NUCLEAR REACTOR ENGINEERING

REACTOR SYSTEMS ENGINEERING

FOURTH EDITION VOLUME TWO

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Contents

Preface

Dr. Samuel Glasstone, the senior author of the previous editions of this book, was anxious to live until his ninetieth birthday, but passed away in 1986, a few months short of this milestone. I am grateful for the many years of stimulation received during our association, and in preparing this edition have attempted to maintain his approach.

Previous editions of this book were intended to serve as a text for students and a reference for practicing engineers. Emphasis was given to the broad perspective, particularly for topics important to reactor design and operation, with basic coverage provided in such supporting areas as neutronics, thermal-hydraulics, and materials. This, the Fourth Edition, was prepared with these same general objectives in mind. However, during the past three decades, the nuclear industry and university educational programs have matured considerably, presenting some challenges in meeting the objectives of this book.

Nuclear power reactors have become much more complex, with an accompanying growth in supporting technology. University programs now offer separate courses covering such basic topics as reactor physics, thermal-hydraulics, and materials. Finally, the general availability of inexpensive powerful micro- and minicomputers has transformed design and analysis procedures so that sophisticated methods are now commonly used instead of earlier, more approximate approaches.

In light of this picture, giving priority to needed perspective, even at the expense of some depth which is now available elsewhere, was considered appropriate. Also, since it was important to keep the length of the book about the same, necessary new material could only be accommodated by deleting some old material. Significant new material has been added, particularly in the areas of reactor safety, fuel management, plant operations, and advanced systems. However, material that generally continues to meet the objectives of the book has been retained, both to preserve its flavor and to keep the revision effort within reasonable bounds. Also, after the passing of Dr. Glasstone, I felt it inappropriate to change the basic approach of the book.

Readers of the book will want to use computer-based methods to supplement the text material, as appropriate. Space did not permit a meaningful presentation of the methodology required. All problems listed at the end of the chapters may be solved with hand calculations. Although I have continued the use of SI units, as begun in the Third Edition, complete adoption by industry has been slower than anticipated. Therefore, some problems utilize English units.

A two-volume format has been adopted for this edition to provide readers with some flexibility. The chapters have been rearranged somewhat to provide volume coherence, with basic material concentrated in the first volume. An Instructor's Manual is also available for qualified instructors, to be ordered directly from the publisher.

The suggestions made by A. L. B. Ho, L. E. Hochreiter, B. K. Malaviya, V. H. Ransom, J. R. Redding, G. R. Odette, and T. G. Theofanous are gratefully acknowledged.

Thanks are due to the Chapman & Hall team that published the book, particularly Marielle Reiter for production administration and Barbara Zeiders for editorial assistance.

Finally, I wish to thank my wife for her help and encouragement during the preparation of the book.

Alexander Sesonske
San Diego, California
March 1994

CHAPTER 8

The Systems Concept, Design Decisions, and Information Tools

INTRODUCTION

8.1. The availability of new, powerful digital computers in recent years has resulted not only in increased sophistication in nuclear reactor engineering but also in the development of other disciplines involving systems and optimal control theory. These subjects have potential applications in reactor engineering. In nuclear power plant design and operation, decisions must be made in applying engineering principles to the problems to be solved. Since plants are complex, aids to the decision-making process are essential. One aid is to recognize certain portions of the plant having a common function as a *system*, a term that we will explain further shortly. Decisions regarding dependencies between systems can be expedited by this type of representation.

8.2. The systems concept as a decision tool has its roots in a subject known as *operations research* [1], which evolved from the strategic planning needs of World War II. As powerful computers became available, sophisticated methods for systems analysis, modeling, and simulation were developed for engineering and management applications. In this computer-based information age, we are witnessing a trend toward greater and greater general use of such systems-related tools for decision making. Computer modeling of nuclear power plant behavior is essential for normal operation as well as for the analysis of postulated accidents to meet licensing requirements (§12.231 et seq.).

8.3. Space here does not permit even an introduction to operations research and modeling procedures. Reliability analysis applications will be mentioned in Chapter 12. However, we do wish to alert the reader to the possibility of using available operations research-based methods for the analysis and design of nuclear reactor systems. A first step here is to identify some typical systems.

