Nuclear Powerplant Standardization: Light Water Reactors

April 1981

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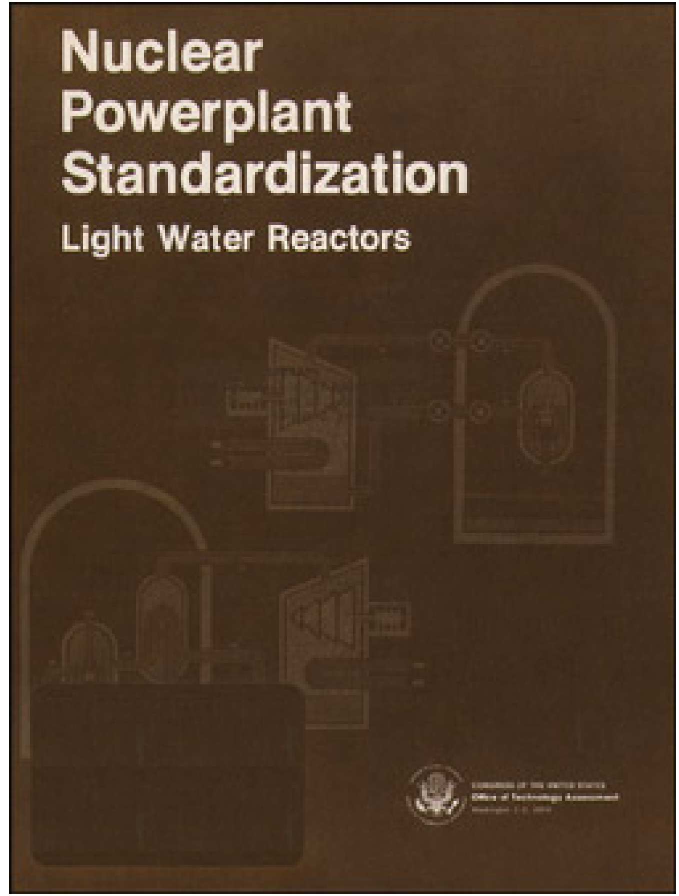

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Foreword

This assessment responds to a request by the House Committee on Interior and Insular Affairs and endorsed by the Senate Subcommittee on Nuclear Regulation to evaluate the extent to which nuclear powerplants can and should be standardized. The assessment provides the essential background material for a broad understanding of the nuclear industry, its institutions and their relationship to standardization. Items presented in the report include specific examples of the current state of nuclear powerplant standardization, and four different concepts of standardization and their potential impact on safety. These concepts represent a wide range of approaches toward standardization and would entail greatly differing requirements for industry and regulators as well as differing implications for safety.

We are indebted to the participants in the workshop, reviewers of the final report, and numerous other individuals who gave extensively of their time and talents in support of this assessment. Also, the contributions of several contractors, who performed background analyses, are gratefully acknowledged.

John H. Gibbons

JOHN H. GIBBONS
Director

Reviewers of the Nuclear Power plant Standardization Report

Robert Civiak
Congressional Research Service

L. C. Dail
Duke Power Co.

Peter R. Davis
Intermountain Technologies, Inc.

Robert L. Ferguson
Washington Public Power Supply System

James F. Mallay
Electric Power Research Institute

David Okrent
University of California, Los Angeles

Oswald F. Schuette
University of South Carolina

G. Thomas Seely
Fluor Power Services

Dee H. Walker
Offshore Power Systems

Abraham Wietzberg
NUS Corp.

E. P. Wilkinson
Institute of Nuclear Power Operations

Workshop on Nuclear Powerplant Standardization

Harold W. Lewis, *Chairman*
University of California

Myer Bender
Oak Ridge National Laboratory

Sidney A. Bernsen
Bechtel Power Corp.

Dale G. Bridenbaugh
MH B Technical Associates

Thomas Cox
U.S. Nuclear Regulatory Commission

Jesse C. Ebersole
Consultant

Darrell G. Eisenhut
U.S. Nuclear Regulatory Commission

Jerry Griffith
U.S. Department of Energy

Joseph M. Hendrie
U.S. Nuclear Regulatory Commission

Stan Jacobs
Stone & Webster Engineering Corp.

Edward O'Donnell
Ebasco Services, Inc.

John Raulston
Tennessee Valley Authority

Don Roy
Babcock & Wilcox Co.

A. Edward Scherer
Combustion Engineering, Inc.

Glenn Sherwood
Nuclear Energy Systems Division
General Electric Co.

William R. Spezialetti
Westinghouse Electric Corp.

Wayne Stiede
Commonwealth Edison

NOTE The reviewers and workshop participants provided advice and comment throughout the assessment, but they do not necessarily approve, disapprove, or endorse the report for which OTA assumes full responsibility

Nuclear Powerplant Standardization Project Staff

Lionel S. Johns, *Assistant Director,* OTA
Energy, Materials, and International Security Division

Richard E. Rowberg, *Energy Program Manager*

Edward C. Abbott, * *Project Director*

Alan T. Crane, *Program Coordinator*

Administrative Staff

Marian Grochowski   Lillian Quigg   Edna Saunders

Supplements to the Staff

William Metz   Dean Eckhoff   Barbara Levi

Consultant

NUS Corp., Rockville, Md.

OTA Publishing Staff

John C. Holmes, *Publishing Officer*

John Bergling**   Kathie S. Boss   Debra M. Datcher

Patricia A. Dyson**   Mary Harvey* *   Joe Henson

Senior Fellow, Advisory Committee on Reactor Safeguards, Nuclear Regulatory Commission
‘‘OTA contract personal

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Glossry

Architect engineer.– A supplier of design and engineering services for construction projects (e.g., powerplants, office buildings, bridges, etc.).

Auxiliary feedwater. –A standby system used to supply the secondary (nonradioactive) side of PWR'S steam generation with cooling water in the event the main source of water fails.

Balance of plant.– The equipment, in addition to the nuclear steam supply system, which is necessary to produce electricity from a nuclear powerplant.

Boiling water reactor. -A power reactor in which water, used as a coolant and moduator, is allowed to boil in the core.

Control rod. –A rod or tube containing a material that readily absorbs neutrons used to control the power of a nuclear reactor.

Decay heat.– The heat produced by the decay of radioactive nuclides or fission fragments.

Fission. —The splitting of a heavy nucleus into two approximately equal parts, accompanied by the release of a relatively large amount of energy and one or more neutrons.

Heat sink.– Anything that absorbs heat; usually part of the environment such as a river, pond, or the atmosphere.

Light water reactor.– A reactor which uses ordinary water as opposed to heavy water as a moderator and/or coolant.

Megawatt. -A unit of energy production or consumption commonly used to describe the generating capacity of a powerplant.

Moderator.—A material such as water used in a reactor to slow down high-velocity neutrons.

Nuclear steam supply system.–An arrangement of equipment with a critical array of nuclear fuel which creates high-quality steam for running turbine generators.

Pressurized water reactors. -A power reactor in which heat is transferred from the core to a heat exchanger by water kept under high-pressure to achieve high temperature without boiling,

Probabilistic risk assessment. –An approach to safety analysis which assesses undesirable consequences and their likelihood.

Radioactivity. —The spontaneous decay or disintegration of a unstable atomic nucleus accompan ied by the emission of ionizing radiation.

Reactor. -A device in which a fission-chain reactor can be initiated, maintained, and controlled.

Safety goal.– A quantitative or qualitative target for either reliability or unreliability (risk).

System.– An arrangement of equipment utilized in a powerplant for a specific function (e. g., the reactor protective system).

Vendor.– The supplier of the design and much of the equipment for the nuclear steam supply system.

Acronyms and Abbreviations

ACRS — Advisory Committee on Reactors Safeguards
AE — architect engineer
AEC — Atomic Energy Commission
AFW — auxiliary feedwater system
ASME – American Society of Mechanical Engineers
ANSI — American National Standards Institute
ASLB – Atomic Safety Licensing Board
BOP – balance of plant
BWR — boiling water reactor
CP — construction permit
DBA – design basis accident
EPA – Environmental Protection Agency
EPRI – Electric Power Research Institute
FDA – final design approval
FSAR – final safety analysis report
GE – General Electric Co,
INPO — Institute of Nuclear Power Operations
LER – licensee event report
LWR — light water reactor
MWe — megawatts electric
MWt — megawatts thermal
NRC — Nuclear Regulatory Commission
NSAC — Nuclear Safety Analysis Center
NSSS — nuclear steam supply system
OL — operating licenses
OPS – Offshore Power Systems
OTA – Office of Technology Assessment
PDA – preliminary design approval
PDDA — preliminary duplicate design approval
PRA — probabilistic risk assessment
PSAR – preliminary safety analysis report
PWC — power-worthiness certificate
PWR — pressurized water reactor
RSS — reactors safety study
SAR — safety analysis report
SDA — standard design approval
SIP — standard information package
SNUPPS – standardized nuclear unit powerplant system
TMI – Three Mile Island
TVA – Tennessee Valley Authority

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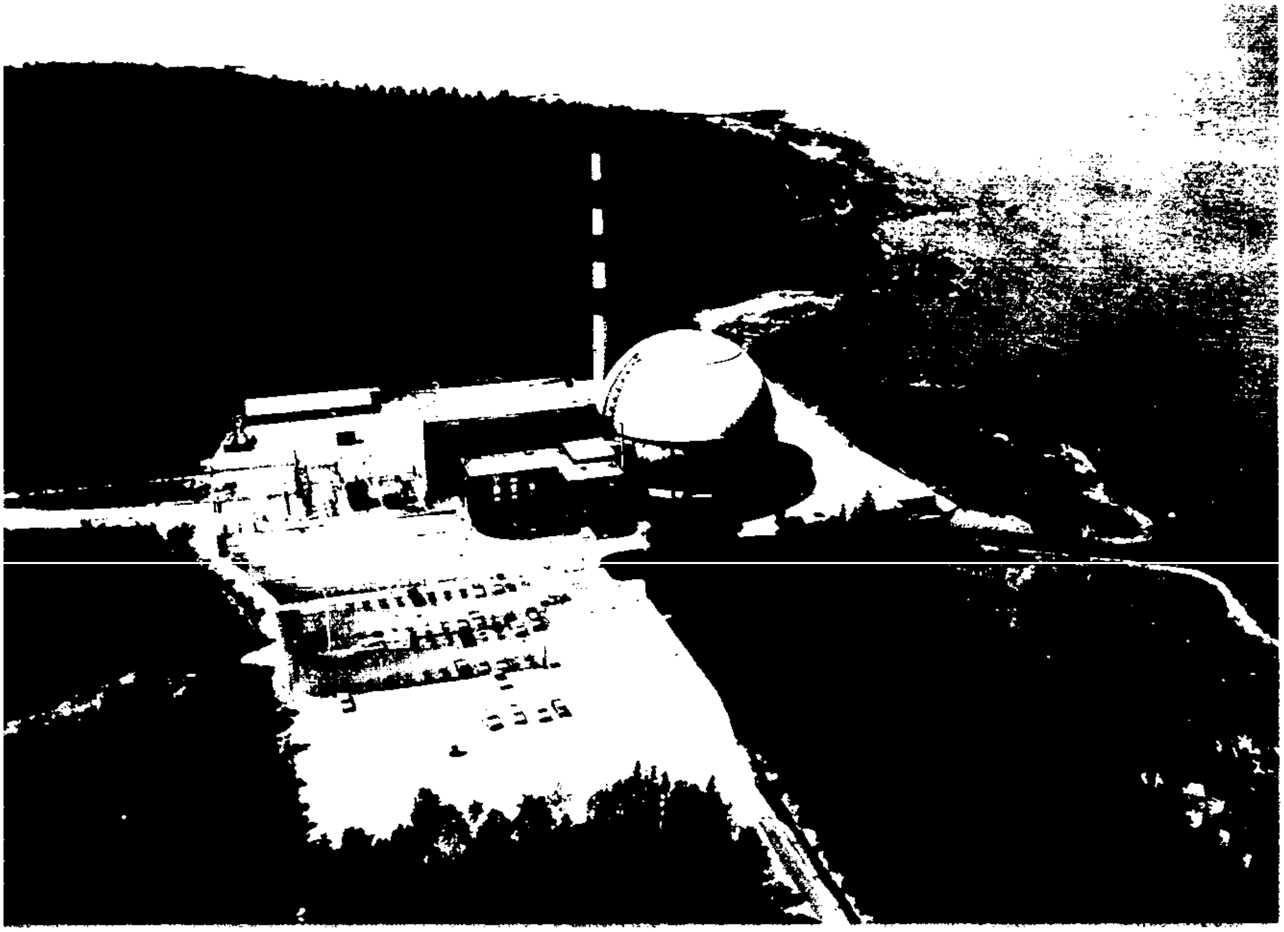
Chapter 1

SUMMARY



Big Rock Poi igh-Power Density, Boiling Water Reactor (70,300 kWe)
December 1962, Charlevoix, Lake Michigan

Chapter 1
SUMMARY

After 25 years of commercial development, nuclear power has entered a period of transition. The results of the accident at Three Mile Island (TMI) have introduced sufficient uncertainties into the industry's licensing and safety practices so that it makes it difficult, if not impossible, to get a new plant approved. At the same time, the unexpectedly low-growth rate that many utilities are encountering has deterred them from ordering any new nuclear plants for the immediate future. However, even zero growth of demand would require some new replacement facilities by the early 1990's to maintain the present generating capacity. If the uncertainties resulting from TM I are resolved soon, the nuclear industry will have a unique opportunity to reevaluate its direction and practices.

One of the peculiarities of the way that the industry has developed is that commercial reactors are built with an unusual degree of variability and diversity. Essentially every reactor, with a few exceptions to date, has been custom-designed and custom-built. The fact that almost every reactor is "one-of-a-kind" has led to excessive difficulty in verifying the safety of individual plants and identifying particular problems in transferring the safety lessons from one reactor to another. It may also account for the escalating costs and long lead-times associated with nuclear powerplants.

Many of these problems can be alleviated if the industry moves away from its "one-of-a-kind" practices toward a degree of standardization in its design, construction, operation, and licensing practices. Several types of standardization are possible, and this report examines them. Some trends in this direction are already occurring; the present lull could be used to lay the groundwork for future standardization.

A minimal level of standardization is the adoption of criteria for performance, reliability, and general design principles. This type of standardization is promoted by groups such as the American National Standards Institute and the American Society of Mechanical Engineers. At the other extreme, some fee I standardization means the selection of one complete nuclear reactor as the "standard" or model, according to which al I other reactors are to be built.

OTA evaluated four different approaches to standardization of the present generation of light water reactors (LWRS). These are:

- The acceleration *of present trends*. This wou Id entail revitalizing and streamlining the Nuclear Regulatory Commission's (NRC) current standardization program and emphasizing one-step I icensing.
- The *procedural standardization.* This means the use of universal "software practices" such as common terminology and format for plant procedures and similar requirements for the training of plant personnel.
- *The standardization of the powerplant's nuclear systern and those systerns critically necessary for the safe shutdown of the reactor— the safety-block concept.* This might include the development of similar designs for auxiliary feedwater and shutdown cooling systems.

  The selection of a single standardized design resulting from a fresh approach integrating the past 25 years of operating experience from various reactors.

This report considers these four representative approaches to standardization and examines the major advantages and disadvantages of each concept.

PRINCIPAL FINDINGS

Standardization *can be an essential element in maintaining a viable and safe program for nuclear energy.* There are relatively few plants built as examples of the approaches to standardization considered in this report, but the present trend in the nuclear industry is toward greater standardization.

Standardization yields safety benefits that are intuitive/y valid even if they cannot be demonstrated unambiguous/y. The common-sense nature of this benefit and its widespread acceptance in the nuclear industry more than counterbalance the paucity of data from the few relevant examples. However, the extreme, "single-design" approach to standardization could pose so many institutional difficulties and generic risks, that the problems would outweigh the safety benefits.

Standardization has c/ear potential for time and cost reductions and for gains in safety for new nuclear p/ants. Several utilities and utility groups have attempted to build standard plants in the hopes of shorter licensing time and reduced design and construction costs. Some improvements have been reported but there have also been problems.

Standardization is not a panacea, and the other elements needed for a safe and efficient nuclear program should not be ignored. Other elements include safe operating practices, programs for effective preventive maintenance, and direction by responsible technical managers.

Standardized plants constructed during different time periods have diverged from their original design due to the changing regulatory requirements, industrial standards, and utilities' preferences. The characteristics of different sites have dictated further divergence from original standardized designs.

The quality of the implementation of standardization is just as important as the concept itself in reaping potential benefits. A custom plant can be safer than a standard plant if it is operated and maintained in an exemplary fashion. Conversely, a standard plant will be safer only if the designers and operators are highly motivated, talented, and technically competent.

