Chapter 4
Alternative Reactor Systems

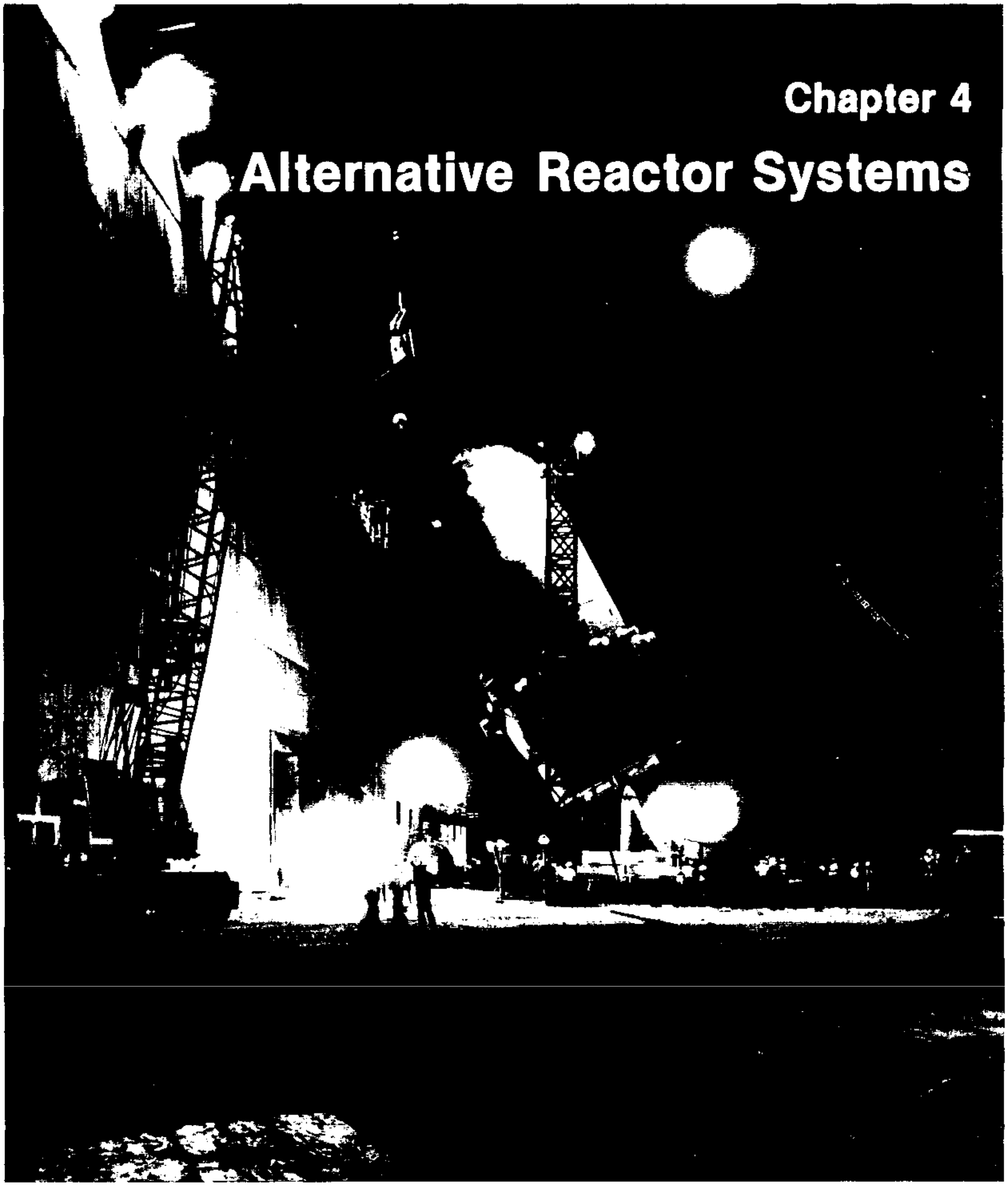

Photo credit: Tennessee ***Valley*** *Authority*

Boiling water reactor vessel being hoisted into a containment building at Browns Ferry nuclear plant

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Chapter 4
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INTRODUCTION

Nuclear power in the United States achieved some remarkable successes in its early years and experienced dramatic growth in the late 1960's and early 1970's. While this rapid growth was seen as a measure of the success of the technology, in retrospect it may have been detrimental. As discussed in chapter **5,** the size and complexity of reactors increased rapidly and there was little opportunity to apply the experience gained from older plants to the design of newer ones. In addition, the regulatory framework was incomplete when many of the plants were designed. As new regulations were formulated, the designs had to be adjusted retroactively to accommodate to changing criteria. With the rush of construction in the mid-l 970's, it was difficult to fully integrate these new requirements into the original designs; hence some portions of the reactor designs emerged as a patchwork of nonintegrated and often ill-understood pieces.

Several changes in design requirements have had far-reaching effects in today's reactors, even though they were not originally expected to have such an impact. For example, new criteria on fire protection in nuclear powerplants have spawned new features and systems to prevent, contain, and mitigate fires. This led to greater separation of safety systems, changes in cable-tray design, requirements for more fire-resistant materials, and changes in civil structures to prevent the spread of fires. Clearly, these modifications can have ramifications for other plant systems. Other regulatory actions concerning seismic design, decay heat removal systems, and protection of safety systems from other equipment failures have also had extensive impacts.

A fresh look at the design of light water reactors (LWRs) could be useful if it more fully integrated the cumulative changes of the past and reexamined the criteria that stimulated those changes. In addition, a new design could incorporate analytical techniques and knowledge that have been acquired since the original designs first were formulated. In fact, it could be beneficial to investigate designs of alternative reactors that have different and potentially desirable characteristics. It is possible that a new design could improve safety and reliability at an acceptable cost and within a reasonable timeframe.

This provides the basic technical reason for reevaluating current nuclear technology as embodied in LWRs. It is important to question, however, the justification for actual changes to the current system. Are there any indications that the current generation of LWR is less than adequately safe or reliable? The public appears to be increasingly skeptical that nuclear reactors are good neighbors. As discussed in chapter 8, more than half of those polled expressed the belief that reactors are dangerous. The same percentage of the public opposes the construction of new plants. While this is not an absolute measure of the adequacy of today's reactors, it does reflect a growing concern for their safety.

The nuclear utilities also have assessed the current reactors in view of their special needs and interests. While they do not believe that LWRs are seriously flawed, the utilities have expressed a desire for changes that would make plants easier to operate and maintain and less susceptible to economically damaging accidents (1 3). Some movement has already been initiated within the nuclear industry in response to utility needs. Most of these efforts focus on evolutionary changes to the current designs and thus represent normal development of LWR technology.

The increasing levels of concern for safety among the public and the utilities has contributed to an interest in safety features that are inherent to the design of the reactor rather than systems which rely on equipment and operators to function properly. The emphasis on inherent safety is reflected to some extent in evolutionary designs for **LWRs, and to a much greater degree** in innovative designs of alternative reactors. In this chapter, LWRs as well as several proposed alternatives will be examined and their relative advantages and disadvantages assessed.

SOME BASICS IN NUCLEAR POWERPLANT DESIGN

To assess the safety and reliability of current reactors and compare them with alternative designs, it is important to understand the basic principles involved in generating power with nuclear technology. At the center of every nuclear reactor is the core, which is composed of nuclear fuel. Only a few materials, such as uranium and plutonium, are suitable fuels. When a neutron strikes an atom of fuel, it can be absorbed. This could cause the nucleus of the heavy atom to become unstable and split into two lighter atoms known as fission products. When this occurs, energy in the form of heat is released along with two or three neutrons. The neutrons then strike other atoms of fuel and cause additional fissions. With careful design, the fissioning can be made to continue in a process known as a chain reaction.

A chain reaction can be sustained best in uranium fuels if the neutrons are slowed before they strike the fissionable materials. This is done by surrounding the fuel with a material known as a moderator that absorbs some of the energy of the neutrons as they are released from the fission process. Several different materials are suitable as moderators, including ordinary water, heavy water,* and graphite.

The heat from the fission process is removed from the core by a continuous stream of fluid called the primary coolant. The reactors examined in this chapter use water or helium as the coolant, although other fluids have been considered. The heat in the coolant can be used directly to produce electricity by driving a turbine-generator, or it can be transferred to another fluid medium and then to a turbine-generator. Both methods have been used effectively in U.S. nuclear powerplants.

There are many possible combinations of fuel, coolant, and moderator that can be used in the design of nuclear reactors. There are advantages and disadvantages associated with the various materials, and no single combination has emerged as being clearly superior to the others.

Several designs have been developed for producing electricity commercially. The most common reactors are known as light water reactors, which use ordinary water as both coolant and moderator. LWR fuel is slightly enriched uranium, in which the percentage of fissionable material has been increased from its naturally occurring value of **0.7 percent to about 3 or 4 percent. After enrichment,** the fuel is shaped into ceramic pellets of uranium dioxide and encased in long, thin fuel rods made of a zirconium alloy.

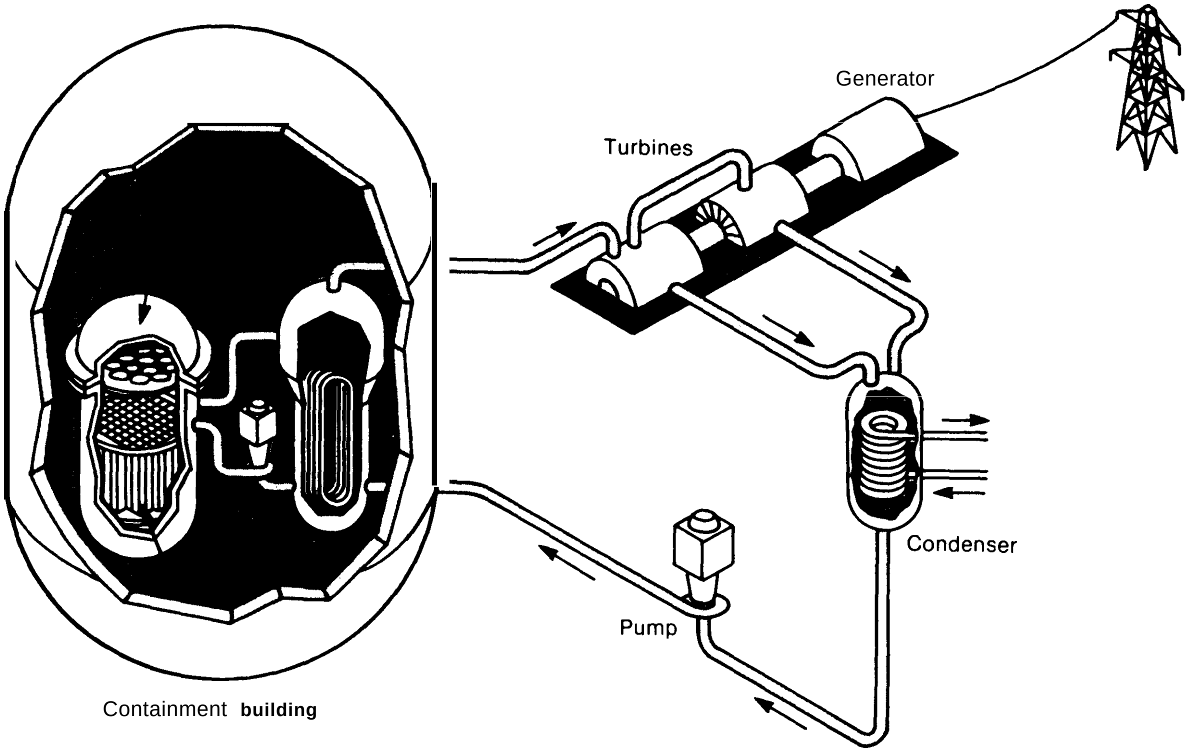
Another commercially feasible reactor is the heavy water reactor (HWR), which is moderated by heavy water and cooled by ordinary water. The fuel in an HWR is similar in form and composition to LWR fuel, but it need not be enriched to sustain a chain reaction. Another design is the high temperature gas-cooled reactor (HTGR), which uses helium as a coolant and graphite as a moderator. The HTGR can use uranium as a fuel, but it usually is enriched to a greater concentration of fissionable material than found in LWR fuel. The fuel form is very different from LWR and HWR fuel, with the uranium shaped into small coated spheres, mixed with graphite to form fuel rods, and then inserted into hexagonal graphite blocks.

