Chapter 3

THE NUCLEAR INDUSTRY TODAY

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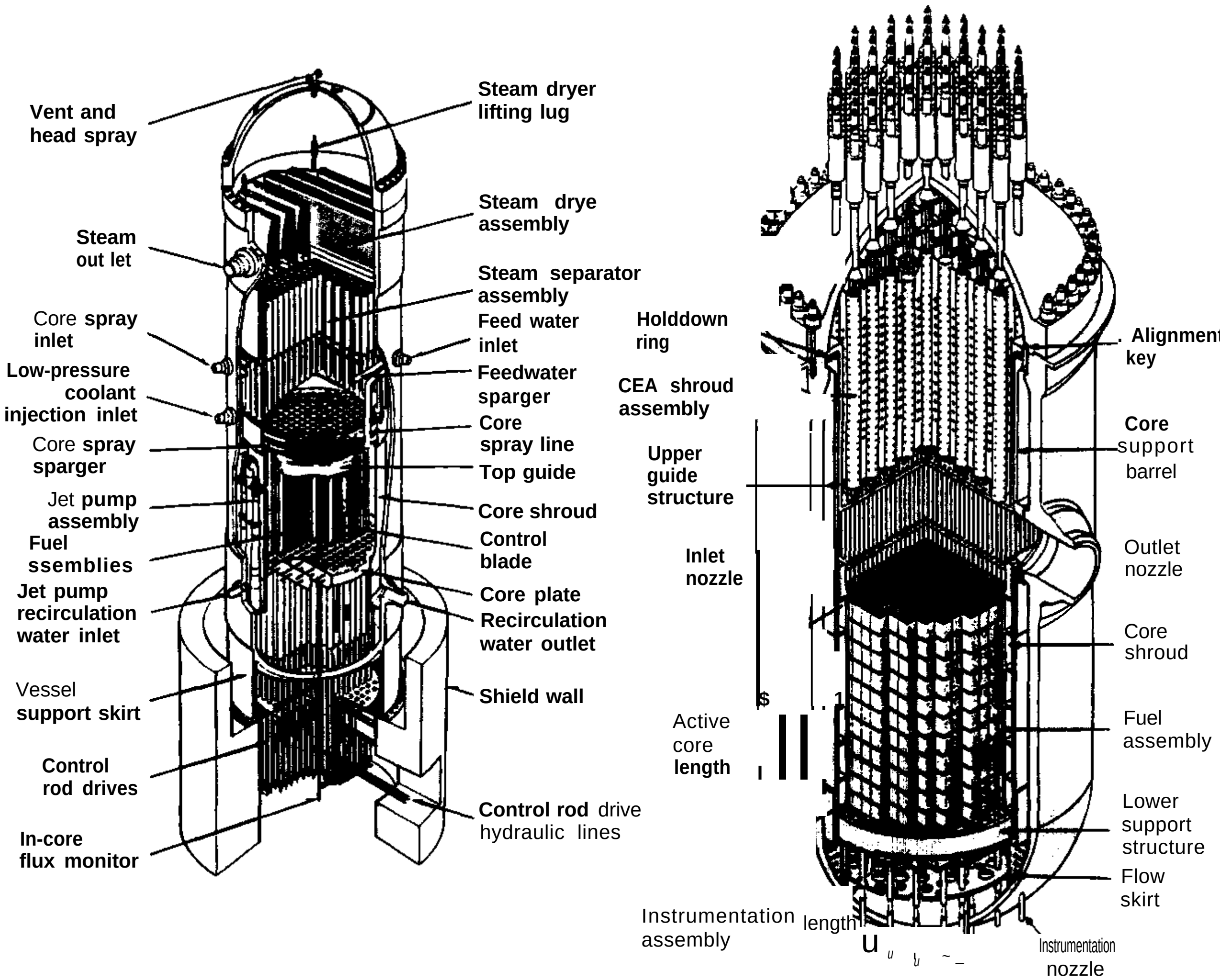
LIGHT WATER REACTORS (LWRs)

To appreciate the degree to which standard i-zation could be improved (costs, savings, and other benefits), present designs and the dif - ferences among them must be understood. A steam electric station converts thermal energy (heat) to mechanical energy and finally to electrical energy. This cycle of energy conversion is common to all central thermal generating stations and results in similar equipment being used amongst facilities (e. g., feed pumps, generators, turbines, heat exchangers, etc.). The different heat sources used are the combustion of fossil fuel (e. g., coal, oil, natural gas) and fissioning of nuclear fuel.

The heat source in a commercial nuclear plant is called a reactor cc core (fig. 1). The core consists of an array of fuel bundles (fig. 2) in a steel cylinder (the reactor vessel) capable of sustaining a controlled nuc ear reaction.

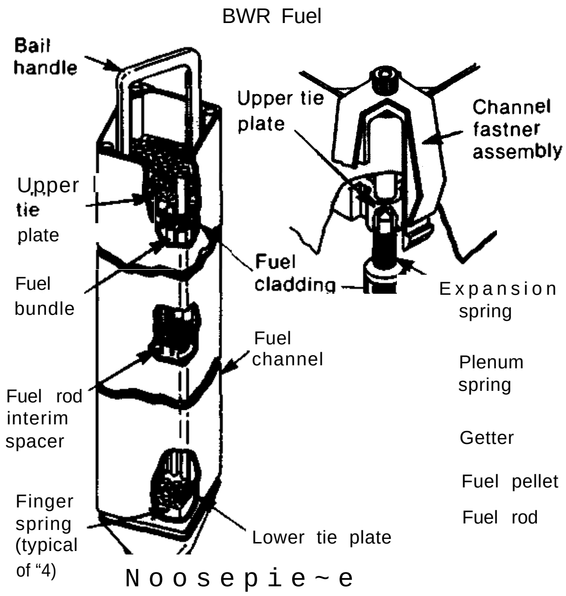
Figure 1 .—Boiling Water Reactor Core and Vessel Assembly



SOURCE: General Electric Co

SOURCE Combustion Engineering, Inc

Figure 2.— Fuel Bundles



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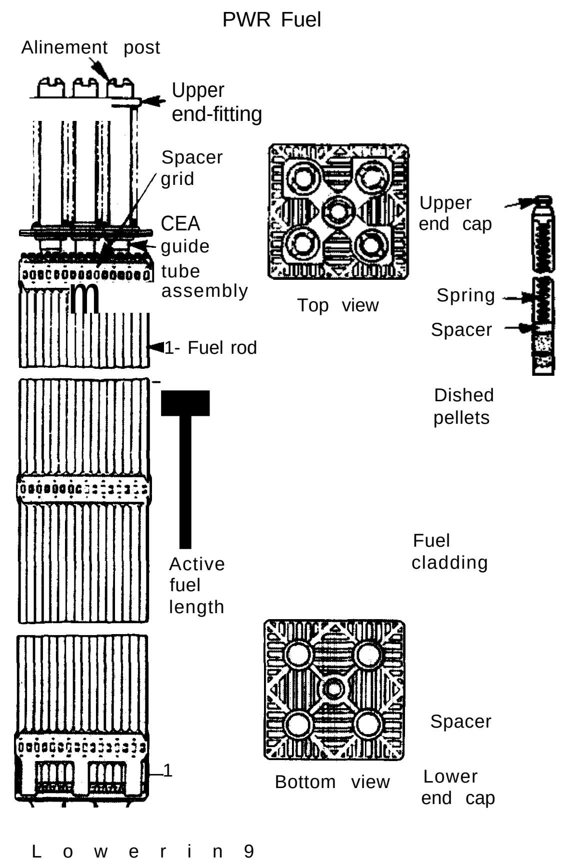