The present trends of the industry toward greater standardization will be great/y encouraged by the implementation of sing/e-stage licensing. Proposals have already been made for the one-step issuance of a standard design approval or "power-worthiness certificate" for nuclear plants, but they have not been implemented.

NRC is current/y devoting little time to the problem of nuclear powerplant standardization. The implementation of the rules and requirements resulting from the accident at TM I is occupying much of NRC's time. If standardization is to succeed at all, NRC must start planning for it now during this period of slack growth in nuclear power. They must develop plans for future standardization, including possible implementation of one-step licensing. In addition, the vendors should realize that domestic orders for nuclear steam supply systems (NSSS) may not occur over the next few years, and they should take this opportunity to review and improve their basic designs.

The adoption of *a* national safety goal is desirable. This would be a stated goal, agreed on by society through some institution — Congress, NRC — as the level of safety acceptable to the Nation. As such, it goes beyond the more general statement in the present law. The adequacy of NRC's response to the accident at TMI, in the absence of such a definition (i.e., how safe is safe enough), is impossible to assess and creates a large uncertainty in the licensing process. NRC must begin to manage its activities in a manner so that prompt and consistent decisions on safety issues can be made. Participants in the nuclear industry agree in principle, on the desirability of a safety goal.

Enhanced standardization increases the likelihood of accurate risk assessment. The only means to assure that a nuclear powerplant has achieved a quantitative safety goal is through the use of probabilistic risk assessment. Improved risk assessment under standardization is primarily due to the increased attention that can be given to a few well-defined assessments rather than many diversified ones.

The safety benefits of *improved procedures, through adoption of uniform reporting practices and industrywide participation in review of operating experience, can easily be obtained now.* Substantial benefits can also be obtained through standardization of training of plant personnel, even when considering the utility's responsibility for a diversity of plant types.

The four approaches to standardization are not necessarily mutualy exclusive and might be explored in parallel. The first two approaches– acceleration of present trends and procedural standard i z at ion — are already being pursued but could be further encouraged. They can be accomplished with little, if any, disruption of the present structure of the industry.

The second two approaches – the unification of "safety -block" systems and the adoption of a single "standard" plant design — could bring about significant and perhaps disruptive changes in the institutions of the commercial nuclear industry, The safety-block approach would transfer design responsibilities for certain safety systems — e.g., the containment — to a section of the industry not traditionally responsible for such systems. The single-standard plant approach would reduce the two major participants in the industry—vendors and architect-engineers (A Es) — to suppliers of components and engineering services for the single national design.

The second two approaches could establish more specific design criteria than currently exist and provide an "idea/" case for measuring future design criteria. The purpose would be served whether or not the more standardized plant design was actually implemented.

The U.S. Navy's experience with standardization is not directly applicable to the commercial nuclear power industry. The naval reactors program is the only U.S. example of a well-standardized program with considerable operating experience, but the principles applied in this program are not directly applicable to the commercial industry, which has a diversity of designers, AEs, and operators who function much more independently than the participants in the Navy program. The Navy's safety record is apparently due to strong central control and the greater attention that can be focused on a smaller number of reactor designs.

Standardization would aid the resolution of some of NRC's generic safety issues, while the resolution of others would be unaffected.

INSTITUTIONAL RESPONSES

Current Nuclear Industry

The tasks of design, construction, and operation are handled by diverse and independent organizations, each with its own distinctive style and mode of business. The 75 commercial reactors now operating in the United States reflect this variety. However, in recent years, the industry has begun to reduce this diversity as designs have matured and to some extent converged.

The two types of companies that together design the systems of a nuclear powerplant are the manufacturers of the NSSS and AE firms. The four NSSS vendors design and manufacture the nuclear-related systems such as the reactor vessel and core, primary cooling system, and reactor protective system. The AE firms (which number about 12) design the balance of the plant, including the piping and electrical layouts, auxiliary feedwater system, and the containment building. Both the NSSS vendor and the AE firm collaborate with the utility to produce a plant that meets the utility's specifications. In most cases, the AE firm also serves as contractor for the plant's construct ion.

In recent years, each NSSS vendor has evolved basically one design for an NSSS which varies little from one order to another. The current variety of designs is due to the larger number of AE firms than the NSSS vendors. Also, satisfying the different utilities' design specifications creates additional variety. Some AE firms have moved toward one design with an interface package to match each of the four NSSS designs. However, the designs have not moved toward greater similarity from one AE to the next. Standardization would reduce the design effort of the AEs. This wou ld not greatly reduce the total cost of the plant since such efforts account for only a low fraction of the component and construction costs, but it would affect the AE's business. Nevertheless, AEs also serve as contractors and accept some form of standardization as inevitable and in the best interests of the industry.

The NSSS vendors, AE firms, and utilities should continue to pursue a cooperative program of standardization, perhaps utilizing the current trade associations. An alternative would be the establishment of a joint utility organization that sets standards and design criteria which are more detailed than the current NRC regulations. Neither of these concepts will become a reality as long as the industry's resources are stretched to meet NRC requirements resulting from the accident at TMI.

Nuclear Regulatory Commission

Since 1973, NRC has had a program for licensing standard nuclear reactors according to one of four definitions of standardization. The industry and utilities have participated actively in hopes of a shorter and more predictable licensing process. The gains in time and manpower effort have only been marginal to date, although it may be premature to judge the program's success.

Industry observers believe that standardization will be hindered until NRC makes definitive rulings regarding which safety concerns are sufficient to warrant a design change in a standard reactor. Until that disciplined approach is achieved, no two "standard" reactors will remain alike.

The same basic criticism is leveled against NRC in both its licensing and regulatory role because it lacks clear direction for making safety rules. A long list of generic safety issues are before the Commission, and several key safety issues await the Commission's ruling. The outcomes will remain unpredictable until NRC establishes a safety goal to guide its decisions. Until regulatory and demand uncertainties are removed, no utility is likely to apply for a new license — custom or standard.

Another step NRC might consider to encourage standardization is the implementation of standard design approval, a concept for one-stage licensing (the current procedure is a two-stage process). NRC has considered the implementation of a standard design approval which would involve submittal of information that is significantly more developed than that now provided for a preliminary design, but somewhat less than that for a final design. The General Electric Co. has proposed a similar one-stage licensing program by which NRC would grant a "power-worthiness certificate" to an acceptable design.

Congressional Role

Although no legislation has emerged from Congress that directs a standardization effort, there remains considerable interest in whether standardization can improve nuclear safety. The findings of this study show that there is no quantifiable demonstration that standardization enhances safety but there is a strong "intuitive" feeling that it will. The issue then becomes the degree to which standardization should be pursued considering the tradeoffs between potential safety gains and possible costs as summarized above and discussed in this report.

If Congress chooses to pursue the third or fourth approaches to standardization, legislation will probably be necessary because neither the industry nor NRC will take these steps voluntarily. If Congress decides that the forces of the marketplace restrained by the numerous industrial standards are sufficient, then legislation mandating greater standardization is probably not necessary. Action that supports this goal, either by legislation encouraging it or setting-up incentives such as one-step licensing, could accelerate the trend and provide a clear policy statement about standardization and nuclear safety. In this connection, establishment of a nuclear safety goal, by Congress, could be an important step in encouraging standardization. Procedural or operational standardization is also being pursued by the industry and utilities. Congressional legislation is probably not necessary to achieve some degree of procedural standardization, but, again, could be encouraged by a congressional statement of national policy.

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Chapter 2
INTRODUCTION

Chapter 2

INTRODUCTION

The nuclear industry that has developed in the United States since 19s9 has grown up with a surprising degree of technical diversity. All but a handful of the 72 plants that are currently licensed for operation have been custom-designed and custom-built. A result of this practice is that the plants must also be individually licensed, since the safety analysis of each is inevitably different. When a utility decides to build a plant, it usually first hires an architect-engineering (AE) firm, then contracts with a reactor manufacturer (one of the four existing "nuclear vendors") to build the nuclear core, vessel, and control mechanisms, which represent about 10 percent of the plant investment. Each vendor has a different design for its nuclear system, so there are four different options. Then the AE designs the balance of the plant (BOP):

- cooling systems;
- feedwater systems;
- steam systems;
- control room; and
- generator systems.

There are about 12 AEs presently designing nuclear plants in the United States, and each has its own preferred approach to these various systems. The AE's approach will be tailored by past experience to be consistent with one vendor's nuclear system, but not necessarily compatible with the systems of all four. In addition to the diversity due to the different architect-vendor combinations, there is also a degree of variability due to the different meteorological, seismic, and hydrological conditions at different plant sites.

Further variability is introduced by the length of the process (12 years) and the piecemeal approach that is taken to both design and licensing. Because safety standards have grown up with the nuclear industry rather than being formulated in full and fixed fashion when the industry began, plant builders and designers have taken a "design-as-you-go" approach to new plants in order to be able to meet upgraded safety standards that might be adopted during the period a plant was under construction. For some years, the industry's practice has been to start construction with the design about 15-percent complete. On the regulatory side, the approach taken — to accommodate changing safety standards due to accrued experience and improved analysis — has been to issue plant licenses in two steps, a preliminary step sufficient to start construction and a final step necessary to start operation. Both of these practices have inevitably increased the variation from one plant to another. Even among plants intended to be identical, but started at different times, significant design differences have occurred in the final plants.

Reducing the diversity that now exists in the nuclear industry would allow increased attention to be given to improving each plant design. It would also increase the amount of operating experience that would be available for a particular design and make it possible for improvements at one plant to be immediately applicable to an entire plant family.

Efforts to encourage standardization however, have met with slow acceptance. Some argue that the many deviations from original designs that now occur before plants operate indicate that neither the technology nor the licensing process is sufficiently stabilized to support standardization. Furthermore, the nonstandardization that now exists in the industry is a direct resu lt of the diversity that exists in the marketplace, and a substantial move toward standardization could result in some restructuring of the nuclear industry.

How substantial any move toward standardization should be is one of the topics of this report. There is such a range of possible options that lie between the two logical extremes—that either all plants be different or all be the same— that four different approaches to standardization merit discussion. The different approaches represent greater degrees of standardization, the last option being a single design identical to all others in both its nuclear and BOP systems. The approaches differ in their technical, institutional, licensing, and safety implications. Some require strong legislative action, while others rely predominantly on trends already underway in the industry.

This study was undertaken by requests of the House Committee on Interior and Insular Affairs and the Senate Subcommittee on Nuclear Regulation. Some committee members expected that standardization would significantly improve the safety of the plants, and help create a stable licensing process in which utilities would have confidence that they would get their reactors approved. The accident at Three Mile Island (TMI) contributed to this expectation because both the local operators and the Nuclear Regulatory Commission (NRC) personnel seemed to lack thorough understanding of the reactor and had failed to learn from similar experiences at related reactors. Ever-increasing licensing delays, especially since TMI, reinforce the need to reexamine the merits of standardization.

Congress is not the only institution interested in standardization. NRC has also encouraged standardization although recent actions indicate that its priority at NRC has been lowered. The NRC Advisory Committee on Reactor Safeguards has maintained a strong interest in the subject. The nuclear industry has also been moving towards standardization as individual companies have filed standardized versions of their own designs with NRC. However, such efforts have been directed more at unifying current practices than at maximizing safety.

Any degree of standardization will require decisions as to the level of specification required. The standard plants that have been filed with NRC specify flow diagrams, design descriptions, and generic information, but does not include all the detailed information required to actually build a plant. Complete standardization would require considerably greater efforts before a design is approved and would allow considerably less flexibility afterwards, but wou ld result in making plants virtually identical.

This is not an exhaustive study of standardization. It is a broad scoping of four kinds of standardizations that could be considered and the major advantages and disadvantages involved in each. In addition, the study examines the standardization of procedures and organizations to see if some advantages can be gained without depending on new designs and plants. The retrofitting of existing plants to enhance standardization or safety has not been considered. OTA had staff and contractors prepare background papers on NRC policy, the U.S. Navy's experience with standardization, several plant systems that could be standardized, and the relation of standardization to safety. These background papers were distributed to the participants of a 2-day workshop held to identify and discuss the issues of standardization. The workshop included representatives of reactor manufacturers, AE companies, utilities, regulators, and concerned observers of nuclear power. This report is the result of the background papers, the conclusions of the workshop, and further information received by the staff. It has been reviewed by the workshop participants and by others.

Chapter 3

THE NUCLEAR INDUSTRY TODAY

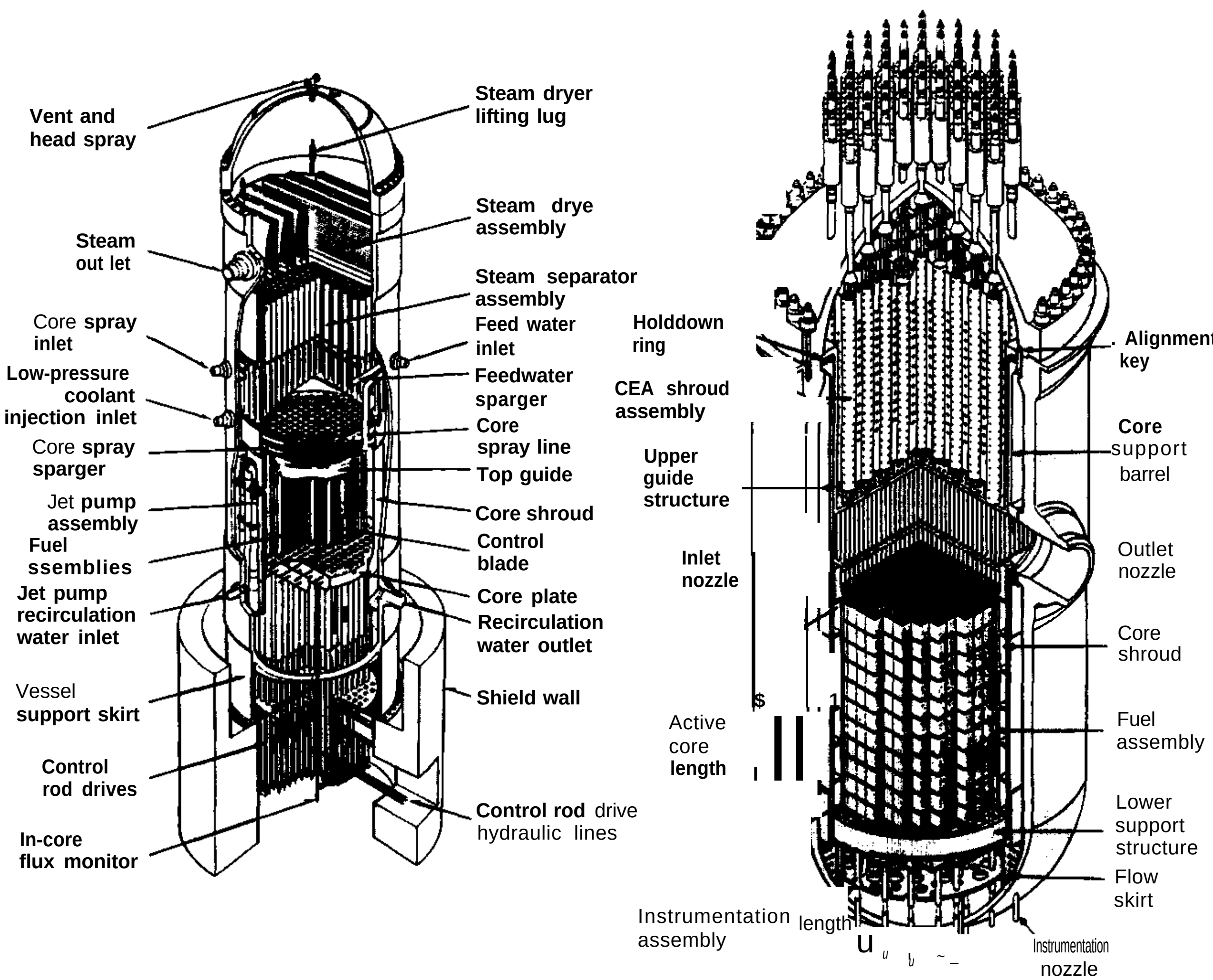
THE NUCLEAR INDUSTRY TODAY

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 elec-trical 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 fis-sioning 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.

