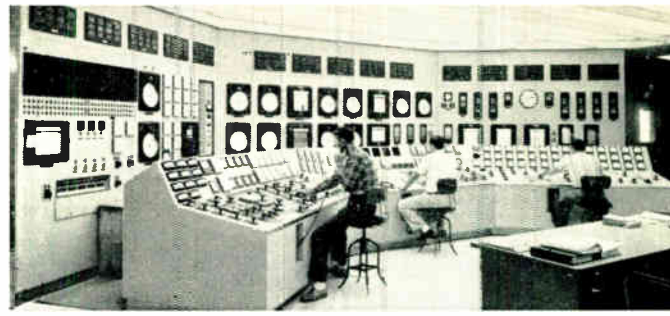
The final part of the plant control system is the *reactor plant monitoring system*. This has two parts—a radiation-monitoring system, and remote-viewing apparatus. The radiation-monitoring system has several functions. It measures radiation levels in various compartments, providing information to enable control of personnel movement within the plant, and furnishes warning to the operator of any defects in the primary-coolant system. The radiation-monitoring system also monitors the waste-disposal system. Another function of radiation monitoring is to furnish data on radiation in areas surrounding the plant site.

Remote viewing is accomplished by two closed-circuit television systems, which monitor conditions in the plant containers. One system enables viewing of areas that are inaccessible during operation; the other views boiler gauge glasses in the containers.

The nerve center of the control system is, of course, the main control room (see photo). The main control console is schematically arranged; all main elements from the reactor core to transmission lines are represented on the console.

control considerations

As mentioned, the reactor system is designed to hold the coolant at a *constant average temperature*. Power control of the



The main control room of the Shippingport plant.

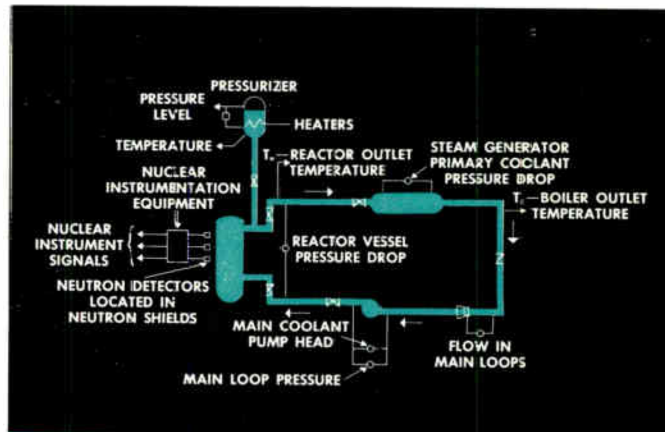


Fig. 1—PRINCIPAL COOLANT LOOP INSTRUMENTATION

Instrumented information is relayed to the following locations:

T_H—REACTOR OUTLET TEMPERATURE
1. Main Control Console—Indicating Recorder
2. Main Instrument Panel—Recorder
3. Auxiliary Instrument Panel—Indicator
4. Reactor Power and Temperature Control System
5. Reactor Protection System (High Temperature signal)
6. Annunciator on Main Instrument Panel (High Temperature alarm)

T_C—BOILER OUTLET TEMPERATURE
1. Main Control Console—Indicating Recorder
2. Main Instrument Panel—Recorder
3. Auxiliary Instrument Panel—Indicator
4. Reactor Power and Temperature Control System

FLOW IN MAIN LOOPS
1. Main Control Console—Indicator
2. Annunciator—Low Flow Alarm

MAIN LOOP PRESSURE
1. Main Control Console—Indicator

MAIN COOLANT PUMP HEAD
1. Main Instrument Panel—Indicator

STEAM GENERATOR PRIMARY COOLANT PRESSURE DROP
1. Main Instrument Panel—Indicator

REACTOR VESSEL PRESSURE DROP
1. Main Instrument Panel—Indicator

NUCLEAR INSTRUMENT SIGNALS
1. Main Control Console
2. Reactor Protection System
3. Reactor Power and Temperature Control System
4. Nuclear Instrument Panel—Recorders and Indicators

REACTOR PRESSURE
1. Main Control Console—Indicator
2. Reactor Protection System
3. Annunciator—Alarm

PRESSURIZER PRESSURE
1. Main Control Console—Indicating Recorder (narrow range)
2. Main Instrument Panel—Indicating Recorder (wide range)
3. Reactor Protection System
4. Annunciator—Alarm
5. Pressure Control

PRESSURIZER LEVEL
1. Main Control Console—Indicating Recorder (narrow range)
2. Main Instrument Panel—Indicating Recorder (wide range)
3. Heater Control
4. Annunciator—Alarm

PRESSURIZER TEMPERATURE
1. Main Console—Indicator
2. Main Instrument Panel—Recorder
3. Heater Control

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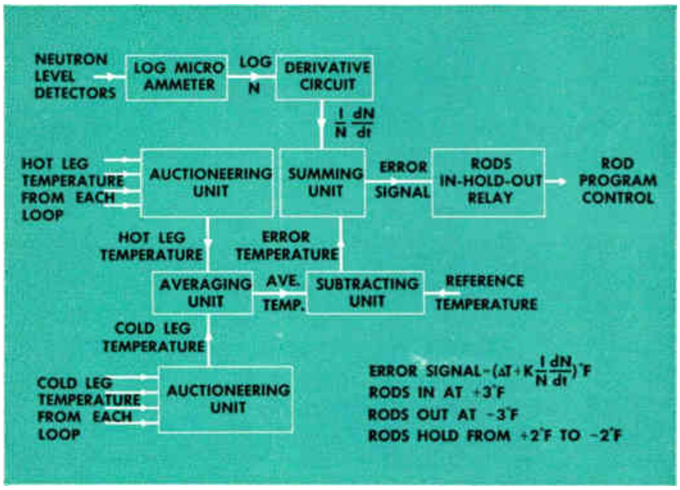


Fig. 2—Power-and-temperature control schematic diagram.

reactor is virtually automatic without external controls because of the negative temperature coefficient. If it were not for the effects of xenon poisoning and fuel depletion, control rods would be needed only for start-up, adjusting power level, and for shutdown. However, xenon poisoning causes the average temperature to drift up or down during load changes. Therefore, some rod motion is required. Rod action is initiated at plus or minus three degrees F error in average coolant temperature, and terminated at plus or minus two degrees error. To prevent overshooting or oscillations, a damping effect is built into the control. The damping effect is provided by the signal from the neutron detectors as described below.

Several features of the reactor power-and-temperature-control system are of particular interest. This system must sense conditions in the reactor plant, evaluate them, decide if action should be taken or if the negative temperature coefficient of reactivity should be allowed to function alone, decide upon the necessary action, then initiate that action. A block diagram of this system is shown in Fig. 2.

The control system receives information from the neutron detectors and from temperature detectors in the hot and cold legs of the primary system. The information from the neutron detectors is translated into a signal that indicates rate of change of power level, which is fed to a summing unit. At the same time, the average loop temperature is measured, compared to a reference temperature, and a resultant error signal sent to the summing unit.

The combination of these power and temperature signals is then considered by the summing unit and a net error signal transmitted to a relay system, which in turn furnishes a signal to the rod-control system indicating whether the rods should be moved in, moved out, or held in position. Consideration of the block diagram and the above brief description shows that the system, while recognizing a temperature error in excess of three degrees F, may decide to "wait and see" because the rate of change of neutron flux gives promise of providing the needed correction.

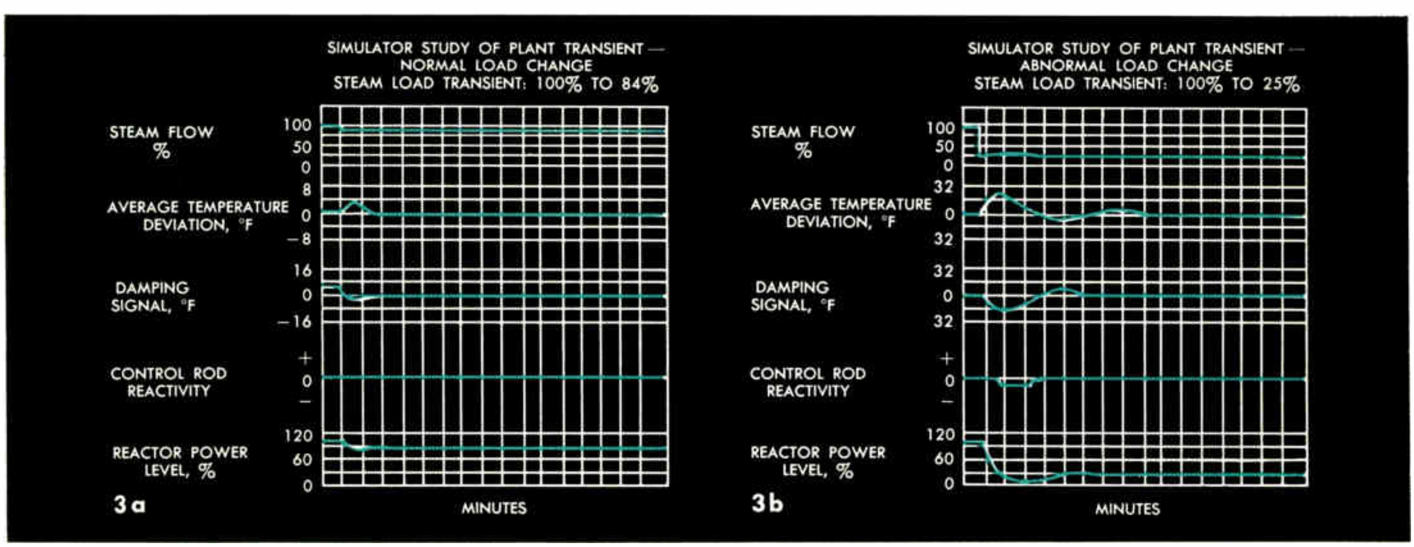
An interesting feature included in the temperature portion of this control is the *auctioneering units.* Hot-leg and cold-leg temperature measurements are obtained from each of the reactor-coolant loops. All measurements from the cold leg are fed to one auctioneering unit, and all measurements from the hot leg to a second auctioneering unit. Each auctioneering unit selects the *highest* of the several temperature indications it receives. These signals, one from each unit, are then fed to an averaging unit, which computes the average coolant temperature and then factors it into the control system.

The fact that each auctioneering unit selects the highest of the several temperatures it receives, prevents an undesirable system action. If only one temperature indication were obtained from each leg, and either temperature instrument failed low, this would lower the computed average coolant temperature, and ultimately result in a rod withdrawal and an unnecessary increase in reactor power level. Since each auctioneering unit selects the highest of several values, any instrument that fails low is, in a sense, ignored. Using this system, should any instrument fail high, the faulty signal would be used, but since this would lead to an average temperature signal that was too high—and thus a rod insertion and lowering of power—this eventuality is of less consequence.