In addition to selecting a fuel, moderator, and coolant, reactor designers also must devise a means to transfer the heat from the core to the turbines. In the United States, two different steam cycles have been developed for LWRs. The pressurized water reactor (PWR) shown in figure 19 maintains its primary coolant under pressure so that it will not boil. The heat from the primary system is transferred to a secondary circuit through a steam generator, and the steam produced there is used to drive a turbine.

The second type of LWR that is in commercial use is the boiling water reactor (BWR), shown in figure 20. It eliminates the secondary coolant circuit found in a PWR. In the BWR, the heat from the core boils the coolant directly, and the steam produced in the core drives the turbine. There

***A molecule of light water is made from one atom of oxygen and two atoms of the lightest isotope of hydrogen. By contrast, a molecule of heavy water is made with the isotope of hydrogen called deuterium, which has twice the mass.**

Figure 19.—Pressurized Water Reactor



SOURCE: "Nuclear Power from Fission Reactors," U.S. Department of Energy, March 1982

is no need for a heat exchanger, such as a steam generator, or for two coolant loops. **In addition, since** more energy is carried in steam than in water, the **BWR requires less circulation than the PWR.**

The two LWRs described above can be used to illustrate another crucial part of reactor design. Since nuclear reactors produce highly radioactive materials as byproducts of the fission process, it is essential that the design incorporates enough safety features to ensure the health and safety of the public. During normal operation, the radioactive materials are safely contained within the fuel rods and pose no threat to the public. The concern is that during an accident the fuel may become overheated to the point that it melts and releases the fission products that accumulate during normal operation.

Safety is designed into a nuclear reactor on several levels. First, every effort is made to prevent minor events from developing into major problems. This is accomplished in part by incorporating inherent features into the design to ensure stable and responsive operation. For example, the physics of the core dictates that most reactors will internally slow down the fission process in response to high coolant temperatures, and thus dampen the effects of problems in removing heat from the core. Both PWRs and BWRs have been designed to respond in this way.

Other features, known as engineered safety systems, operate in parallel with, or as a backup to, the inherent physical safety features. They are designed to ensure that the chain reaction is interrupted promptly if there is a problem in the plant and to remove heat from the core even under extreme circumstances. This is necessary because radioactive decay continues to produce heat long after the reactor has been shut down. If decay heat is not removed, the core can overheat to the point of melting. In the event that the shut-

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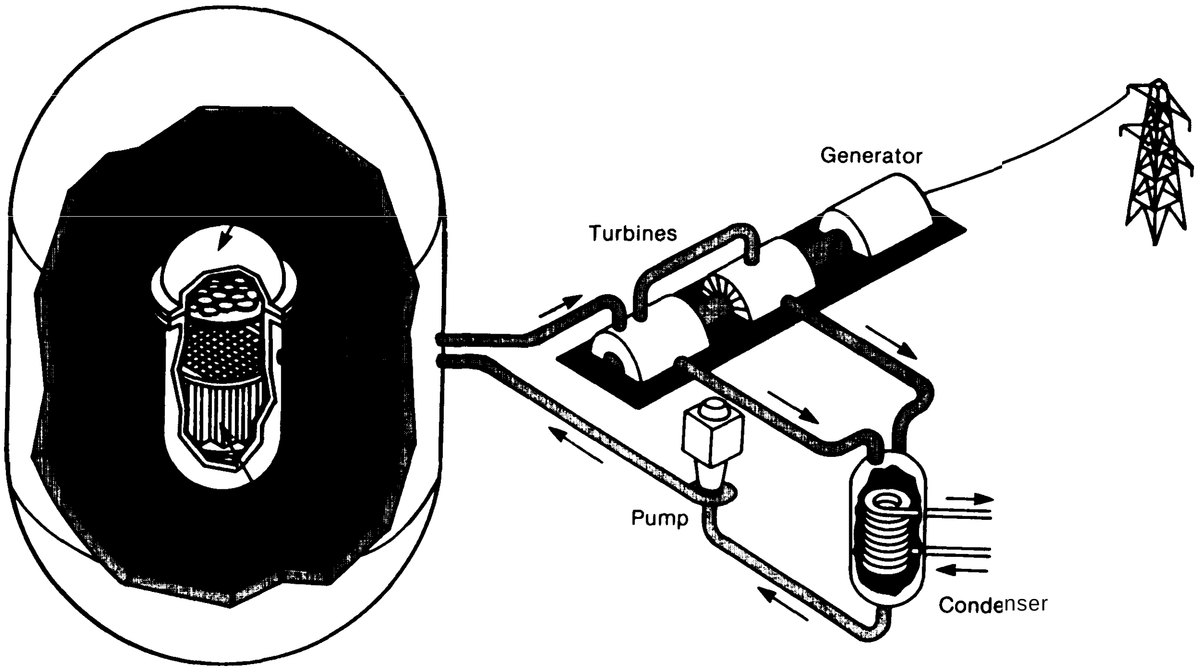
Figure 20.—Boiling Water Reactor



SOURCE: "Nuclear Power from Fission Reactors," U S. Department of Energy, March 1982.

down or decay heat removal systems fail, additional safety systems prevent the escape of fission products to the atmosphere.

Rapid interruption of the nuclear chain reaction is accomplished by inserting control rods which contain neutron-absorbing boron into the core. The control system is designed to shut down the reactor automatically in the event that abnormal conditions develop in the core or primary coolant system. Even after the chain reaction is interrupted, however, the coolant must continue to circulate to remove decay heat. If the coolant pressure drops in a BWR or PWR–indicating that some of the coolant has been lost from the primary system—the core is automatically flooded by an emergency core cooling system (ECCS). If the secondary cooling system fails in a PWR, an auxiliary feedwater system is designed to take over. Other backup cooling systems in these plants include high- and low-pressure injection pumps and spray systems. These safety systems are designed to operate automatically, with no requirement for action by the plant operators. They are dependent on human action only insofar as they must be designed, constructed, and maintained to function correctly.

The final step in the design for the safety of a nuclear powerplant is to incorporate features that prevent the release of fission products in the event of a fuel-melting accident. This is done using the concept of "defense in depth," that is, providing successive barriers that radioactive materials must breach before endangering the public. The barriers in LWRs are the fuel cladding, the heavy steel of the reactor pressure vessel, and the thick concrete of the containment building that encloses the pressure vessel and other components in the coolant system.

These examples necessarily oversimplify the complex designs and interactions of safety systems. Many safety systems play a role in the routine operation of the plant as well. This sampling serves as background for the subsequent discussions of safety features of LWRs and of alternative designs.

THE SAFETY AND RELIABILITY OF LIGHT WATER REACTORS

Overview of U.S. Reactors

Of the 84 nuclear reactors with operating licenses in the United States today, about two-thirds are pressurized water reactors. They are offered by three companies—Babcock & Wilcox Co., Combustion Engineering, Inc., and Westinghouse Electric Corp. The remaining reactors (with the exception of one HTGR) are boiling water reactors, sold by General Electric Co. These four companies all supply the nuclear steam supply system (NSSS), or the nuclear-related components of the reactor. The balance of the plant consists of such items as the turbine-generator, the auxiliary feedwater system, the control room, and the containment building. The balance of plant design typically is supplied by an architect-engineering (AE) firm, any one of which might team up with a vendor to provide a reactor plant that meets the needs of a particular utility at a specific site. So far, no completely standardized plant design has emerged, although some convergence has occurred among the designs of each nuclear steam system vendor. There is still a great deal of difference among the designs of similar components (e.g., steam generators) and system configurations. This is not surprising considering the various combinations of vendors and AE firms that have been involved in powerplant design. Furthermore, the **utilities themselves may customize a reactor design to meet specific site requirements.**

Even without the benefits of a standardized design, the LWRs that have operated in the United States for more than 20 years have had good safety and reliability records. There never has been an accident involving a major release of radioactivity to the environment, and the operating performance, while not spectacular, has been comparable to that of coal-fired powerplants. Still, doubts linger about both the safety and reliability of these LWRs. This section examines the reasons for such concerns, including particular features of these reactors that contribute to **concern.**

Safety Concerns

The occurrence of several widely publicized accidents such as those at Three Mile Island and Browns Ferry nuclear plants have underscored the potential for a catastrophic accident. These accidents shook some of the confidence in our understanding of nuclear reactors. For example, the scenario that unfolded at Three Mile Island had not been stressed prior to the accident: it involved the loss of coolant through a small leak in a pressure relief valve, whereas safety analysis had previously concentrated on large loss-of-coolant accidents. Most studies of these serious accidents have faulted the plant operators more than the reactor hardware (1 O), which indicates that LWR designs are not as forgiving of human error as they might be.

Safety concerns also arise because nuclear powerplants have encountered hardware malfunctions in virtually every system, including control rods, steam generators, coolant pumps, and fuel rods. The majority of these hardware problems have been resolved by retrofits, changes in methods of operation, and redesign. Some problems are expected as a new reactor matures, but many of the LWR problems have persisted. Others continue to surface, some because of the intense scrutiny of plants following the Three Mile Island accident and others because of the aging of the earlier reactors. Most of the difficulties probably have technically feasible solutions, but it is not always clear that they would be cost effective to implement. Meanwhile, the discovery of new problems and the slow resolution of old ones continues to erode confidence in the safety of LWRs.

Confidence in LWRs might be enhanced if there was an objective standard for judging the safety of these plants. As a step in this direction, the Nuclear Regulatory Commission (NRC) has proposed a set of qualitative and quantitative safety goals for nuclear powerplants on a 2-year trial basis (4). These safety goals will provide a means for answering the question, "How safe is safe enough?"

There is a fundamental problem with specifying standards for safety: there is no technique for quantifying the safety of a nuclear powerplant in an objective and unambiguous way. One attempt to define nuclear safety is probabilistic risk assessment (PRA), which outlines sequences of events that could lead to accidents and then assigns probabilities to each basic event (12). PRA is becoming a useful tool for such tasks as comparing certain design options in terms of their safety impact. However, the technique is still in its infancy and the results vary widely from one practitioner to the next. The variations occur because the users of PRA must put in their own assumptions about factors contributing to accidents and their probabilities of occurrence. More research is required to establish reasonable and standard assumptions and to refine the process of assessing risk.

Another important component of safety analysis is the consequence of an accident. This depends on the amount of radioactive material that can be released to the environment following a nuclear reactor accident, otherwise known as the nuclear source term. Recent findings indicate that the source terms now used in regulation and risk analysis may overestimate the magnitude of potential fission product releases (5). Only further analysis can tell whether reductions in the source terms can be fully justified, and, if so, the magnitude of the appropriate reduction for each fission product and for each accident scenario. Modeling and analysis programs are now being conducted by NRC and by the Electric Power Research Institute (EPRI), the American Nuclear Society, and by the Industry Degraded Core Rulemaking Program. These studies should eventually produce realistic estimates of fission product releases, but the task is complex and likely to be lengthy.