8.4. In modern engineering practice which utilizes the power of the computer to accomplish most tasks, information of various kinds is a necessary tool for decision making. Nuclear engineers have available to them a variety of data bases, computer codes, and other information from many sources. An introduction to these resources is useful as a complement to the presentation of relevant reactor engineering topics in this book. Also, we have drawn attention to appropriate information sources in various other chapters.

SYSTEMS

Introduction

8.5. A nuclear power plant consists of a number of systems which interact with one another in various ways. Knowledge of these intersystem interactions is essential for the design and operation of the plant. The systems themselves represent applications of the principles described in the previous and subsequent chapters of this book. We believe that it is useful at this point to introduce the system concept as applied to nuclear power plants and to indicate some of the intersystem dependencies. As energy transport, fuel management, safety, and other topics are presented in subsequent chapters, the reader can then be alert to the integration of the subject matter in plant applications.

8.6. The term *system* may be defined as a group of *interrelated* elements, or subsystems, forming the "whole" in a regular, organized manner [2]. In engineering design, the concept lends itself to mathematical modeling in which the interrelationships can be expressed formally for purposes of design optimization. Nuclear engineers also use such a systems modeling approach for probabilistic risk analysis (§12.206 et seq.).

8.7. Just what constitutes a system or subsystem depends on the need for analysis of the interdependencies that may exist. A system may have many subsystems or parts which, in turn, could have within them "sub-subsystems," usually referred to merely as subsystems. Where does this subdivision stop? The key question is whether the needs of the problem at hand require further subdivision. If not, the constituent part is referred to as a component.

8.8. Unfortunately, the term *system* is often used loosely to refer to any collection of components even if the grouping is, in fact, a subsystem. For example, manufacturers' literature for some pressurized-water reactors describes the reactor coolant "system" as within the Nuclear Steam Supply System. Also described are auxiliary fluid systems and electrical, instrumentation, and control systems. Systems for waste processing, fuel storage, and ventilation are also covered. To be consistent with such practice, we will refer to both specific systems and subsystems as "systems" but indicate their hierarchial relationship as appropriate. The nature of some of these systems will be reasonably clear based on the material already presented. Others will become clearer as subsequent chapters are read.

PWR Nuclear Steam Supply System

Reactor core system

8.9. Although not normally labeled as such, the reactor core is indeed a system, with such components as the fuel assemblies and control elements. Here, we see an example of the need sometimes to define the boundaries of a system arbitrarily since the control elements could be included in the control system.

Reactor coolant system

8.10. Strictly speaking, the reactor coolant system includes those components associated with the flow of the primary coolant, such as the coolant pumps, pressurizer, and associated piping and valves. However, the reactor vessel and steam generators could also be included arbitrarily.

Instrumentation and control system

8.11. This system includes the following major subsystems:

1. Nuclear instrumentation that indicates power level
2. In-core instrumentation to provide nuclear flux distribution
3. Process instrumentation for nonnuclear measurements in steam system supply components
4. Reactor protection system for initiating safety in response to abnormal conditions (§12.22)
5. Control room (§14.25)

Engineered safeguards system

8.12. Countermeasures in the event of an accident are provided for by this system. Included in this system are several subsystems, such as the safety injection systems, containment sprays, the emergency feedwater system, and other safety features (§12.33 et seq.).

Other systems

8.13. The Nuclear Steam Supply System also includes a number of other subsystems that we need not consider here, such as those for auxiliary fluids, containment, fuel handling, waste management, and chemical control.

THE COMPUTER AS A DECISION TOOL

System Modeling

8.14. The usefulness of the systems concept as a decision tool is enhanced if the system and its behavior can be described in a clear manner. Graphical representation is a classical approach. However, with a wide variety of computers now available, some type of representation that permits mathematical manipulation is preferred. This could take various forms, which could involve graphics, analytical descriptions, or combinations of each. For example, in risk management, graphical representation is useful but manipulation follows the rules of Boolian algebra (§12.218).