As can be surmised from this auctioneering unit, the entire control system is fail-safe insofar as possible. Where failure will not be indicated to the operator by system action or annunciation, visual monitoring of the components will be accomplished once each eight-hour shift.

Auctioneering systems are used in cases where several signals are available, and all are designed to select the signal that represents the greatest danger to the plant. By this and other means safe and reliable operation is assured.

Fig. 3—Simulator curves showing effect of several load changes; at left, for a small change, and at right for a much larger change.



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computer studies of operation

Extensive computer studies have been made of the effectiveness of the control system. In effect, the complete system has been set up on an analog computer, and various operating conditions imposed. Two of these illustrate graphically the system performance expected under operating conditions.

The curves in Fig. 3a show a normal load change, from 100 percent of full power to 84 percent. This change is within the ability of the negative temperature coefficient to accommodate without undue system disturbance. In this study the temperature coefficient used was only half of the minimum value expected in the actual plant, which demonstrates the stability to be expected.

The curves in Fig. 3b show the effect of a much larger change in turbine throttle opening—from 100 percent of full power to 25 percent in one step. In this case, rod motion is employed to limit the temperature excursion. Note that in a very few minutes the control rods return to their original position and the temperature coefficient stabilizes the plant.

The total instrumentation and control in the PWR plant is obviously large. However, a significant part is included either for "double-safe" operation or for information gathering purposes. ■

Table 1—Reactor Protection System Inputs Causing SAFETY SHUTDOWN OPERATION

PLANT PARAMETERS	TYPE SENSING ELEMENT	INSTRUMENT RANGE	SHUT-DOWN	COMMENTS
Primary Coolant Temperature (T_h)	Resistance Thermometer	475°F to 575°F	T_h > 555°F any one T_h sufficient	
Pressurizer (Steam) Pressure	Twisted Bourdon Tube	0 to 3000 psig	< 1600 psig	Cut out during cold start-up until pressure is above 1600 psig.
Reactor Vessel Pressure	Twisted Bourdon Tube	0 to 4000 psig	< 1600 psig	Cut out during cold start-up until pressure is above 1600 psig. Auto. restored > 1800 psig.
120 V A-C Regulated for Reactor Protection	Type SV Relay (Half Wave) Type SG Relay (Undervoltage)	Drop out, approx. 15 cycles after loss of voltage or reduction (½ wave)	0.25 second, 2 of 3	Relays are supplied by bridge rectifier from voltage regulating transformer.
120 V A-C Regulated for Nuclear Instrumentation Channels.	Type SG Relay	Drop out, approx. 15 cycles after loss of voltage	0.25 second, 2 of 3	Relays are supplied by bridge rectifiers from A-C supplies to nuclear instrumentation channels.
Neutron Flux (Power Range)	Compensated Ion Chamber	1–150% rated full power	140%. See comments	Safety insertion and shutdown depend on reactor power to reactor coolant flow ratios (2 out of 3 coincidence).
Primary Coolant Pump Speed	Aux. Contacts on Speed Selector Sw.	Contacts closed for slow speed connection	See comments	Used in determining when pumps are not at full speed.
Primary Coolant Flow (Pump Power)	Potential and Current Transformers to a watt-type relay	0–800 w balanced 3 phase (120 V pt & 5 a, ct)	< 50% pump power. See comments	Insertion & shutdown depend on power to flow ratios. Loss of flow is detected when timing relays drop out after pump power relays restore.
Primary Coolant Flow (Main-Loop Stop Valves)	E-Core Valve Position Indicator	Open or not fully open	½″ valve travel. See comments	Insertion & shutdown depend on power to flow ratios. Valve not fully open indicates assoc. loop not operating.
Manual Shut-down	Manually Operated Switch	See comments	Manually Controlled	Manual safety shutdown switch may be used to shut down reactor at any time.
Seed Fuel Element Temperature	Thermocouples	500°F to 1500°F	Approx. 1050°F. See comments	Exact setting of alarm and shutdown points, if used, will depend on analysis of system performance. This protection may be connected in place of T_h.

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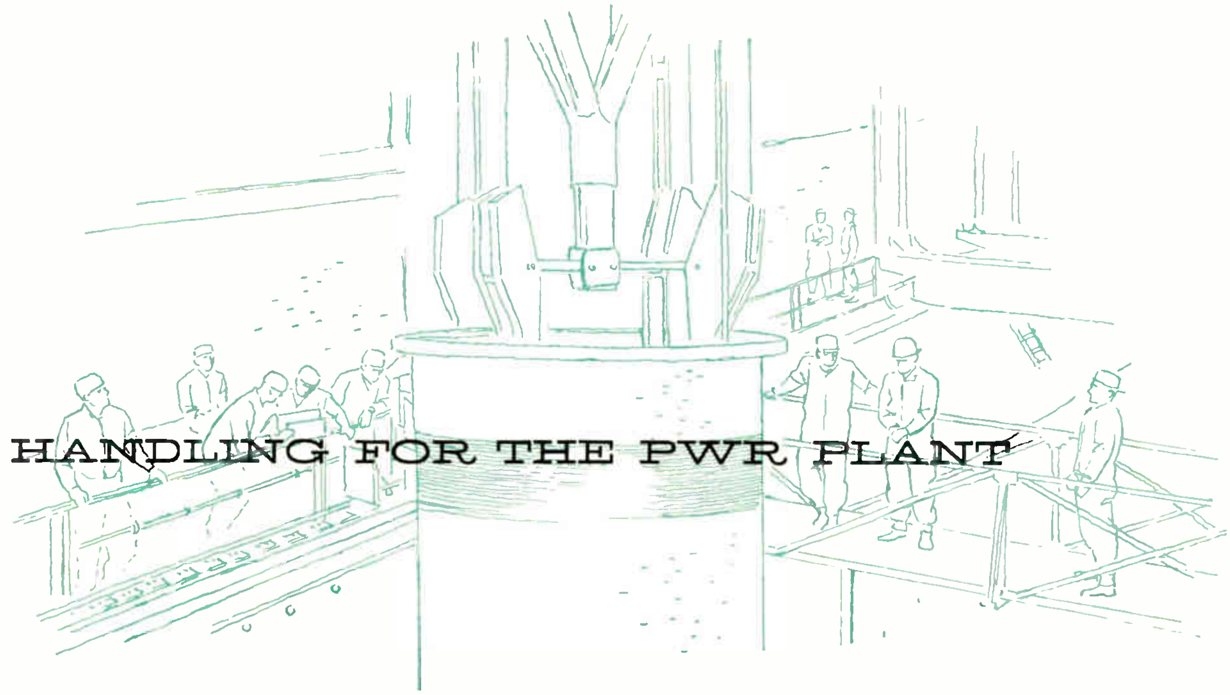

FUEL HANDLING FOR THE PWR PLANT

The nuclear core of the PWR reactor is a meticulously constructed assembly composed of hundreds of accurately built components. Not only must the outside dimensions of the 70-ton, eight-foot diameter core be held to close tolerances, but hundreds of channels and passages between core components must be maintained at accurate spacings, for both nuclear and mechanical reasons. The core is not a particularly fragile device, but it can be damaged. And because large amounts of time and money are involved, it must be handled like a delicate mechanism from the time it is moved out of the manufacturing plant until it is completely installed in the reactor. When a spent core is removed, the problem of radioactivity is also introduced. All this adds up to extensive planning in the design of fuel-handling facilities and procedures.

Actually, fuel handling breaks into two parts. Initially, of course, comes the installation of the first core; and, subsequently, the problem of refueling. Although used in a somewhat different fashion, the basic fuel-handling facilities are the same for both operations, although not all equipment is used in each situation.

fuel-handling equipment

Located above the reactor compartment of the plant is a fuel-handling canal (Fig. 1). Several areas of the canal can be separated from the others by locks, so that they can be isolated, and drained or filled with water. Various areas of the canal serve different functions; included is an area where fuel can be stored, another area where two cores can be dismantled, a service lock for a refueling tool, a transfer canal, and the reactor pit. The reactor pit is directly above the reactor com-

1a

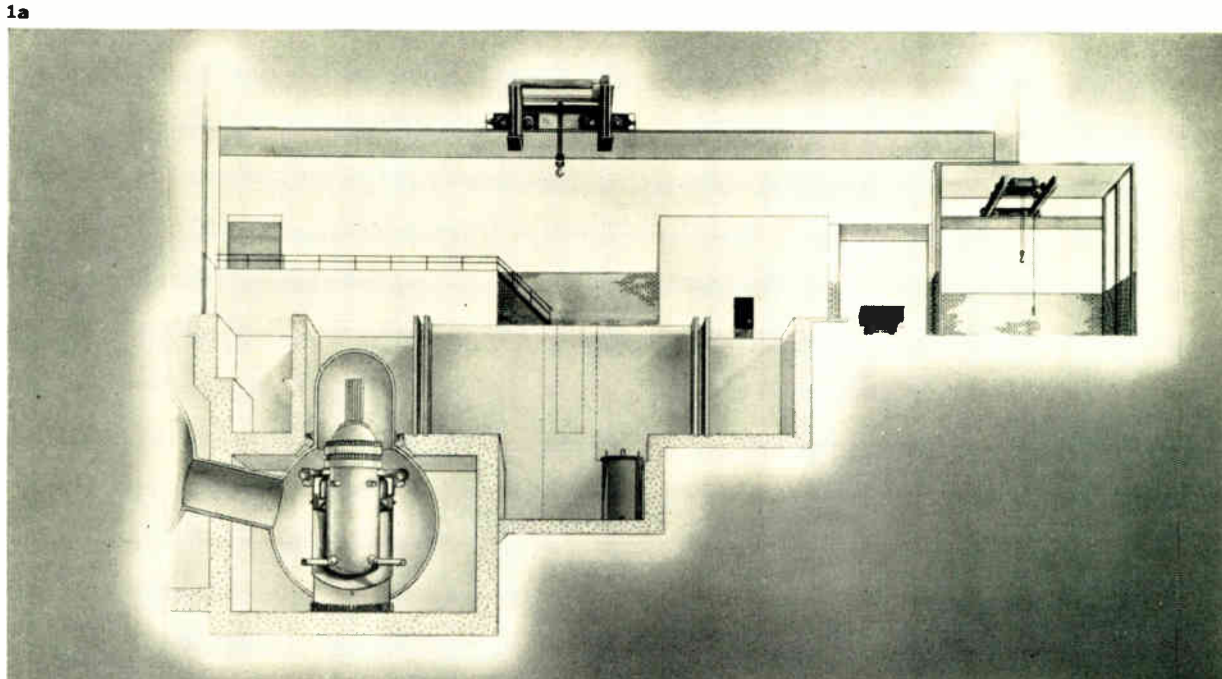

Fig. 1a and 1b—Cutaway sketches of fuel-handling canal.

Fig. 2—Artist's sketch of refueling through ports.

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partment of the plant; access to the reactor is achieved by removing a dome that seals the compartment from the canal.

When installing the initial core, this canal will not contain water, since the core is not radioactive at this point. When any refueling is done, the canal will be filled, and components will be handled under water to protect operating personnel from radioactivity.