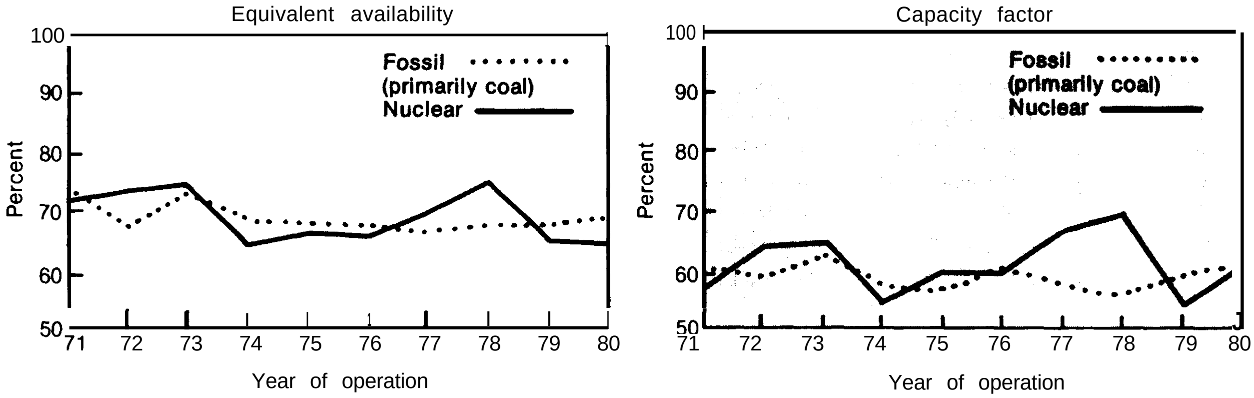
Reliability Concerns

Reliability and safety concerns are closely related, since the same factors that create concern about the safety of LWRs also raise questions about their reliability. If a safety system malfunctions or threatens to do so, the plant must be shut down for a lengthy and often expensive period of maintenance. On the other hand, chronic reliability problems are likely to indicate or contribute to fundamental difficulties that could reduce safety.

The reliability of LWRs is easily quantifiable, in contrast to the difficulties in defining safety. Detailed data on reactor performance have been collected since the beginning of the nuclear era, and they can be analyzed to determine trends. Two measures of performance are commonly used—availability and capacity factor. The availability is defined as the percentage of a time period during which the reactor was available for operation (whether or not it was actually in service). The capacity factor is the ratio of the actual amount of electric generation to the total theoretical output of the plant during the same time period. Each of these quantities has some drawbacks as a measure of plant reliability: the capacity factor is affected by the demand for electricity and the plant availability is insensitive to the capability of the plant to operate at full power. Since nuclear powerplants usually are baseloaded, the capacity factor is generally a better measure of reliability. Both capacity and availability are shown in figure 21 as a function of time for all years from 1971 through 1980 **('17). To** provide a basis for comparison, reliability records are also shown for coal-fired plants larger than 400 megawatts electrical (MWe). It can be seen that the average availability for the two types of plants has been nearly identical at about 69 percent. The average capacity factor for nuclear plants over the same time period was 60 percent, which was 3 percentage points better than for coal. Thus, nuclear plants operate reliably enough compared with their closest counterparts, even though the average performance has not been as outstanding as anticipated by the original nuclear powerplant designers.

it is instructive to reexamine performance data for groups of reactors as well as the industry as a whole. Capacity factors are shown for each reactor type and vendor in table 13 (27). When comparing the data on a lifetime or cumulative basis, it can be seen that there are only slight differences among reactor vendors or types. It also

Figure 21.—Comparison of Fossil Units (400 MWe and Above) to All Nuclear Units



Year

Equivalent availability	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Avg.
Fossil	72.2	67.8	73.2	68.3	67.7	67.2	66.7	67.4	68.0	69.5	68.6
Nuclear	71.6	73.7	74.4	64.0	66.7	65.6	69.8	74.3	64.8	64.5	68.9

Year

Capacity factor	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Avg.
Fossil	60.0	59.6	62.4	57.7	57.3	69.4	57.4	55.3	56.3	59.5	56.6
Nuclear	58.9	63.1	64.1	53.6	59.4	59.0	65.2	68.3	52.9	59.5	59.8

SOURCE. National Electric Reliability Council, "Ten Year Review 1971-1980 Report on Equipment Availability "

Table 13.—Comparison of Lifetime Capacity Factors for U.S. Reactors

	Lifetime capacity factor
Reactor type:	
PWR	61 .OO/o
BWR	58.70/o
Reactor supplier:	
Westinghouse	62.7%
Combustion Engineering	59.70/0
Babcock & Wilcox.	57.0 "/0
General Electric Co.	58.7%
All plants	60.2%

SOURCE: A. Weitzberg, et al , "Reliability of Nuclear Power Plant Hardware — Past Performance and Future Trends," NUS Corp., Jan.15, 1983.

should be noted that these averages can mask substantial spreads in the performance of individual plants. As discussed in chapter 5, the cumulative capacity factors of the worst plants are as low as 40 percent while those of the best are as high as 80 percent.

The hardware problems discussed above have contributed to low availabilities in some plants. These and other hardware problems have been responsible for lengthy periods of downtime as discussed in detail in volume II. It is concluded there that most of the these problems have been or soon will be resolved (27).

Despite signs of progress, LWRs still are not operating trouble-free. The steam generators in several plants have degraded to the point that it has been necessary to replace them. This repair is estimated to cost between $60 million and $80 million in addition to the cost of purchasing replacement power. Other plants may have to undertake expensive retrofits or modify operation to mitigate concerns over pressurized thermal shock (26).

Another impediment to achieving high availability is the stream of retrofits that has followed the accident at Three Mile Island. The Three Mile Island action plan contains about 180 requirements for changes in operational plants; these changes, of course, could not be incorporated into the basic powerplant design, but had to be added to existing systems. This type of retrofitting is seen as a problem by both the nuclear industry as well as its critics since it introduces the possibility of adverse safety consequences. In fact, in some cases, new requirements might reduce rather than enhance safety. This could happen if unanticipated interactions arise or if there is an inadequate understanding of the system the requirement is intended to improve.

The revision in NRC requirements for seismic restraints on piping is often cited as an example of retrofit problems. The restraints in nuclear powerplants are designed to preserve the integrity of pipe by limiting vibrations even if an earthquake should occur. Many plant operators and designers complain that these restraints are expensive to install and that they hold the pipes too rigidly to allow for thermal expansion. Furthermore, some critics of the current seismic requirements feel that piping actually may be more prone to failure in an overconstrained system. These critics assert that today's requirements for seismic restraints result from an attempt to make it easier to analyze conditions in plants rather than from an identifiable need (l).

On the balance, retrofits probably have improved the safety of operating nuclear powerplants. in fact, one assessment of plants before and after the Three Mile Island retrofits concludes that the probability of an accident has been reduced by a factor of 6 in PWRs and by a factor of 3 in BWRs, with the core melt probability for PWRs now only slightly higher than for BWRs. These improvements are attributed primarily to higher reliability of feedwater systems and regulatory and inspection procedures that reduce the probability of human error (19).

Examples of Specific Concerns

Since 1978, NRC has been required by Congress to prepare a list of generic reactor problems. This list is revised annually to reflect new information and progress toward resolution. Each time a new safety issue is identified, NRC assesses the need for immediate action. In some cases, action such as derating (reducing the approved operating power) certain reactors, is taken to assure public health and safety. in other cases, an initial review does not identify any immediate threat to the public, and further research is conducted. Many generic safety issues have been resolved and removed from NRC's list of significant safety items (26).

Table 14 summarizes the 15 most important unresolved safety issues as determined by NRC in 1982. A few of the items on that list will be examined here as examples of the types of concerns that remain about LWRs and some of the factors preventing their resolution.

One of the most widely publicized safety problems is the potential in PWRs for fracture of the reactor vessel from pressurized thermal shock. Reactors are designed to be flooded with relatively cold water if a loss of coolant accident occurs. The sudden temperature differential causes surface strains, known as thermal shock, on the thick metal wall of the reactor vessel and imposes severe differential stress through the vessel wall. While plant designers have understood and accounted for this phenomenon for 'years, they have only recently discovered that two other factors may make the effect more acute than anticipated. One is that the emergency cooling system is likely to be actuated following a small-break accident (e.g., the one at Three Mile Island) when the reactor vessel is still highly pressurized. In such a situation, the stresses due to thermal shock would be added to those due to internal pressure. The second factor is that the weld and plate materials in some older reactor vessels are becoming brittle from neutron exposure faster than had been expected. Such embrittlement increases the vulnerability of the vessel to rupture following pressurized thermal shock. While the possibility

Table 14.—Unresolved Safety Issues

issue/Descridion

Water hammer: Since 1969 there have been over 150 reported incidents involving water hammer in BWRs and PWRs. The incidents have been attributed to rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage has been relatively minor.

Steam generator tube integrity: PWR steam generators have shown evidence of corrosion-induced wastage, cracking, reduction in tube diameter, and vibration-induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions with adequate safety margins.

Mark I containment long-term program: During a large-scale testing program for an advanced BWR containment system, new suppression pool loads associated with a loss of coolant accident were identified which had not been explicitly included in the original design of the Mark I containment systems. In addition, experience at operating plants has identified other loads that should be reconsidered. The results of a short-term program indicate that, for the most probable loads, the Mark I containment system would maintain its integrity and functional capability.

Reactor vessel material toughness: Because the possibility of pressure vessel failure is remote, no protection is provided against reactor vessel failure in the design of nuclear facilities. However, as plants accumulate service time, neutron irradiation reduces the material fracture toughness and initial safety margins. Results from reactor vessel surveillance programs indicate that up to 20 operating PWRs will have materials with only marginal toughness after comparatively short periods of operation.

Fracture toughness of steam generator and reactor coolant pump supports: Questions have been raised as to the potential for Iamellar tearing and low fracture toughness of steam generator and reactor coolant pump support materials in the North Anna nuclear powerplants. Since similar materials and designs have been used on other plants, this issue will be reassessed for all PWRs.

Systems interactions in nuclear powerplants: There is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to the effect on the redundancy and independence of safety systems.

Determination of safety relief valve pool dynamic **loads** and temperature limits for BWR containment: Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or structures located within the pool.

Seismic design criteria: While many conservative factors are incorporated into the seismic design process, certain aspects of it may not be adequately conservative for all plants. Additional analysis is needed to provide assurance that the health and safety of the public is protected, and if possible, to reduce costly design conservatism.

issue/DescrilMion

Containment emergency sump performance: Following a loss of coolant accident in a PWR, water flowing from a break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the tank water is depleted, a recirculation mode is established by drawing water from the containment floor or sump. This program addresses the safety issue of the adequacy of the sump and suppression pool in the recirculation mode.

Station blackout: The loss of A.C. power from both off site and onsite sources is referred to as a station blackout. In the event this occurs, the capability to cool the reactor core would be dependent on the avail ail it y of systems which do not require A.C. power supplies and the ability to restore A.C. power in a timely manner. There is a concern that a station blackout may be a relatively high probability y event and that this event may result in severe core damage.

Shutdown decay heat removal requirements: Many improvements to the steam generator auxiliary feedwater system were required after the accident at Three Mile Island. However, an alternative means of decay heat removal in PWRs might substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and thus reduce the overall risk to the public.

Seismic qualification of equipment in operating plants: The design criteria and methods for the seismic qualification of equipment in nuclear plants have undergone significant change. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads may vary considerably and must be reassessed.

Safety implications of control systems: It is generally believed that control system failures are not likely to result in the loss of safety functions which could lead to serious events or result in conditions that cannot be handled by safety systems. However, indepth plant-by-plant studies have not been performed to support this belief. The purpose of this program is to define generic criteria that may be used for plant-specific reviews.

Hydrogen control measures and effects of hydrogen burns on safety equipment: Reactor accidents which result in degraded or melted cores can generate large quantities of hydrogen and release it to the containment. Experience gained from the accident at Three Mile Island indicates that more specific design provisions for handling large quantities of hydrogen releases maybe appropriate, particularly for smaller, low-pressure containment designs.

Pressurized thermal shock: Neutron irradiation of reactor pressure vessel weld and plate materials decreases fracture toughness. This makes it more likely that, under certain conditions, **a crack could grow to a size that might threaten vessel integrity,**

SOURCE: US. Nuclear Regulatory Commission, "Unresolved Safety Issues Summary," Aug. 20, 1982, of a pressure vessel failure is peculiar to only a few older reactors, it is of concern that such a potentially severe condition was not recognized sooner. Measures to mitigate the problem of pressurized thermal shock include reducing the neutron flux near the outer walls, increasing the temperature of emergency cooling water, heating the reactor vessel at very high temperatures to reduce brittleness, and derating the plant (1 5,27).