SOURCE Combustion Engineering, Inc

The fuel bundles consist of square arrays of 50 to 250 fuel rods about 1/2 -inch in diameter and 12 feet long. Each rod is filled with 1/2-inch-long fuel pellets containing slightly enriched uranium dioxide, and 200 to 500 fuel bundles arranged in a circular array form the core.

A nuclear reaction is initiated by the absorption of a neutron in the nucleus of a fissionable atom (e. g., uranium-235, plutonium-239). The fissionable atom splits, releases energy and more than one neutron. These extra neutrons are then available to produce more fissions and continue the reaction and the release of energy. This release of energy produces heat within the fuel which in turn is released to the cool ing water flowing through the core.

In a boiling water reactor (BWR), the type shown in figure 3, this coolant is allowed to boil. The steam thus produced drives a turbine, which in turn yields electrical energy. In a pressurized water reactor (PWR), shown in figure 4, the water that circulates through the core (the primary coolant) is kept under pressure and not allowed to boil. Instead, it transfers its heat in a steam generator to a secondary cooling loop. Water in this steam generator then boils, and its steam drives a turbine. In both BWRS and PWRS, the steam emerging from the turbine is discharged to the main condenser where the steam condenses and the waste heat is rejected to a heat sink such as a cooling pond or tower, The condensed steam or water then returns to the reactor vessel (in a BWR) or to the steam generator (in a PWR) to begin the cycle over again. The conversion of steam to electrical energy with turbines and generators is similar to nonnuclear steam electric stations. The systems used in this conversion are referred to as power generation or nonsafety-related. The major systems required for the nuclear heat source — including some, but not all, of the safety-related systems— are defined. by the industry as the nuclear steam supply system (N SSS).

The byproducts from the fission process include unstable nuclei (fission products) which decay to more stable nuclei by emitting an energetic particle or gamma ray. This decay proc-

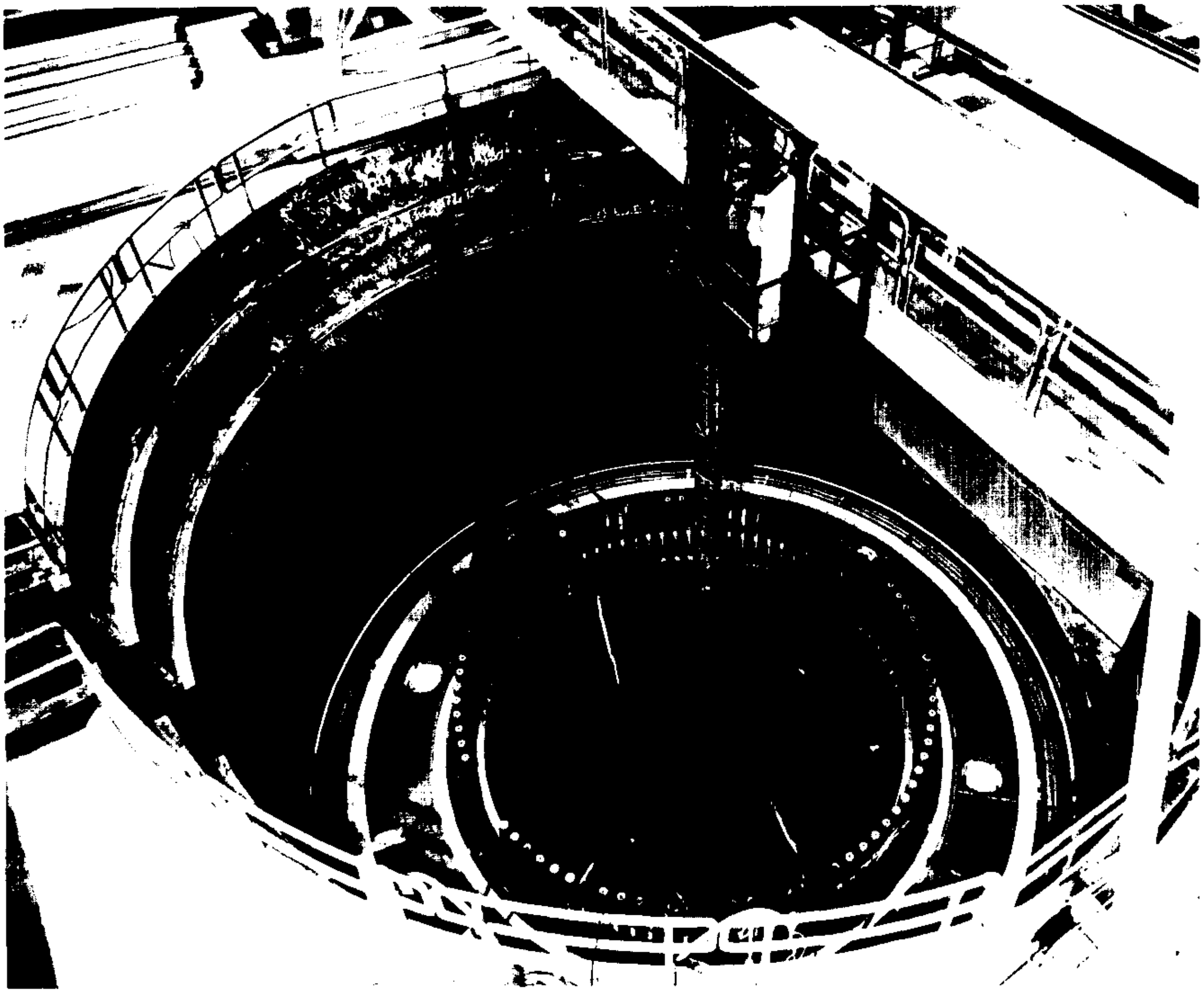


Photo credit Atomic Includes Forum, Inc

A refueling crane operator lowers a fresh fuel bundle into the core of a boiling water reactor. To the right of the fuel bundle are two of the four vessel penetrations that route steam from the reactor to the turbine

ess produces heat at a much lower rate (several percent of the fission process), but it continues even after the reactor is shut down.