The first core will be installed as a unit. When refueling is necessary, this can be accomplished in two ways: the reactor head can be removed and all or any part of the core removed; or individual fuel elements can be removed through fuel ports in the reactor closure head without removing the head itself.

For removing individual assemblies, a long slender grappling device is inserted through the fuel ports and can be controlled to remove any fuel assembly. For handling the entire core, a 100-ton crane is available, plus a special grapple and jigs for handling the core.

initial core installation

To insure mechanical accuracy, the entire core was assembled at Bettis before shipment to the Shippingport site. The elements were assembled with a cadmium "poison" rod locked in each seed element. After assembly, the cooling and rod passages were checked dimensionally, and each control rod tried in its respective seed passage. After checking, the poison rods were again locked into place and the core disassembled for shipment.

Fuel elements were shipped to the site in special shipping racks, over a carefully planned route, and with elaborate precautions against any form of accident.

Accidents in the reassembly and installation of the core must be guarded against—not only from a nuclear standpoint, but also because of the large monetary investment. However, as far as nuclear hazards are concerned, several facts about safety were established by experiments. As long as it is dry, the assembled core cannot go critical, even if the control rods are withdrawn. In water, the complete core can be kept subcritical by the control rods. Without the blanket assemblies, the remaining portion of the core (i.e., the seed portion) cannot go critical, even in water and with the control rods withdrawn. Since control or poison rods are at all times locked in place, and every precaution is taken to keep the core dry, no nuclear hazard is present during installation of the first core.

After the core was reassembled at Shippingport, the poison rods were removed and the control rods replaced and held in position. The core-hoisting equipment, shown in Fig. 3, then was attached and the core lowered into place. Although this seems at first a simple procedure, in reality it is a ticklish process. Only six-hundredths of an inch clearance is available between the core and the thermal shield, and at this point the clearance cannot be observed. This requires some means to detect binding or friction, which is accomplished by a load scale on the crane. Actually, a dry run was first made with a simulated core to check procedures and make sure that all the keys and keyways were aligned.

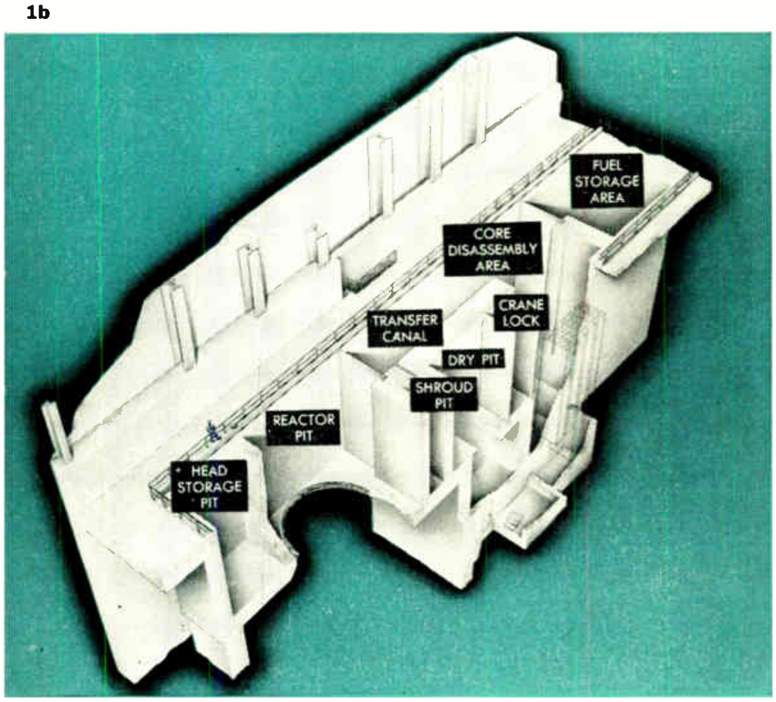
refueling methods

As mentioned in previous articles, the seed portion of the nuclear core must be replaced at least once to get the full useful life out of the blanket. This necessitates means of removing portions of the core without taking out the whole core structure. The individual removal of fuel elements is also desirable from another angle. Should any element fail, it could be removed without removing the whole core. While core elements with minor defects probably would not be removed until the reactor is refueled, a major defect might make replacement of an individual element desirable.

As mentioned, removal of spent fuel can be achieved in two ways in the PWR plant. Individual elements can be removed through the fuel ports of the reactor closure head, without removing the head; or the head can be completely removed for refueling. During both these operations, the fuel-handling canal will be filled with water.

Refueling Through Fuel Ports—Individual elements can be withdrawn or replaced by a fuel-handling tool, shown in Fig. 2. This tool is inserted through the fuel port, and can be indexed to remove any fuel element. The tool is motor driven and is mounted on a structure that spans the canal. It is controlled

1b



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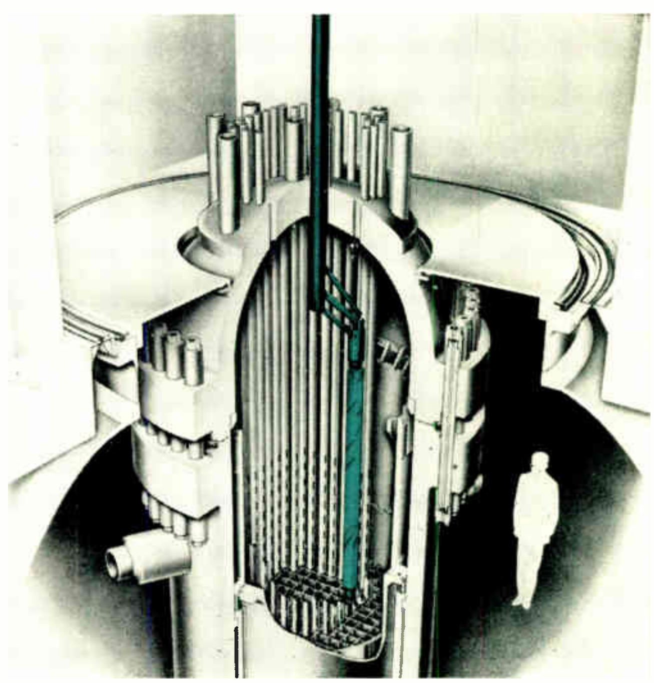

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by an operator at a control console, who has variable speed control of five functions—bridge, trolley, hoist, azimuth, and arm extension—only one of which actions can be performed at one time.

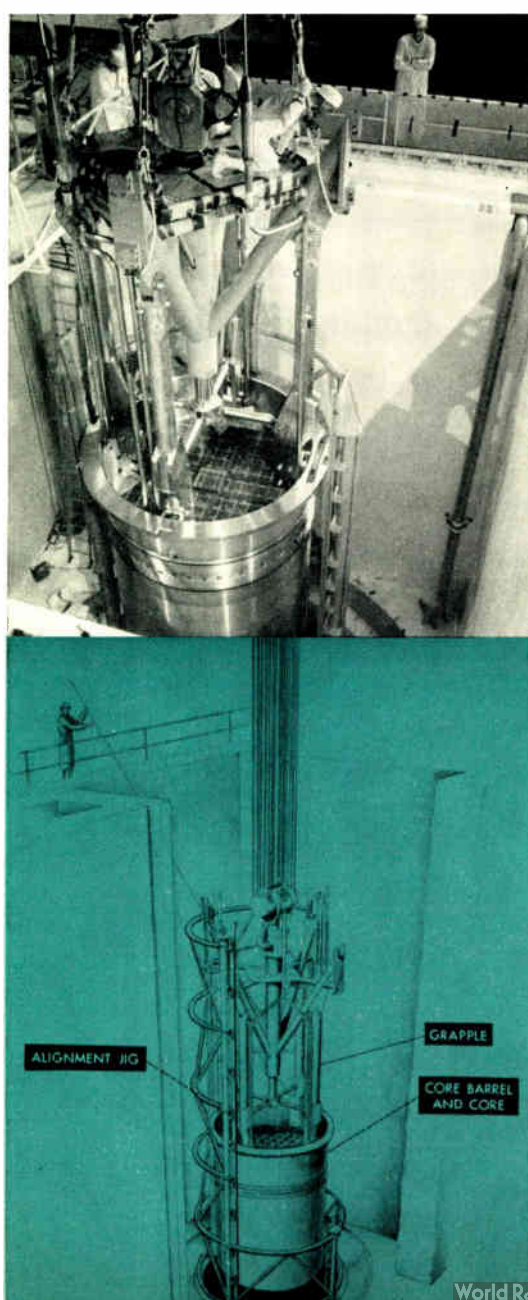
Initially, an underwater television camera, plus a tv screen near the operator are used to position the arm for each fuel element. These positions are recorded so that they can be used to reposition the tool without the camera.

The extraction tool is aligned and inserted in the fuel port. When it reaches a point where it is safe to extend the arm that contains the tool head, a light on the operator's console is energized. The tool is then indexed to select the proper fuel element. The tool head is so designed that it can unlock a fuel element from the core; once the element is unlocked from the core it is locked to the tool. Latching or unlatching of the fuel-handling tool can only be accomplished when the tool head is properly inserted in the grid of the fuel element.

Once firmly attached to the tool head, the fuel element is raised vertically until it clears the core, then the tool head is realigned with the fuel port and the whole structure withdrawn from the reactor vessel. The fuel-extraction tool is so designed that it cannot raise a fuel element within ten feet of the surface of the canal water, thus preventing any radiation hazard to operating personnel.

Fig. 3a Photo of core being installed.

3b Artist's sketch of core-hoisting equipment.



In this whole process, protection is provided against damage to the core or any other part of the reactor. Load on the refueling head is indicated by strain gauges, so that any variation from normal that might be caused by jamming or other reason can be quickly detected. These strain gauges also indicate any unbalance in horizontal forces on the fuel element; the direction and magnitude of the unbalance are shown on an oscilloscope so that the tool can be repositioned to balance the forces. These and other protective devices prevent damage to the core or the fuel element itself. Once removed, the fuel elements are moved under water through the canal to a fuel storage area, where they can be properly packed for shipment to a reprocessing area.

Refueling with Reactor Head Removed—With the reactor head removed, either individual fuel elements or the whole core can be replaced. The removal of the core is similar to its installation, except that water is, of course, in the canal at the time of replacement. An extension of the crane hook is added to prevent hoisting of the core above a safe shielding level in the canal water. An alignment jig guides the grapple into proper position and also guides the removal. As shown in Fig. 3, one side of the jig is open to allow lateral movement of the core to the core storage pit.

Special Fuel-Handling Equipment—In addition to the tools for inserting and removing the core or individual fuel elements, other equipment must be provided to handle elements once they are removed. Spent fuel, for example, must be shipped to a processing plant. This requires special shielded containers to hold the units for shipment. Individual fuel elements are loaded into the special container underwater in the canal, then the container is removed from the canal and placed on a special railway car for shipment. This car has additional built-in shielding, and a heat-removal system for decay heat.