BWRs are not susceptible to pressurized thermal shock, but they have been plagued by a problem known as intergranular stress corrosion cracking. This problem, which involves defects in the reactor coolant piping, is now listed by NRC as resolved, but it continues to be the subject of extensive and costly research programs throughout the industry. Most of the service piping sensitive to such cracking has been designed out of the later BWRs, but reactors currently under construction will have recirculation loop piping with some susceptibility to this phenomenon (15,27).

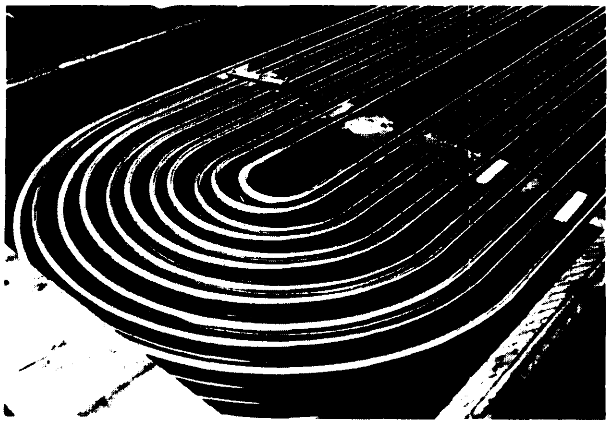
Another problem on the list of unresolved safety issues deals with the corrosion or fatigue cracking of steam generator tubes (15). This is of concern because these tubes separate the primary coolant from the secondary system, and there is some question whether degraded tubes will be able to maintain their integrity under accident

Photo credit: Atomic Industrial Forum

These four steam generators are awaiting installation in a large pressurized water reactor. In this type of reactor, water flows through the core to remove heat and then through narrow tubes in the steam generator to transfer it to a secondary coolant loop.

conditions. NRC estimates that steam generator degradation has accounted for about 23 percent of the non refueling outage time in nuclear reactors. The corrosion has been attributed to a combination of inappropriate water-chemistry treatment and poor quality materials in the steam generators. The result has been wastage, cracking, reduction in tube diameter, separation of cladding from the tube sheet, and deterioration of the metal plates that support the tubes inside the generators. The severity of the problem varies with steam generator design and water treatment methods. Much of the corrosion has been brought under control, but plant operators continue to inspect their steam generators regularly and plug the degraded tubes when necessary. Operators of several nuclear units–Surry 1 and 2 in Virginia and Turkey Point 3 and 4. in Florida—have already had to replace their steam generators. Other units may face expensive and lengthy overhauls in the future. The fatigue cracking appears to result from flow-induced vibrations, and resolution of this problem may require design modifications.

Two general safety issues deal with uncertainties over the behavior of the complex systems found in nuclear powerplants. One concern is that the interactions among the various systems are not fully understood and could contribute to an accident under some circumstances. in particular, it is possible that some system interactions



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could eliminate the redundancy or independence of safety systems. Another concern is that the failure of a control system might aggravate the consequences of an accident or prevent the operator from taking the proper action. These concerns relate to the need to understand the entire reactor and its interrelated systems rather than to any specific feature of its hardware. Resolving either of these concerns probably would require analysis on a plant-specific basis, but NRC is attempting to identify generic criteria. Both of these issues contribute to a larger question: are LWR designs more complex than necessary and could a simpler, but equally safe, reactor be designed?

Another safety concern is that a nuclear plant may develop a serious problem and fail to shut down automatically in an incident known as an "anticipated transient without scram." **In such a situation, the emergency cooling system would** have to remove not only the decay heat, but also the heat generated by full-power operation. NRC has removed this item from its list of generic safety issues, but many critics feel that it continues to represent a valid concern. This issue was highlighted by the discovery at the Salem plants in New Jersey that faulty circuit breakers prevented the scram systems from activating automatically (16).

The resolution of several of the unresolved issues may require adding new equipment to LWRs in operation or under construction. One of these items is the provision for an alternative means of removing decay heat. NRC is examining whether an additional system might substantially increase the capability of the reactor to deal with a broader range of accidents or malfunctions. Another issue is whether the installation of a means to control hydrogen is required to prevent the accumulation of dangerous levels of the gas in the containment vessel. These concerns are an outgrowth of lessons learned from the Three Mile Island accident. Either measure discussed above would be very costly to retrofit on existing plants but might be easily designed into a new plant.

These examples indicate that there are still unresolved generic safety questions concerning the LWR, and that the resolution of these issues will involve a complex tradeoff between cost and safety. While no single issue has been identified as a fatal flaw, neither is there a clear indication that current LWR technology is fully mature and stable.

Lessons Learned From Operating LWRs

The utilities operating LWRs have gained knowledge of both the strengths and weaknesses of their plants. A recent survey of the utilities reflects the concerns about safety and reliability mentioned above and gives specific recommendations about features that might mitigate the problems (13). The following recommendations were made:

- safety and control systems in new LWRs should be simpler and easier to operate, test, and maintain. Utility personnel expressed preferences for passive and fail-safe characteristics whenever possible;
- safety systems should be separated from nonsafety systems and dedicated to single functions. plant operators worry that overlapping functions may lead to adverse impacts in a complex accident scenario;
- the response of existing plants to abnormal occurrences should be slowed. Because of the low inventory of primary and secondary cooling water in current LWRs, the time between the start of a transient and the onset of melting in the core can be short. If all safety equipment works as designed, it is likely that transients would be brought under control quickly, regardless of the cause. If certain important safety components fail, melting could begin after 40 minutes in a PWR. The failure of all safety systems, which is not considered a plausible scenario, could cause core melting in less than 1 minute;
- containment buildings should be larger to provide adequate space for maintenance. Current structures do not allow enough room to easily handle equipment that is being disassembled for maintenance; and
- the potential for retrofits on future plants should be reduced as much as possible by taking a fresh look at the LWR. Such an effort should integrate the retrofits into a new

design rather than piece them on top of an existing one.

Both NRC's list of unresolved safety issues and the survey of utility executives provide concrete examples of reasons for concern over the safety and reliability of current plants. Existing LWRs have serious, although resolvable, problems with important hardware components; the interrelated safety and control systems hinder a deep understanding of their behavior; complexity has been increased by the large number of safety-mandated retrofits; and current reactors are somewhat unforgiving of operator errors. Despite this less-than-perfect record of the LWR, many in the industry are reluctant to abandon it. They argue that they have made appreciable progress along the learning curve that would have to be repeated with an alternative reactor concept.

ADVANCED LIGHT WATER REACTOR DESIGN CONCEPTS

Improvements could be made to the current generation of LWRs by redesigning the plants to address the safety and operability problems outlined above. Furthermore, the entire design could be integrated to better incorporate various changes made to LWRs in the past decade. If such a redesign effort were successful, LWRs would probably continue to be the preferred option in the future. LWRs have the advantage of being familiar, proven designs with a complete infrastructure to support manufacturing, construction, and operation.

Advanced LWR designs are being developed by both Westinghouse Electric Corp. and General Electric Co., and they should be available to U.S. utilities before any new reactors are ordered. General Electric is designing a new BWR that is an evolution of the most advanced plants currently under construction. The newer reactor has been modified to enhance natural circulation of the primary coolant and hence improve passive cooling. This increases the ability of the coolant system to remove decay heat in the event that the main circulation system fails. The design further provides for rapid depressurization of the primary system so that both low- and high-pressure pumps can supply water to the reactor vessel. This safety feature provides additional assurance that emergency coolant would be available in the event that primary coolant is lost (6).

Westinghouse Electric Corp. is developing an advanced version of its PWR. The company has undertaken a joint program with Mitsubishi Heavy Industries, the Westinghouse licensee in Japan. It is expected that the design development will be completed by 1984, and there are plans to initiate verification testing by 1986. Westinghouse is reviewing this design effort with NRC so that the advanced PWR could be readily licensed in the United States.

The Westinghouse design focuses on reducing risks, improving daily operations, anal controlling costs *(8,11)*. It attempts to reduce economic risk to the owner and health risk to the public by incorporating several new features. The coolant piping has been reconfigured and the amount of water in the core has been increased to reduce the possibility that a pipe break could drain the primary coolant enough to uncover the core. Protection against other accidents that could uncover the core has been provided by safeguarding against valves that could fail in an open position. Additional risk reduction efforts have been focused on the response of plant operators and systems following an accident, with the goal of requiring less operator action and more automatic responses.

Improvements in normal plant operations are another important feature of the Westinghouse advanced design. The reactor has been redesigned to operate 18 to 24 months on a single batch of fuel, rather than the current 12 month cycle. The longer fuel cycle is made possible by enlarging the diameter of the reactor and reducing the power density and neutron flux. This is combined with a different way to moderate the neutrons at the beginning of the cycle so that more plutonium is produced; the extra fuel is burned at the end of the cycle when the fissionable uranium is depleted. Other efforts have been made to reduce the amount of time required to refuel and inspect the reactor when it is shut down. The combined effect of these final two features is to increase the availability of the reactor, perhaps to as much as 90 percent.

Efforts have also been directed toward increasing the reliability of the advanced PWR. For example, the steam generators have been redesigned so that there will be fewer stresses on the tubes and less potential for contaminants to collect. Other improvements have been made to increase the reliability of the nuclear fuel and its support structure.

Operational improvements have been made in the area of occupational radiation exposure. The new plants have been designed for easy access to reduce worker exposure. In addition, radiation shielding has been added to areas where high radiation fields can be expected. Finally, ease of maintenance and repair have been factored into the redesign of the steam generators; this should reduce the large occupational exposures that have resulted from steam generator maintenance in current PWRs.

An area that is closely related to both safety and operation is the effort to increase the design margins of the advanced PWR. The increased amount of coolant in the core and the reduced core power density make the reactor less sensitive to upsets. Furthermore, the physics of the design dictates that the core will be even better at slowing the fission process in response to a rapid rise in temperature.

The capital cost of the advanced PWR proposed by Westinghouse probably would be greater than that of current PWRs. An effort has been made to hold down the capital cost by simplifying fluid systems, making more use of multiplexing, and completing at least three-quarters of the design before construction is initiated. Moreover, other features of the design should compensate for the increased capital costs, resulting in lifetime costs that should be comparable to those of today's LWRs. Significant fuel savings are expected, with reductions in both uranium and enrichment requirements. This is due to the reduction in specific power, the more effective use of plutonium created within the core,

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An intermediate step in the redesign of the PWR has been taken by the Central Electricity Generating Board (CEGB) in Great Britain (22). CEGB did not redesign the basic PWR; rather, it added safety features to the Standardized Nuclear Unit Power Plant System (SNUPPS), which was developed and marketed in the United States in the 1970's. Most of the changes increase the separation and redundancy of safety systems, such as adding a steel containment shell to the normal concrete containment structure, increasing the number and capacity of pumps in the emergency-cooling system, and increasing the number of independent diesel generators to provide electrical power if the normal supply is interrupted. These changes are likely to reduce the probability of a core meltdown, and the probability for the release of fission products is much lower still. These safety improvements will not be inexpensive; the new design is estimated to cost 25 percent more than the original SNUPPS plant.

HEAVY WATER REACTORS

If the new designs for LWRs are perceived as being less than adequate to ensure safe and reliable operation, it is possible that alternative designs will become more attractive to utilities and investors. The HWR is a potential candidate because it is the only other type of reactor that has been deployed successfully on a commercial scale. The HWR has been developed most extensively in Canada, where the CANDU (Canada deuterium uranium) reactors produce all the nuclear-generated electricity. In addition, HWRs have emerged as competitors with LWRs in several other nations, including India, Korea, and Argentina.