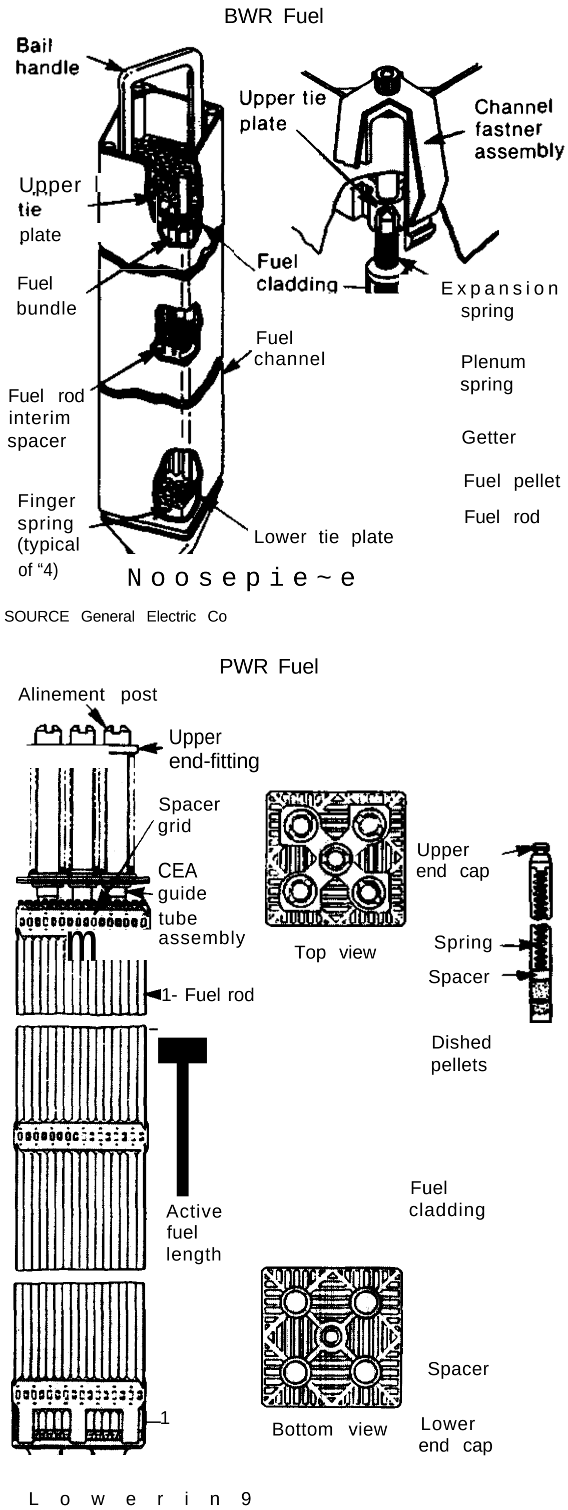
Figure 1 .—Boiling Water Reactor Core and Vessel Assembly



SOURCE: General Electric Co

SOURCE Combustion Engineering, Inc

Figure 2.— Fuel Bundles



SOURCE General Electric Co

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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-

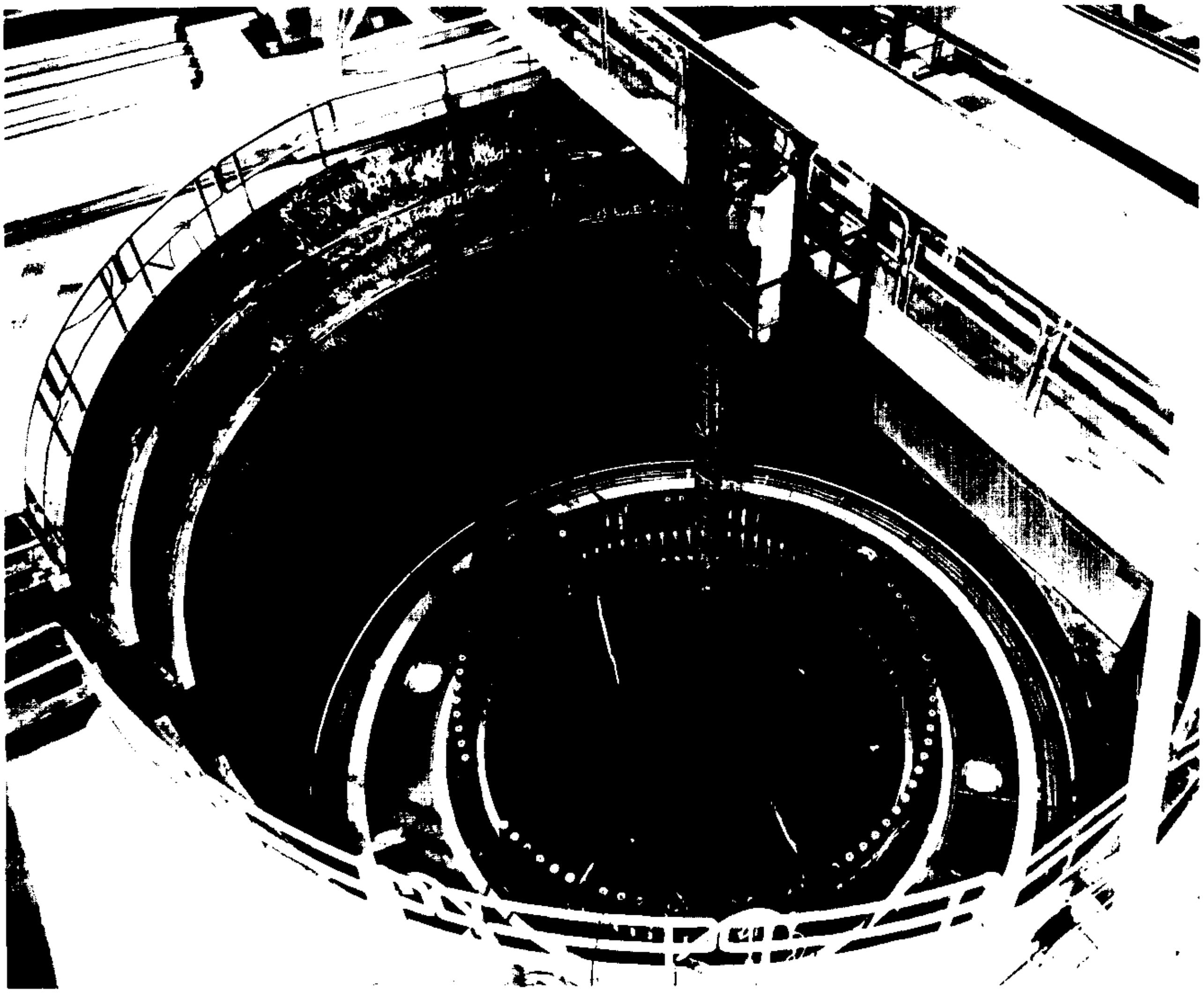


Photo credit Atomic Includes Forum, Inc

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.

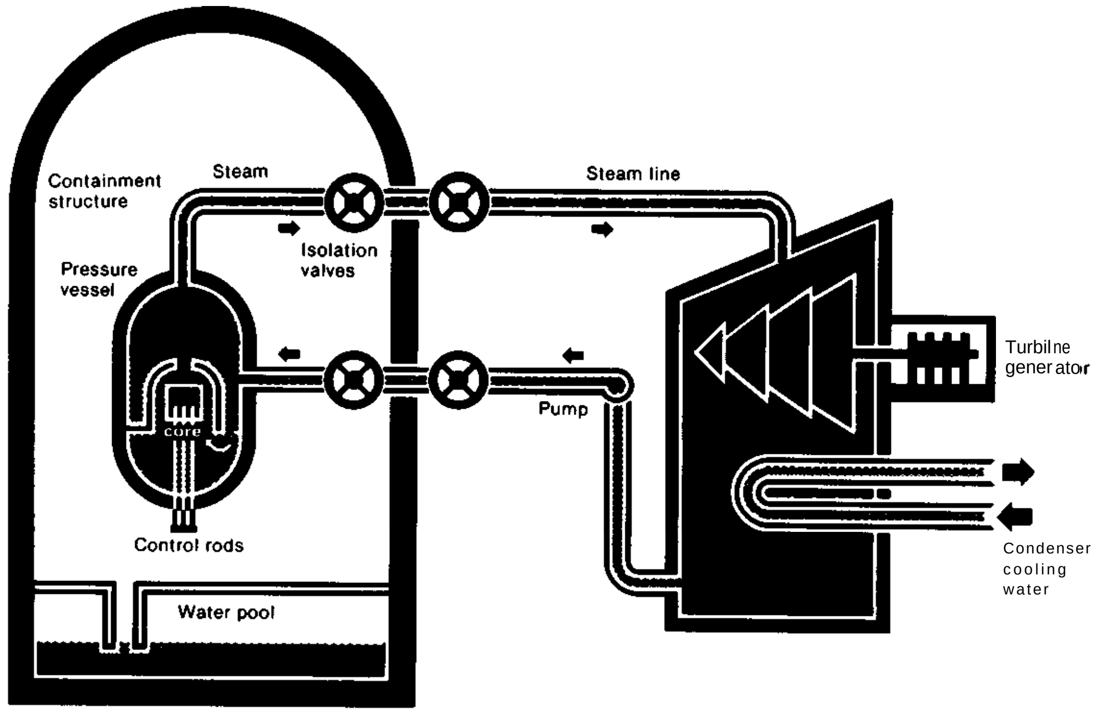
The fission process carries the unique problems of fission rate control, fission product containment, and decay heat removal. Systems normally associated with these processes are known as "safety-related" systems since they are the ones depended on to prevent or control accidents that could endanger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

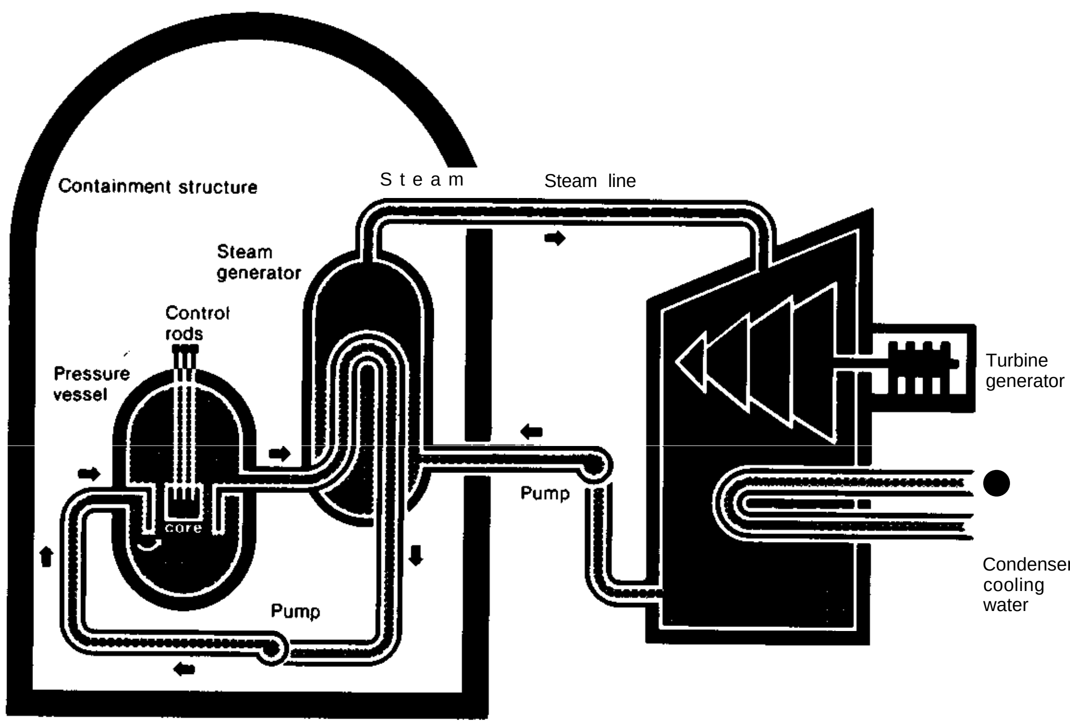
The term "control rod" refers to a mechanical device containing an absorber with a fixed

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

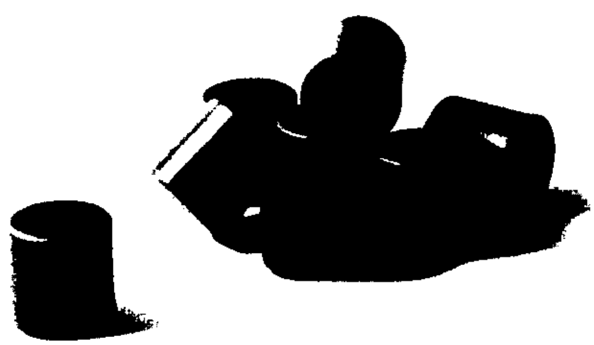


Photo credit Atomic Industrial Forum, Inc

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, NLIc lear 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, " VOl 1, NUREG-0660, January 1980

'Electric Power Research Institute, op cit

'Nuclear Regulatory Commission, "Generic Evaluation of Small Break Loss of Coolant Accident Behavior in 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

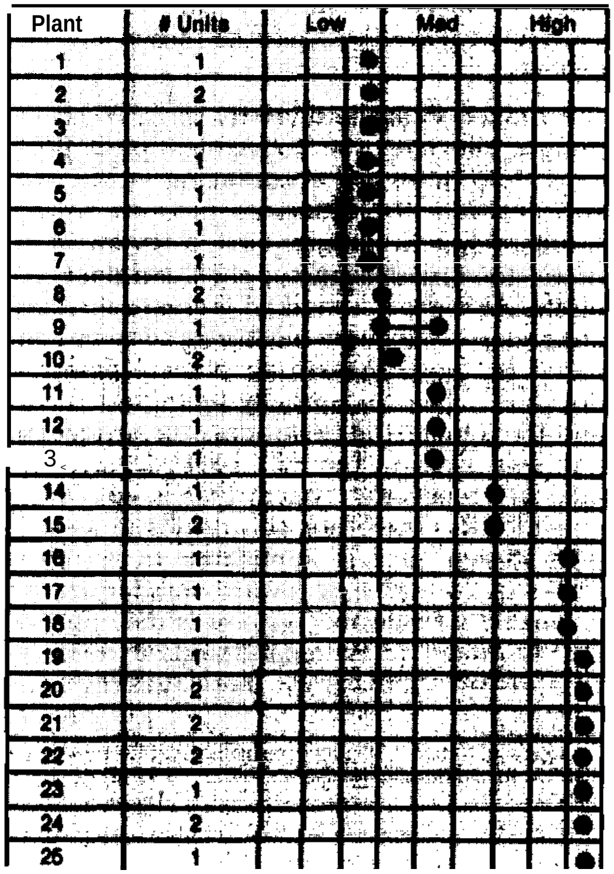
[7] American National Standards Institute, "Auxilary Feedwater Systems for Pressurized Reactors, " AN SIIANS 51 10, November 1979

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 1 motor-driven	Steam: 575 gal/rein @ 2,800 ft Motor: 575 gal/rein @ 2,800 ft	Automatic
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 2 motor-driven	Steam: 600 gal/rein @ 2,437 ft Motor: 300 gal/rein @ (each) 2,437 ft	Manual
Palisades	Bechtel	1 steam-driven 1 motor-driven	Steam: 415 gal/rein@ 2,730 ft Motor: 415 gal/min @ 2,730 ft	Manual
St. Lucie 1	Ebasco	1 steam-driven 2 motor-driven	Steam: 500 gal/rein @ 1,200 lb/in^2 Motor: 250 gal/rein @ (each) 1,200 lb/in^2	Manual

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 Office of Technology Assessment, September 1980

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

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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

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,

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

Table 2.—Nuclear Reactor Suppliers

Manufacturer	Commercial plants		Under construction		On order	
	Number	MWe	Number	MWe	Number	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	71[a]	53,297	89	97,396	19	22,174

[a] Does not include Indian Point 1 or Humboldt Bay

SOURCE U S Department of Energy

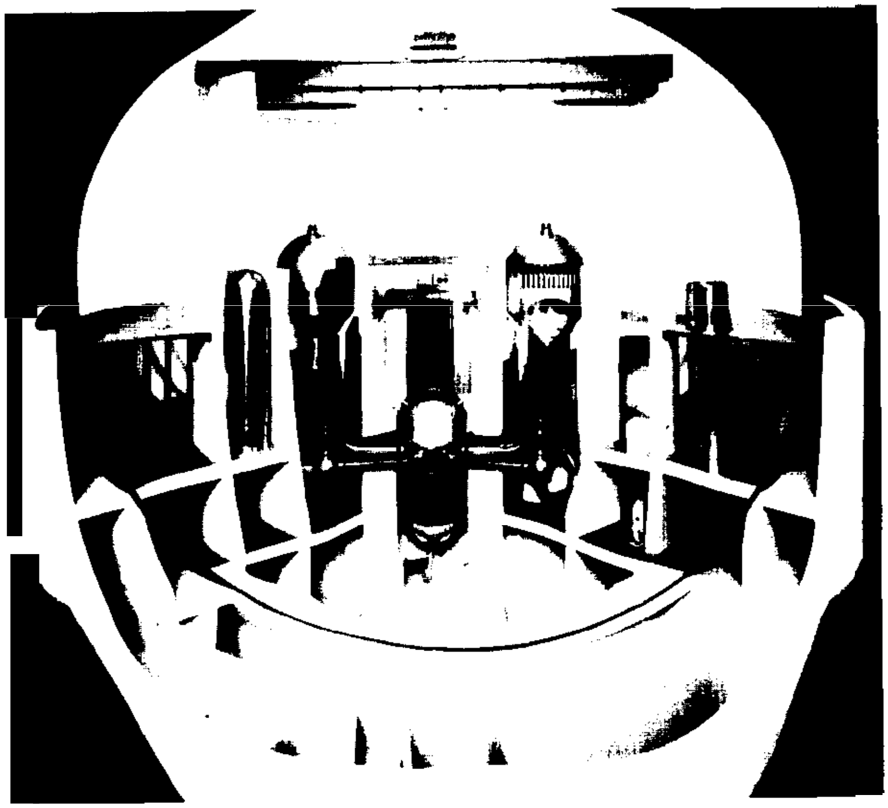


Photo credit: Combustion Engineering Inc.