8.15. System behavior is normally described by quantitative relationships. For example, coolant circulating pumps may be described in terms of performance specifications such as flow rate, pressure differential, etc. On the other hand, some systems may involve qualitative or subjective inputs in addition to quantitative relationships. For example, in developing a model for a radioactive waste storage facility, technical requirements such as the desired amount and activity of radioactive isotopes are essential. However, there is also the likely need to input items such as local acceptance considerations and regulatory requirements. Economic factors may also require input into the model.

8.16. In analytically modeling a system in such a way that associations between systems can be manipulated by computer, quantitative relationships can be handled in a straightforward manner. However, to permit similar computer manipulation of subjective modeling inputs, they must be quantified in some way, often a significant challenge.

System Interactions

8.17. The systems concept is a useful design tool for the analysis of how component and subsystem failures can propagate and affect performance.

For example, in a PWR, a reactor coolant pump could fail as a result of an electrical fault. The resulting reduction in flow would, in turn, raise the core water temperature, possibly cause boiling, then raise the fuel temperature, all of which would affect the reactivity as determined by the appropriate coefficients. This, and similar chains of events, can be treated analytically. Thus, a systematic study of intersystem dependencies provides the designer with a thorough examination of various modes of system response which would not otherwise be apparent.

8.18. Fault tree and event tree analyses (§12.213) are related analytical procedures for assessing the reliability of safety-related systems. As will be discussed in Chapter 12, these approaches are valuable decision tools for identifying "weak links" in intersystem functional dependencies.

Design Interactions and Intersystem Dependencies

8.19. In the design of various parts of the Nuclear Steam Supply System (NSSS), the effect of one parameter on another must be considered. For example, in a PWR, the fuel rod diameter and spacing affect both the moderator/fuel ratio and the coolant pressure loss through the core (§9.110). Throughout the NSSS, there are numerous design parameters which form an interrelated matrix which must be addressed systematically during the design process in order to develop specifications. A simplified indication of the need for integrating various parametric contributions in the design process is shown in Fig. 8.1. To simplify the figure, we have omitted the equally important areas of safety design and economics-related design, which also interrelate with the three areas indicated.

Sensitivity Analysis and Design Parameter Interactions

8.20. After the system has been modeled and intersystem dependencies identified, it is desirable to carry out a *sensitivity analysis*. Assume that the system in question is described by *input* parameters that may be varied and by *output* functions which characterize the behavior of the system and might, in turn, serve as input to another system. We would like to know how sensitive the output functions are to adjustments to the input parameters. Should some parameters have little effect on the output, the model could be simplified. Also, should one or more input parameters be found to have an overriding effect, such input parameters would be identified as being critical to the design. At any rate, the sensitivity analysis step provides insight to the workings of the system and is a decision tool.

Feedback

8.21. As part of our study of reactor control, we learned how *feedback* could affect the behavior of a system, particularly its stability (§5.132).