The interest in this type of reactor originally derived from the effective way in which heavy water moderates neutrons, with a resultant increase in fuel economy when compared to LWRs. There are also various inherent safety features of the HWR that make it an attractive alternative to the LWR. in addition, the current generation of HWRs has compiled an excellent operating record. In fact, the HWRs in operation worldwide have the most impressive reliability record of any commercial reactor type. As shown in table 15, HWRs operated at an average capacity factor of 71 percent in 1982, which far exceeds the records of either PWRs or BWRs. Moreover, in 1982, 5 of the top 10 best performers internationally were HWRs, even though this type of plant accounts for only 5 percent of the total nuclear capacity. Both lifetime and annual capacity factors for the world's best power reactors are shown in table 16.

Table 15.—World Comparison of Reactor Types (150 MWe and larger)

Type of reactor	Number of reactors	Average annual load factor (percent)
Pressurized heavy water reactor	13	71
Gas-cooled reactor[a] (Magnox)	26	57
Pressurized (light) water reactor	106	59
Boiling water reactor . .	57	60

aTh, natural. uranium gas-cooled reactors referenced here differ Significantly from the HTGR discussed in this report.

SOURCE: "Nuclear Station Achievement 19S3," *Nuclear Engineering International*, October 1983,

The design of an HWR is somewhat similar to a PWR in that primary coolant transfers heat from the core to a secondary coolant system via a steam generator (3,6,21). In many other ways, the design of an HWR differs significantly from that of a PWR. As implied by its name, heavy water is used to moderate the neutrons generated in the chain reaction. This is a more effective moderator than ordinary water, and so the core can be composed of less concentrated fissionable material than in a PWR. As a result, the HWR can operate with unenriched or natural uranium. Nations that have not developed enrichment technology perceive this as an advantage, but in the United States uranium enrichment is readily available. it is likely that U.S. utilities would elect to operate HWRs with uranium that is slightly enriched. On such a fuel cycle, the HWR would require only 60 percent of the uranium used in

Table 16.—World Power Reactor Performance (150 MWe and larger)

Reactor type	Unit	Nominal rating gross (MWe)	Capacity factor (percent)
Annual[a]			
PHWR	Pickering 2	542	96
PHWR	Bruce 3	826	96
PHWR	Bruce 4	826	95
PHWR	Pickering 4	542	91
BWR	Muehlberg	336	91
PWR	Turkey Point 3	728	90
PWR	Beznau 2	364	89
BWR	Garona	460	88
PWR	Grafenrheinfeld	1299	88
PHWR	Bruce 1	826	87
Cumulative[b]			
PHWR	Bruce 3	826	88
PHWR	Bruce 4	826	85
PWR	Beznau 2	364	85
PHWR	Pickering 2	542	83
GCR	Hunterston A	169	83
PWR	Stade 1	662	82
PHWR	Pickering 4	542	81
BWR	Muehlberg	336	80
PWR	Obrigheim	345	80

[a]From July 1982 through June 1983.
[b]From start of operation through June 1983

SOURCE "Nuclear Stat Ion Achievement 1983, " Nuclear *Engineering International,* October 1983

an LWR to produce the same amount of electricity.

The use of heavy water instead of ordinary water as a moderator and coolant provides a fuel-economy advantage, but it also suffers from a disadvantage. Heavy water is expensive, and the initial inventory of heavy water represents about 20 percent of the capital cost of HWRs. Total capital costs of HWRs are probably comparable to those of LWRs, but fuel cycle costs over the life of the plant should be lower.

Another feature of the HWR that distinguishes it from a PWR is the use of hundreds of small pressurized tubes instead of a single large pressure vessel. The pressure tubes in an HWR enclose the fuel assemblies and heavy water coolant which flows through the tubes. They are positioned horizontally in a large unpressurized vessel known as a calandria, as shown in figure **22.** The calandria is filled with heavy water, which acts as a moderator and is kept at low temperature and pressure. The heavy water in the calandria surrounds the coolant tubes but is isolated from the fuel.

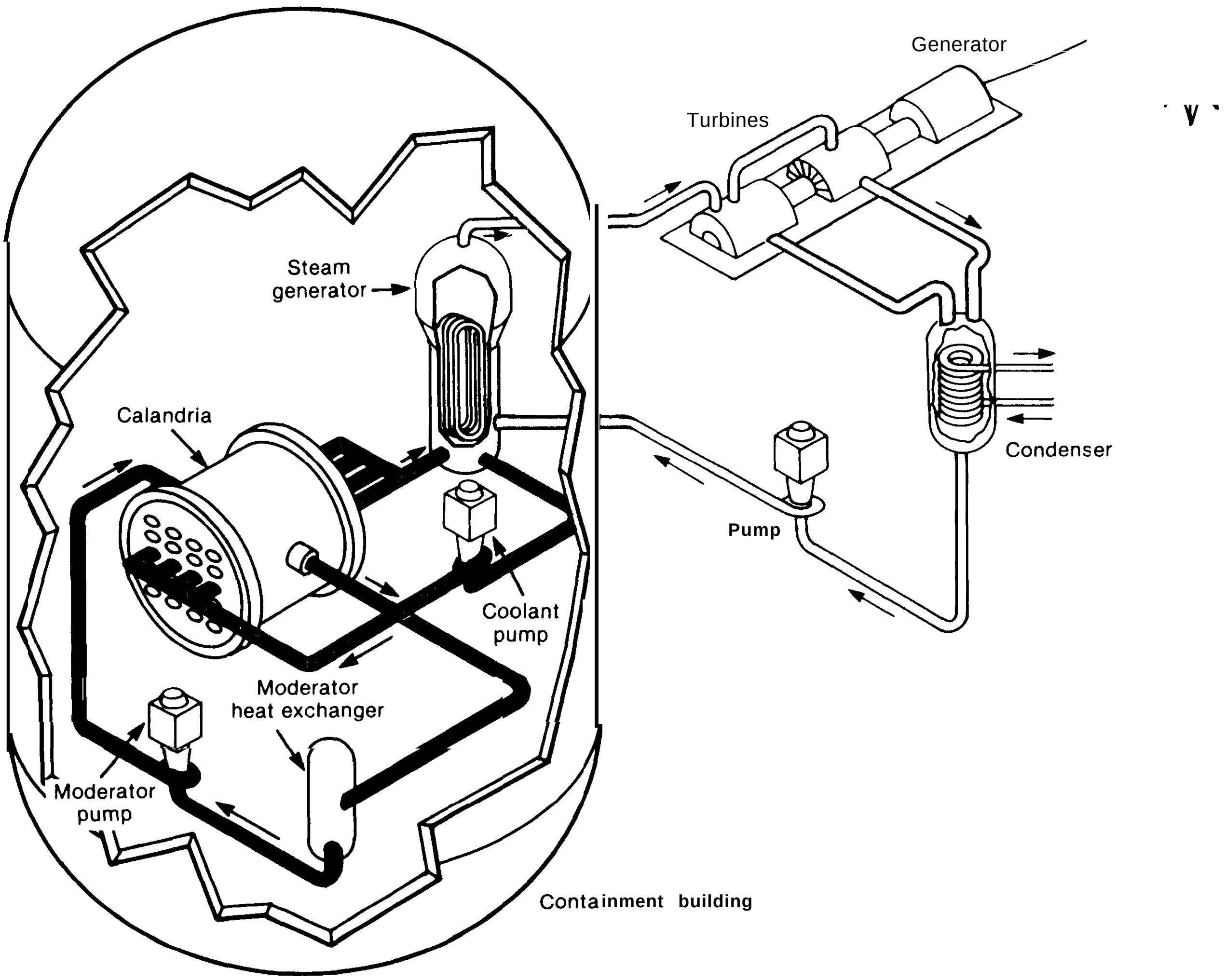
A disadvantage of the pressure tube configuration is that the thin walls of the tubes restrict the temperature of the coolant more than the heavy steel pressure vessel of an LWR. Hence, the overall efficiency of energy conversion is somewhat less in an HWR. Efficiency is further limited in the HWR by the heat that is deposited in the heavy water moderator. Current HWRs achieve an efficiency of only 29 percent, while LWRs typically achieve a 33-percent efficiency.

The pressure tube configuration has some advantages in that it separates the moderator from the coolant. The reactivity control devices and safety systems are located outside the primary coolant loop and cannot be affected by a loss of coolant accident. Moreover, the moderator acts as a backup heat sink that could cool the fuel if the primary coolant system fails. This reduces the necessity to develop elaborate systems to provide emergency core cooling and decay heat removal, which have been a primary concern in LWRs.

The HWR differs from the LWR in its method of refueling. In an LWR, the reactor must be shut down, the cover of the pressure vessel removed, and the fuel rods exchanged in an operation that lasts for several weeks. HWRs can utilize online refueling because they use pressure tubes rather than a single pressure vessel. This means that the reactor can continue to operate while depleted fuel assemblies are removed and replaced with **fresh fuel. This method of refueling contributes to the overall availability of the reactor since there is no need to shut** it down. It also enhances rapid identification and removal of fuel elements that leak radioactive materials into the cooling water.

As discussed above, the HWR appears to have certain safety and operational advantages with respect to the LWR. However, there are other factors that make it unlikely to be considered as a viable alternative to the LWR in the United States. As with all alternatives to the LWR, the heavy water reactor suffers from lack of familiarity and experience in the United States. There are no vendors in the United States which offer HWRs, and hence there is no established domestic infrastructure to build and service them. Furthermore, there are no utilities with HWR experience. These factors would not necessarily prohibit the introduction of an HWR into the United States, since the manufacturing require-

Figure 22.—Heavy Water Reactor



SOURCE: Office of Technology Assessment.

ments, design, and operational skills for the HWR and LWR are similar. However, any alternative design would have to overcome barriers before being accepted. It would have to be clearly superior to the LWR, but in some areas that have proved to be most troublesome for **LWRs** for instance, operational complexity, the HWR may actually add to the uncertainty.

In addition to lack of familiarity, the HWR offers no capital cost advantages to the LWR. Even though lifetime costs may be lower, it is unlikely that utilities would be willing to assume large capital debts without a clear demonstration of the ad-vantages of an alternative reactor.

Another issue relates to uncertainties in the licensing process. The HWR is a fully commercial and licensable reactor in Canada and other nations. In the United States, however, it would be necessary to modify the licensing procedures to match a new reactor. It also is likely that design modifications would have to be made to the **HWR in order to accommodate to the U.S. regulatory structure; there is a spectrum of opinions on how extensive these changes would be (24). Cost and availability uncertainties would be in-**