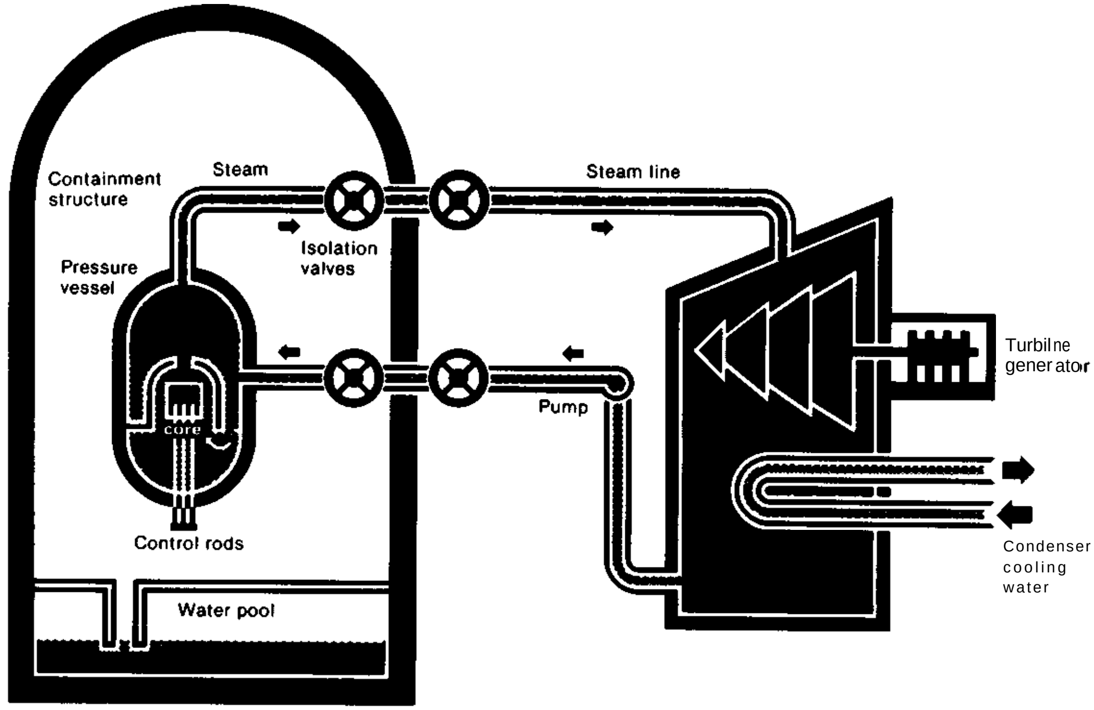
The fission process carries the unique problems of fission rate control, fission product containment, and decay heat removal. Systems normally associated with these processes are known as "safety-related" systems since they are the ones depended on to prevent or control accidents that could endanger the public. Several safety-related systems are discussed here with the purpose of understanding the relationship of safety to standardization.

Fission Rate Control

The rate of the fission reaction is controlled by materials that absorb neutrons without fissioning and, therefore, absorb the neutrons available for fission. These absorbers are commonly referred to as "poison s."

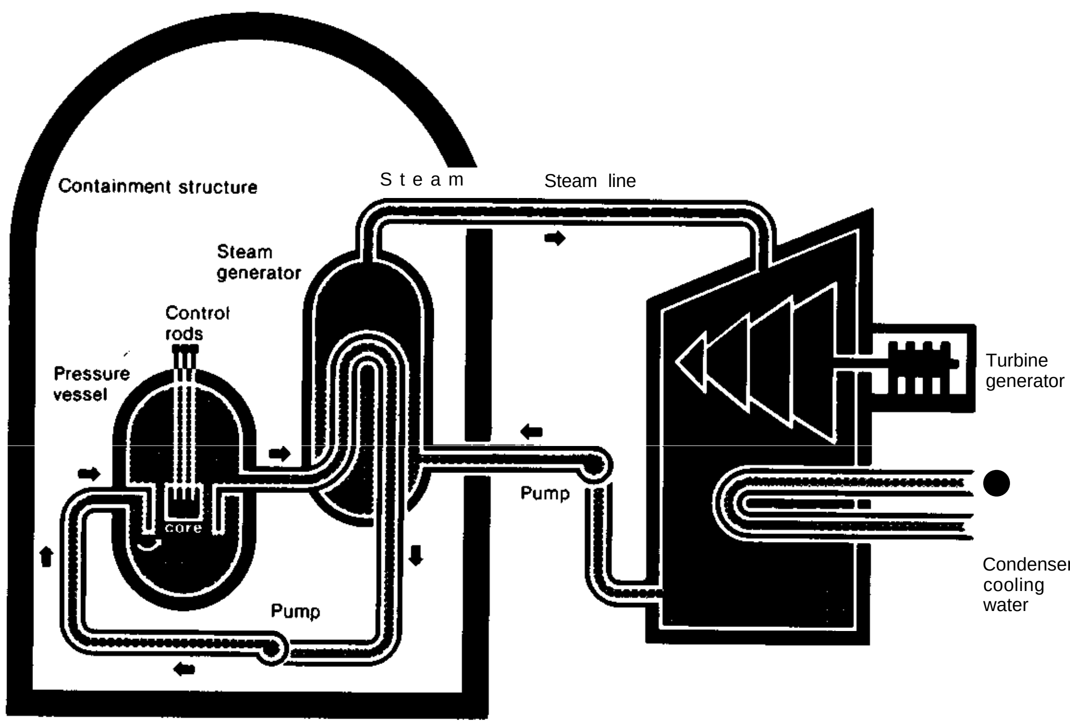
The term "control rod" refers to a mechanical device containing an absorber with a fixed

Figure 3.— Boiling Water Reactor (BWR)



SOURCE: Atomic Industrial Forum, Inc.

Figure 4.—Pressurized Water Reactor (PWR)



SOURCE: Atomic Industrial Forum, Inc.

Photo credit Atomiic Indusrial Forum, Inc

The major portions of the power conversion train are located within the turbine building. 1) Main turbine, converts steam's thermal energy to rotational mechanical energy. The thermal energy is generated in the core by fissioning nuclear fuel. 2) Main generator, converts rotational mechanical energy to electrical energy. 3) Generator alterix, maintains the generator's rotating electric field

geometric shape. Another form of poison is soluble in water and added to the primary coolant. I n pressurized water reactors these soluble poisons are used in both safety and power generation systems. In boiling water reactors they are only used in safety systems.

There are differences in designs between reactor vendors in both the control rod and its mechanical drive. PWRS use tubular control rods that are inserted into the fuel bundle. I n BWRS the control rod is in the shape of a cruciform which is inserted between fuel bundles. In either case, the rod and its mechanical drive are a "standard" design peculiar to each vendor.