At times, a fuel element may have to be removed and returned to the laboratory for examination. A special container handles one irradiated fuel element, and provides shielding and cooling during shipment. This container is about $4\frac{1}{2}$ feet square, 12 feet long, and weighs 25 tons. The cavity is large enough to hold one fuel element plus cooling water. This cooling water is, in turn, cooled by stainless-steel coils capable of removing 40 000 Btu per hour.

It a ruptured fuel element is to be removed from the core, a special enclosure prevents fission products from being spread throughout the canal. This is a sheet-metal enclosure with a door in the bottom; the ruptured fuel element is drawn into the enclosure by the extraction tool, then the door is sealed, thus preventing release of fission products.

In addition to these special provisions, other equipment must be provided for disassembling the blanket elements, and to allow inspection or replacement of individual fuel rods. As mentioned, an underwater television camera is also used in the refueling process. This can be used inside the reactor, or to view any underwater operation outside the reactor; periscopes are also available for observing underwater operations, and cameras can be attached to the periscope viewers if desired.

No hazard exists in the refueling operations unless operating procedures are grossly violated. Either poison rods or control rods are in the reactor core at all times, so no nuclear incident is feasible. Individual fuel elements, or the core itself, cannot be withdrawn from the canal water in the reactor pit, so any radiation danger is eliminated. As in the case of other aspects of the PWR plant, painstaking plans have been made and every hazard has been surveyed to prevent accidents. ■

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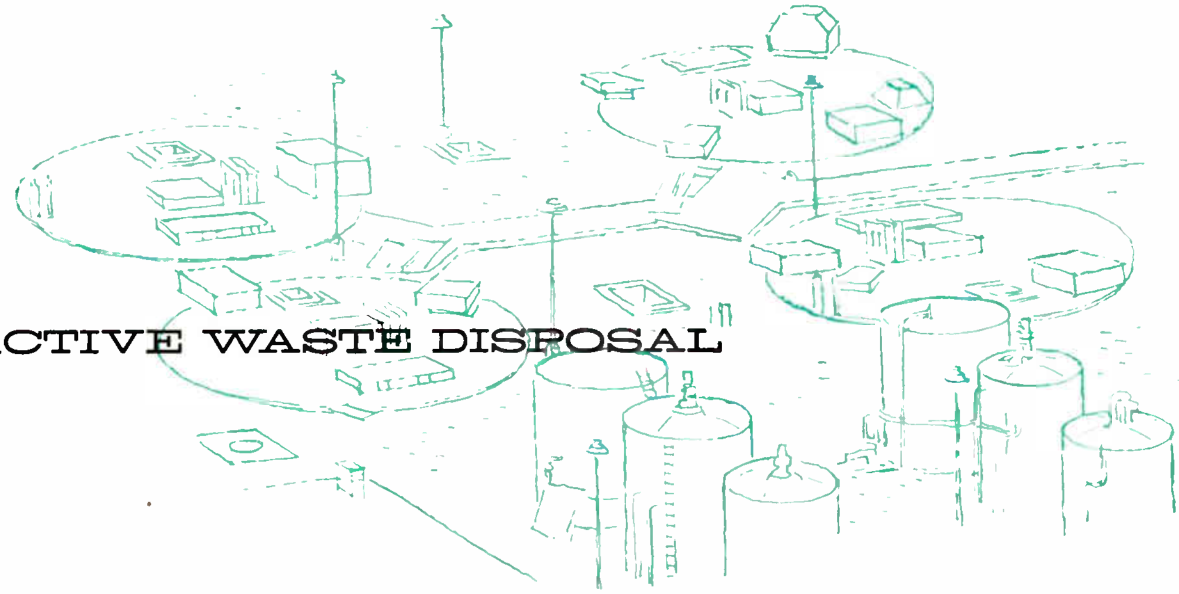
RADIOACTIVE WASTE DISPOSAL



■ Potentially, the major source of radioactive wastes from the PWR plant is the primary coolant. These wastes could be produced in two ways. First of all, any corrosion products and trace elements in the coolant can become radioactive; and second, fission products released by failure of fuel elements would be radioactive. These radioactive liquids and gases are collected from various parts of the primary system by the coolant discharge and vent system and fed into the waste-disposal system.

In planning the system a prime consideration was the number of failed fuel elements that might be expected, since this would be the major source of radioactivity. The waste-disposal system is designed for the worst feasible case—that at any one time the core contains 1000 failed elements. This is a pessimistic design condition, since the plant contains means of detecting, locating, and removing failed elements, so the presence of that many at one time is highly unlikely. Other sources of radioactive wastes are from the laboratories, laundry, showers, and decontamination facilities.

Wastes in the PWR are disposed of in one of three ways—by natural decay of their radioactivity, by dilution, or by concentration and storage, or any combination thereof. All liquids, gases, or solids that might be radioactive are processed by the waste-disposal system to assure that no hazard exists either at the plant site, or in the surrounding area.

liquid and gaseous wastes

The handling of liquid and gaseous wastes depends upon the magnitude and type of radioactivity involved. Thus all liquid effluent flows first to underground stainless-steel surge tanks, surrounded by a concrete enclosure, where it is monitored for radioactivity. From here on, it may go through any one or more of several steps. If the type of radioactivity is such that it has a relatively short life, the liquid merely may be held in the tanks until the radioactivity decays. Or, if not, it may be passed through demineralizers and a gas stripper. Or, if the liquid is already within permissible limits, it may be mixed with the condenser cooling stream for discharge into the Ohio River.

Mixed-bed demineralizers remove all soluble radioactive impurities as well as particles. Any dissolved fission gases are removed in the steam stripper, and are stored in steel tanks until they are safe to discharge to the environment. After the liquid is processed in this manner, it is delivered to test tanks, where it is again sampled to assure that when mixed with discharge water it will fall within safe limits.

Both liquids and gases are monitored before and after release to the environment, and at various test stations outside the plant site to assure safe disposal.

Liquid wastes containing a high solids content, such as fluids used for decontaminating equipment and facilities, are processed through a vapor compression evaporator. The distillate from the evaporator is sent to the liquid surge tanks and the concentrate will be mixed with cement and drummed for burial at sea.

solid wastes

The solid wastes from the PWR plant can be disposed of in one of three ways, depending on their nature. They can be burned or stored underground at the site, or removed from the site and buried at sea.

Contaminated wastes that are combustible—paper, rags, clothing, etc.—are burned in an incinerator. Gases from the incinerator are scrubbed and filtered to remove particles: these particles and the waste water are then delivered to the resin storage tank.

Spent resins from the demineralizer are delivered to permanent underground storage tanks. These tanks are stainless steel, surrounded by a waterproof concrete enclosure. Here the solids are allowed to settle out, then the liquid is removed for further processing, if necessary, and ultimate disposal. The tanks have been sized to contain the volume of depleted resin expected to accumulate in five years at the maximum resin consumption rate.

During this five-year period, the resin in the tanks will be sampled and monitored. After the fission products contained on the resin have decayed to the point where the resin can be safely handled, it will be removed from the tanks and handled as other noncombustible wastes.

Noncombustible solid wastes—tools, small pieces of equipment, etc.—are sealed in metal drums. These drums are then placed in larger drums, the area in between filled with concrete for shielding, and the whole container shipped to the coast for disposal at sea.

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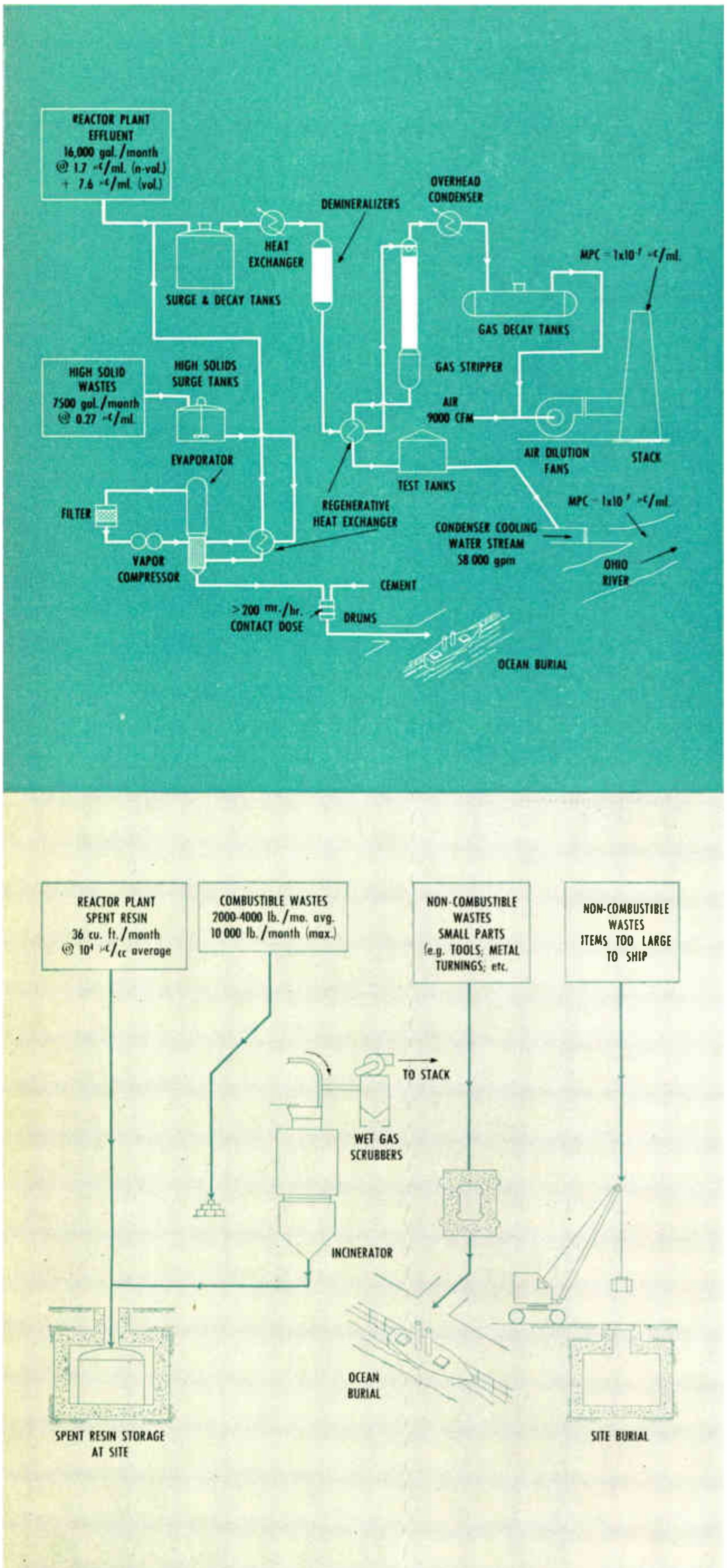


At top, a process flow diagram of waste-disposal facilities for liquid and gaseous wastes. **At bottom,** a flow diagram for solid wastes.

control of the process

System—Provisions are made to sample the waste streams as they enter the waste-disposal system, before and after each waste-treatment process, and before discharge from the plant. Sampling facilities are also provided so that diluted waste streams, both liquid and gaseous, can be monitored after dilution. The system provides flexibility in that any waste stream can be reprocessed if necessary to reduce the radioactivity to a level where subsequent dilution will yield a stream at or below the maximum permissible concentration chosen for discharge from the plant. This flexibility, along with the basic batch type of processing in the system and the large number of sample locations will minimize the possibility of discharging plant wastes that are "over-tolerance."