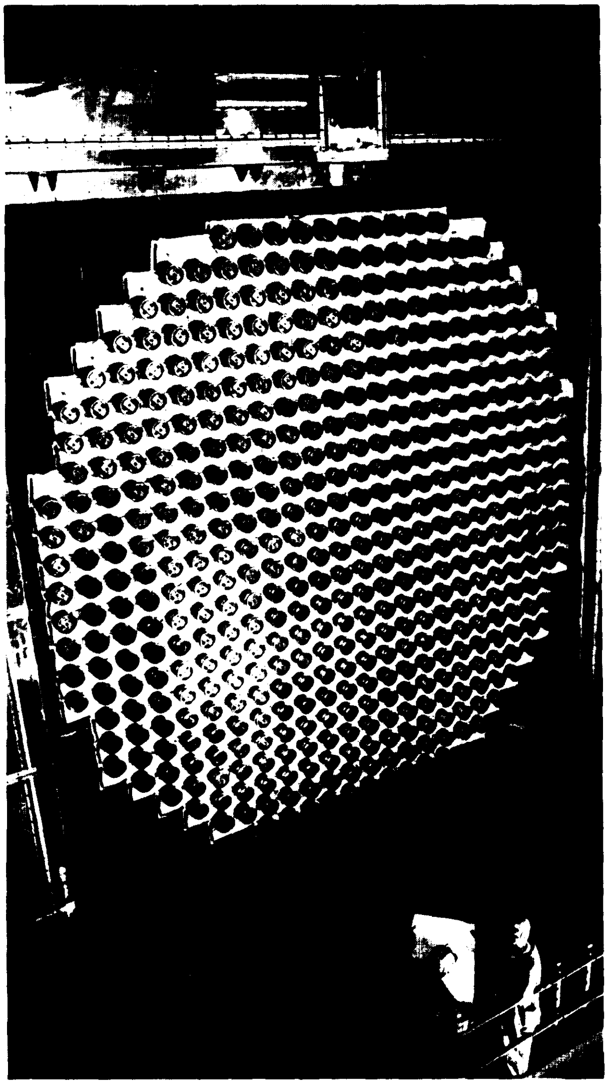


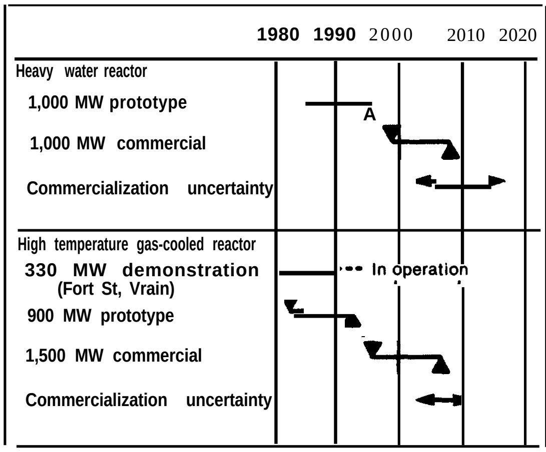
Photo credit: Ontario Hydro

In a heavy water reactor, natural uranium is **used as the fuel.** The fuel bundles are loaded horizontally, each in its own tube containing pressurized water. The pressure tubes in a heavy water reactor replace the large steel vessel in light water reactors

troduced into a system that already is less predictable than desired. Initial capital costs might be increased by design modifications to improve efficiency or meet stringent seismic requirements. The commercialization process could be extended if significant changes are required in the licensing and design of this reactor. One possible schedule for development was developed by EPRI and is shown in figure 23.

In the United States, the HWR is not perceived as offering enough advantages to abandon LWR technology. Nonetheless, the HWR, may become a source of electricity for U.S. utilities. It is possible that Canadian reactors operating near the U.S. border might significantly increase the export of power to U.S. grids if the HWRs in Canada continue to operate safely and reliably.

Figure 23.—Schedules for Alternative Reactors



▼ Preliminary safety analysis report filed

▲ Completion of construction

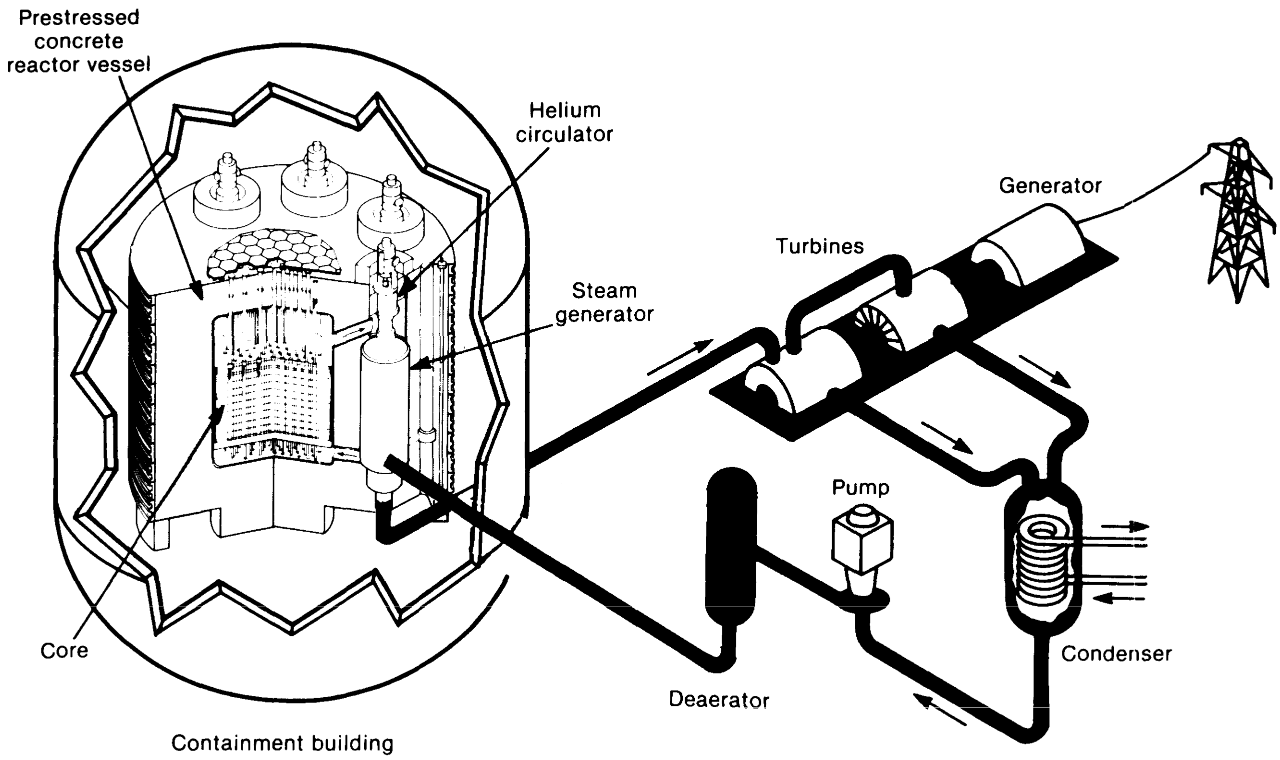
◄► Uncertainty in achieving commercialization target

SOURCE: "Alternative Nuclear Technologies, " Electric Power Research Institute, October 1981.

THE HIGH TEMPERATURE GAS-COOLED REACTOR

The HTGR is cooled by helium and moderated by graphite, and, as shown in figure 24, the entire core is housed in a prestressed concrete reactor vessel (PCRV) (2,6,25). The reactor uses enriched uranium along with thorium, which is similar to nonfissionable uranium in that it can be transformed into useful fuel when it is irradiated. Because helium is used instead of water as a coolant, the HTGR can operate at a higher temperature and a lower pressure than an LWR. This results in a higher thermal efficiency for electricity generation than can be achieved with the

Figure 24.—High Temperature Gas-Cooled Reactor



SOURCE: GA Technologies and Office of Technology Assessment.

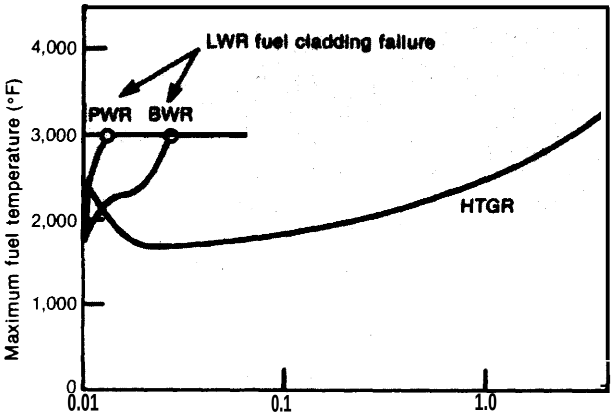
other alternative reactors discussed in this chapter; it also makes the HTGR particularly suited for the cogeneration of electricity and process steam.

The HTGR has several inherent safety characteristics that reduce its reliance on engineered devices for safe reactor operation. First, the use of helium as a primary coolant offers some advantages. Because helium is noncorrosive in the operating temperature range of the HTGR, it causes little damage to components. Furthermore, it is transparent to neutrons and remains nonradioactive as it carries heat from the core. The use of graphite as a moderator also has some advantages. It has a high heat capacity that greatly reduces the rate of temperature rise following a severe accident, and hence there is less potential for damage to the core. As a result, HTGRs do not require a containment heat removal system.

The design of the fuel and core structure for the HTGR also has inherent safety features. The core is characterized by a low power density in the fuel, only about one-tenth that of a light water reactor. As discussed above, the core has a large thermal capacitance due to the presence of graphite in the core and support structures. As a result, temperatures would rise very slowly even if the flow of coolant was interrupted, The operators of an HTGR would have a great deal of time to diagnose and correct an accident situation before the core is damaged. Figure 25 compares the 10-hour margin to fuel failure for an HTGR with conventional LWRs, in which the margins are measured in minutes.

Another safety feature of the HTGR is the PCRV, which contains the entire primary coolant system. The PCRV provides more shielding from the radioactive materials in the core than a steel vessel since the thick concrete naturally attenu-

Figure 25.—Comparative Fuel Response Times



SOURCE: Gas-Cooled Reactor Associates.

ates radiation. in addition, catastrophic failure of a PCRV is regarded to be much less likely than failure of a steel vessel. The vertical and circumferential steel tendons that are used to keep the concrete and liner in a state of compression are isolated from exposure to damaging neutron fluxes and extreme temperatures. Furthermore, they are independent of one another and can be readily inspected; it is extremely unlikely that many of the tendons would fail simultaneously.

The entire primary coolant loop, including the steam generators, helium circulators, and other auxiliary equipment, is included within cavities in the PCRV. The advantage of such a configuration is that pipe breaks within the primary loop cannot result in a rapid loss of coolant. As a result, the only engineered safety system needed to protect the core from overheating in the event the main core cooling fails is a forced circulation, decay heat removal system. In the HTGR, this is provided by a core auxiliary cooling system, which is dedicated to decay heat removal and incorporates three redundant cooling loops for high reliability. This is enhanced by a PCRV liner cooling system that provides an additional heat sink for decay heat.

Because of the high thermal efficiency of this reactor and the safety features discussed above, the HTGR has long been considered as a possible alternative to the LWR for commercial power generation. Work on the HTGR was initiated in the United States soon after the LWR was developed. The concept was successfully demonstrated in 1967 when a 40-MWe reactor was placed in commercial operation. This prototype unit, Peach Bottom 1, was constructed by Philadelphia Electric Co. and was the world's first nuclear station to produce steam at 1,000° **F.** The plant operated at an average availability of 88 percent before being decommissioned in 1974 for economic reasons.

Research and development (R&D) leading to commercial-sized systems continued after the Peach Bottom 1 demonstration. A cooperative effort of Public Service Co. of Colorado, the Atomic Energy Commission, and General Atomic Co. led to the construction of the 330-MWe Fort St. Vrain reactor in Colorado. This plant introduced several advanced features relating to the design of the fuel, steam generators, helium circulators, and PCRV. The plant first generated power in 1976 but experienced problems in its early years that contributed to a disappointing availability record. However, the majority of the systems in the Fort St. Vrain reactor have performed well. Furthermore, radiation exposures have been the lowest in the industry, even though extensive modifications were made to the primary system after the plant started operating. The Fort St. Vrain experience also demonstrates that the HTGR is manageable and predictable, even under extreme conditions. In the past 7 years, forced circulation cooling has been interrupted 17 times at the Fort St. Vrain reactor. The operators generally were able to reestablish forced circulation within 5 minutes, with the longest interruption lasting 23 minutes. Even with so many loss of flow incidents, there has been absolutely no damage to the core or any of the components.

Fort St. Vrain also experienced some unexpected technical difficulties-slow periodic fluctuations in certain core exit temperatures were observed. The fluctuations were associated with small movements of fuel in the core, caused by differential thermal expansion of fuel blocks. The problem was resolved at Fort St. Vrain by maintaining the spacing between fuel regions with core restraint devices. The next generation of reactors will avoid such problems by redesigning the fuel block.