If the fission rate increases above a predetermined level (greater than the rate at which heat can be removed by the coolant), the fission process is stop ped by the rapid insertion of the control rods this function is commonly called a "scram"). The systems that sense power excursions o actuate the protective systems and scram the reactor are called "reactor protection system s." These systems have undergone a careful evolutionary design change with changes in state-of-the-art electronics — e.g., one vendor has changed the system's analog signal processor to one using digital computers. Although these designs are standard to each vendor, they have not been "locked-in" to one design insulated from advances in the applicable technology.

Fission Product Containment

The radioactive fission products must not be released to the environment in excess of Federal regulations because they could harm the general public and the plant's personnel. Several barriers exist between the fission fragments and the environment. They are:

- fuel pellet;
- fuel rod (i. e., cladding);
- reactor vessel and primary coolant piping;
- primary containment; and

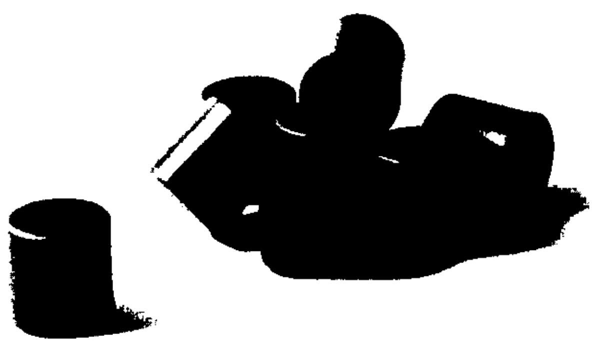


Photo credit Atomic Industrial Forum, Inc

One-half-inch long fuel pellets (< 1/2 **in.** diam.) containing slightly enriched uranium dioxide

- secondary containment (on BWRS and some PWRS).

Each barrier is a backup to the one before in the event of failure —e. g., failure of the fuel rod as a boundary is mitigated by the reactor vessel and associated piping. In addition, penetrations in the primary containment (e. g., for ventilation ducts, piping, etc.) have isolation valves (normally two) which close automatically on signals indicating potential fuel failures. The barriers listed can generally be described as passive (e. g., the fuel rod has no active components), or active (e. g., the isolation valves require motive power to shut and require process signals for automatic actuation).

During the Three Mile Island (TMI) accident a hydrogen explosion caused a pressure pulse that actuated the containment isolation system. The system's sensors and relays changed electrical states and signaled the containment isolation valves to shut. The signal was of short duration (4 minutes) and eventually cleared, allowing the operator to "reset" the containment isolation system, thereby returning the electric portion of the system to its previous "standby" state. [1] On resetting, the containment isolation valves for the containment sump opened, allowing contaminated water from inside the containment to flow to the auxiliary building. This may have caused an inadvertent release of gaseous activity into the environment through the exhaust ventilation in the auxiliary building. The simple resetting of the isolation signals should not have caused the containment valves to open.

A post-TMI requirement was to review this problem and ensure that each containment isolation system would not automatically open isolation valves when the initiating signal was reset. ~ A review of a selected number of responses to this requirement shows that this was a problem at some reactors but not at others.

This lack of standardization in containment isolation systems required a detailed review of each plant's containment isolation system and resulted in a unique fix for each similar problem that was discovered. The Nuclear Regulatory Commission (NRC), in turn, had to stretch its limited resources to review each design to determine whether or not a modification was required. This lengthened the time and reduced the depth of the review.

Auxiliary Feedwater Systems (PWRS only)

Auxiliary feedwater (AFW) systems are designed to remove decay heat when the reactor is shut down but at high pressure (normally greater than 400 lb/in ^{2}). The design criteria for them are usually established by the NSSS vendor while the detailed design responsibility usually rests with the architect engineer (A E). AFW systems are required to be available on loss of main feedwater. The inadvertent isolation of this system was a possible contributor to the accident at TMI. Valves on the outlet of the pumps were found shut and they isolated the pumps from the steam generators. The operator eventually opened these valves (approximately 7 minutes into the accident). [3]

In addition, the unavailability of a plant's AFW system is an important and significant contributor to the overall risk of any particular PWR. As mentioned earlier, the generation of heat from fission products must be removed or dissipated to ensure that the integrity of the passive containment boundaries is maintained. I n a PWR, the methods available at high reactor pressures for decay heat removal are the AFW system; some PWRS are also able to use an alternative method incorporating the high--pressure injection pumps. ' The former method is preferable because the AFW system is on the nonradioactive side of the plant. The latter method is often called "feed and bleed" and may requ ire discharging radioactive primary coolant onto the containment floor. The latter

' E lectrlc Power Research Institute, Nuclear Safety Analysis Center, "Analysis of Three Mile Island, Unit 2 Accident, " NSAC-1, july 1979

'Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TM I-2 Accident, " VOI 1, NUREG-0660, January 1980

'Electric Power Research Institute, op cit

'Nuclear Regulatory Commission, "Generic Evaluation of Small Break Loss of Coolant Accident Behavior ın Babcock & Wilcox Design 177-F Operating Plants, January 1980 was the primary heat removal mechanism during the initial phases of the accident at TMI. [5]

In response to TMI, NRC conducted a detailed review of AFW systems in PWRS to identify deficiencies in existing systems by assessing their relative reliability under loss of main feedwater. The results of a portion of the study are presented in table 1 and figure 5.[6] Table 1 shows the diversity in an AFW system for one PWR vendor. Note that only one plant had automatic system initiation and most plants differ in the number of pumps of each type. A direct result of this diversity is shown in figure 5, Quantitative reliability assessments on 33 existing AFW systems show there is a wide spread in the likelihood that the AFW system will fail on the interruption of main feedwater. As with the primary containment isolation problem, the design solutions to this problem are many and have very few elements in common. In addition, the acceptability of the system is impossible to judge in the absence of a specific reliability goal. Therefore, the design solutions are unique to each plant and subject to arbitrary judgment. If these systems were more standard than they are today, there would not be such a wide divergence in reliability; therefore, mandated engineered fixes to the design would be easier to implement and review.