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.

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

"U S Department of Energy, op cit

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 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. 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 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. 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 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. 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 Inservice inspection

SOURCE. Nuclear Regulatory Commission

Table 4.—Architect Engineering Responsibility for Nuclear Powerplants

Architect Engineer	Commercial plants		Under construction		On order	
	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.
bDoes 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.

''U S Department of Energy op clt

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

Chapter 4

THE NUCLEAR REGULATORY COMMISSION'S ROLE

Chapter 4
THE NUCLEAR REGULATORY COMMISSION'S ROLE

The Nuclear Regulatory Commission (NRC) licenses all commercial nuclear reactors and monitors them for safe operation. Thus, NRC is a natural agency both to promote standardization and to benefit from it. NRC recognizes the advantages of standardization: from its viewpoint, it would expedite the licensing process and save staff time and attention; it would enhance public health and safety; and it might benefit construction through the earlier availability of final design documents and construction experience.

A standardization program was first instituted in 1973 by the former Atomic Energy Commission. The program, with some changes, still operates under NRC today, as will be described further. Vendors, architect engineers (AEs), and utilities have participated in the standardization program, however, it has had only marginal success in reducing leadtimes or manpower efforts. At the present time, standardization is accorded low priority at NRC. In fact, all licensing at NRC is at a virtual standstill, not only because of decline of new plant orders but also because of the many uncertainties over the outcome of unresolved safety issues. These topics will form the content of this chapter.

NRC'S CURRENT STANDARDIZATION PROGRAM

All plants currently must be reviewed at both the preliminary safety analysis report (PSAR) stage and at the final safety analysis report (FSAR) stage. Thus, both custom and standard plant applications follow a two-stage review process.

For the custom plant, the utility applicant must submit a PSAR, including a general plan for the plant and many details about the particular site. If the PSAR is approved, the utility is granted a construction permit (C P). In the second stage, the utility applicant must file an FSAR that describes in greater design detail the reactor as it is actually being built. The FSAR has considerably more detail about the types and characteristics of the actual components of the balance of plant (BOP) than does the PSAR. Acceptance of the FSAR and inspection of the completed plant result in the issuance of an operating license (O L).

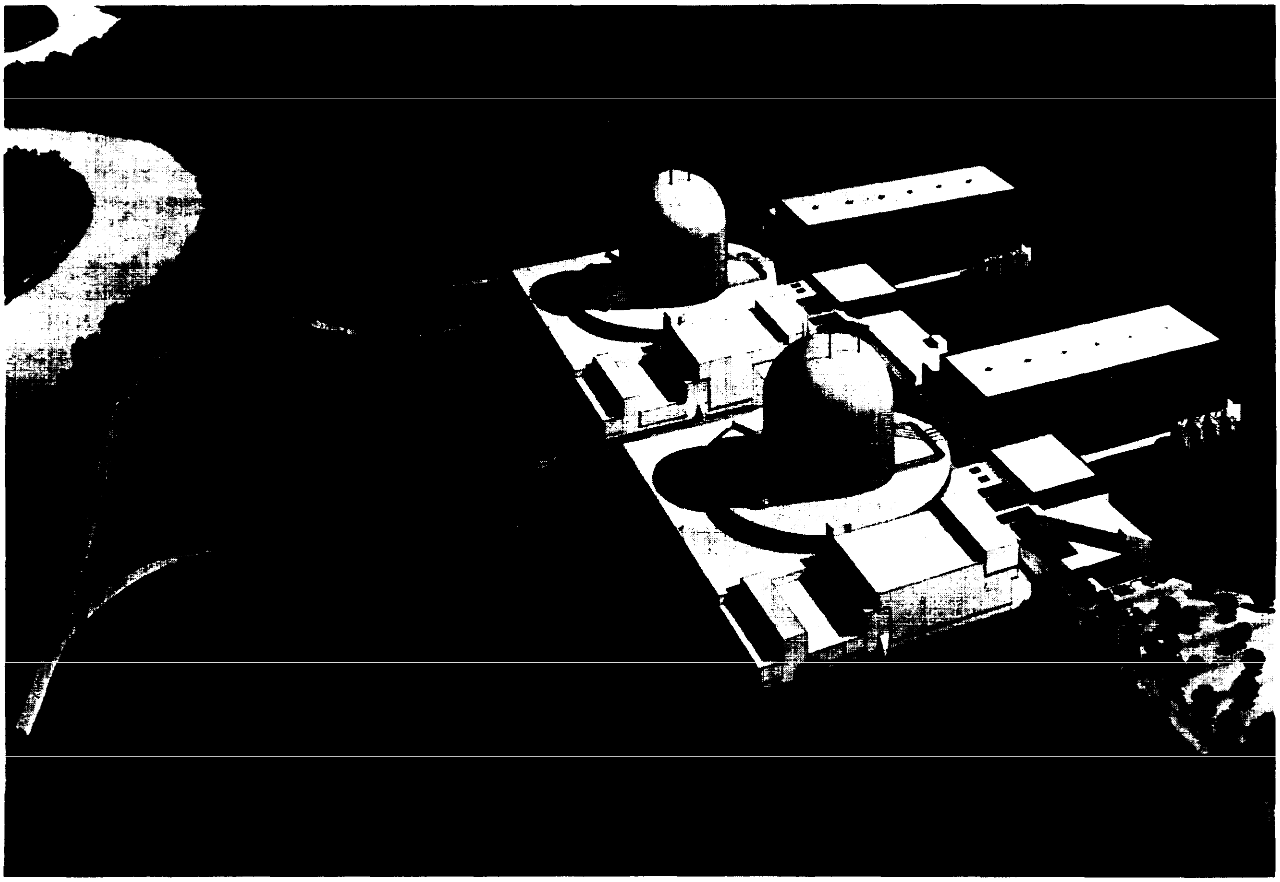
The licensing for a standard plant is also a two-stage process but may take a shortcut by one of the following four methods. [1]

Reference Plant Concept

Under this concept, a vendor or an AE firm may apply for approval of an entire facility, or a major portion of it, outside the context of a particular utility application. Once NRC reviews and accepts the reference system design, it issues a preliminary design approval (PDA). The PDA can then be referenced by a utility to build a specific plant at the CP stage.

A similiar procedure exists for the vendor or AE firm to obtain a final design approval (FDA), which can then be referenced in the FSAR submitted by the utility applicant at the OL stage. Once the FDA is issued for either a nuclear steam supply system (NSSS) andlor BOP, the PDA is no longer needed. The way licensing would then work is that the utility would reference the FDA for a CP. The utility's OL for the plant would, therefore, only require an audit of the constructed plant to assure compliance with the FDA. (The licensing of the Palo Verde plants in Arizona should give Combustion Engineering an FDA for its standard NSSS.)

[1] Nuclear Regulatory Commission, "Review of the Commission Program for Standardization of Nuclear **Power Plants and Recommendations to Improve Standardardizations Concepts," NUREG 0427, February 1978**



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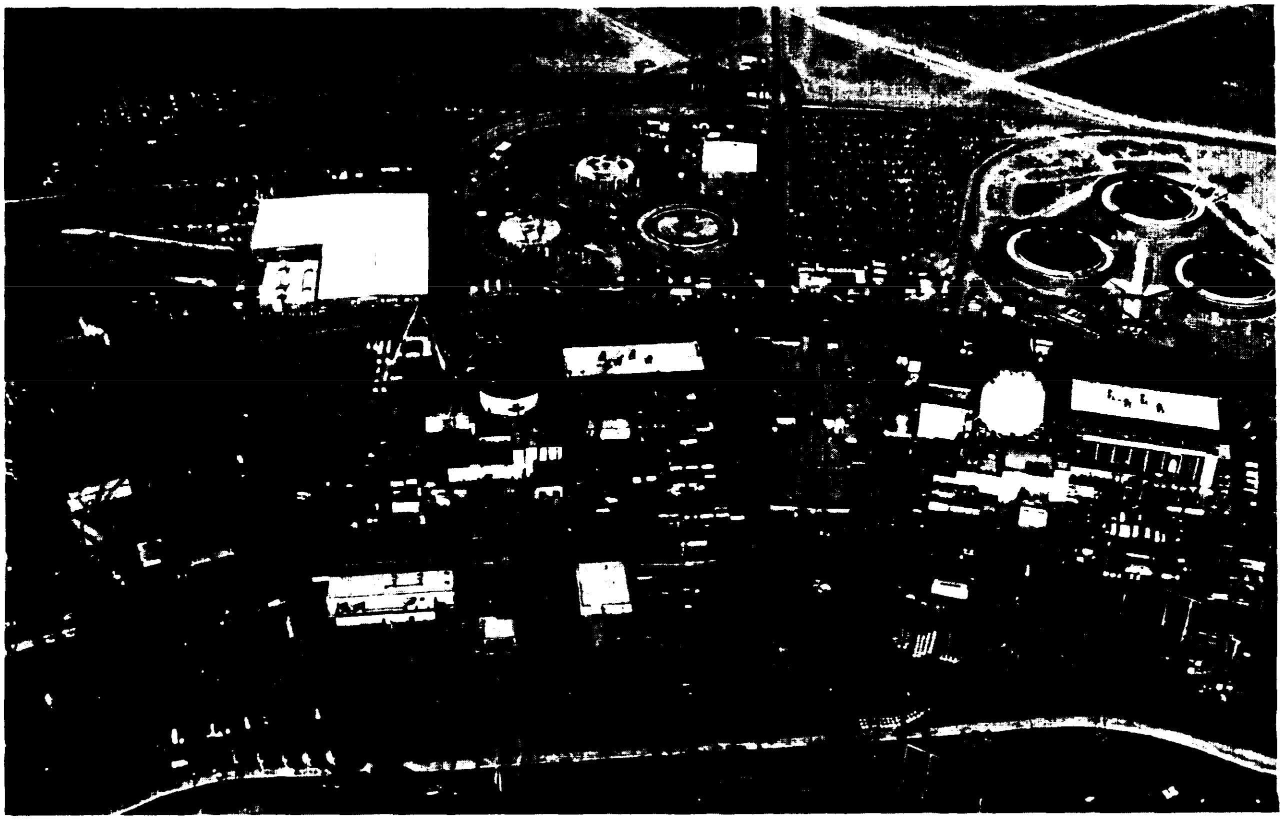
Duplicate Plant Concept

Under this concept, NRC receives a number of applications for construction and operation of nuclear powerplants of essentially the same design to be built at different sites by one or more utilities. Initially, the concept applied to applications received within a few months of one another. As modified, the concept allows NRC staff to issue a preliminary duplicate design approval (PDDA) for the first duplicate plant approved at the CP stage and a final duplicate design approval (FDDA) at the OL stage.

Manufacturing License Concept

The manufacturing license concept involves the submittal of an application for a number of identical nuclear powerpl ants which would be manufactured at one location and moved to a different location for operation. An application for a manufacturing license is submitted by a vendor and includes a report that is similar to a safety analysis report (SAR) except that it is designated a design report. The utility-applicant and site-specific information are reviewed on each application that references the manufacturing license application.



This concept is specifically applicable to the Offshore Power Systems (OPS) approach. The approach includes the NSSS and BOP, manufactured in Florida, towed to a permanent site for mooring and connection to the electrical grid.

Replicate Plant Concept

The replicate plant concept involves the submittal of an application by a utility applicant for a nuclear powerplant of essentially the same design as one in which the staff's review has resulted in the issuance of a safety evaluation report. The nuclear powerplant previously reviewed by the staff is referred to as the base plant, and the new plant is referred to as the replicate plant.

NRC has considered, but not yet implemented, a program by which a standard plant could be reviewed only once before it is licensed. This involves the concept of a standard design approval (SDA). NRC staff believes that single-stage licensing review is desirable from the standpoint of the public, industry, and NRC. The advantage of a single-stage licensing review from the NRC staff's viewpoint is that it is based on more complete information and a single set of regulatory requirements.

From the utility applicant's standpoint, construction can proceed on the basis of an NRC staff-approved design that will not be subjected to a second review. From the public's viewpoint, a more complete understanding of the facility is available at the beginning. Intervenors should be able to frame more specific contentions based on the more detailed design.

One problem in the formulation of any single-stage licensing procedure is that the AE firms would have difficulty in supplying, at an early stage, the level of detail typical of an final design. As one AE firm put it, they believe one-stage licensing:

> . . . can be an effective tool in increasing licensing predictability if executed at the proper level of detail so as not to tie up the AEs, the utilities, and the regulators in paperwork that would result from the inevitable changes necessary to complete the design and construction of a plant.

To make a single-stage review applicable to the entire plant design, NRC would issue an SDA in lieu of a combined PDA and FDA. The SDA concept involves the submittal of information that is significantly more developed than that now provided for a preliminary design but somewhat less than that for a final design. The SDA would of necessity be limited in some areas to complete functional specifications rather than to actual design drawings and specifications to avoid possible antitrust problems with equipment suppliers. A supplementary NRC staff-audit function would be required during plant construction to verify that the actual components—features installed or constructed —adequately meet the approved functional specifications. To date the SDA single-stage review concept has not been implemented.

The General Electric Co. (GE) has proposed the similar concept of a "power-worthiness certificate" (PWC), in analogy to the air-worthiness certificate granted to aircraft by the Federal Aviation Administration. The major difference between PWC and SDA concepts is the scope of hardware licensed. The minimum scope for the PWC is the NSSS plus the other radiologically significant systems and structures. This is contrasted to the NSSS *or* BOP minimum for the SDA.

EXPERIENCE WITH THE NRC STANDARDIZATION PROGRAM

Under the reference plant concept there has been quite a lot of activity for NSSSS, but this is of marginal value for standardizing reactors because the vendors' NSSS systems are already fixed in design. Five AE firms have submitted BOP designs under the reference system concept, but none of these have yet been used.

Under the duplicate plant concept, two major projects have been undertaken. With three plants being planned at each of three sites, Duke Power has more experience with this concept than other utilities. Duke is also unusual in that it serves as its own AE. A consortium of utilities (Standardized Nuclear Unit Power Plant System (SNUPPS)) is also making considerable use of the duplicate plant concept. Originally planned at six plants, the group has now cut its number of plants planned under the concept to three. Four other applications for pairs of plants under the concept have been made.

OPS is the only applicant that has requested a manufacturing license, which is an application for a license to manufacture eight identical plants. The licensing process has been completed except for the new requirements imposed by NRC as a result of the TM I accident. This post-TMI review has been held up by NRC as it has for other pending CP. OPS presently has no plants on order and probably must obtain a manufacturing license before it can accept orders.

Initially, five utilities applied for licenses under the replicate plant concept, but only one of these applications remains active. Single-stage licensing has not been implemented.