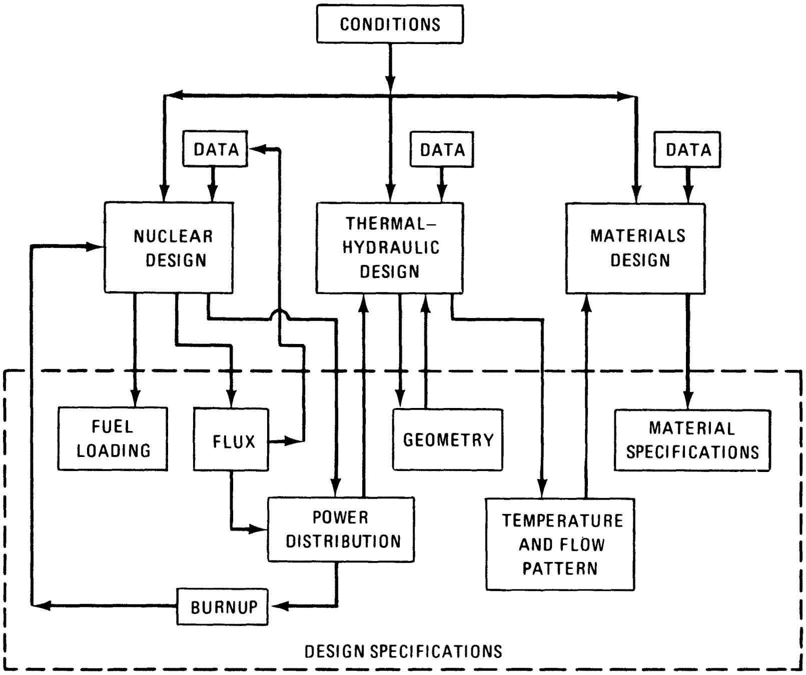

Fig. 8.1. Integration of parametric contributions.

Should the output of our system serve as the input to a second system or subsystem and then have one of the second system outputs return as input to the original system, we have a feedback effect. In the case of reactor control, the second system could either reinforce or inhibit an original disturbance signal, leading to an unstable or a stable condition, respectively.

8.22. In the reactor design logical flow diagram shown in Fig. 8.1, the required fuel loading, flux pattern, and power distribution evolve from the nuclear design, and some feedback is provided to both the nuclear data and thermal-hydraulic design activities. Thus, we have a typical procedure initiated by a *trial* design which is then refined by feedback. Although not shown in the figure, the process involves an evaluative stage in which results are compared with goals. This procedure lends itself to automation using standard computer programming practice.

Optimization

8.23. In engineering design, there is often the desire not only to use the systems concept to develop a *feasible* solution to a given problem, but to use various formal methods to find an optimum or *best* solution. To accomplish this, the various parameters controlling the system are adjusted to yield "behavior" that is best in relation to a prescribed so-called "figure of merit" or design goal.

8.24. Application of various formal optimization approaches to nuclear engineering problems, particularly to those in fuel management, has attracted the attention of researchers for many years [3]. However, since most methods require extensive iteration by computer, approximations in modeling the core in fuel reloading problems (§10.40) have been necessary to maintain a practical level of calculation effort. Therefore, the resulting uncertainty in the results has limited commercial interest in applying existing methods. This picture could change as improved core modeling methods are developed utilizing newer computers, particularly those based on parallel processing. Separate from the fuel reloading problem, the optimum design of the total plant to provide energy at minimum cost has been standard practice for many years. In the case of the plant, the number of significant parameters is modest compared with the reload core. Thus, computer modeling of the plant design problem is easily managed.

Expert Systems and Artificial Intelligence [4]

8.25. As the power of computers increased over the years, their use to accomplish so-called "thinking" processes which simulate certain human activities became the subject of a subdiscipline known as artificial intelligence (AI). Applications have varied from simple game playing (chess) to speech recognition. Normally required are a bank of stored knowledge and a search procedure in accordance with specified rules that lead to a decision. The boundaries of these efforts are not well defined. For example, many optimization methods follow a similar approach, although optimization is not normally considered an AI procedure. Our initial attention will be focused on "expert systems," a so-called single-purpose AI procedure that is now widely used commercially [5].

8.26. An *expert system* is a computer program that incorporates the knowledge and reasoning of human experts as a decision tool. A typical expert system includes a *knowledge base*, and *inference engine*, and a system–human *interface*. The knowledge base includes available relevant information from data bases as well as various rules-of-thumb facts developed from experience by so-called "heuristic" processes. The inference engine manipulates the knowledge base information by programmed rules of logic to reach the desired conclusions. Translation of input into computer language and then from the computed results to output is accomplished by the interface feature.

8.27. The major advantage of this type of program is the ability to apply the wisdom of experts systematically in a repeatable manner to the solution of a problem without the experts being present. Prescribed logic paths for decision making are incorporated in the inference engine. "Reasoning" techniques may be forward "chaining" starting from a set of known facts or backward chaining in which the system works backward from goals and tries to establish needed supporting evidence. Common features are "if-then," "and," and "or" decision gates. Inprecise or "fuzzy" information may be incorporated into the program through the use of "certainty factors," which represent judgments of the validity of a fact on a numerical scale of 0 to 1, where 1 means complete certainty.