A reduction in the diversity of AFW system designs alleviates the above-mentioned problems. Two items are encouraging in this area and illustrate the industry's progress toward standard system designs. First, a review of existing standard designs supplied and docketed by the AE's show a marked increase in standardization of auxiliary feedwater systems compared to those in existing plants—design is docketed when it is formally submitted to NRC and the administrative process for review and approval begins. Ten AEs have designed an auxiliary feedwater system that is applicable to all PWRS. Therefore, this results in a single AE's design that is applicable to all three PWRS. A further step toward standardization of auxiliary feedwater systems is the approval by the American National Standards Institute of a design standard for these systems. [7] This

[5] Electric Power Research Institute, op cit

[6] Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transient\ and Small Break Loss of Coolant Accidents in Combustion Engineering Design Operating Plants," NUREG-0635, January 1980

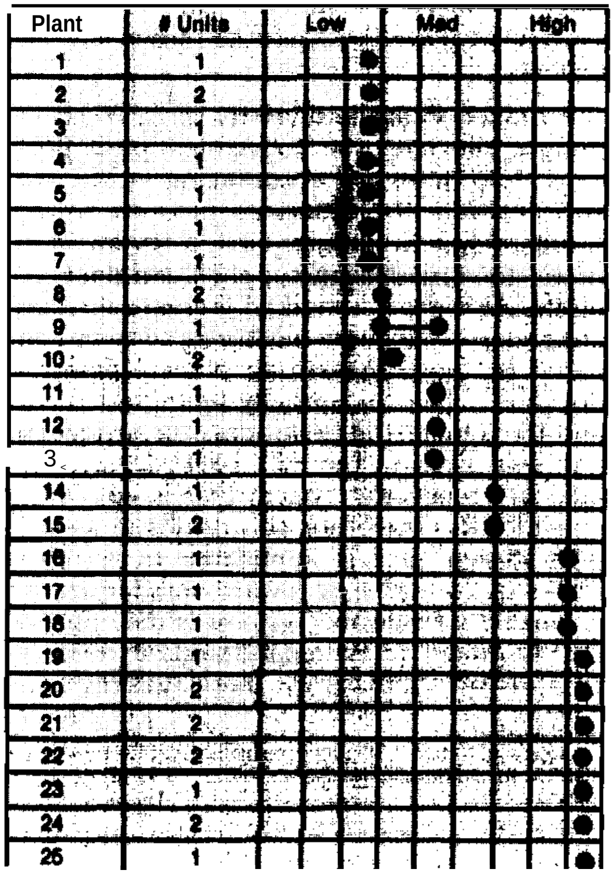
[7] American National Standards Institute, "Auxilary Feedwater Systems for Pressurized Reactors," AN SIIANS 51 10, November 1979

Table 1.—Auxiliary Feedwater Systems

Plant	AIE	Number of pumps/type of drive	Capacity	AFW system mode of initiation
Arkansas Nuclear One, Unit 2	Bechtel	1 steam-driven	Steam: 575 gal/rein @ 2,800 ft	Automatic
		1 motor-driven	Motor: 575 gal/rein @ 2,800 ft	
Calvert Cliffs 1 & 2	Bechtel	2 steam-driven per unit	700 gal/rein @ 1,100 lb/in2a each	Manual
Ft. Calhoun 1	Gibbs & Hill	1 steam-driven	Steam: 260 gal/rein@ 2,400 ft	Semiautomatic motor-driven
		1 motor-driven	Motor: 260 gal/rein @ 2,400 ft	Pump manually connected to diesel generator
Maine Yankee	Stone & Webster	1 steam-driven	**Steam: 500 gal/rein@ 1,100 lb/in2g**	Manual
		2 motor-driven	Motor: 1,500 gal/rein @ (each) 1,100 lb/in2g	
Millstone 2	Bechtel	1 steam-driven	Steam: 600 gal/rein @ 2,437 ft	Manual
		2 motor-driven	Motor: 300 gal/rein @ (each) 2,437 ft	
Palisades	Bechtel	1 steam-driven	Steam: 415 gal/rein@ 2,730 ft	Manual
		1 motor-driven	Motor: 415 gal/min @ 2,730 ft	
St. Lucie 1	Ebasco	1 steam-driven	Steam: 500 gal/rein @ 1,200 lb/in^2	Manual
		2 motor-driven	Motor: 250 gal/rein @ (each) 1,200 lb/in^2	

SOURCE Nuclear Regulatory *Commission*

Figure 5.—Comparisons of Auxiliary Feedwater System Reliability on the Loss of Main Feedwater System (LMFW)



SOURCE: Nuclear Regulatory Commision.

standard was approved late in 1979 and took about 3 years to develop through the "consensus" process. As encouraging as these items appear, they lack the quantitative reliability criteria needed to remove the arbitrariness in regulatory judgments regarding their adequacy.

Decay Heat Removal

At low-reactor pressures (less than 400 lb/in^2), redundant methods of decay heat removal prevent the uncontrolled heatup of the core. The systems remove decay heat by continuously circulating water through the core and rejecting the heat through heat exchangers to the ultimate heat sink (e. g., cooling tower, pond, lake, etc.),

The heat removal function operates in two modes: 1) "emergency core cooling" during accident conditions, and 2) normal "shutdown cooling" when the plant is not producing electricity. In the emergency core-cooling mode, the systems operate automatically to provide cooling. In the shutdown cooling mode, the operator sets up the system manually in accordance with the procedures for shutting down the plant. The design responsibility of these systems rests with the vendor. There is very little difference between plants of the same vendor. For light water reactors (LWRS) there are four basic residual heat removal designs which are standardized. These designs all comply with the "general design criteria, " which are part of the Federal code (10 CFR) governing the design, construction, and operation of commercial reactors.

However, critics of these designs have pointed out that, due to the lack of specificity in the requirements, the fundamental problem of decay heat removal during the normal shutdown cooling mode has been overlooked. [8] instead, the operator is required to use his wit and ingenuity to overcome built-in design complexities for the simple purpose of removing decay heat during plant malfunctions when a loss-of-coolant accident does not occur. A well-publicized example of this is the Brown's Ferry fire where decay heat removal depended on nonsafety-related equipment arranged in a manner not previously considered necessary for shutdown conditions. Even though these standardized residual heat-removal systems exist for both PWRS and BWRS and conform to the existing design criteria, their adequacy under nonaccident conditions is questionable. In fact, the West Germans have added to their American-designed PWRS an extra "bunkered" decay, heat removal system independent of the safety-related systems used during loss-of-coolant accidents.

As the various criteria for decay heat removal illustrate, NRC's general design criteria (supplemented by the existing standards and regu-

[8] E P Epler, "Common Mode Failure of Light Water Reactor Systems. What Has Been Learned," Institute for Energy Analysis, May 1980

lations) may not be adequate for routine operations during adverse plant conditions (e. g., a plant fire). Some suggest this deficiency results from the lack of specificity in the criteria. Therefore, standardizing designs, without increasing the level of detail in the criteria and accounting for past operating experiences, may not make future standard plants any safer than the existing operating ones. New NRC rulemaking actions in the wake of the accident at TM I point this out.

Control Room Design

Because the accident at TM I highlighted concern over operator error, greater attention is being placed on the control room design. I n the past, control room designs have varied a great deal from plant to plant. One reason was the considerable input from the utilities, which have preferred to maintain a degree of similarity between their nuclear plants and other types of power-generating plants. Even before the TMI accident, control room designs for future plants incorporated some of the following features: [9]

- consideration of functional grouping of the reactor control panels;
- location and layout of individual controls on each panel in a logical common sense manner;
- compliance with regulatory criteria for separation and installation of safety-grade control equipment; and
- utilization of state-of-the-art computer and display technology to aid the operator in the evaluation and control of the plant's condition.