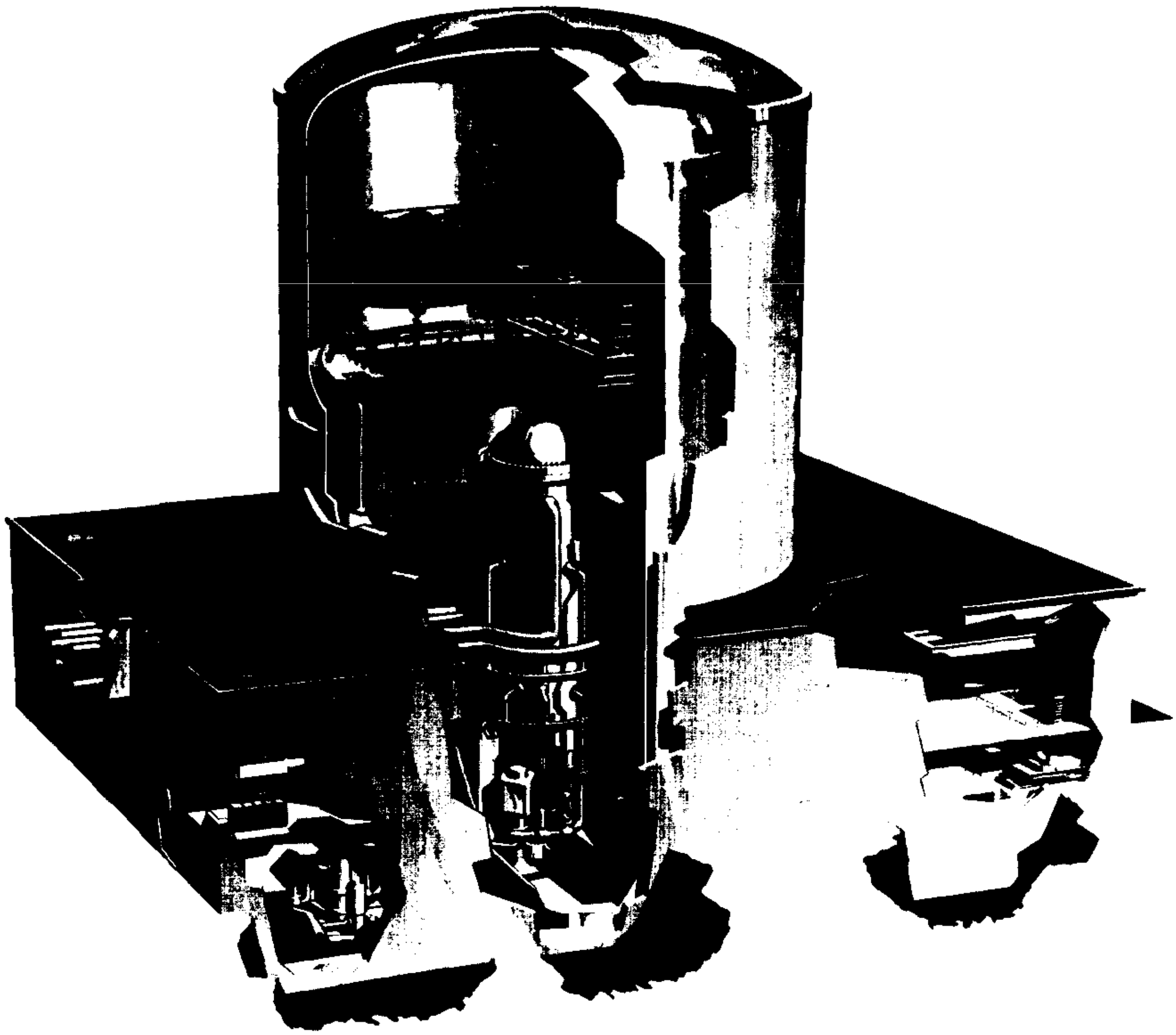


illustration credit General *Electric Co*

Cutaway view of the reactor building for the BWR nuclear island. The nuclear island contains those structures of radiological significance

In summary to date, there has been participation in NRC's standardization program by all four vendors, by five AE firms, and initially by 10 utilities (though some have since canceled). It should be noted that all BOP construction done under NRC standardization programs has been under the duplicate plant concept. No doubt most participants did so in the hope of reducing licensing times, increasing predictability that designs would be accepted for licensing, and lessening construction costs.

By and large, it is too early to judge whether many of these hopes will be realized. Duke Power has encountered some difficulties with its duplicate plant efforts (see ch. 3, pp. 23-24). However, SNUPPS does report it has cut construction costs about 10 percent by standardizing.

An NRC study of the standardization program revealed that savings in the effort needed to review applications, the primary objective of NRC's program, have been minimal so far.[2] The number of questions asked during a review is considered a key indicator of the difficulty of processing a nuclear plant application. The standard reviews over the period studied took as many as 12.6 man-years and as many as 1,060 questions, compared to 6.3 man-years and 700 questions for a custom design. Note however, that the standard design review included a review of the basic design; subsequent reviews of referencing applications should be expected to be much shorter. Data for duplicate plant reviews indicate a substantial reduction in staff and industry effort.

In an interview of the NRC staff, OTA learned that the staff strongly supports the current approach to standardization but hopes that fewer standard plant designs than the number submitted to date will eventually result from the efforts of NRC. The staff expressed little or no support for a single standard nuclear plant designed and supported by Federal Government agencies. They did express great interest in single-stage licensing as represented by SDA but recognized that it poses greater problems to implement for BOP designs than for the NSSSS , mainly because of the traditional engineering procedures followed by the AEs and their utility customers.

The NRC staff feel that the nuclear power industry would be improved by having both Government and industry maintain a firm commitment to l imit changes to an approved standard design to those clearly needed for public health and safety reasons.

The staff felt that nonstandard designs resulted in confusion in understanding accident conditions such as those experienced at TMI. If that plant had been a standard design, the accident could have been analyzed with far less confusion and with more certainty.

Finally, the NRC staff believes that the present hearing process is a large impediment to the full realization of the benefits of standardization. Under present procedures, a standardized design with safety features reviewed in public licensing hearings and accepted by the NRC staff can be reevaluated and perhaps changed in future hearings. This process of adversary cross-examination may cause the industry and regulators to perform better and more thorough safety evaluations of proposed nuclear powerplants than would be performed in the absence of public scrutiny. To reduce the opportunity for public rehearing would enhance efforts toward standardization at the expense of public input.

The alternative in the NRC current standardizations program is the submittal of a final design by any qualified applicant to the Commission for rulemaking.[3] This allows NRC to review and approve a plant design without having received an individual CP application. Once approved, utilities could reference this design and start construction after demonstrating the acceptability of the proposed site. A procedure similar to this is being used for the manufacturing license although a final NRC review and possible hearing will be required prior to the towing of the initial floating nuclear plant from the manufacturing facility. Each site for a floating nuclear plant must be licensed. Since a public hearing is mandatory during rulemaking on the design, later hearings at the CP and OL stages would be limited to issues unrelated to the approved design (e. g., environmental impact and utility competence). Presently, no applicant has requested rulemaking under the current standardization program, probably because of the expense involved and the possible public perception that rulemaking on a design was a means of bypassing the statutory requirements of the Atomic Energy Act for public hearings.

Nuclear Regulatory Commission, op c it

[3]Code of Federal Regulations, title 10, pt 50, app O (40 FR-2977), Jan 17,1975

CURRENT STATUS OF LICENSING

Since TMI, new plant orders have disappeared. Meanwhile, NRC has been under intense pressure to investigate or rule on many safety issues. These two factors have combined to force standardization programs into the background. Currently, such programs are under the Standardization and Special Projects Branch at NRC and are manned by a very small staff with virtually no budget. However, the work that must be done includes review of Combustion Engineering's application for an FDA, which is the first one to be requested under the reference system concept [the 3-unit Palo Verde application references the Combustion Engineering FDA). Another important action awaiting the standardization branch is the complete acceptance review of the application by GE in March 1980 for an FDA for its nuclear-island designs. Six plants now under construction reference this design.

Although other matters do require heavy demands on its staff, NRC should be aware that current steps must be taken, both to consolidate the gains begun under the standardization program and to plan for a possible future of renewed interest in nuclear power. In particular, NRC should be giving more attention to the implementation of some form of single-stage licensing.

NRC'S FUTURE ROLE

One criticism sometimes leveled at NRC is that it is not disciplined or consistent in its decisions regarding which safety concerns are sufficient to warrant design changes or even reactor retrofits. industr^y observers, in particular, worry that unless NRC is more disciplined, reactors initially designed as similar plants may grow apart because of changing regulations. Adoption of a safety goal would certainly help NRC arrive at consistent and more predictable decisions regarding design changes. As generally used now, the concept of a safety goal — which might be either quantitative or qualitative— is the definition of an optimum level of safety as a focus for the licensing process. It would consider both individual and societal risk, and include some method of measuring the effectiveness of the safety standards prevailing at any particular time.

In the licensing for either custom or standard plants, NRC has currently introduced an atmosphere of uncertainty. Many safety issues await rulemaking by NRC. Until NRC rules on the issues or unless NRC adopts interim criteria, nuclear plant designers will be uncertain how to design a plant that can be licensed. One example of these current safety issues pertains to degraded cores. The objective of the degraded core rulemaking, to commence some-time during the second half of 1981, is to determine whether fundamental changes are required in reactor design to prevent or mitigate a melted core from penetrating the containment and entering the outside environment. NRC has not provided any interim guidelines for what designs are acceptable until the rulemaking is completed.

As another example, NRC has recently ruled that applicants analyze all accidents of a certain class (called "class 9" accidents). Unfortunately, NRC has not defined these accidents well enough or sufficiently narrowed that class of accidents for them to be reasonably analyzed.

These two examples are a few of many that indicate that NRC is not managing its activities effectively at this time. Uncertainties or ambiguities such as those mentioned will impede *all* licensing— standard or otherwise. The adoption of a safety goal might facilitate the many decisions NRC has to make.

Chapter 5

THE NUCLEAR INDUSTRY'S EXPERIENCE WITH STANDARDIZATION

Chapter 5

THE NUCLEAR INDUSTRY'S EXPERIENCE WITH STANDARDIZATION

THE NAVAL REACTOR PROGRAM

The Naval Reactors Program under Adm. Hyman Rickover has responsibility for 125 operating nuclear-powered ships, with 36 authorized or under construction. The U.S. Navy has attempted to maintain as much standardization as practicable, with particular emphasis on the similarity of control rooms, instrumentation, operating procedures, and training programs, All operators attend the same nuclear power school, and manuals used for training are of the same type as those on shipboard. However, the specific propulsion-plant designs may vary because of the different sizes and military functions of the vessels they must power. The Navy's nuclear-powered ships include attack submarines, ballistic missile submarines, aircraft carriers, and cruisers. At least 11 classes of ships are built or authorized under the Naval Nuclear Propulsion Program. Even within a single class, some variation has resulted as new technologies develop and become installed on later models of a given class.

The designs are formulated at one of two, Government-owned, contractor-run laboratories—the Bettis Atomic Power Laboratory (operated by Westinghouse) and the Knolls Atomic Power Laboratory (operated by General Electric (GE)). Designers at these laboratories must obtain the approval for their designs from Naval Reactors headquarters, and the designers are held responsible for study and improvements in the designs even after the reactor has been built. Presently, submarines are built at two commercial shipyards.

With regard to commercial nuclear powerplant standardization, Rickover made two observations before the President's Commission on the Accident at Three Mile Island on July 23, 1979. The first was on the desirability that the utilities "unite to establish a separate technical organization which could provide a more coordinated and expert technical input and control for the commercial nuclear power program than is presently possible for each utility with its limited staff. "

Among the things such an organization could do are:

- develop the standards and specifications utilities should require for design and construction of their plants;
- provide direct, indepth technical assistance to utilities in design, construction, and operational questions;
- establish recommended staffing requirements for operation of nuclear plants—e.g., at times there may be only a single operator with no supervision present in the control room of an operating plant. Also, that operators may be assigned and actually carry out unrelated duties while on watch. These are contrary to Navy practice;
- develop a comprehensive training and retraining program, including lesson plans, qualification requirements, etc., for utilities to use in training their people. This must be based on what is needed and not geared solely to passing licensing examinations. It should cover al l types of personnel, not just operators;
- provide trained technical teams to perform periodic audits of nuclear stations and critically evaluate the plants and the qualifications and performance of personnel; and
- advise utilities on technical safety questions.

In the same testimony before the Three Mile Island (TMI) Commission, Rickover recommended that "plant designs, equipment, control rooms, training, etc., should be standardized insofar as practicable. " Rickover noted that much more standardization seems practical on new plants than old ones (where it might nevertheless be possible to achieve some degree of standardization of control rooms, instrumentation, etc.), and that standardization should have two distinct benefits. First, he noted, that better designs should result because a larger number of engineering man-hours could be applied to standardized designs, and, with a larger number of identical operating systems, operational experience would "provide a valuable source of information that can be used to improve the design and procedures and establish a more effective preventive maintenance program for all plants. " Second, he noted, the use of standard designs would make it possible to train operating and inspection personnel more effectively.

However, Rickover did not advocate the most extreme form of standardization. "In advocating more standardization I am not saying that there should be one single design. I have standardized in my program as far as practicable. Even then we have a number of designs to suit the different power ratings and ship types and to take advantage of new developments and technology which have become available. "

With regard to a new technical organization, the utilities have jointly funded the Institute for Nuclear Power Operations (I NPO), which is undertaking to prepare models for operator training programs, and will establish training program criteria, accrediting industry training programs, and performing in-plant evaluations. INPO hopes these programs will be more specific than those of the Nuclear Regulatory Commission (NRC). The models will be recommendations, not requirements, for the utilities. Another collective organization funded by the utilities is the Nuclear Safety Analysis Center (N SAC), recently created by the Electric Power Research Institute to provide more technical assistance to the utilities. The commercial nuclear industry has, therefore, strengthened its organizations along the lines suggested by Rickover although none has the total authority that the Navy exercises over its reactors' program. The benefit of these organizations is difficult to judge because of preoccupation with the implementation of the requirements resulting from the accident at TM I and the short length of time (about 1 year) of their existence. Their success will depend on the quality of the personnel in the organization and the willingness of the utilities to accept their assistance responsibly.

With regard to standardized plant designs, the currently available standard designs docketed with NRC represent an improvement in decreasing the number of designs that are commercially offered. A greater reliance on a joint utility organization that sets design standards and criteria that are more detailed than those in NRC regulations is desirable. The implementations of such a concept in the near future may be extremely clifficult due to the current high level of regulatory activity in areas other than standardization.

STANDARDIZED NUCLEAR UNIT POWERPLANT SYSTEM

The Standardized Nuclear Unit Power Plant System (S NUPPS) is a consortium of utilities organized to build identical nuclear plants at different sites across the country. It is the closest project to full standardization with plants under construction. SNUPPS project management is handled by a contractual arrangement with Nuclear Projects, Inc., and a hierarchy of utility companies. The five utility companies have entered into separate but nearly identical contracts with Bechtel Power Corp. (lead architect engineers (AE)), Westinghouse (supplier of the nuclear steam supply system (N SSS)), General Electric Co. (supplier of turbine generators), and Nuclear Projects, Inc.

SNUPPS originally was a consortium of power utilities that made an application for six units at four sites. One unit was withdrawn shortly after application. Another (the Sterling unit) was canceled because of a lessening of

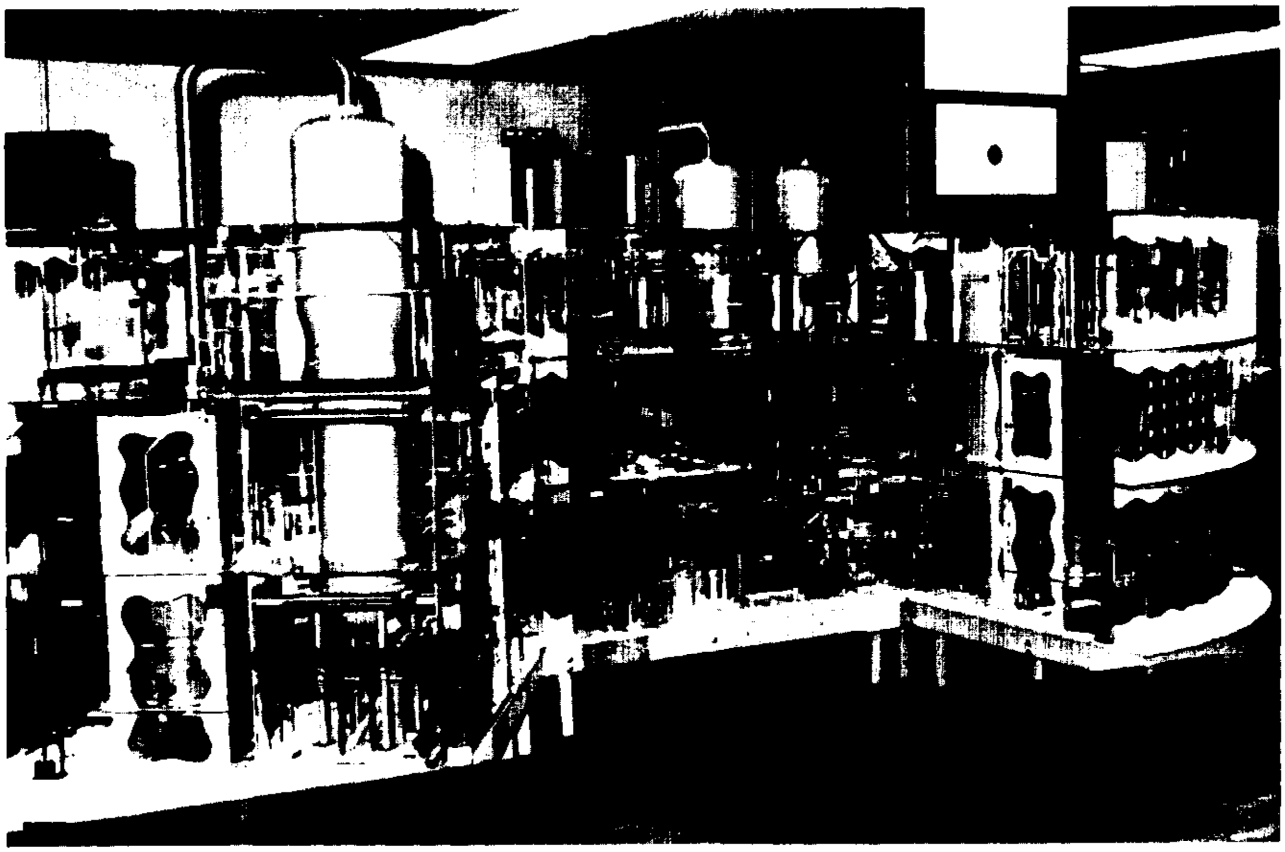


Photo credit Bechtel Power Corp

A mock-up of the equipment inside the containment is used to minimize problems with equipment layout and pipe or cable tray interference. It also serves as a planning aid during construction of the plant. The model seen here is of SNUPPS. The long cylindrical vessels with the "J" shaped tube at the top are the steam generators. The reactor vessel mock-up would be surrounded by the portion of the containment seen here

demand, restrictions of local governments, and uncertainties in the Federal regulatory procedures. The design and construction of both Callaway 1 and Wolf Creek units are over 60-percent complete, however, both have suffered time delays and substantial cost increases. The time delays have resulted from financial considerations and Federal regulatory concerns, while cost increases have occurred primarily due to recent unusually high inflation rates.