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Photo credit: Gas-Cooled Reactor Associates (GCRA)

The high temperature gas-cooled reactor uses helium as a coolant instead of water. Helium offers some advantages since it is less corrosive than water and does not become radioactive. The helium circulator for the Fort St. Vrain reactor is shown above

Since 1977, the HTGR program has focused on the development and demonstration of a commercial-sized plant. The emphasis is currently on designing a four-loop, 2240 megawatt thermal (MWth) reactor that can generate electricity at high efficiency or be applied to the cogeneration of electricity and process steam. This design incorporates the lessons learned from the operation of Fort St. Vrain, experience from the earlier design of commercial HTGRs, and information obtained from cooperative international programs. Key design changes have been made to simplify the plant, improve its licensability and reliability, correct specific component-design deficiencies, and increase performance margins (7). The design work has been sponsored by the U.S. Department of Energy (DOE).

Another design effort is directed toward developing a much smaller gas-cooled reactor, known as the modular HTGR (14). This concept capitalizes on small size and low power density to produce a reactor that may be able to dissipate decay heat with radiative and convective cooling. In other words, it would not require emergency cooling or decay heat removal systems. Such a reactor would be inherently safe in the sense that no operator or mechanical action would be required to prevent fuel from melting after the reactor is shut down. The design for this type of reactor is still in a preliminary stage, but its potential for walk-away safety may warrant further investigation.

In addition to the domestic effort to develop the HTGR, there are also international programs to design and construct gas-cooled reactors. The Federal Republic of Germany has operated a 15-MWe prototype plant since 1967 with great success, achieving an average availability of more than 85 percent. A 300-MWe demonstration plant is scheduled for startup in 1984, and work has been initiated on the design of a 500-MWe HTGR. Japan also has been involved in the development of a very high temperature gas-cooled reactor for process heat applications. The Japanese program is directed at designing a 50-MWth reactor to begin construction in 1986.

INHERENTLY SAFE REACTOR CONCEPTS

incentives for developing a more forgiving reactor arise from several sources. As previously discussed, the design of current LWRs has developed in a patchwork fashion, and there are still a number of unresolved safety issues. The accident at Three Mile Island increased the incentive to develop a foolproof reactor when it became clear that LWRs are susceptible to serious accidents arising from human error. A more forgiving reactor design became desirable in terms of investment protection as well as public health and safety.

The modular HTGR discussed above is an example of an effort to develop an inherently safe reactor. Another example of a reactor that attempts to improve safety dramatically is based on LWR technology. [t is known as the Process Inherent Ultimately Safe (PIUS) reactor, and it is being developed by the Swedish nuclear firm ASEA-ATOM (6,9,23). The design goal is to ensure safety, even if the reactor is subjected to human error, equipment failure, or natural disaster. In the PIUS reactor, this goal was translated into two primary design objectives: first, to ensure safe shutdown and adequate decay heat removal under any credible circumstances; and second, to use the construction and operating experience of current LWRs as a basis for development. This eliminates some of the uncertainties associated with a *new* design.

The safety goal was paramount in the design of the PIUS reactor concept. The nuclear island was designed to ensure that the fuel would never melt, even if equipment failure were to be compounded by operator error. To accomplish this, two conditions have to be met. First, it is necessary to ensure that the core always remains covered with water. In LWRs, engineered safety systems ensure that cooling water is available to the core. However, confidence in these systems was shaken by the accident at Three Mile Island when the fuel was exposed long enough to melt.

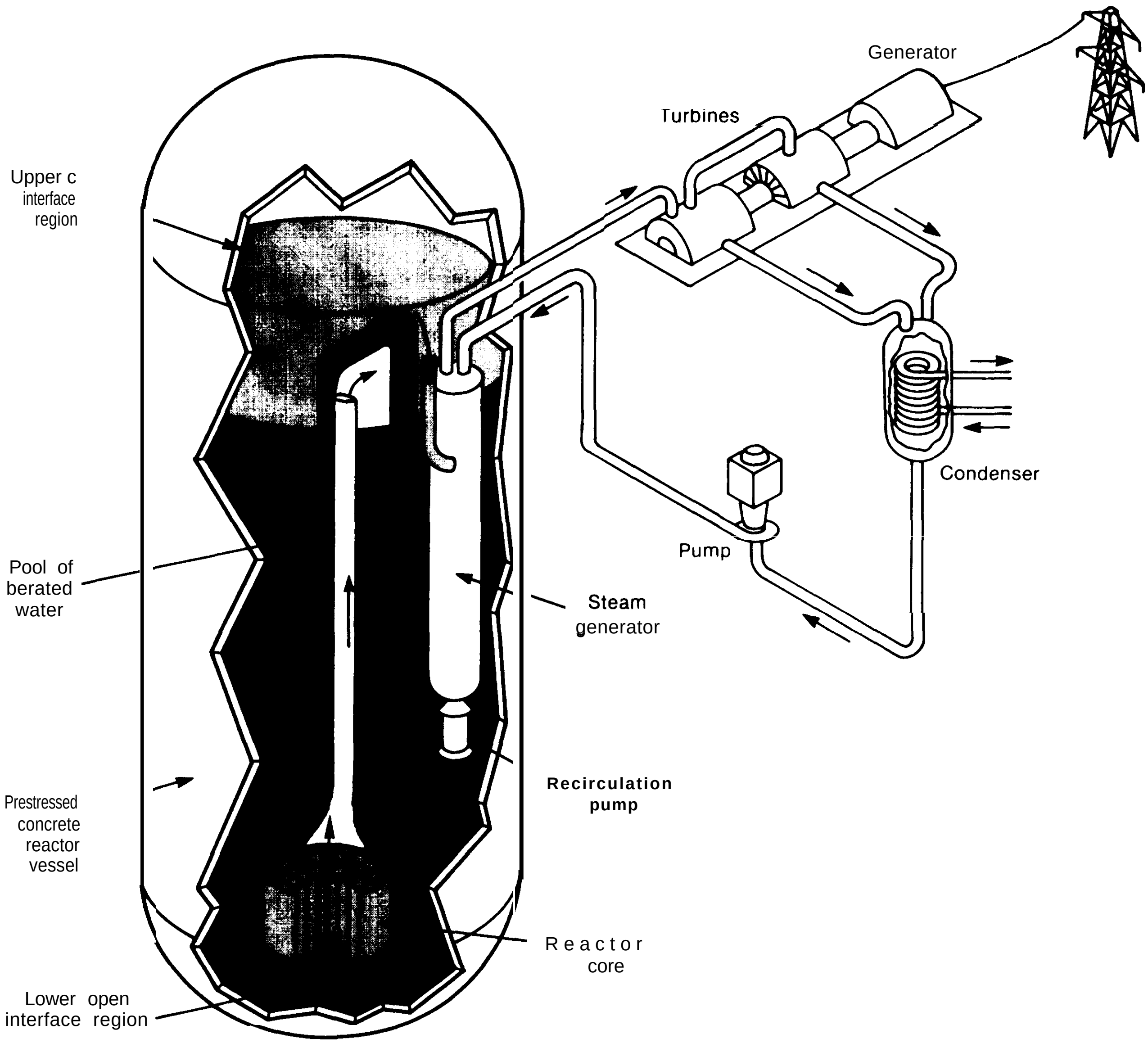
The basic configuration of the PIUS reactor is similar in many ways to that of a conventional PWR. It employs a primary loop of pressurized water that transfers heat to a secondary steam loop through a steam generator. The main differences between the two reactors are the size of the pressure vessels and the location of the primary system components. In a PWR, the fuel is surrounded by water and enclosed in a pressure vessel that is slightly larger than the core; the primary system pump, steam generator, and piping are located outside the vessel. In the PIUS reactor, the core is located at the bottom of a very large pool of water. As shown in figure 26, the pool and core are enclosed in an imposing concrete vessel (13 meters in diameter and 35 meters high) that is reinforced with steel tendons. In the PIUS reactor, the other primary components are submerged in the same pool of water that contains the fuel, and all penetrations to the PCRV are located at the top of the vessel. With such a configuration, it is impossible for any type of leak or equipment malfunction to drain off the cooling water.

During normal operation, primary coolant is pumped through the core and primary loop. At the bottom of the plenum under the reactor core there is a large open duct extending down into the pool. The pool water ordinarily is prevented from flowing into the core circuit by the difference in density between the hot water within the core and the cool pool water. At the static hot-cold interface in the duct, a honeycomb grid helps prevent turbulence and mixing between the hot and cold fluids. If for any reason the core temperature should rise to the point at which steam is formed, the pressure balance at the interface would be upset, and the pool water would automatically flow upward and flood the core. Natural thermal convection through the pool would provide enough circulation to cool the core, and the pool water would keep the core covered for about a week. This system relies completely on thermohydraulic principles and is totally independent of electrical, mechanical, or human intervention. The principal uncertainty in this reactor concept is in the stability of the pressure balance between the hot primary circuit water and the cold berated water.

The PIUS reactor is designed to automatically shut down as soon as the pool water begins to flow through the core. This is guaranteed by maintaining a high concentration of boron in the pool water; boron is a strong neutron absorber and automatically interrupts a chain reaction. This feature of the PIUS reactor is attractive because the possibility of a transient without a subsequent reactor scram is virtually eliminated.

If the PIUS reactor proves to be reliable, it might resolve some of the troublesome problems with LWRs. With inherent mechanisms for automatic shutdown, natural convective cooling, and a

Figure 26.—Process-inherent Ultimately Safe Reactor



SOURCE: ASEA-ATOM and Office of Technology Assessment

large heat removal capacity, there do not appear to be any credible mechanisms for uncovering or overheating the core of the PIUS reactor. However, the changes from the standard LWR design create new problems and uncertainties. The design of the **PIUS is different enough from current LWRs that further development will be required** for components, materials, and procedures. The design and construction of the PCRV will pose a problem, since it is larger than any other similar vessel. The steam generator in the PIUS reactor also will require additional development since it will be of a different configuration than in conventional PWRs. More importantly, maintenance may be very difficult since the components will be submerged in a pool of berated water. It is therefore essential that all primary system components perform reliably and with little maintenance. The submerged components and piping pose another problem—since the pool water will be about 380° F less than the primary system coolant, it is necessary to insulate the primary coolant loop from the borated pool water. The insulation for such an application has not yet been fully developed or tested. Another potential difficulty relates to fuel handling. Nuclear fuel assemblies are removed and exchanged routinely in current LWRs, but a similar operation in the PIUS reactor is complicated by having to work at a distance of 80 feet.

Another serious concern relates to the thermal hydraulic response of the reactor. There is considerable uncertainty about the flow patterns in the lower interface region between the pool water and the primary coolant. If the boundary is not stable, normal operations could be interrupted unnecessarily by the inflow of borated pool water through the core, shutting the reactor down. Computer simulations have been performed by ASEA-ATOM to determine the characteristics of the interface region. The Tennessee Valley Authority has supplemented this with a small-scale test to observe flow patterns. The uncertainties associated with the liquid-interface region can only be resolved with larger and more definitive experiments, such as th 3MWm test planned by ASEA-ATOM.

In many respects, the PIUS design builds on demonstrated LWR technology. The fuel for the PIUS reactor is essentially the same as for LWRs. The PIUS core is designed to use burnable poisons to maintain a constant reactivity throughout the fuel cycle. These have been used extensively in BWRs, and the experience is directly applicable to the PIUS reactor. The water chemistry and waste handling systems for the PIUS are also very similar to today's LWRs. Finally, most of the materials that would be used in the PIUS are identical to those in conventional LWRs. In fact, the temperature, pressure, and flow conditions in the PIUS reactor would be less severe than in LWRs.

Overall, the PIUS reactor represents a fairly dramatic departure from conventional LWRs. One consequence of this reconfiguration is that the economics become far less certain. Because of the requirement for a large PCRV, the nuclear island of the PIUS is likely to be significantly more expensive than that of an LWR. Furthermore, the remaining technical uncertainties relating to component and materials development could be costly to resolve. The originators of the PIUS design suggest that other plant costs might be reduced by easing or eliminating the safety qualifications for the balance of plant systems because reactor safety would not be dependent on them. I n current LWRs, safety qualification of secondary and auxiliary systems contributes significantly to the overall cost of the plant. If nuclear regulators agree to such a reorientation, it is conceivable that the overall cost of the PIUS reactor would be comparable to today's PWRs or BWRs. It should be noted, however, that many nuclear-grade systems would still be required to remove, process, and return radioactive gases and liquids from the pressure vessel.