Since TMI, NRC has required all operating reactor licenses and applicants for operating licenses to perform a detailed control room design review to identify and correct deficiencies.[10] These reviews, which are expected to take 1 year, are to include, among other things, an assessment of control room layout and consideration of human factors that influence operator effectiveness.

These requirements may indirectly lead to greater standardization of control room designs. Three of the four vendors offer specially designed control rooms that incorporate "human engineering" features. Most recent control room designs by AE firms incorporate some human engineering. Utilities are likely to find it too expensive to custom-design their own control rooms for new facilities. Thus, the number of different control room designs should be reduced in the future.

Causes for Variations Among LWRS

Aside from the two major types of light water reactors (BWRS and PWRS), there are many possible variations in design ranging from minor deviations in piping layouts to different numbers of steam generators to the different types of heat sink. Some of the major variations stem from the varied designs evolved by the three vendors supplying PWRS and from the range of reactor sizes desired to be built –e.g., Westinghouse reactors all have one standard loop design for the primary coolant system, and plants of different sizes result by including two, three, or four such loops in parallel. By contrast, Combustion Engineering and Babcock & Wilcox have two loops in every plant but meet different power requirements by varying the size of pumps and steam generators.

Other variations in design result from site-specific factors. Reactors built in regions subject to earthquakes must be designed for higher reaction loadings for such features as the containment structure and mechanical and electrical equipment. The meteorology associated with a particular site affects plant design (possibly mandating a secondary containment) because of concern over the patterns of dispersion of any radioactive gases released from the plant. Flood and tornado hazards may also have some effect on plant design. Duke Power has cited an example of site-specific require-

[9] "Supporting Information for the Background Papers on Nuclear Powerplant Standardization," submitted to the Off Ice of Technology Assessment, September 1980

[10] Nuclear Regulatory Commission, "NRC Act Ion Plan Developed as a Result of the TMI 2 Accident," op cit

ments that caused major divergences between two standard units that were built at its McGuire site and two units intended to be identical, but built later at its Catawba site. Differences between plant characteristics at the two sites were forced by: 1) rulings of the Environmental Protection Agency (EPA) and NRC, and 2) changes in industry standards during the period of design.

EPA required Catawba to use cooling towers rather than once-through-cooling. Cooling towers are less efficient. The additional power consumed by the fans and the higher temperature of the cooling water in the condensers affected the design of other plant systems. The overall power rating will be reduced from 1,180 to 1,145 MWe.

Duke Power also intended the decay heat removal systems for these standard plants to be the same, however, the EPA ruling cited above forced the Catawba decay heat removal heat exchangers to be larger than those at McGuire ire. In addition, NRC took a new regulatory position requiring the Catawba units to have an independent suction from the reactor coolant system for each of the two trains of decay heat removal. The McGuire ire units have a single suction supplying both trains. Finally, the industry standards changed in the time period of design of the four units, causing variation in the characteristics of such items as pumps and relief valves. ' These are a few of the many examples of similar modifications. However, such design changes may not be great enough to inhibit some of the benefits of standardized plants.

[1"]'Supporting Information for the Background Papers on Nuclear Powerplant Standardliation, " op cit

THE NUCLEAR POWER INDUSTRY

The major participants in the process of designing and constructing a nuclear power plant are the:

- electric utility;
- NSSS vendor;
- AE; and
- construction company.

The total number of companies involved may well be in the hundreds, but these four effectively control the major decisions.

If and when a utility determines that it needs new control-station generating capacity, it usually hires an architect-engineer firm to help estimate costs and other considerations of the various options. Eventually, alternative power systems —e.g., solar, wind, etc. — may be considered, but at present few utilities have any options other than coal or nuclear for large, new power supplies. The cost comparison includes fixed-price bids from some or all of the four NSSS vendors. The utility then contracts with one of the NSSS vendors to supply the nuclear components, and an AE firm (usually, but not always, the same one) to design the balance of the plant (BOP). The utility also hires the construction company (often, but not always, the AE firm). The AE and construction companies work on a cost-plus basis since it is impossible to predict in advance exactly what level of effort will be required. In some cases (usually large utilities), the utility may act as its own AE or constructor or both.

The process outlined above and the participants described below represent the industry as it operated several years ago. No plants have been ordered for several years and few are expected for the next few years. Some changes may be expected if a resurgence of orders occurs, particularly if a policy of standardization is enforced. For instance, Offshore Power Systems, a subsidiary of Westinghouse, offers a complete nuclear powerplant. In this case, the Westinghouse reactor is mounted on a barge and sold to the utility complete with all systems required to operate the reactor and generate electricity. The only AE involvement would be in site preparation. A somewhat similar scope of supply will be available at General Electric Co, (C E), which expects to offer a complete "nuclear island. " The nuclear island consolidates the GE BWR and auxiliary equipment into one standard design and includes all of the buildings and structures that have radiological significance. Some AEs offer standard BOP designs which interface with the standard NSSS.

Vendors

Four companies manufacture NSSS for nuclear LWRS. These are listed in table 2 together with the number of plants built and on order, and the total generating capacity of these plants. [12] GE makes a BWR while the other three companies make PWRS. BWRS and PWRS are clearly quite different facilities that will call for quite different systems, components, and layouts. However, the three PWRS are also quite different. The number of loops for a given power level may vary as can the size of the reactor, the means of controlling it, and the design philosophy of the systems servicing it. All three PWRS are the end product of two decades of somewhat divergent evolutionary development. Even though conceptually similar, the engineering approaches to the various design problems have been so sufficiently different that each NSSS is quite distinctive.

Since the NSSS is only one part of a large complex of systems comprising a powerplant, design of other systems may be assigned to either the vendor or the AE at the discretion of the utility owner. I n recent years however, a uniform scope of responsibility has come about through actions by NRC. When NRC [formerly the Atomic Energy Commission) was beginning to encourage standardization of nuclear powerplants in the early 1970's, it formulated a detailed program for docketing standard plants for review and approval. The vendors at that time decided to limit their scope of design responsibility to those components which they planned to market as a standardized responsibility (i. e., those components that *were* proving competitive). As a result, NRC developed the list of systems shown in table 3 as the NSSS standard plant scope to be docketed by each vendor. Note that the list of systems is largely the same for each vendor. [13]

[12] U S Department of Energy, "Nuclear Power Program information and Data, " May 1980

[13] "Nuclear Regulatory Commission, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants, " WASH-1 341 and amendment 1, August 1974

Architect Engineering Firms

The remaining systems necessary for a functioning plant are referred to as the BOP. Some AE firms in accordance with NRC's program submitted standard plant designs for the BOP. Each firm's BOP design is matched to the NSSS through "interface criteria. " The BOP designs vary from one firm to another, but each firm's BOP design is generally applicable to any PWR by adjusting parameters (e.g., pressures and flow rates) to meet the interface criteria. BWRS require a separate class of BOP designs.