The SNUPPS project is based on identical units with no shared systems. If two units were to be constructed at the same site they would be identical but separate units. For each plant, Westinghouse produced a standard information package in order that Bechtel could design and engineer the balance of plant with minimum changes to the NSSS. This approach facilitates the orderly progression of design drawings and the ordering of equipment.

All plants will have identical portions called the "power block", this consists of the reactor building (containment), fuel building, turbine building, hot machine shop, auxiliary building, diesel generator building, control room building, and radwaste building. Structures and components outside the power block differ for the various plants and are not under control of SNUPPS.

In the licensing process, the project is managed by a single project manager and review team within NRC. In addition, the Advisory Committee on Reactor Safeguards assigned a subcommittee to review the standard portions of SNUPPS and when the Atomic Safety and Licensing Board hearing for two of the units were held, previously resolved issues were not re-reviewed. This sharing of licensing resources allows more licensing personnel to provide a greater indepth review than would have been possible with a customized application for several plants. In addition, there was a reduction in the questions asked by NRC from an average of 700 for a customized plant to an average of 150 per single SNUPPS unit.

During the procurement for the units, only proven materials, equipment, and systems are to be used; American National Standards institute and other appropriate standards are to be strictly followed. Power block purchases are centralized — i.e., with few exceptions the same supplier and the identical item for a particular function are used for all plants. This allows interchangeability of parts between plants. These are common industrial practices.

During construction, a considerable amount of standardization is maintained. Detailed models and photographs of the models of the standard plants are used in the construction effort. This method has eliminated much interference and many delays while providing a considerable surety of proper construction techniques. Construction equipment common to SNUPPS plants is shared by the construction crews.

Standard preoperational procedures, start-up, and functional operating procedures are being prepared for the SNUPPS plants. Also, simulators will be available for the SNUPPS plants and operating experiences will be shared among SNUPPS utilities' personnel.

The participants in the SNUPPS program claim the SNUPPS plants will be built for about 1(1-percent less than if they were customized plants. Further, they feel the plants will be safer because of the standardization effort. However, there are no hard data to substantiate this claim, only an intuitive feeling that the more man-hours spent on a particular system design the safer it will be. '

'NIcolax A Petrick, "Progress Report on the SNUPPS Nuclear Stat lons," Nuclear Projects, Inc , Nuclear Englneerlng Internatlonal, November 1977

THE FRENCH NUCLEAR PROGRAM

The French have developed a consensus of government energy policy makers that is supported, almost totally, by all four major political parties. The French nuclear program has some of the same problems as other nations — e.g., opposition by organized citizen groups, some difficult public relation situations, and some technological shortfalls; however, they have maintained a firm commitment to their policy of "tout nucleaire" (i. e., decommissioning oil-fired electricity generation plants and building coal-fired, hydrostorages and mostly nuclear powerplants). The French policy was formulated by their perception for the need to reduce dependence on foreign supply of oil (which in 1973 supplied France 67 percent of its energy needs). Further, the French have only very limited supplies of oil, natural gas, and coal within their boundaries. The French condition is quite different from the United States— i.e., there is no clear political consensus on the need for nuclear power in the United States, partly because there is an indigenous supply of oil, natural gas, and coal within U.S. borders.

The choice between the two commercial types of light water reactors —e. g. (boiling water reactor (BWR) and pressurized water reactor (PWR)) — using enriched uranium was made on the basis of price. The French entered into a competitive program between European holders of licenses for the manufacture of American designed plants. Framatome held a Westinghouse license and Alsthom, a GE license. The latter group had a significant dis-

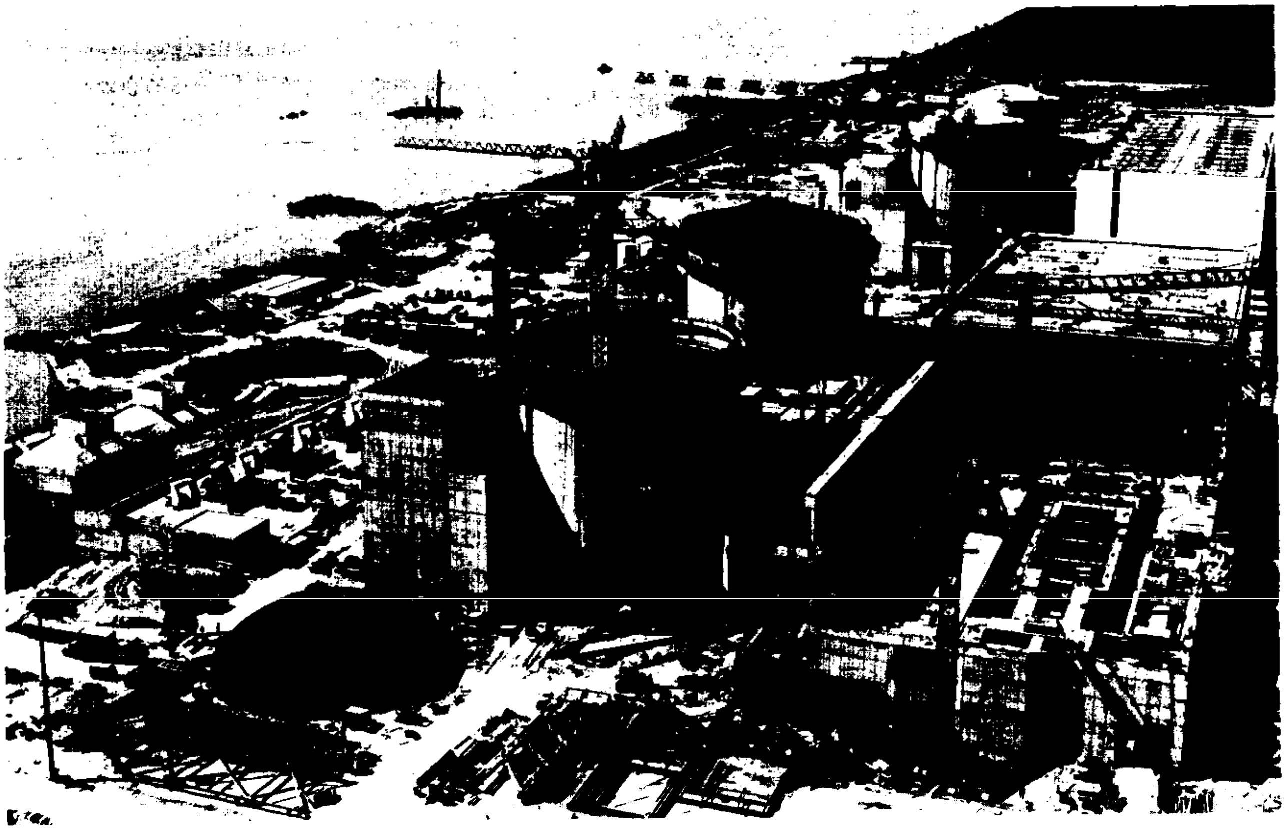


Photo credit Electricite de France

Four identical nuclear units are shown under construction in Blayais, France. The first unit, as seen in the background, is scheduled to produce electricity in 1981, just 6 years after construction started. By 1985 there will be 80 such units supplying 52,000 MWe for an area no larger than the State of Texas

advantage in the competition by the fact that it does not own heavy forging facilities for reactor vessel construction. For this reason, Alsthom either had to call upon their competitor, Framatome, or contract abroad. The BWR line was therefore dropped not because of the PWRS technical superiority but to ensure a sufficient workload for the French industrial group in charge of construction. The French industry was, therefore, restructured into one constructor of nuclear steam supply components (Framatome) and one constructor of turbine generators (Alsthom). In addition, the national electric utility is the only French AE, thereby making the standardization of nuclear powerplants easier in France than it would be in the United States.

The French recognize four major safety-related advantages for standardization:

1. a more thorough investigation of safety-related matters is possible when multiple units are involved;
2. experience in design, manufacture and construction, and operation can be transferred from unit to unit;
3. more designer time becomes available to spend time working with a new generation standardization series; and
4. regulators can spend more time inquiring about site-specific considerations, the need for new units, and the ability of the utility owner to operate the unit.

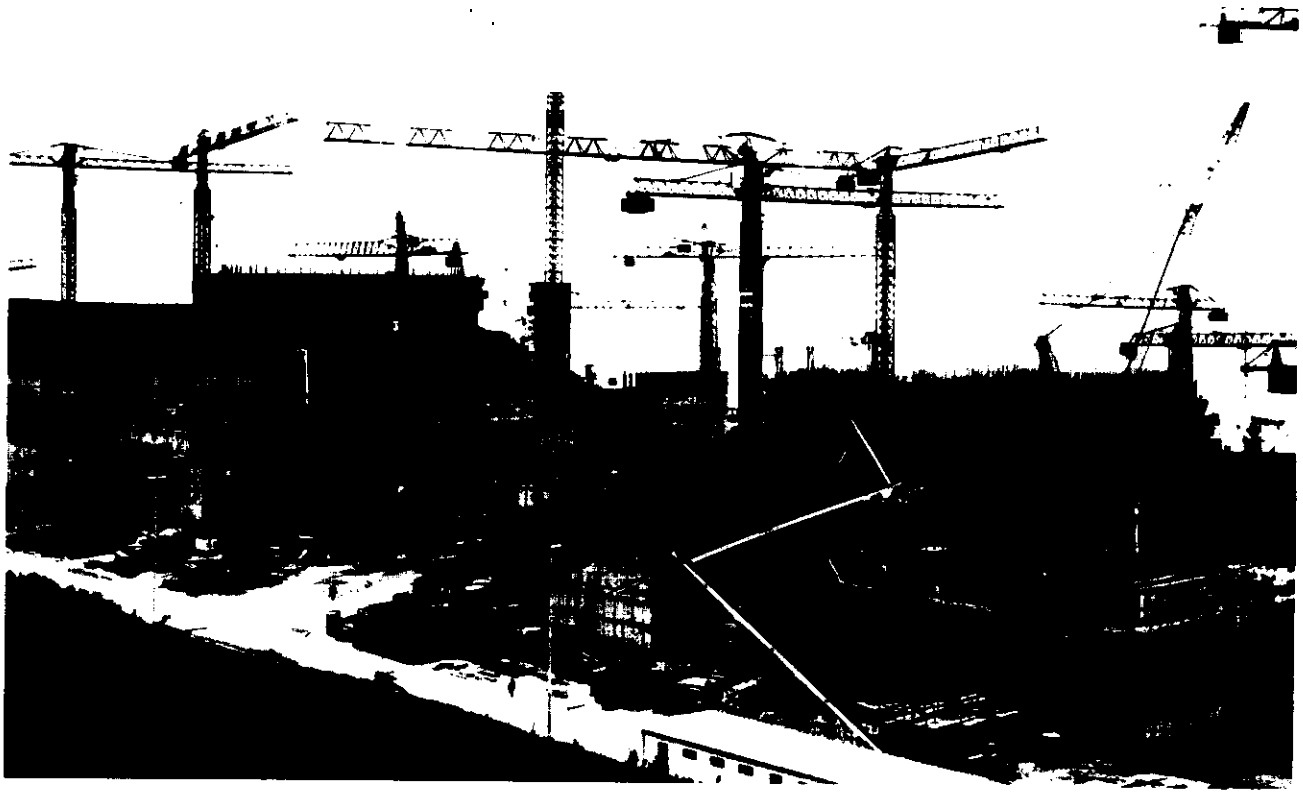


Photo credit Electricite France

The Paluel site, Normandy, France consists of four 1,300-MWe units. The concrete walls of the containment and auxiliary buildings were erected during the early stages of construction. The first unit should produce electricity y sometime in 1983

Also, the French recognize at least three major difficulties with standardization of nuclear powerplants:

1. problems involved with one unit of a series propagates to other units in the series and may require expensive and time-consuming redesign and back-fitting;
2. site considerations may require substantial design differences between units of a standardized series; and
3. the optimal balance between design stability and technological upgrading is difficult to determine (i. e., a definition is needed of safety enhancement or cost reduction required before a new technological achievement can be incorporated .2

Overall, the French are satisfied with their choice and consider that the advantage of standardization (especially those related to safety and economics) far outweighs those dif - ficulties.

'Mlchel Pecgner, "How,One Organization Runs the Whole industry, " Commissarat a L Energle Atomique (CEA), Nuclear Englneerlng International, December 1976

THE WEST GERMAN OPERATOR TRAINING PROGRAM

A possible model for standardization of training and certification of personnel in commercial nuclear powerplants is the West German program. The West Germans train and certify their operators for both conventional and nuclear powerplants in a powerplant school called the Kraftwerkschu le. This is a joint organization of owners of large powerplants with 116 members from six different countries. The primary purpose of the school is to provide professional and advanced training in six different technical areas for powerpl ant personnel in maintenance and operation. The school was founded by a parent organization called the Technical Association of Power Plant Operators, formed as a result of a severe boiler explosion in 1920.

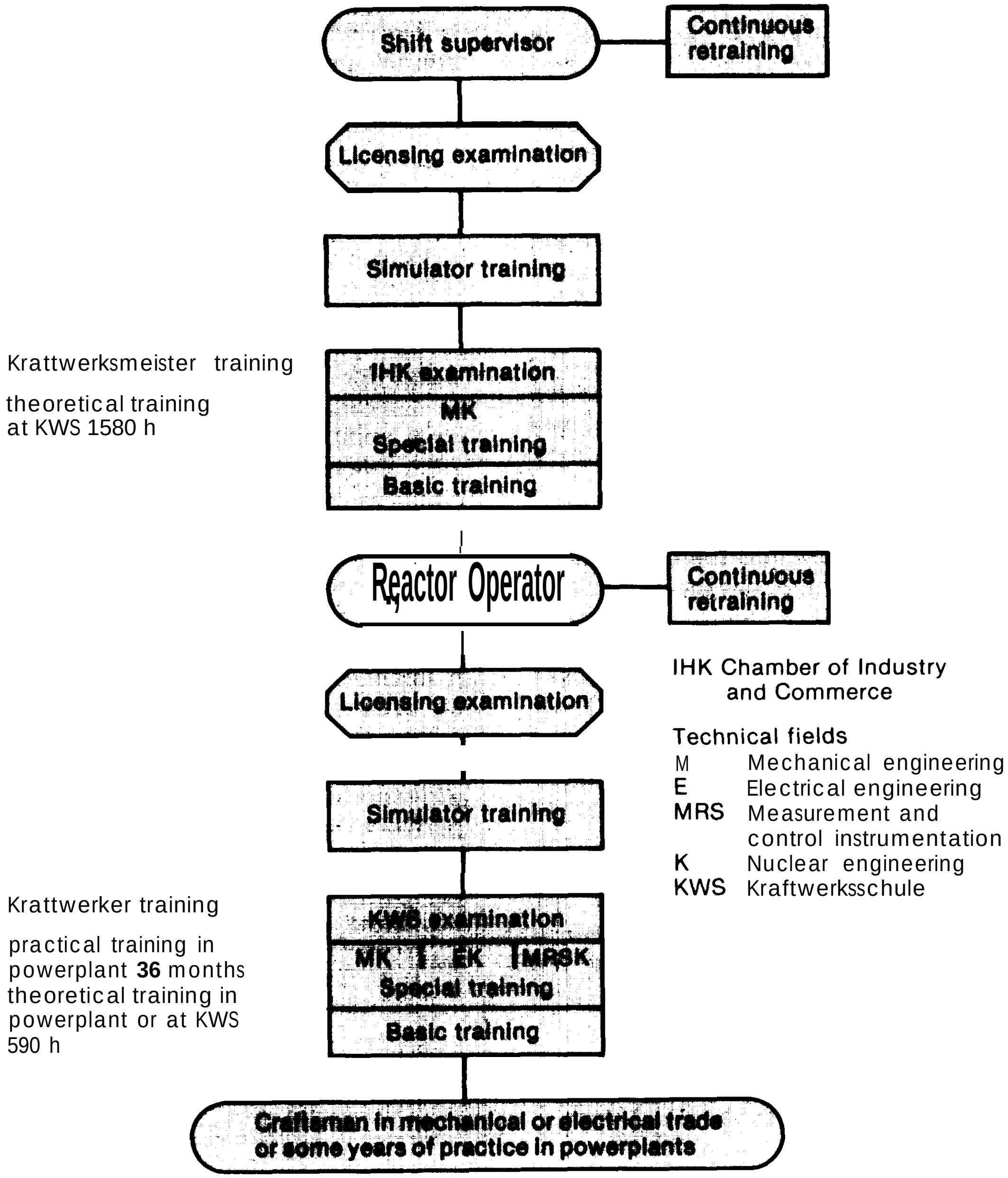
The formal training for a plantworker takes 3 years and is integrated into the operation of the powerpl ant. Training consists of theory and practice with a final exam for certification in the operation of powerplant systems. Figure 6 shows the progression for a nuclear plantworker from initial certification by the Kraftwerkshule to shift supervisor.