In addition to the numerous sampling provisions provided for the system, effluent activity monitors are installed on the main liquid effluent header and on the stack. These monitors will detect and inform the operator of accidental discharge of either liquid or gaseous wastes which are over-tolerance.

Off Site—A site-monitoring program was started early in 1956 and will continue during operation of the plant. The purpose of the pre-operational phase of this program was to determine the types and amounts of radioactive materials that occur naturally in the environment around the PWR plant site and to determine the variations in the amounts of these materials over a period of approximately 1½ years prior to start-up of the nuclear portion of the plant. This will allow an evaluation of the environmental radioactivity so that similar studies conducted after plant operation will ensure that the operation of the nuclear plant meets allowable tolerances. Evaluations are being made on (1) soil in the general vicinity of the plant, (2) the Ohio River water both above and below the site, (3) well water within a one-mile radius of the site, (4) vegetation in this general area, and (5) the air in the area.

River water analyses are being obtained from two locations above and one location below the site. Another monitoring station is currently being installed at the "first point of use" below the plant, which is the water-intake service to the city of Midland, Pa.

To properly evaluate the presence of radioactive dusts and gases in the air, and to measure the normal background radiation in the vicinity, a group of five "mobile monitoring stations" are being operated. Four of these units are located at fixed stations while the fifth monitor will transverse the general area.

The four fixed locations were selected on the basis of wind directions and velocities that have been determined by the U.S. Weather Bureau over the past year. The stations are placed so as to indicate as accurately as possible the concentrations of radioactivity in the air both downwind and upwind of the plant site.

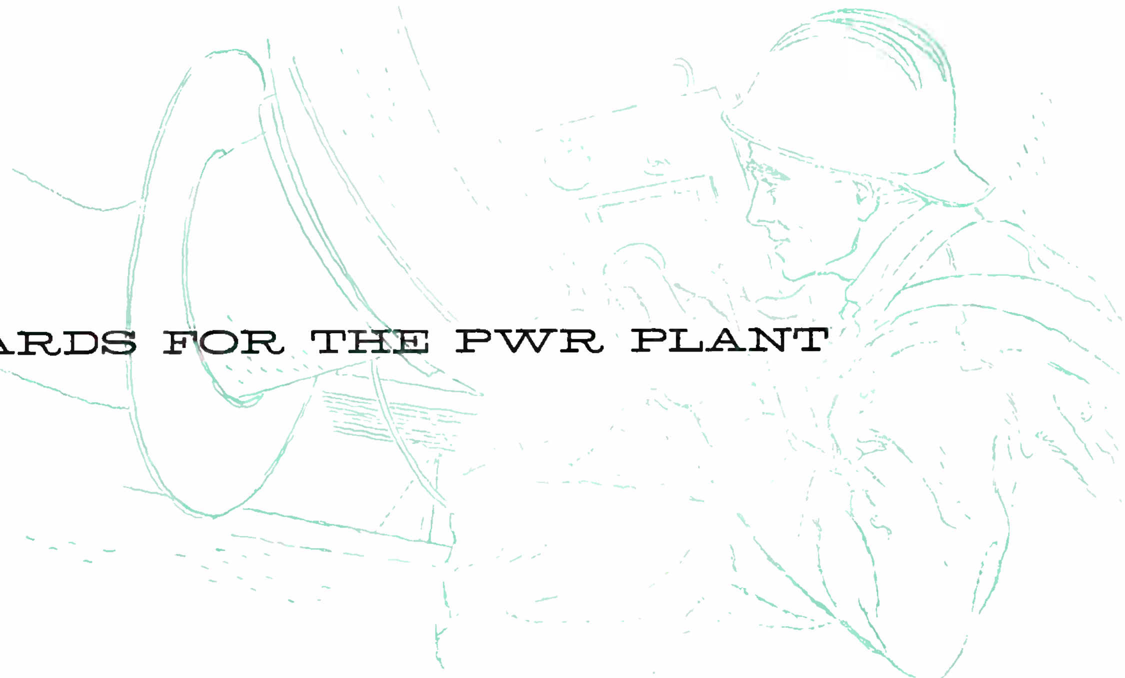
The fifth station will be moved around to different locations to determine the relationships between the four fixed stations and other areas that may be of interest.

Currently mud samples are being taken from the river bottom from pools behind dams near the plant site. Samples of river algae are also being obtained. Data from these samples will be compared to data from similar samples after radioactive wastes from the plant are being discharged into the river. This will determine how these wastes are distributed in the river.

The results of these and other tests will be of immeasurable value in determining whether the waste-disposal facilities are satisfactory or whether modifications should be made. ■

World Radio History

SAFEGUARDS FOR THE PWR PLANT



▪ Few things in history have been designed and constructed with as stringent and elaborate safety precautions as has the PWR plant. In fact, by normal safety standards, this plant is undoubtedly overdesigned—and deliberately so. Obviously, a prime consideration is to prevent off-site radiation hazards. In any training, operation, or testing at the station, this will be a prime requirement.

The safeguards mentioned here are all over and above the normal precautions taken to protect personnel and equipment. A secondary reason for the extensive safeguards is to provide the information necessary to evaluate the protection necessary in future plants.

Broadly speaking, a nuclear source of heat introduces a potential hazard not found in conventional utility plants. This is the possible biological hazard to people off the site from the accidental release of radioactive particles or fission products. Release of fission products can be prevented by proper design and construction of the plant.

The PWR plant will not present any hazard outside the plant site as a result of any type of plant explosion—physical, chemical, or nuclear. Furthermore, the PWR plant is designed so that no plausible sequence of events could cause release of hazardous quantities of fission products beyond the plant-site boundary.

These statements do not ignore the possibility of accidents, either through human error or mechanical failure; in fact, they take into account the worst possible chain of events that could plausibly occur.

operational safety

Despite the extreme care taken in design, manufacture, and installation of the components and systems incorporated in the PWR plant, their failure-proof operation is not assumed. Elaborate instrumentation and control will provide ample warning of any trouble.

However, any fission products, to reach the outer atmosphere, would have to pass through three consecutive barriers built into the plant. Fission products are developed in the fuel itself; the cladding or container for this fuel constitutes the first barrier. Should any fission products get through this first barrier, they would circulate in the primary coolant system—but this is an all-welded system designed for zero leakage. The sealed primary system thus constitutes a second effective barrier. The third barrier is the plant container, a group of interconnected, vapor-tight steel pressure vessels.

Despite the inherent safety furnished by these barriers, no stone was left unturned in providing for every plausible accident. While all the possibilities cannot be considered here, a few examples may illustrate the nature of the planning necessary in developing a nuclear plant. Two of the most serious accidents that could conceivably occur are loss of coolant in the primary system, and reactivity accidents. Consider these possibilities briefly.

loss of coolant

Based on existing experience with pressurized systems of many types—and at much higher pressures and temperatures—and because of the care exercised in manufacture of components, a major rupture of either piping or vessels is extremely unlikely. However, since a major rupture might offer a potential hazard to the surrounding area, this eventuality was studied in considerable detail.

The exact sequence of events that would follow a rupture depends upon its size and location. However, for any rupture large enough to be serious, the water level in the pressurizer, as well as the reactor plant pressure, would immediately begin to fall. An alarm would sound either when the pressurizer water level reached the low-level alarm point, or when the pressure dropped to 1850 psia, whichever condition occurred first. The operator then would initiate an emergency procedure. When pressure reached 500 psia, the safety-injection system would start (see p. 60). If the rupture were above the level of the core, the injection system would cover the core in a short time, preventing any significant melting of the core or release of fission products. If, however, a major rupture occurred below the core, the safety-injection system would take some time to re-cover the core. This would allow some melting, releasing some fission products within the plant container. In either event, after cooling water reached the top of the core, it would be supplied continuously until either the leak was

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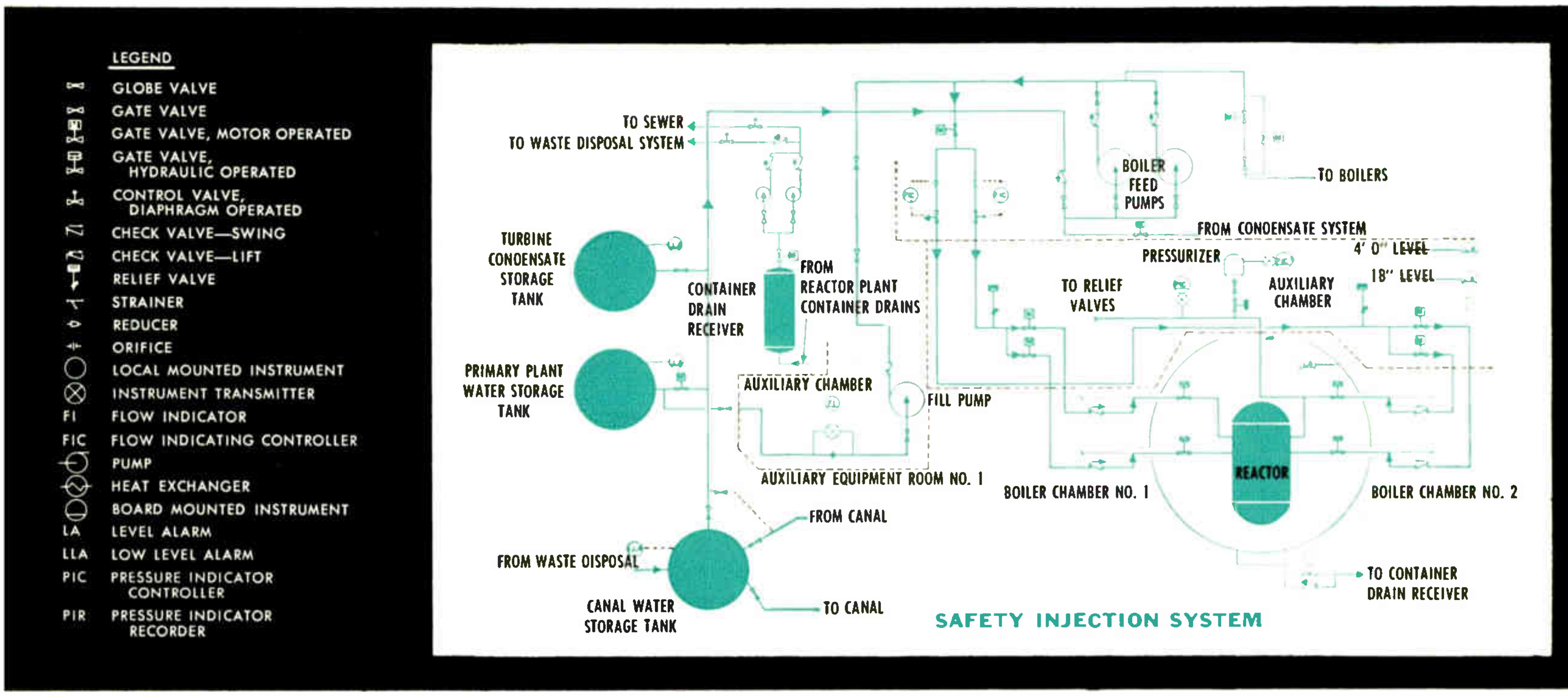