The NSSS represents about 10 percent of the total plant, and the AEs design the remaining 90 percent. The cost of the plant design is about 10 percent of the total plant cost. There are also considerably more AEs than vendors,

Table 2.—Nuclear Reactor Suppliers

Manufacturer	Commercial plants Number	Commercial plants MWe	Under construction Number	Under construction MWe	On order Number	On order MWe
Westinghouse	27	20,063	38	41,454	3	2,590
General Electric	24	17,758	28	30,101	7	8,304
Combustion Engineering	8	6,361	15	17,893	6	7,490
Babcock & Wilcox	9	7,885	8	7,947	3	3,790
Other	3	1,230	—	—	—	—
Total	71a	53,297	89	97,396	19	22,174

aDoes not include Indian Point 1 or Humboldt Bay

SOURCE U S Department of Energy

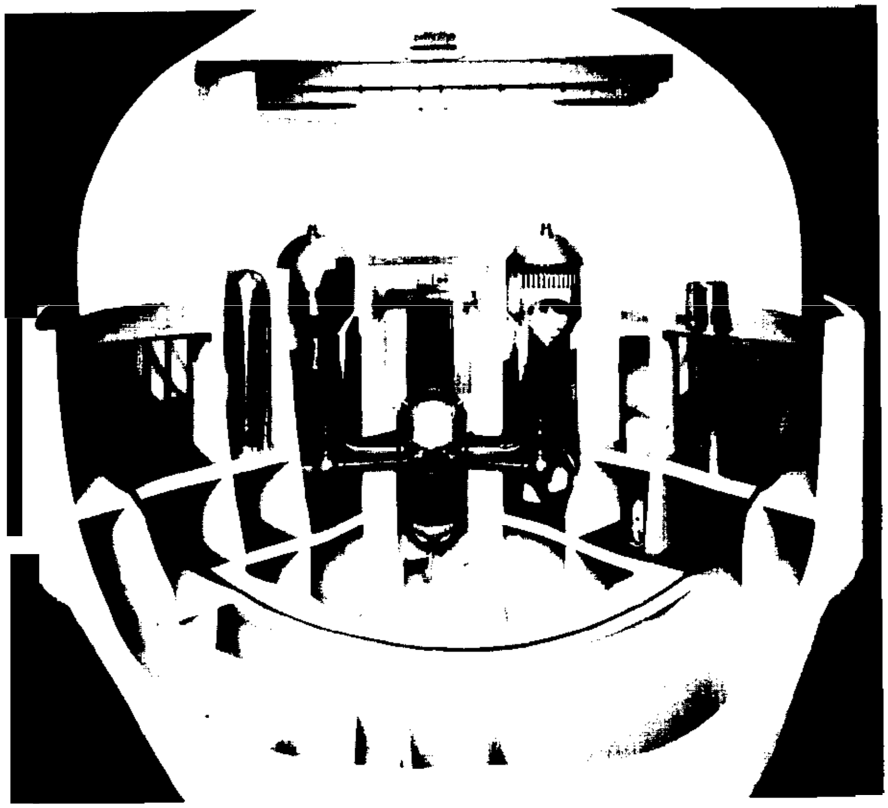


Photo credit: Combustion Engineering Inc.

Basic primary loop configuration of standard NSSS (3,800 MWth class) illustrating standard features of design

so the present diversity of designs is due more to the AEs. The AEs and their share of the business is indicated in table 4.

Experienced AEs have preexisting designs that can be tailored to specific site characteristics and utility needs. As will be discussed further, some of these designs have achieved sufficient maturity; the AEs have developed standard plants for some or all of the NSSSS. The use of such standard plants reduces the required design effort (which as stated before is a moderate fraction of the total cost) and also reduces the schedule and uncertainty of l icensing. These gains become questionable if the utility insists on too many modifications to suit its particular desires. Greater standardization would affect the relationships of involved firms by reducing the role of the utilities more nearly to that of the purchaser of a stock item.

“U S Department of Energy, op cit

Construction Companies

With only a few exceptions, the companies that build nuclear powerplants are the same AE firms that design them. Thus, they will not be identified separately here. The role of the construction company is to build the plant according to the design and specifications of the AE and the NSSS vendor. Theoretically, two plants built to the same design would be identical, but in actual fact, minor differences develop at the work site. A subcontractor may deviate slightly from his blueprint due to unforeseen interferences, buildup of tolerances, problems with field fits, or the unavailability of a component. These changes are performed under the supervision of a responsible engineer

Table 3.—Content of an NSSS Standard Design Application[a,b]

Babcock & Wilcox

A. Reactor
   1. Fuel assemblies
   2. Reactor vessel internals
   3. Control assemblies
   4. CRDMS

B. Reactor Coolant System (including layout and analysis)
   1. Reactor vessel
   2. Reactor coolant pump
   3. Steam generator (not beyond nozzles)
   4. Main piping
   5. Pressurizer (including safety valves)
   6. Pressurizer relief system
   7. Inservice inspection
   8. Equipment supports (not including embedded anchorage)

C. Emergency Core Cooling Systems

D. Instrumentation and Controls for the NSSS[c]
   1. Main control room panel board (including all integral equipment)
   2. I&C equipment racks and panels
   3. Reactor control and protection systems (including actuation systems)
   4. Nuclear Instrumentation system
   5. Process I&C (including control valves)

E. Electric Power[c]
   1. CRDM power supply
   2. Pressurizer heater controls

F. Auxiliary Systems
   1. Special handling equipment for fuel and reactor vessel internals
   2. Makeup and purification system
   3. Chemical addition and boron recovery system
   4. Steam generator circulating system
   5. Decay heat removal system

G. Startup Test Program for NSSS Items

Combustion Engineering

A. Reactor
   1. Fuel assemblies
   2. Reactor vessel internals
   3. Control element assemblies
   4. Control element drive mechanisms

B. Reactor Coolant System (including layout and analysis)
   1. Reactor vessel
   2. Reactor coolant pump
   3. Steam generator (not beyond nozzles)
   4. Main piping
   5. Pressurizer (including safety valves)
   6. Inservice inspection
   7. Equipment supports (not including embedded anchorage)

C. Emergency Core Cooling Systems

D. Instrumentation and Controls for the NSSSC
   1. Main control room panel board (including all integral equipment)
   2. I&C equipment racks and panels
   3. Reactor control and protection systems (including actuation systems)
   4. Neutron monitoring system
   5. Process I&C (including control valves)

E. Electric Power[c]
   1. Control element drive mechanism power supply
   2. Pressurizer heater controls

F. Auxiliary Systems
   1. Special handling equipment for fuel and reactor vessel internals
   2. Chemical and volume control system
   3. Shutdown cooling system

G. Startup Test Program for NSSS Items

General Electric

A. Reactor
   1. Fuel assemblies
   2. Reactor vessel internals
   3. Control assemblies
   4. CRDMS
   5. Control rod drive hydraulic system