The professional competence of the operators and shift supervisor is regulated by official government guidelines. The minimum personnel complement for a nuclear powerplant control room is a shift supervisor, a deputy shift supervisor, and a powerplant reactor operator. The shift supervisor must be an engineer and his deputy must be at least qualified as a Kraftwerkmiester (see figure 6). All three require additional special nuclear training including simulator training, and practical nuclear powerplant experience.

As in the United States, the plant's superintendent is responsible for the selection and training of the powerplant team. The superintendent assesses the workers' capabilities and determines who will eventually be qualified as a plant attendant, plant operator, or shift supervisor. Unlike his counterpart in the United States, the West German superintendent picks his candidates from a pool of workers who have completed a standardized training program established by the owners of the powerplants under the guidelines of the government. In this country, the closest organization to the West German program that has uniform training and certification for its reactor operators is the U.S. Navy. Many utilities rely heavily on the Navy for qualified plant operators. This dependence can create manpower shortages in an area vital to the national defense and allows the utility to abrogate some of its responsibility for a complete and total training program for new operators with no nuclear experience.

'0 Schwarz and G Schiegel, "Combining Theory **and Practice in West German y,"** Nuclear Engineering International, March 1980

Figure 6.—Training Patterns for a West German Reactor Operator



SOURCE Nuclear Engineering International

Chapter 6

POLICY IMPACTS OF FOUR APPROACHES TO STANDARDIZATION

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The situation presented in the previous chapters is one of an industry which has been slowly evolving toward a greater degree of similarity in its products. Stringent standardization is very difficult in the commercial nuclear industry where the tasks of design, supply and construction, operation, and regulation are undertaken by multiple and often independent organizations. Nevertheless, the designs formulated by the nuclear steam supply system (NSSS) vendors and architect-engineer (AE) firms are slowly converging toward a single design for each company. Several utilities and utility consortia have attempted to construct multiple reactors based on a single design. The Nuclear Regulatory Commission (NRC) has for some years defined special licensing for four categories of "standard" plants defined in chapter 4. These steps have been taken voluntarily over a 10-year period, because the industry perceives they will produce lower costs, shorter licensing times, and more reliable plants. Increasingly, both Government and industry personnel have concluded that a more rapid move to standardization may increase the safety of nuclear plants. They also recognize that the industry will not move more rapidly toward standardization unless external forces push it in that direction. Four representative approaches to standardization are used here to provide a framework for this analysis.

FOUR APPROACHES

Acceleration of Present Policies. –An incentive program to accelerate the present trends in the industry could reduce the number of designs substantially. In the first place, such a program could reduce the number of designs to one for each designer— i.e., 4 NSSS designs and 4 to 12 balance-of-plant (BOP) designs, depending on the number of AEs that remain active in the nuclear field. Only a few AEs have developed BOP designs for the boiling water reactor (BWR) and General Electric Co. 's (GE) completed design for a nuclear island approach based on the BWR make it likely that future BWRS will be of one design. For the pressure water reactors (PWRS) produced by the other three vendors, each AE would have basically the same BOP design tailored to meet the various interface criteria. Thus, the possible number of different reactor plants would be in the range of 5 to 13. The lower number could result if the utilities agreed on design features and specific criteria for a standard BOP. Any AE could design a BOP conforming to these agreed-on criteria and the existing regulatory requirements. NRC could then offer one-step licensing for any utility referencing this "standard" in a license application. The time to implement this level of standardization would equal the time to formulate the criteria and implement one-step licensing– about 1 to 3 years.

Procedural and Organizational Standardization.– One advantage of standardization would be that it would allow personnel training, operating procedures, terminology, etc., to be specified in greater detail for a larger body of plants. Adoption of more universal practices would allow operators of different plants to learn more from the experiences of one another and would facilitate audits. Even without identical hardware, for the existing generation of powerplants, the "software" practices could be made more alike. NRC has some standards for such practices and private groups, such as the Institute for Nuclear Power Operations (IN PO) and the Nuclear Safety Analysis Center (NSAC), are currently evaluating operating practices with a view toward upgrading them. If the Government wished to do more, a starting point would be to examine NRC's current standards to see if they could be more precisely specified and more universally applied. An examination of the West German standardized training and certifications program for nuclear powerplant personnel discussed earlier, might be appropriate.

Standardization of the NSSS Design Plus a Safety Block. –One of the major reasons or standardization is to allow more attention to a smaller number of designs and especially to safety-related systems (e. g., auxiliary feedwater and containment solation systems). One possible approach to standardization is to define those portions of the BOP that are necessary to bring the reactor safely to a cold shutdown condition and to allow only four variants (one for each NSSS) of this so-called "safety block. " Under this approach, the safety block would include 25 to 50 percent more equipment and hardware than the present NSSS. This version of standardization represents a significant deviation from the current mode of doing business and would require either a redefinition of responsibilities as now specified by NRC and perhaps some legislative action. To achieve this level of standardization, either the Government or industry would have to define what components belong in the safety block, subject the particular designs to some criteria of safety and reliability, and transfer responsibility for them from the AE firms to the NSSS vendors or to a design team composed of both.

Critics of the approach suggest this transfer of responsibility for safety systems, normally under the control of the AE firms, may place a burden on some of the NSSS vendors for which they are neither qualified nor prepared and thereby significantly alter the present structure of the nuclear industry. Some of the essential safety systems (e. g., the containment) require design and construction skills for which the AEs are uniquely qualified. The safety-block approach is similar to that proposed by GE, with its "nuclear-island" concept; this would take about 3 to 5 years to implement.

One Single-Standard Plant. –The ultimate in standardization would be to select only one plant design according to which all future reactors would be built. Such standardization would have to be accomplished by legal statute and would completely alter the present structure of the commercial nuclear industry. To implement this concept of standardization one must decide who would have overall responsibility for the design, what the design criteria should be, and what would be the criteria and time scale for incorporating modifications into the standard design. It would require from 6 to 10 years to design and an equivalent time to construct this single, national reactor.

Even for a single-design approach, site-specific factors such as seismology, meteorology, and hydrology would require modifications in some of the reactor plants.

SAFETY BENEFITS

Almost all of the potential safety benefits of standardization are proffered on the basis of intuition rather than experience. Few relevant examples of standardization exist, and none demonstrate unambiguously that the safety achieved results from the standardization rather than from other factors — e.g., the safety record of the naval nuclear reactors program probably results as much or more from the U.S. Navy's central control and other factors as from any similarity among its various reactor plants. Some of the arguments for the safety benefits may break down in the extreme case of standardization — e.g., the one single-standard plant concept is seen by some as an opportunity for a fresh objective look at commercial reactor design while it is viewed by others as a dangerous commitment to a possibly flawed, single design. The following discussion is an examination of the arguments in favor of standardization and the extent to which these arguments apply to the four previously defined approaches to standardization.

Enhanced Design Review. –Most people in the nuclear industry or within NRC concur that the attention given to a particular design should increase as the number of designs decreases. The incentive program towards standardization should allow more concentration of attention within the designer firms. Moving towards a safety-block concept or single-standard design would primarily benefit regulators such as NRC by greatly reducing the number of different reactors it would have to understand and regulate. Those advocating a single national design feel that its major advantage would be the design attention devoted to it. Designers could start afresh, yet benefit from the experience gained during the many years of operation with light water reactors (LWRS). Similarly, design attention to a safety-block design may lead to a safer product. One should keep in mind that the quality of attention paid to a design is as important as the quantity of designers or safety analysts studying it. It is also possible that the reduction in the number of reactor designs might merely result in a proportional reduction in the number of designers.

A design-review mechanism known as probabilistic risk assessment (PRA) has received considerable attention since the Three Mile Island (TMI) accident. The use of this technique in assessing auxiliary feedwater system reliability was discussed earlier in chapter 3. On a larger scale, PRA involves the steps of identifying hazards, hazardous activities and accident sequences, and quantifying the probability of accident sequences and the magnitude of their consequences. The determination of risk for a nuclear plant involves all parts of the plant and its operation. The NSSS, the BOP (e. g., the control room, containment, power conversion system, and electrical systems), and utility-operator aspects (i. e., the operating and maintenance procedures and the electrical grid), all are important in determining overall plant accident risks.

What sequences dominate risk can be strongly dependent on the details of plant design and operation. Subsequent to the reactor safety study (RSS) (WASH-l 400) which considered two reactors in detail, NRC sponsored an RSS methodology applications program which looked at four additional reactors. While the results of this work have not been published, preliminary results indicate that considerable differences in accident sequences exist compared to the one considered in WAS H-1 400. These differences are due to:

- safety systems unique to the plant studied;
- safety systems performing functions different than in WASH-1400; and
- multiple success options for a given function requiring different levels of system success.

Not only were unique plant sequences found, preliminary results indicate that the dominant sequences vary from plant to plant.

Therefore, the major impact of standardization on probabilistic risk assessment would be to avoid industry manpower limitations in the evaluation of all plants to the degree needed to maximize plant reliability and safety. The fewer number of plants needing evaluation the greater the quality and detail of the risk assessment for a given amount of resources. In addition, a greater understanding of the insights particular to risk assessment would be obtained. In retrospect, the RSS (WASH-1400) yielded considerable insight to the TM1-type accident (e. g., a small break, loss of coolant accident), to the recent Browns Ferry partial scram and to the contributions of human errors to reactor accidents in general. If it were applicable to all reactors, these design problems might have been anticipated and therefore prevented by early corrective action.

Increased Awareness and Applicability of Operational Experience. –This possible safety benefit should be realized to various degrees for any of the four approaches to standardization. Naturally, the fewer the differences among reactors, the more the overlap of experience. The accident at TM I provides a positive example, by which reactors of similar design have learned to watch for a similar sequence of events. On the other hand, many incidents—

'Nuclear Regulatory Commission, "Reactor Safety Study An **Assessment of Accident Risks in Commercial Nuclear Powerplants, " NUREG-75/014, WAS H-1 400, October 1975**

such as the Brown's Ferry partial scram — are still caused by specific piping or instrument errors which may be peculiar to that plant alone.

One central mechanism by which various nuclear plant operators learn from the experience of others is by the Licensee Event Reports submitted to NRC.[2] The greater the similarity among plants—even if it is only more similar terminology or procedures—the easier it should be to understand these events and to decide to which other plants they potentially relate.

There is no inherent reason why operators of custom plants should learn as much from operating standard plants as other plants, but more interpretation is required to decide where each experience is relevant. It has been reported that an incident at the Davis Besse plant, was a precursor to the TM I accident, but no warning was issued. Standardization would not eliminate such omissions automatically but could ease the burden of deciding which reportable events were especially important to which plants.

The feedback provided by the naval nuclear reactors program is a key element in the safety of their program, and it is achieved despite considerable variation among naval reactors. Currently, NRC and the industry are striving to improve the feedback of plant experience. NRC has established the Office for Analysis and Evaluation of Operational Data. The Office reviews all reportable events from reactors and users of byproduct material. NSAC has created a communication and evaluation network used by operators of commercial reactors to inform one another of significant operational occurrences.

Regardless of the organization, one of the difficulties experienced with reviewing operating data is that of interpreting the relevance of a specific component failure at one plant to the safety of another plant using a similar but not identical component. The interpretation may be easier if the component used is identical in all plants, but the plants themselves differ significantly. Experience to date has shown that emphasis on feedback of operating data by the reactor vendors **(GE,** in particular) has markedly improved plant availability. One characteristic of responsible plant management is its willingness and ability to identify and to correct the generic or recurrent problems underlying all unusual occurrences in its nuclear powerplants. In a more standardized nuclear industry there would be no question about the importance of taking the broad view of all identified problems. A more standardized industry would potentially permit a relatively small group of experienced engineers to review the data generated by operating experience, looking for the generic implications of apparently "random" failures. At present, the heterogeneous nuclear industry provides generic assessment of operating experience by means of various user groups. Examples include the BWR Mark I containment owners' group; and the GE, Westinghouse, Babcock & Wilcox, and Combustion Engineering owner's groups; The formation of these groups results in part from an interest in the free flow of information on solutions to their common problems.

While increased standardization would further help in the identification and resolution of safety issues, it would also increase the risk of systematic oversight of potential problems. As a matter of policy, electric utilities plan diversity into their generating mix, both fossil and nuclear, and among the several reactor designs. This course has been amply vindicated by the many generic shutdowns that have occurred without loss of a major part of the nuclear generating capacity. A nonnuclear analogy would be the obvious consequence of having a standardized U.S. jumbo jet, such as the DC-10, grounded when a generic engine-mounting defect is discovered. The degree of nuclear standardization needed to produce optimum benefits is a subject for further evaluation.

The greatest increase in health and safety comes from the review and evaluation of operating and construction experience on one sin-

2 nuclear Regulatory Commission, "Reporting of Operating Information – Appendix, A Technical Specification," Regulatory Guide 116 (revision 4), August 1975

gle-plant design. However, the institutional barriers and the possibility of systematic oversight of safety problems may outweigh any safety benefits accrued through the feedback of data on one "accepted" design. With regard to procedural and organizational standardization, the benefit acheived through uniform reporting and review practices can be easily obtained with little if any disruption in the institutions regulating and operating commercial reactors.

Improved Training for Plant Personnel

The impact of the approaches to standardization of improving plant training is easily analyzed by considering three of the concepts under one heading "hardware standardization." The order of increasing hardware standardization would be:

1. acceleration of present trends;
2. NSSS plus safety block; and
3. single-plant design.

The other approach, procedural standardization, is considered by itself as the standardization of the management processes as distinct from hardware. In addition, other institutional factors not normally considered part of an idealized, formal training program must be taken into account.

The basis for the procedures for design, construction, and operation of a nuclear powerplant is the Code of Federal Regulations, industry standards, and NRC's rules and regulations. Each applicant for a license establishes a set of administrative procedures that implement the letter and intent of these rules and regulations. For an operating reactor, one part of there administrative procedures deals with the selection, training, and qualification of the plant's employees —e. g., these procedures describe the general employee training requirements as well as those for technicians and operators. Each member of the plant staff is subjected to some training with different degrees of intensity and depth according to the position filled. Currently, there is wide diversity in the training programs resulting from the way the utilities interpret the basic requirements when establishing their administrative procedures —e. g., the requirements for a licensed operator to requalify on a yearly basis include the performance of 10 major changes in the plant's status from the operator's console. Some utilities meet the requirement by simply counting the startups or shutdowns the operator has performed over the past year. Others send the operator to a plant simulator for as long as 2 weeks for intensive retraining. New requirements resulting from the accident at TM I have specified in detail the types of manipulations necessary for this requalification.[3] In addition, these manipulations will require the use of a plant simulator.

Greater standardization in operator training programs than what currently exists would ease the administrative burden on implementation and auditing of this new requirement. Also, the effectiveness of the requirement over the next few years would be easier to judge if the change were made from training programs which had more in common. Greater hardware standardization would make the detailed procedural level of these training programs more alike but would be unlikely to increase their effectiveness or ease the administrative burden.