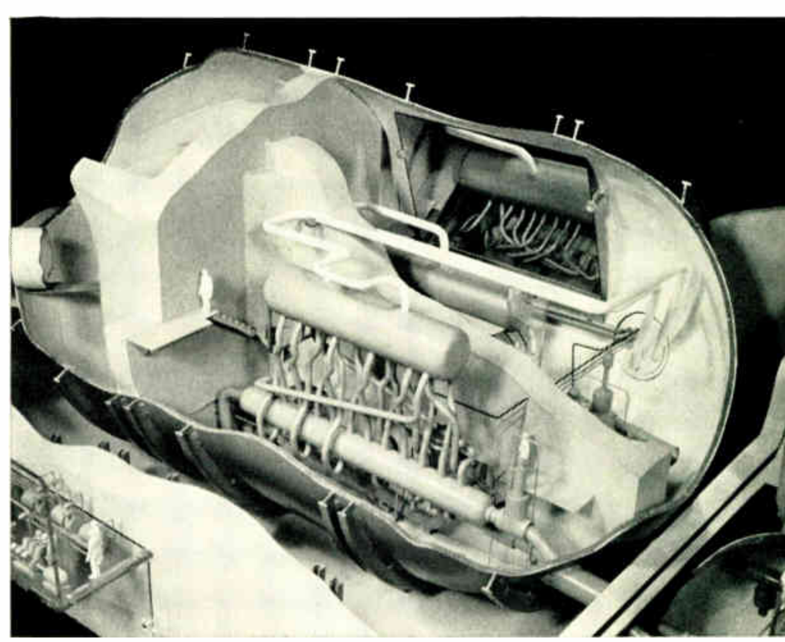
isolated or decay heat was reduced sufficiently to prevent melting of fuel elements.

Certain conclusions must be drawn before a loss-of-coolant accident can be completely analyzed. Among the more important of these for the PWR plant are:

Can brittle fractures occur in any part of the primary-coolant system? Brittle fracture in any of the major piping or equipment acting as the primary pressure boundary for hot reactor coolant is not considered a feasible accident. The pipe and most equipment enclosing the reactor coolant is austenitic steel, which is not subject to brittle failure at or above normal plant ambient temperatures. The reactor, pressurizer and one type of boiler are of ferritic steel, but will never be under pressure at a temperature at which their ductility is so low as to be in the brittle range.

Can sufficient radiation damage occur to components to cause serious embrittlement? Nowhere in the PWR plant are pressure-containing walls subjected to sufficient radiation fields to reduce ductility appreciably. The reactor vessel itself is the only component exposed to significant neutron flux; however, the relatively high operating temperature causes a continuous annealing action, which reduces the amount of radiation damage to a minimum.

How large a rupture can feasibly occur? Because of the un-



Right, a cutaway model of one vapor container. Note that two loops are isolated from each other by a concrete shield.

Center, a cutaway sketch showing three of the plant containers and the fuel handling canal.

Far right, curves showing the minimum times to uncover core for various pipe break sizes.

World Radio History

usual care taken in its design and fabrication, a major rupture of the reactor vessel is considered implausible. All experience with steam piping indicates that major failure is highly improbable, and perhaps not even feasible. While experience shows that sometimes cracks occur in stainless-steel piping, such as used here, in most cases these failures have occurred at much higher temperatures and under more severe service. No record of such cracks causing a major rupture has been found. Small piping, on the basis of all industrial experience, is the most apt to rupture.

However, to assure a conservative design, the assumption was made that it was possible for a major coolant pipe (15-inch inside diameter) to develop a longitudinal split of an area equivalent to twice the cross-sectional area of the pipe. This is considered the worst feasible case of rupture.

How will pipes rupture? Studies of industrial pipe failure indicate that a sudden circumferential shearing of the main pipe is not a feasible accident. For this and other reasons, the longitudinal split is considered to be the possible form of piping rupture.

What other accidents, if any, could cause overstressing of the pressure boundary of the reactor plant, which might result in a loss-of-coolant accident? No feasible accident can push the pressure beyond design limits. Even if the turbine throttle tripped when the plant was producing 100 mw of electric power—and the reactor failed to scram—the reactor plant pressure-relief valves would discharge coolant fast enough to prevent pressure from going above the 2500-psi design pressure.

Will electric power be available during the accident, and will the safety-injection system work? Studies show that a loss-of-coolant accident will not result in a complete loss of power to the station, and that a loss-of-power accident will not cause a loss-of-coolant accident. Loss of coolant and loss of all power *simultaneously* from unrelated causes is not believed to be a plausible accident. Duquesne Light Company records show that complete loss of power to a station happens once in 25 years. For these and other reasons, electric power is used to operate the safety-injection system pumps, and this system will be available during the unlikely occurrence of loss-of-coolant accidents.

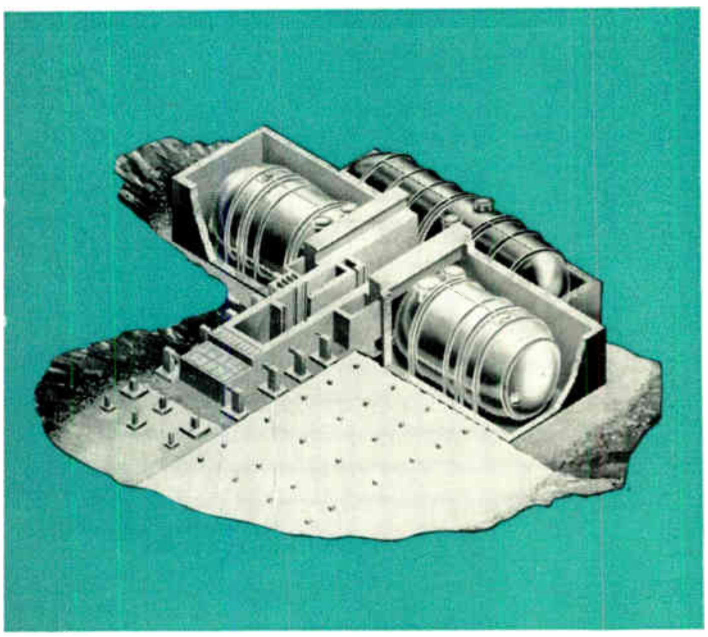
Is the reactor plant container necessary, i.e., will a loss-of-coolant accident be severe enough to melt the core and release significant quantities of fission products? This question is important not only from a safety standpoint but also from a cost viewpoint, since the container is an expensive structure. The decision to include a plant container was actually made before the detailed design of the PWR plant was started. Subsequent investigations raise considerable doubt as to the necessity for the container from a safety standpoint. The container would serve a safety function only for a loss-of-coolant accident, and then only for the worst possible case. However, in this first plant, ultraconservatism was practiced extensively, and the plant container was included.

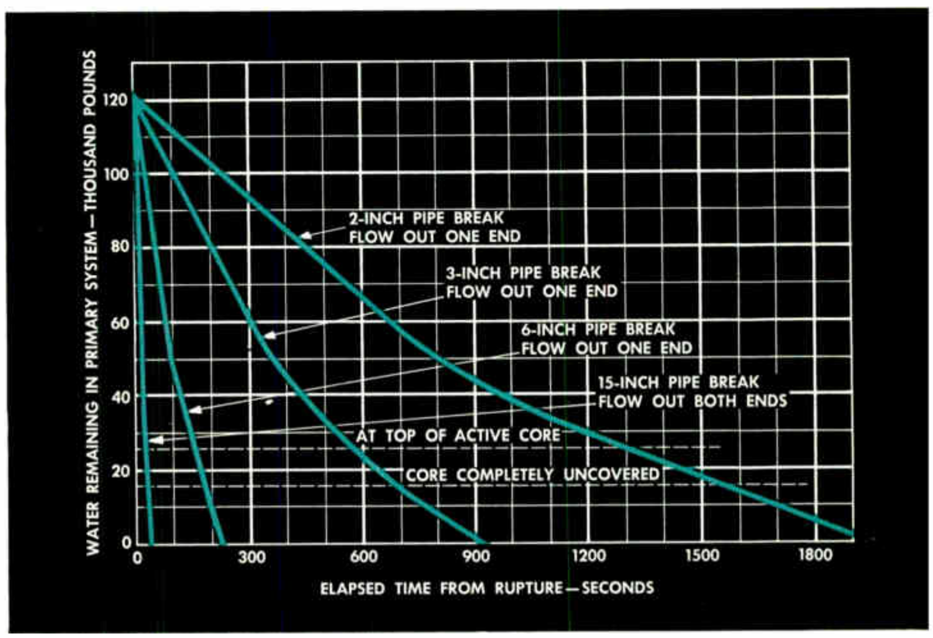
Based on these conclusions, every plausible loss-of-coolant accident was fully explored. Sequences of events, interactions within the plant, and all the various other factors were explored in detail and with painstaking care.

As a result, certain conclusions can be drawn. In the event of a loss-of-coolant accident, if the core does not melt, the release of 100 percent of the reactor coolant directly to the outside atmosphere would not result in a biological hazard at the plant-site boundary. And this conclusion holds true even if the core had been operated for 3000 hours at 270 megawatts and the coolant contained the maximum radioactivity that could be caused by 1000 defected blanket-fuel elements. This, incidentally, is the maximum number of blanket elements expected to become defective during the lifetime of the core. This conclusion, of course, assumes complete release of coolant to the atmosphere, an unfeasible accident because of the plant container. Should the worst possible accident occur, which includes some core melt-down and fission-product release, the plant container is fully capable of containing such products, and no hazard would exist at plant site boundary.

reactivity accidents

A reactivity accident is one in which more reactivity is inserted in the core than is needed for a normal rise in power. In such a case, reactor power level would increase faster than desired, and, if the rise were not stopped, local heating of the fuel elements might occur with resultant melting of the clad-





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ding and release of fission products to the primary coolant. Before considering the different types of reactivity accidents and their results, one fact should be made clear. *Even if all protection circuits failed, the reactor would shut down in any conceivable reactivity accident.* Steam would form in the coolant channels, and since steam is not an effective neutron moderator, the fission rate would decrease. The formation of steam might result in local heating and perhaps damage the core, but no radioactive products would find their way out of the primary-coolant system. Consider, then, three ways in which excess reactivity could be inserted in the core, and the handling of each situation.