B. Reactor Coolant System (including layout and analysis)
   1. Reactor vessel
   2. Recirculation pumps
   3. Recirculation piping and MSL piping (including but not beyond second isolation valve)
   4. Safety/relief valves
   5. Inservice inspection
   6. Equipment supports (not including embedded anchorage)

C. Emergency Core Cooling Systems

D. Instrumentation and Controls for the NSSSC
   1. Main control room panel board (including all integral equipment)
   2. i&C equipment racks and panels
   3. Reactor control and protection systems (including actuation systems)
   4. Nuclear instrumentation system
   5. Process I&C (including control valves)

E. Auxiliary Systems
   1. Special handling equipment for fuel and reactor vessel internals
   2. Standby liquid control system
   3. Reactor core isolation cooling system
   4. MSLIV leakage control system
   5. Reactor water cleanup system
   6. Residual heat removal system
   7. Pressure regulation system

F. Startup Test Program for NSSS Items

Westinghouse

A. Reactor
   1. Fuel assemblies
   2. Reactor vessel internals
   3. Control assemblies
   4. CRDMS (including missile shield and ventilation)

B. Reactor Coolant System (including layout and analysis)
   1. Reactor vessel
   2. Reactor coolant pump
   3. Steam generator (not beyond nozzles)
   4. Main piping
   5. Pressurizer (including relief and safety valves)
   6. Pressurizer relief tank
   7. Inservice inspection
   8. Equipment supports (not including embedded anchorage)

C. Emergency Core Cooling Systems

D. Instrumentation and Controls for the NSSSC
   1. Main control room panel board (including all integral equipment)
   2. I&C equipment racks and panels
   3. Reactor control and protection systems (including actuation systems)
   4. Nuclear instrumentation system
   5. Process I&C (including control valves)

E. Electric Power[c]
   1. CRDM power supply
   2. Pressurizer heater controls

F. Auxiliary Systems
   1. Special handling equipment for fuel and reactor vessel internals
   2. Chemical and volume control system
   3. Boron recycle system
   4. Emergency boration system
   5. Residual heat removal system

G. Startup Test Program for NSSS Items

[a]The items to be addressed in an NSSS SSAR are listed by major systems, components, and structures Items more detailed in nature will be handled on a case-by-case basis

[b]For each item listed, the NSSS SSAR should present the functional description, design requirements, drawings and diagrams. safety evaluation, and Interface requirements With the exception of the layout. analysis, and supports for the reactor coolant system, other design aspects such as layout, structural considerations, supports. piping analysis, protection against flooding, pipe whip, missile protection, cabling layout, ventilation requirements, Instrument cabling and piping, etc should be addressed in the BOP SSAR.

[c]Includes the equipment items only for the NSSS, not the Interconnecting piping and cabling

[d]Design provisions to accommodate Inser'vice inspection

SOURCE. Nuclear Regulatory Commission

Table 4.—Architect Engineering Responsibility for Nuclear Powerplants

	Commercial plants		Under construction		On order	
Architect Engineer	Number	MWe	Number	MWe	Number	MWe
Bechtel	27	20,099	21	22,564	6	7,494
Burns & Roe	4	3,184	2	2,163	1	350
Black &Veatch	—	—	—	—	2	2,300
Brown & Root	—	—	2	2,500	—	—
Ebasco	4	2,676	8	8,003	1	1,150
Gilbert/Commonwealth	—	—	3	3,310	—	.
Gibbs & Hill	1	457	2	2,222	—	—
Gilbert Associates	3	2,114	—	—	—	—
Fluor Power Services	3	1,595	—	—	—	—
Sargent & Lundy	8	5,626	13	13,310	2	2,240
Stone&Webster	9	5,859	11	10,797	4	4,800
United Engineers	4	3,480	4	4,836	—	—
Tennessee Valley Authority	4	4,343	13	15,896		
Utility owner [a]	4	3,864	10	11,795	3	3,840
Total	71[b]	*53,297*	*89*	*97,396*	*19*	*22,174*

[a]*Includes Niagara Mohawk Power Corp. Public Service Electric&GasCo.,* American Electric Power Service Corp Pacific Gas & Electric Co..and Duke power Corp.
[b]Does not Includ Indian Polnt I or Humboldt Bay

SOURCE" Off Ice of Technology Assessment

Photo credit Atomic Industrial Forum, Inc

A milestone in the construction on a nuclear powerplant is the setting of the reactor vessel within the containment. In this photo, a PWR vessel is being lowered into position. The steam generators have already been set in place in the background

which prevents the subcontractor from arbitrarily changing the design. However, stringent levels of standardization might frustrate such practices and lengthen the time required for construction of the plant.

Industry Trends

The roles of the utility and these three participants are not fixed. Some utilities do some or all of the AE design work themselves; the Tennessee Valley Authority (TVA), Duke Power, and American Electric Power are examples. The utility is responsible for licensing, but it can delegate the bulk of this task to the AE and vendor if it chooses. Standardization would tend to diminish utility involvement in licensing. AEs would also have a less pivotal role.

Current trends in standardization will be discussed in the following chapters, but it should be noted from table 4 that the dominance of several AEs may help ensure a certain degree of standardization even in the absence of any official action. Only four AEs (not counting the utilities) have more than four projects underway: Bechtel, Ebasco, Sargeant & Lundy, and Stone & Webster. The current experience and expertise of these four (plus one or two others) will likely attract utilities to them when and if they begin to order new plants. Any resumption of orders is likely to be at a relatively slow rate compared to the peak years of the late 1960's and early 1970's. These four to six AEs could probably handle all the renewed business, and the utilities would most likely concentrate their orders on them. In that event, the number of different possible combinations of BOP plus NSSS would be sharply reduced.

Table 5 shows the present combinations of NSSS vendors and AEs for LWRS under construction or on order. Instead of being 56 possible combinations, there are 22 NSSSIAE, plus **4** NSSS/TVA, and 2 other utility designs. ” If GE succeeds in completing licensing its nuclear island, if AEs having a smaller share of the market are excluded, and if most of the remaining ones have approved standard designs, the total number of combinations could be less than 10.

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Table 5.—NSSS/AE Combination of Light Water Reactors Under Construction or On Order

	Westinghouse	General Electric	Combustion Engineering	Babcock & Wilcox
Bechtel	6	10	6	5
Burns & Roe	—			
Black & Veatch			—	
Brown & Root	2	—	—	—
Ebasco	4		4	—
Gilbert/Commonwealth.				
Gibbs & Hill	2	—	—	—
Gilbert Associates				
Utility Owner	7	—	6	—
Fluor Power Services	—	—	—	—
Sargent & Lundy	8	7	—	—
Stone & Webster	5	6	2	2
United Engineers	2	—	—	2
Tennessee Valley Authority	3	6	2	2

SOURCE ffice of Technology Assessment