Standardization of hardware would make selected improvements possible in training plant personnel. One area in which this could occur is the use of plant simulators. A simulator consists of a mockup of the control room with indicators, gages, and other instruments and devices driven by a computer. The operator's manipulations of the switches in the mockup are monitored by the computer, which simulates the reactions of the plant on the mockup instrumentation. If greater hardware standardization were used in the nuclear industry, more plant operators could use the same simulator and fewer plant-specific simulators would be needed. Standardization of the hardware would also increase the analytical capability of simulators to deal with off-normal transients when a transient occurs at one

[3] Nuclear Regulatory Commission, NRC Action Plan Developed as a Result of the T MI-2 Accident, N U R E G-0660, " May 1980

plant and operators at other plants need to be trained for possible reoccurrence of the same type of event. Another benefit is that the incorporation of an actual event into the simulator's computer would be easier –e.g., all actual transients could be incorporated into the simulator without the necessity of incorporating specific differences in plant operating characteristics resulting from different designs. The difficulty encountered by the various vendors in simulating the TM I accident on their own simulators (as an aid to operator training) was an example of of this.

However, all of these advantages must be viewed in the context of the existing mix of generation common to most utilities and regional differences in the utilities' service areas. The additions of several nuclear powerplants of standard design may not simplify the utilities training program if the current program is determined by the diversity in existing operating units. Among most electrical utilities, any "standard" plant would be *unique* as a source of power generation because it would be different from existing plants. It would complicate rather than simplify the existing training program. Unless a utility makes a substantial use of a single design in its operating system, the value of hardware standardization in improving the utility's training program will be minimal.

Procedural standardization in personnel selection, training, and requalification may be difficult if there are significant differences in State labor laws, union contracts, or State regulatory requirements. However, considering the current generation mix of each utility, this standardization approach appears to be the easiest to implement with substantial benefits in personnel training.

RELEVANCE TO A NATIONAL SAFETY GOAL

The question of the need for quantitative safety goals to ensure that adequate levels of nuclear powerplant safety are achieved is a longstanding one. The Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974 established the legislative basis for NRC regulation to ensure the safe use of commercial nuclear power. In response to the legislative mandate, NRC regulations require, as a part of issuing a nuclear powerplant construction permit, that a finding be made that "the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public'" and as a part of issuing an operating license that a finding be made "that there is reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public. "5

The principles used by NRC are based on a "defense-in-depth" approach to the plant design. Reactor safety as practiced in accord with these principles is defined in NRC's regulations, safety guides, branch technical positions, and related industry standards. These provide an extensively documented licensing process that has helped the nuclear industry to achieve an impressive record with regard to public health and safety. In this process, many safety requirements and calculational methods have been identified. Following NRC rules establishes that plants adequately meet specific safety requirements and satisfy the requirements of the legislative mandate. This deterministic process is based on implied but unstated probabilities. For instance, a qualitative probabilistic judgment was made many years ago that the large rupture of a reactor pressure vessel in LWRS was unlikely enough that it did not have to be considered in the design. In the intervening years a quantitative basis has been provided to support that qualitative judgment, The NRC licensing process is now considering other factors that arise from accidents of greater severity than the design-basis accidents (DBA). Consideration of such

[4]CFR 1050, sec 5035
[5]CFR 1050, sec 5057

accidents will require a different type of analysis than the traditionally conservative approaches taken in the assessment of DBAs. The use of PRA techniques is rapidly coming into use for this purpose. Quantitative criteria for acceptable levels of risk, or safety goals, are needed if all the benefits of PRA are to be realized. PRA is an acceptable quantitative method of showing compliance with a well–defined safety goal.

U.S. activities relating to the establishment of a national safety goal are going on within the NRC, the Advisory Committee on Reactor Safeguards (ACRS), the nuclear industry in general, and the national technical and scientific community. There are also international activities in this area. Possible variations in goal forms that have been considered include: single v. multiple goals, quantitative v. qualitative goals, and individual v. societal goals. [G] The goal-setting process can be divided into two broad phases, the initial phase in which a wide range of goal elements and alternative strategies are identified, and the second phase in which the effort is directed toward winnowing down the elements and strategies for more indepth analysis and decisionmak ing.

in demonstrating compliance with any safety goal, a high level of confidence in the related risk assessments will be necessary. A high level of confidence will also be necessary to achieve public acceptance. PRA techniques are relatively new and there are too few ski l led practitioners for it to be applied routinely for reactor safety assessment. If design standardization were to result in a large reduction in the number of designs to be reviewed, PRA could be applied more comprehensively to show compliance with a safety goal. By the same token, as the development of PRA techniques continues, confidence in their application will increase and the number of skilled practitioners will become very much larger. It may then be possible to address a wider range of designs and this aspect of standardization would be less important.

6S Levine, "TM I and the Future of Reactor Safety, " Atomic In-Cfu$trlal Forum International Publ{c Affairs Workshop, Stockholm, Sweden,] une 1980

THE IMPACT OF STANDARDIZATION ON RESOLUTION OF GENERIC ISSUES

A December 1977 amendment to the Energy Reorganization Act of 1977 required NRC to submit to Congress a list of unresolved safety issues and plans for their resolution. Progress on resolution is to be included in NRC's annual report to Congress. Prior to that, NRC had developed task-action plans for a multitude of outstanding topics, many of which were not considered unresolved safety issues. 1 n January 1979, NRC submitted a report to Congress identifying 17 unresolved safety issues and their related task-action plans. [7] A more recent plan updates the status of these issues and plans.[8] The 17 issues are listed in table 6.

As a result of the many investigations of the TM I accident, NRC published an action plan in May 1980.[9] This report contains actions to be carried out by each nuclear plant owner and the NRC. One might consider these as generic safety issues; however, they are resolved issues in that specific action is called for. Also, these actions are applied to all operating plants, as well as those under construction. Thus, standardization would not have changed these action plans.

As an example of the effect of standardization on a safety issue, consider item 1 of table 6, "water hammer. " The phenomenon is similar to the banging of steam-heated radiators commonly found in old homes or office build-

'Nuclear Regulatory Commlsslon, " Identification of Unresolved Safety Issues Relating to Nuclear Powerplants, " NUREG-O51O, January 1979

"Nuclear Regulatory Commission, "Task Action Plan for UnreSOlVed Safety Issues Related to Nuclear powerplants, " NUREG-0649, February 1980

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

Table 6.—Unresolved Safety Issues

1. Water hammer
2. Asymmetric blowdown loads on the reactor coolant system
3. Pressurized water reactor steam generator tube integrity
4. BWR Mark I and Mark II pressure suppression containment
5. Anticipated transients without scram
6. BWR nozzle cracking
7. Reactor vessel materials toughness
8. Fracture toughness of steam generator and reactor coolant pump supports
9. System Interactions in nuclear powerplants
10. Environmental qualification of safety-related electrical equipment
11. Reactor vessel pressure transient protection
12. Residual heat removal requirements
13. Control of heavy loads near spent fuel
14. Seismic design criteria
15. Pipe cracks in boiling water reactors
16. Containment emergency sump reliability
17. Station blackout

SOURCE: Nuclear Regulatory Commission.

ings. Occurrences have been attributed to rapid condensation of steam pockets, steam--driven slugs of water, pump startup with partially empty lines, and rapid-valve motion. Much of the problem might therefore be resolved by piping arrangement to assure filled lines and prevent steam pockets. This would, of course, be easier to resolve in standardized layouts as opposed to those of differing plant designs. Although there has been no release of radioactivity outside the plant's boundary because of a water-hammer incident, the frequency of such events and the potential safety significance of the systems involved caused NRC to consider the water-hammer problem significant. Were most plants of standardized design, modifications to prevent recurrence of many safety-related problems could be carried out more rapidly as fewer designs need be examined.

Resolution of another issue, related to containment emergency sump reliability, would also be quicker if designs were standardized. Although NRC has issued guidance for containment sump design and testing, there are still concerns about blockage of sump filters and loss of ability to draw water from the sump. With fewer designs to investigate, the emergency sump reliability issue could be settled much quicker.

The previous discussion indicates that standardization would have facilitated resolution of some of the unresolved safety issues and therefore improved nuclear powerplant safety. On the other hand, there are issues that would be unaffected by standardization. For instance, the disclosure by Virginia Electric Power Co. that asymmetric loads in the reactor vessel supports and vessel internals caused by a PWR pipe break could cause a safety problem, was the result of studies with computer codes using more detailed analytical models. In other words, advances in the state of the art uncovered a problem. In that case the discovery would have occurred at about the same time in the advancement of the technology, whether or not standardization had been implemented.

Finally, several situations have occurred where similarities in plant standardization resulted in many nuclear plants experiencing the same problem — a lesser degree of similarity (i. e., less standardization) could have limited the number of plants involved. One example of this was the realization that hydrodynamic loads on the suppression pool associated with loss-of-coolant accidents and safety-relief valve discharge were not considered in the design of Mark I and Mark II BWR containment. These loads affected 24 Mark I and 11 Mark II plants. Another example is the BWR nozzle-cracking problem associated with feedwater systems of many BWRS of similar design—18 of 21 units inspected had cracks in feedwater nozzles.

For the most part, these generic issues arose when operating experience or advances in the state of the art uncovered a problem, a discovery which would have occurred at about the same time in the advancement of the technology, with or without standardization. Resolution of some of the issues would be expedited if affected nuclear plants were more standardized, while resolution of other issues would not be affected had standardization been more prevalent.

STANDARDIZATION AND ANTITRUST

As noted in chapter 3, the AEs normally enter into a contract with the utility to provide engineering services for the proposed nuclear plant including procurement of material for the BOP. However, the utility selects the NSSS from the four available vendors based on competitive bidding. The reactor, much like the turbine generator, is considered for the purpose of procurement as a large single piece of equipment. The utility normally does not involve itself with the selection of the vendor's supplier other than to assure they are qualified. In many cases, the vendor may have already completed procurement through existing contracts with its suppliers. On the other hand, the BOP equipment and materials are procured by competitive bidding for each plant to satisfy the State agencies regulating the utilities.

In order to perform safety reviews of proposed nuclear plants, the NRC staff prefers to have as much detailed design as possible. The level of detail provided by the vendors is sufficient for this purpose, even before actual construction of the plant begins. However, the AE cannot supply as detailed a design as can the NSSS vendor because the procurement of material and detailed design work has generally not been completed at the time the CP is issued.

The exclusion of any qualified supplier of plant equipment due to licensing requirements for a standard design is a breach of antitrust law. Increasing the level of detail in design for the BOP to the same level found in the NSSS would exclude qualified suppliers from the market place, due to the differences in business methods.

By taking into account the antitrust due process in the setting of standards for plant systems and equipment, the antitrust problem can be eliminated. Due process in standards-making according to the Department of J ustice includes:'"

- adequate notice of the proposed adoption of a standard;
- standards development meetings should be open to the public;
- the standards-setting body should have an affirmative obligation to seek consumer and small business opinion; and
- membership on standards development committees should represent a balanced cross-section of all affected parties.

The development of standards which specify sufficient detail to perform a safety review by knowledgeable engineers under the above guidelines should be sufficient to satisfy the concern over anticompetitive practices and protect the health and safety of the public. A subcommittee of the Atomic Industrial Forum is currently working on a proposed revision to the current NRC guidance on information required for a safety analysis report for single-stage licensing. In addition, at least two AEs and one vendor are considering similar proposals.

Of the four standardization approaches considered, the continuation of present policies with refinement already being considered by the industry is the least likely to create problems with antitrust. The safety-block concept would not create any more difficulties than the acceleration of present policies, although it would place more of the total plant under the design control of the vendors to the exclusion of the AE. However, the AE's role as an engineering services contractor would be affected since design work encompasses only about 10 percent of the total cost of the facility. The "national single design" could force one or more NSSS vendors from the marketplace. The specifications for the design could be written to allow the vendors to remain competitive suppliers under contract to the utility for equipment and systems. Each vendor would have to evaluate its interest in the supply business, based in part on the similarity of the national design components to its own. However, the single-design standardization approach has the greatest antitrust problems due

[10] John H Sherrefield, **Department of Justice, "Standards** for Standards-Makers (Washington, D C Department of Justice, American National Standards Institute, March 1978)

to the reduction of the NSSS vendors to suppliers and the possible exclusion of large portions of their product line from the national single-plant design.

UTILITIES AND STANDARDIZATION

A utility which operates and maintains a nuclear powerplant is uniquely responsible to the Federal and State Governments for the protection of public health and safety. In addition, the utility is responsible to the stockholders for the efficient operation of the plant and the protection of plant investment in equipment and fuel supply (i. e., the reactor's core). These are not mutually exclusive goals and measures which protect the core, increase plant availability, and protect the public. Because of this unique relationship between the utility, its stockholders, and government, nuclear utilities should actively participate in the formulation of any standard design or approach to standardization.

Over the past 25 years, some utilities that have purchased nuclear powerplants have had minimum influence on their design due in part to the lack of expertise in nuclear design engineering. Therefore, these utilities placed heavy reliance on the judgment of the AEs and vendors to protect their financial and regulatory interests. Other utilities, such as Duke Power and Tennesee Valley Authority have acted as their own designers and have maintained a strong influence in the design and construction of their plants. It is also this latter group of utilities which have maintained a strong commitment to standardization as evidenced by their recent construction record for duplicate plants. However, having only a few utilities committed to standardization may not be enough to reap its benefits if a resurgence in new plant orders occurs.

A utility organization could, over the next 2 or 3 years, develop standards and criteria for new plants which incorporate the cumulative operating experience of the industry. These criteria should concentrate on safe, conservative designs and reemphasize the past practice of simply meeting licensing requirements. This effort would result in a set of criteria for everyone (e. g., designers, operators, and regulators) and lend consistency to their actions, Common, understandable objectives could be established which concentrate on the real issues of safety and reliability. The effort should include input from AEs, vendors, and perhaps NRC. Inclusion of NRC should be limited to their role as regulators not designers or operators.

Once the criteria are set, standard designs could be developed. Future construction dockets could then be limited to these designs and thereby allow the marketplace to limit the number. Single-stage licensing would be a considerable inducement to the whole process.

FEASIBILITY

Of the approaches to standardization considered, the acceleration of present trends and procedural standardization are the most feasible to achieve. These approaches work within the existing structures and motivations of the commercial nuclear industry. Organizations such as NSAC and INPO have already been established as a result of the TM I accident and are in excellent positions to develop and promote these forms of standardization. In addition, these institutions were established by the utilities and the utilities are solely responsible for their success or failure. Such utility organizations could fill the role described previously for the development of design standards and criteria. The burden for standardization should rest with the utilities as they are ultimately responsible for commercial nuclear power and also have the most to lose in the event of an accident.

As discussed earlier, trends in the industry over the past 25 years have led to some standardization. This trend can be greatly accelerated by implementing single-step licensing (or NRC's standard-design approval) and regulating the industry in a consistent well-defined fashion. The development and implementation of a safety goal would certainly assist the regulation of the industry. However, its absence should not deter the development of the standards and criteria necessary for the next generation of nuclear powerpl ants.

Under the safety-block concept, the vendor, either alone or in conjunction with an AE, would develop and obtain regulatory approval of a standard design which consolidates in a single design certain parts of the plant which traditionally have been split between the vendor and the AE. This would enable one designer or design group to have total system responsibility for the entire nuclear part of the plant and to better anticipate the impact of various events on the entire plant. This approach would eliminate a number of interfaces that create difficulties in design and licensing, since all the systems crucial for licensing would be inside the safety-block portion of the plant. Approval of the power-generating systems should be wholly routine. The safety block approach should therefore facilitate the licensing process and allow a more thorough design approval to take place. In either case, the AE firms would retain the bulk of their function. This concept would require the vendor and perhaps the AE to expand their scope of design responsibilities and accept the resulting additional liability, The utility, therefore, would have to accept a lower degree of involvement than under the acceleration of present policies.

The single-standard design would require creating an entirely new design organization. This has the very real possibility of disrupting the existing institutions which design, construct, operate, and regulate nuclear plants. Given the possibility of replicating an undetected safety flaw in all the plants of a single-standard design and the necessity of relating operating experience to the mixed set of plants already in place, the safety benefits of such an approach are doubtful. The single-design approach has the greatest problems with antitrust as well. The existing Atomic Energy Act would have to be drastically modified to enforce this approach and would transfer the incentive and responsibility for design improvements from the industrial participants, who now have the responsibility, to an umbrella design organization. There is no private industry in the United States that has undergone such a radical change. The net effect of imposing a single design on the utilities is impossible to judge.

An alternative approach is to have a separate body go ahead with the design of a "national reactor" or "yardstick" design, even without a commitment to actually build them. This exercise would allow a comparison with existing designs and possibly would bring improvements to them. Such a design would have to recognize the problems associated with combining components or systems in ways not previously done and without any operational experience base for its performance. Such a yardstick could more easily be achieved by tightening the existing criteria to meet the utilities requirements for availability, reliability, and safety. This yardstick could then be used outside the licensing and regulatory framework to measure the relative weaknesses or strengths of existing designs.

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