Excessive Rod Withdrawal—This accident assumes that rod withdrawal, once initiated, continues beyond the desired point. This is most likely to occur at start-up. Several factors guard against this type of accident. One automatic protective device is the start-up rate limit, which overrides all withdrawal signals; if the rate of change of neutron flux exceeds an established figure, this control inserts the rods.

A backup protection to this control is a scram control that automatically inserts the rods if power goes much beyond normal operating levels. At start-up, pumps are normally operated at half speed; therefore, one scram action is set to be effective at 70 percent power. Another scram device is set at 140 percent power, and is independent of the number or speed of the pumps.

The maximum rates of rod withdrawal possible are also a limitation on excessive rod withdrawal. A relatively long time is required to insert sufficient reactivity to cause damage, therefore the possibility of the accident occurring without cognizance of the operator is slight. The operator, except during start-up, does not need to adjust control rods frequently. Normal load changes are taken care of by the negative temperature coefficient of reactivity. Also, about two seconds of rod motion are required for every degree of temperature rise, and the normal change effected by an operator is but a few degrees. Thus any withdrawal accident would require rod motion for an out-of-the-ordinary length of time, a situation not likely to occur by operator action.

Excessive Heat Withdrawal by Steam-Plant Valves—Several protective systems prevent hazards in the event that any steam-plant valves suddenly withdraw an excessive amount of steam from the system. For example, if the plant is operating normally and one of the controlled steam-relief valves should open, an automatic system that monitors the ratio of power to flow cuts back the reactor power output. As backup to this, two scram signals, set to operate at different power levels, would shut down the reactor before damage could occur. However, in general, steam-valve malfunctions with the reactor at power are of little consequence, since the heat-transfer capabilities of the boiler prevent any serious overpower.

During start-up, when the reactor has not reached sufficient power for control by the negative temperature coefficient, four separate control features provide protection. The first control inserts the rods at a normal speed if neutron flux increases faster than a predetermined rate; this control, of course, overrides any outward motion of the rods by the operator. As backup to this protection system, the same power-to-flow control mentioned above cuts back reactor power if this ratio is not proper in the system. If the power-to-flow ratio departs substantially from the proper figure, another control scrams the reactor. The final, or fourth, control system scrams the reactor if the nuclear level reaches a preset point. Note that two of these controls act at normal speeds to cut back power, the other two act at rapid speed to scram the reactor.

Cold-Water Accidents—If a sizable quantity of cold water were suddenly inserted in the reactor, the negative temperature coefficient would cause an undesired increase in reactor power output. The only source of sufficiently cold water to cause any difficulty is a loop that is out of service. Such an accident is prevented by two independent protection systems. One device measures water temperature in the out-of-service loop and compares it with water temperature in the hot loops. If the difference is greater than a preset value, a relay fails to energize and the pump cannot be started. The second protective device is an interlock on a circuit breaker in the rod-control system, which prevents opening of the main coolant valves of an out-of-service loop when that circuit breaker is closed. Thus once the valves to any loop are closed, that loop cannot be put into service again while the reactor is critical.

potential hazards from waste disposal

The same "worst case" approach was used in analyzing potential hazards from the waste-disposal plant as was applied to the primary-plant system. Consider two extreme cases, one involving release of gaseous wastes, and the other liquid wastes.

Suppose, for example, that one gas storage tank suddenly ruptured and released its entire contents in one "puff." Assume also that this tank held the maximum amount of radioactivity expected—i.e., all the radioactive gas contained in the reactor coolant with 1000 defected fuel rods, and after the reactor had operated 3000 hours at full power. These, of course, are extremely pessimistic assumptions.

Under the above conditions, and with average weather conditions, the dosage received by a person due to this cloud would be extremely small: 7×10^{-2} roentgens at 100 meters (about 325 feet) and decreasing rapidly at greater distances. Even under bad weather conditions, i.e., with a large inversion, the dose would be only 3 roentgens at 100 meters, again decreasing rapidly with distance. Considering the severity and improbability of these conditions and the relatively small dosages received, no significant radiation hazard would exist from this source.

In the case of liquid waste, what would happen if an operator violated established operating procedures and emptied the contents of one surge and decay tank directly into the river? Add a few more pessimistic conditions: the quantity of the liquid is equal to that contained in the reactor plant, its reactivity is the maximum expected in the reactor coolant, the liquid is not processed through any other waste-disposal equipment, and it has been allowed to decay for only one day instead of the usual 45 days. For these highly pessimistic and improbably conditions, the radioactivity of the liquid *at the point of discharge* into the river would be about 1.8 times the maximum permissible concentration. However, even in this event, the waste would be diluted to well below tolerance level before it reached the first point of use down the river (about a half mile).

Both of these situations are highly improbable, not only because so many operational faults and conditions are exaggerated, but also because of the safety devices and monitors used on the system. Thus the radioactive waste-disposal system presents no hazard to surrounding communities.

In addition to the operational protective systems, of course, safety is built into the plant at every stage of design and manufacture. Every element, every combination of elements undergoes exhaustive tests. Thus, even though the PWR plant is the first of its kind, safety is such an overriding factor in every element of the plant that no plausible accident could cause a biological hazard to off-site personnel. ■

World Radio History

PERSONNEL AND TRAINING

▪ While the operation of a nuclear power plant is probably no more difficult than a conventional station, two important facts about operating personnel stand out. Even the most highly skilled operating man is faced with problems he has never before encountered in a conventional station; and because the station is experimental and will be used to gather information as well as produce power, more people are required to operate this plant.

For these reasons, the Duquesne Light Company chose plant personnel for the Shippingport Station very carefully, and has provided extensive training, both in classrooms and in on-the-job facilities.

organization of staff

The selection and training of personnel to operate and maintain the Shippingport Station began in 1954. The staff includes sufficient people to operate and maintain all equipment, including the nuclear portion of the plant; to maintain all site property; to perform all the necessary industrial-hygiene services; to provide plant security; and to perform the necessary clerical services. The general organizational set-up is shown on p. 64. The top man at the station is the station superintendent; and in his absence, the chief engineer is responsible for the plant.

This staff is, of course, organized to serve a dual purpose. As well as operation and maintenance, it will have the task of testing and evaluating the plant. The total number of people planned for the station is about 135. About 26 of these people will be involved full time in testing and gathering information. Although the rest of the staff is primarily for operation and maintenance, some will also be required for testing purposes on a part-time basis.

After operating experience has been gained, this force can probably be reduced, to perhaps 81. This compares with 66 people for a conventional coal-burning plant. At this stage of the game, however, the manpower for the station has been planned on a conservative basis.

training of the staff

The operation and maintenance of the Shippingport Station requires people skilled in both generating-station and nuclear-plant operation. Such people simply were not available; therefore an elaborate training program was undertaken. In general, the approach was to take experienced utility men and train them in the nuclear aspects of the plant.

Several different training methods were used in this program. On-the-job training was conducted at various Atomic Energy Commission installations, and at the station itself. Formal classes were held on various aspects of the plant operation, and inspection trips to equipment test and assembly facilities were conducted. The exact nature and duration of the training varied, of course, with the individual's function in the nuclear station.

On the job training was conducted at the Naval Reactor Facility and the Materials Testing Reactor at Arco, Idaho, at the Savannah River Plant, and at the Bettis Plant.

A total of 48 station employes received 171 man-months of training at the NRF installation in Idaho, which is also used to train personnel for the Navy's nuclear program. Here the personnel obtained on-the-job experience in reactor-plant chemistry, health physics, maintenance, operation, instrumentation and control, and testing. In many phases of this training, the personnel were integrated with the operating personnel at the facility, where they could gain actual operating experience.

The Materials Testing Reactor was used largely to train a few employees in health physics and chemistry aspects of reactor operation. The nuclear instrumentation and control engineers had about six months on-the-job experience at Savannah River.

At the Bettis Plant, both formal and on-the-job training was undertaken. Seven senior members of the operating staff were assigned to the Bettis Plant for a year. Included in their assignments were working on the reactor, shielding, and fluid systems designs; plant analysis; and operation of the Bettis Test Facility and the PWR critical experiment. The station chemists, health physicist, and reactor engineers also received training at Bettis.

On-the-job training is also being conducted at the Shippingport Station itself. This consists of an orientation program, a series of lectures, and actual practice in operating various parts of the plant during preliminary operation.

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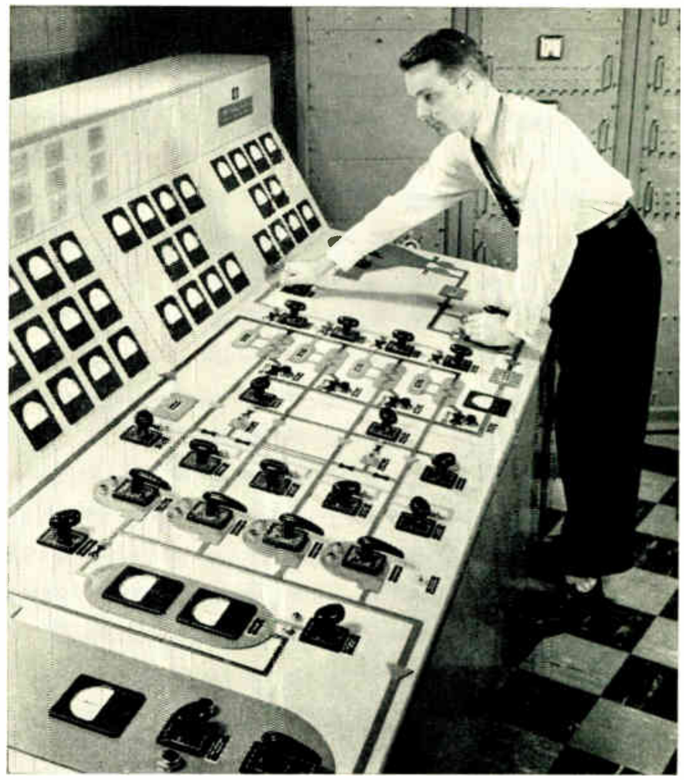
An interesting training aid is a simulator that contains the reactor and turbine control consoles in abbreviated form (see photo). The design of the simulator is based on the parameters of the stations, and the functions of the systems and components.

Operators use this board in much the same fashion as they would the actual control console. However, signals from the console are fed to a computer, which returns operating data and other indications to the console instruments. Thus, the operation of the plant can be simulated realistically.

Formal training was largely conducted by professors recruited from local college faculties and was intended to give certain personnel a working background in specific subjects, such as nuclear physics, applied electronics, and reactor technology.