Technical and economic uncertainties are significant factors in the decision to develop any new reactor concept. **In spite of these unknowns, further development of the PIUS reactor might be warranted** due to its potential safety advantages. These advantages might restore public confidence in the safety of nuclear power, but they must be tested further before any final judgments can be made.

Regardless of the merits of this particular design, the PIUS concept illustrates that innovative revisions of the standard LWR design can emerge if designers are not constrained in their thinking. Even if the PIUS concept itself turns out to have some insurmountable problems, exploratory research might continue concerning the basic concept of inherent safety.

THE SMALL REACTOR

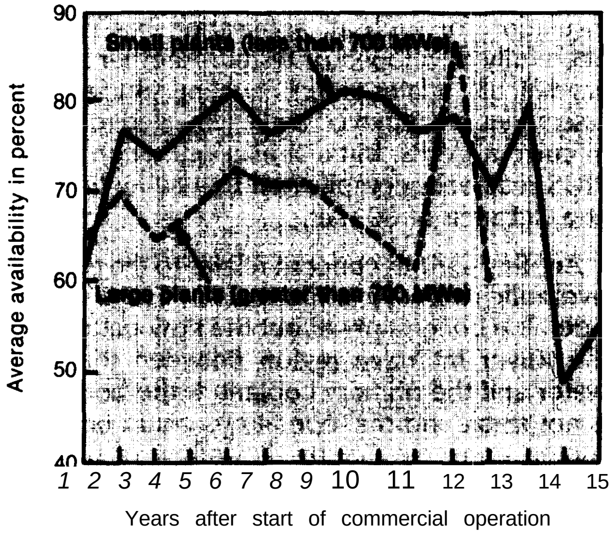
Considerable sentiment is often expressed in favor of reactors that are smaller than the 1,000- to 1,300-MWe LWRs that now represent the norm. When the current generation of reactors was being designed in the late 1960's, it seemed natural to continue scaling up the size because larger nuclear units were cheaper to build per kilowatt of capacity. Moreover, utilities were growing rapidly and needed large increments of new power. The situation is very different now. Many seem to feel that a carefully designed small reactor might be easier to understand, more manageable to construct, and safer to operate. Although many of these claims seem intuitively convincing, they are difficult, if not impossible, to substantiate. OTA sponsored a search for evidence that small plants have any advantages over large plants in terms of safety, cost, or operability (20). This search revealed no firm statistical data in support of the small reactor, although it summarized some of the arguments that make it an attractive concept (see vol. II).

Utilities may find small plants especially appealing today because they allow more flexibility in planning for the total load of the utility. In addition, the consequences of an outage would have a smaller impact on the overall grid. Furthermore, reducing the size of plants would limit the financial exposure of the utility to loss and increase overall system reliability. Initially, small plants appear to suffer a disadvantage in unit construction costs since they cannot realize the full benefits of economies-of-scale. However, more of the plant could be fabricated in the factory rather than constructed in the field, and this could result in large cost savings if the market is large enough to justify investment in new production facilities. Moreover, the construction times for small plants would probably be much less than for their larger counterparts. Overall, it is not clear that small plants would necessarily be more expensive than today's large ones.

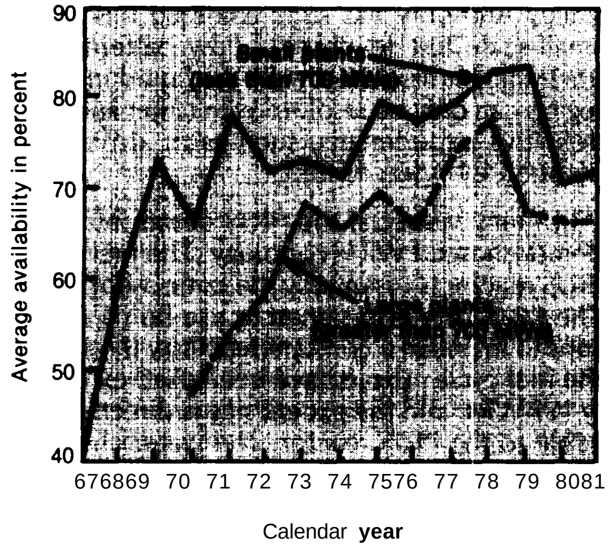
The operability of different sized plants maybe compared on the basis of availability. **As** shown in figure 27A & B, the availability of small plants generally exceeds that of currently operating larger plants, although only by about 5 percent. This trend surfaces both when availabilities are plotted as a function of the number of years after start of operation (which compares plants of the same age) and as a function of calendar year (which compares plants operating in the same environment). These differences could be due to either the number or duration of outages at smaller plants, which indicates that small plants

Figure 27.—Nuclear Reactor Availability

A.—As a Function of Age



B.—As a Function of Calendar Year



may be easier to control or maintain. However, this comparison is not conclusive since most of the smaller reactors were designed and built in the 1960's and have not been affected by as many design changes as the newer, larger plants.

It is difficult to compare the safety of small and large reactors, but one indication is the occurrence of events that could be precursors to severe accidents. It appears that the frequency of such events is independent of reactor size. However, the initiating events that do occur at small plants may be easier to manage than similar events at large plants. This result is based on too small a sample to be conclusive, but it may warrant further study. Another safety comparison can be made on the basis of the consequences of an accident. The worst-case accident in a small plant would be less damaging because the fission product inventory is much less in a smaller plant. This effect, however, might be offset by the larger number of small units needed to comprise the same generating capacity.

Small reactors are unlikely to be able to compete commercially with their larger counterparts unless R&D that specifically exploits the potential for modular, shop fabrication of components is sponsored. This would allow small plants to take full advantage of the increased productivity and quality of work in a factory setting.

THE STANDARDIZED REACTOR

The concept of a standardized reactor has been widely discussed for years in the industry but has yet to become a reality (18). The advantages are many: more mutual learning from experience among reactor operators, greater opportunity for indepth understanding of one reactor type, and sharing of resources for training operators or developing procedures. Since much of the concern over current reactors centers on their management rather than on their design, the opportunity to concentrate on learning the correct application of one well-understood design is appealing.

Utilities and vendors would be especially enthusiastic about standardized designs if that concept were coupled with one-step or streamlined licensing. The simplification of the licensing process might bring concomitant benefits in reduced capital costs. If the plants were smaller than those of the current generation, larger numbers of small plants would be built to meet a given demand, and this would facilitate standardization.

A major barrier to designing a standard plant is the difficulty in marketing identical reactors, given the current industry structure and regulatory climate in the United States. There are many opportunities for changes in today's plants, such as to match a particular site, to meet the needs of a specific utility, and to accommodate NRC regulations. In addition, the existing institutional structure does not lend itself easily to industry-wide standardization. There are currently five reactor suppliers and more than a dozen AE firms. While each reactor vendor is moving toward a single standardized design, balance of plant designs by the AEs continue to vary. It is unlikely that a single dominant plant design will arise out of all combinations of vendors and AEs, which implies that there may not be industrywide standardization. However, it is possible that a few prominent combinations of the more successful reactor suppliers and AEs will join forces to produce a more manageable number of standardized designs.

CONCLUSIONS

No single reactor concept emerges as clearly superior to the others since the preferred design varies with the selection criterion. If safety is of paramount concern, the reactors that incorporate many inherent safety features, such as PIUS or the modular HTGR, are very attractive. In such reactors, the critical safety functions of reactor shutdown, decay heat removal, and fission product containment are provided by simple, passive systems which do not depend on operator action or control by mechanical or electrical means. The full-scale HTGR is also attractive in terms of safety since it provides more time than any of the water-cooled concepts for the operator to respond before the core overheats. The remaining reactors appear to be roughly comparable regarding safety features. The HWR has the lowest inventory of radioactive materials, and the independent moderator loop serves as a passive, alternative decay heat removal system. In addition, the HWR has compiled a superb record in Canada. Advanced LWRs incorporate the benefits accrued from many years of extensive operational experience. Finally, small and/or standardized reactors may have operational advantages resulting from a better understanding of and control over their designs.

If the reactors were to be ranked on the basis of reliable operation and easy maintenance, a different order results. The advanced LWR is very attractive because these criteria have heavily influenced its design. Small reactors also appear high on the list because their size and shop-fabricated components may facilitate operation, maintenance, and replacements. HWRs rate high because they have performed well to date, and they do not require an annual refueling shutdown. The few HTGRs that are in operation have had mixed performance records, but the newest design addresses some of the problems that contributed to poor reliability. One factor enhancing overall performance is the ease of maintenance in an HTGR resulting from inherently low radiation levels. There are many uncertainties associated with the PIUS concept. It is likely to pose maintenance problems. It is also possible that the behavior of the PIUS will be erratic in normal transients, thus increasing the difficulty of operation. In other ways, however, the PIUS could be simpler to operate.

Any attempt to rate these reactor concepts on the basis of economics is very difficult. Experience with LWRs indicates that the price of facilities of the same design can vary by more than a factor of 2, so estimates of costs of less developed reactors are highly suspect. Only a few speculative comments can be made. Small reactors suffer a capital-cost penalty due to lost economies-of-scale, but it is possible that this could be reduced by fabricating more components in factories and keeping construction times short. HWRs are expected to have comparable capital costs, but their lifetime costs may be lower than those of LWRs since the **HWRs have lower fuel costs. Standardization of any of the reactors discussed would reduce costs, if the reactors could be licensed and constructed more quickly. The HTGR appears to be comparable in cost to LWRs, but there are greater and different uncertainties associated with it. It is premature to estimate the cost of a PIUS-type reactor for several reasons. First, it is still in the conceptual design phase, so types and amounts of materials cannot be determined precisely nor can construction practices and schedules be accurately anticipated. In addition, the PIUS designers are relying on low costs in the balance of plant to compensate for the higher costs of the nuclear island. It is not clear whether the balance of plant systems can be decoupled from their safety functions; the regulatory agencies obviously** will have a major impact on this decision, and hence the cost of a PIUS-type plant.

A final criterion applied to these reactors might be the certainty of our knowledge of them. How predictable will their performance be? The ranking here is almost the reverse of that for safety. Advanced LWRs are clearly superior in terms of familiarity because they have evolved from plants that have operated in the U.S. for more than 20 years. HWRs have also compiled a Icing record, but design modifications might have to be made before the reactor could be licensed in the United States. There is much less experience with HTGRs in the United States, with only a single facility in operation. The PIUS concept lags far behind the other reactors in terms of certainty since it has never been tested on a large scale.

This survey has examined many reactor concepts and found that none were unambiguously superior in terms of greater safety, increased reliability, and acceptable cost. Most represent a compromise among these factors. A few could not be adequately compared because so many uncertainties surround the design at this stage. The present lull in nuclear orders provides an opportune time to reduce the uncertainties and expand our knowledge of the less well-tested concepts. A demonstration of advanced LWRs may soon occur in Japan, and the results should be valuable input to future decisions on the LWR concept. If continued, the Department of Energy's development program on HTGRs will continue to provide information and experience that could make the HTGR a viable alternative to the LWR. It may also be valuable to examine the operation of Canada's HWRs to determine if any of their experience can be applied to U.S. reactors. If considerable sentiment continues to be expressed in favor of small reactors, some initial design work may be appropriate. Finally, a preliminary investigation of the PIUS reactor would teach us still more about a concept that is very promising. Work on this or another "fresh look" design would require government support since the existing reactor designers do not see a big enough market to support new research programs.

Until the results of future investigations are in, nothing on the horizon appears dramatically better than the evolutionary designs of the LWR. There is a large inertia that resists any move away from the current reactor types, in which so much time has been spent and from which so much experience has been accrued. However, if today's light water reactors continue to be plagued by operational difficulties or incidents that raise safety concerns, more interest can be expected in alternative reactors.

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