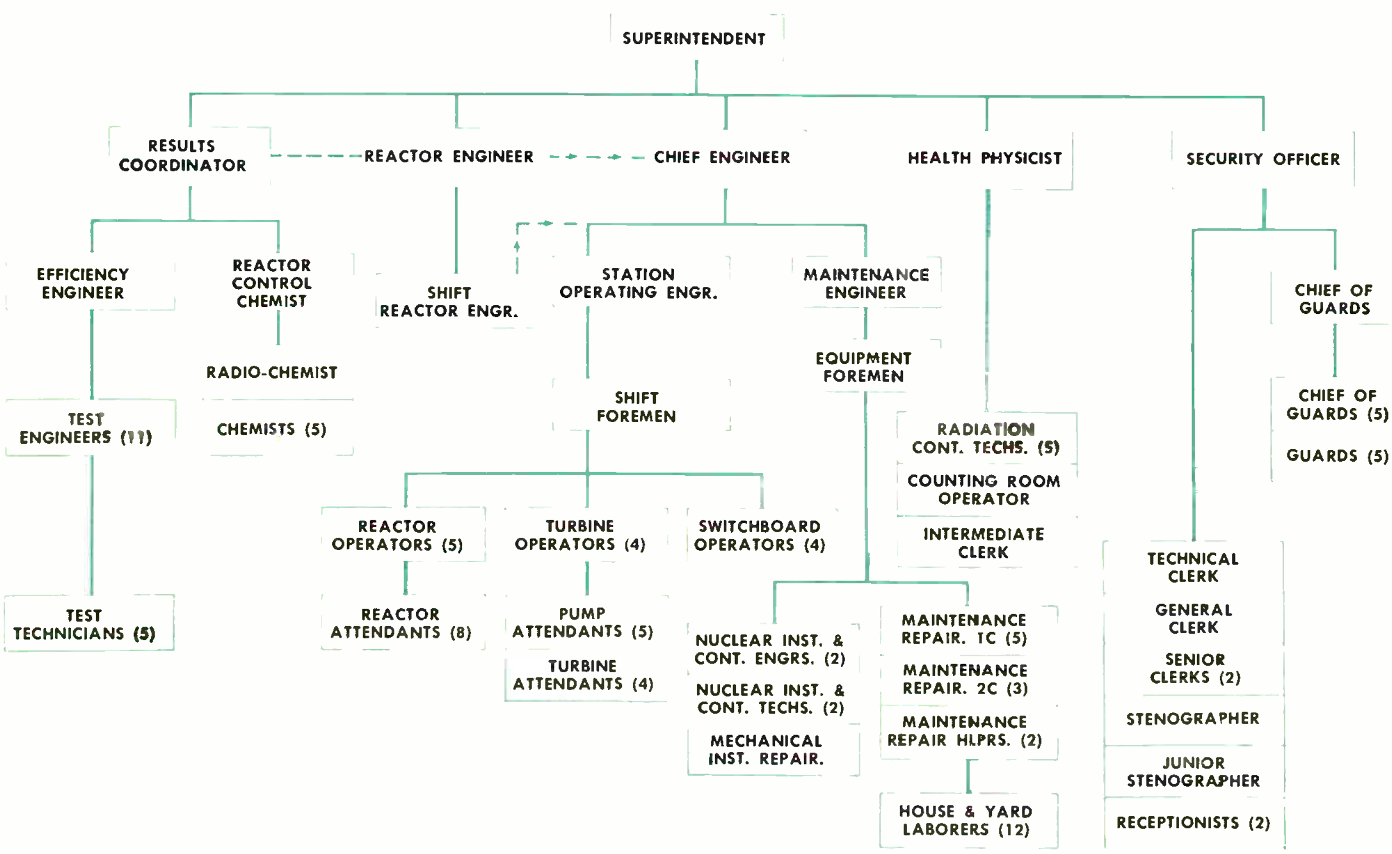
In addition to these forms of training, selected personnel were also taken on inspection trips to manufacturing plants to observe such things as the manufacture and testing of instrumentation and control, fuel elements, rod-control mechanisms, and core-handling equipment.

Through such training, the Shippingport personnel have received extensive training in the functions, operations, and maintenance of the station they will run. As compared to a conventional station, both the numbers of people and their training is considerably greater. As yet it is still far too early to tell the number of people that will ultimately be needed in other nuclear plants. However, the operation of this station will provide much needed information as to the requirements of future stations. ■



The control simulator used to train Shippingport personnel.

SHIPPINGPORT ATOMIC POWER STATION ORGANIZATION



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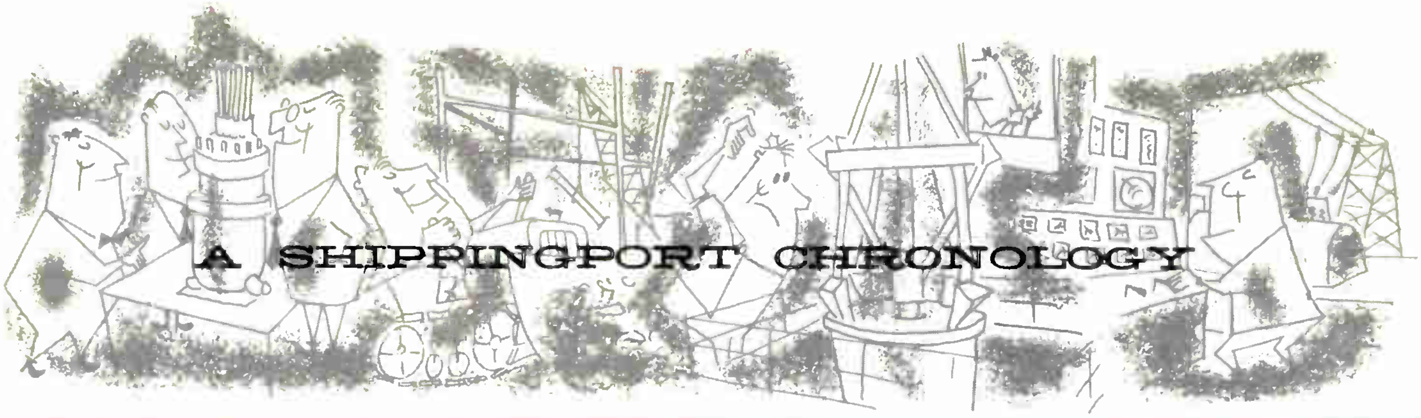

A SHIPPINGPORT CHRONOLOGY

JULY 1953	PWR design contract awarded to Westinghouse by the AEC.
JANUARY 1954	Preliminary design of reactor selected. Duquesne Light Company's proposal selected.
MARCH 1954	Shippingport plant site announced.
MAY 1954	Procurement of major components, started (steam generator).
SEPTEMBER 1954	Ground breaking ceremony by President Eisenhower.
JANUARY 1955	Seed-and-blanket fuel concept selected.
APRIL 1955	Selection of blanket fuel (UO_2).
AUGUST 1955	First concrete poured at site.
DECEMBER 1955	Core manufacture started.
MAY 1956	First major non-nuclear component, the feedwater heaters, installed.
JULY 1956	Waste-disposal construction started.
AUGUST 1956	First major nuclear component, a steam generator, installed.
OCTOBER 1956	Initial welding of primary system piping. Reactor vessel installed.
NOVEMBER 1956	Number of construction personnel reaches its peak (1779).
DECEMBER 1956	First pump volutes installed.
JUNE 1957	Primary loop piping (18 inch) completed.
AUGUST 1957	Primary system underwent hydrostatic testing (3750 psi).
SEPTEMBER 1957	Hot flush of primary loop at 525 degrees F with filter.
OCTOBER 1957	Turbine-generator installation completed. Nuclear core inserted. Initial fill of reactor vessel.
DECEMBER 1957	Critical operation. Power operation.

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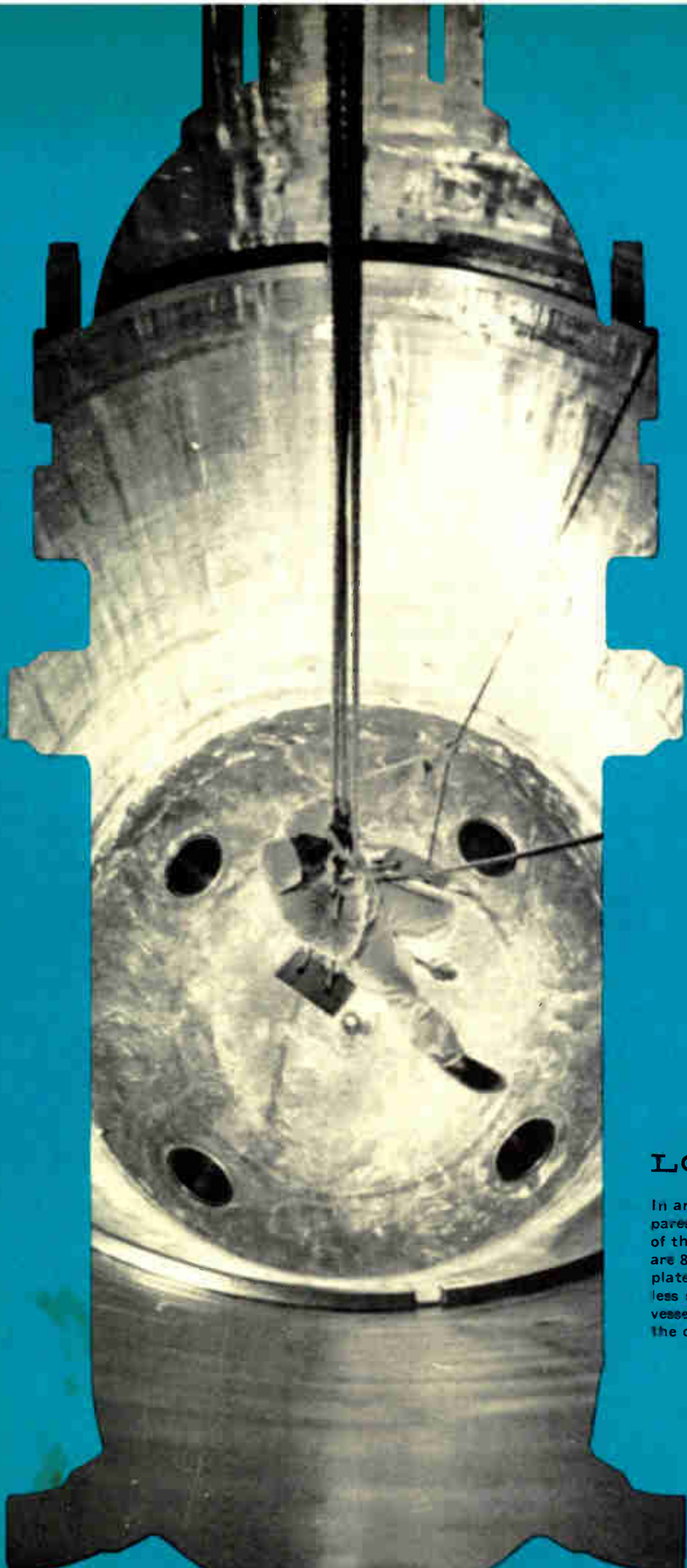

LOWER AWAY!

In an improvised bosun's chair, an inspector prepares to examine the shiny, stainless steel interior of the pressure vessel for Shippingport. The walls are 8½ inches thick and are made of carbon steel plates and forgings with a one-quarter-inch stainless steel cladding. Shown at the bottom of the vessel are the inlet nozzles and near the top are the outlets.

World Radio History