8.28. In nuclear reactor control rooms, expert systems can provide the operator with guidance regarding measures to be taken in the event of unanticipated incidents, thus reducing operator error. However, the applicability of an expert system is limited by the scope of information in the knowledge base. Therefore, development of expert systems to cover a very wide range of emergency situations is taking place. Other uses include the monitoring of the performance of various plant components and recommending operating changes; various training situations, including the analysis of trainee performance; and many management-related applications.

8.29. *Neural network* procedures utilize a type of computer architecture in which the processing elements are interconnected so that a great many calculations can be carried out in parallel, imitating the way that neuron cells in the brain process information. The theory and descriptions of some neural network applications are available in a number of references [6]. In general, neural networks consist of processing units, arranged in layers, with connections between units. These connections have "weights" which have a memory capability and may be "trained" to adjust to changing conditions. Information is processed in a distributed manner rather than by a conventional computer serial approach. Hence a general characteristic is the ability of a network quickly to recognize the various conditions or states of a complex system once it has been suitably "trained." Thus, applications in pattern recognition, language translation, and speech understanding are typical.

8.30. Applications to the operation of nuclear power plants are being developed. For example, efficient pattern recognition can be a useful diagnostic tool in interpreting control room data. Another area is the modeling of nonlinear systems with application to process dynamics [7].

Computer Code Sources

8.31. The computer is an essential tool for modern nuclear engineering practice. Although hand calculations are valuable for instruction, conceptual visualization, scoping, and preliminary design, the complexity of nuclear systems demands computer calculations for final design. Hence, there is a need for computer programs to meet a wide variety of requirements. Although writing a custom program may be necessary for some applications, it is first desirable to explore the availability of developed codes that will meet the need. Therefore, familiarity with the sources of available computer codes is a useful engineering tool.

8.32. An essential first step in code use is to become thoroughly familiar with the problem to be solved and to determine what codes are best for the given task. In addition to developing background from the literature, consultation with experts experienced in solving similar problems may be necessary. In the process, it is likely that code availability will be determined. Our purpose here is merely to point out some common code sources should it be necessary to use them.

8.33. The National Energy Software Center at Argonne, Illinois serves as a central computer program library in the areas of reactor physics, reactor engineering, and design. Periodically, it publishes a compilation of code abstracts. The center has available for distribution to qualified users well over 1000 programs.

8.34. A second major resource is the Radiation Shielding Information Center at Oak Ridge, Tennessee (§6.159). Although the emphasis in the computer code collection is on radiation transport and shielding, much information is also provided on data analysis. User code packages are provided by the Center, which has a collection of several hundred programs.

8.35. The Electric Power Research Institute (EPRI), the research arm of the electric utility industry, is responsible for significant code development. Software packages for EPRI-sponsored work are available to member utilities and others under a licensing arrangement from the Electric Power Software Center, operated by the Power Computing Co., a commercial information service in Dallas, Texas.

8.36. For software not available from central collections, it is often possible to obtain packages directly from the code developer by making appropriate arrangements. However, proprietary considerations may limit availability.

INFORMATION AS A DECISION TOOL

Introduction

8.37. Familiarity with information resources is an important decision tool for the nuclear reactor engineer. Traditionally, the published literature has been the first step in developing background to solve a given problem. Although an understanding of relevant principles is still essential, convenient access to a variety of data bases and other information can expedite problem solving. Our purpose here is to draw attention to some of these resources. A useful guide to nuclear engineering resources is given by Jedruch [8].

8.38. Most practicing engineers recognize the need to update their expertise continually through routine scanning of professional society news magazines and journals devoted to their speciality. We assume that they have access to a technical library and know how to use it to find articles in the archival literature and books of interest. Computerized catalogs are available in most major libraries. Therefore, extensive searches can be carried out quickly. Our objective here is to remind the nuclear reactor engineer of several other important information sources.

Abstracts

8.39. *Atomindex*, published by the International Atomic Energy Agency, provides comprehensive abstract coverage of the entire field of nuclear engineering and science. *Energy Research Abstracts* published by the U.S. Department of Energy provides coverage of nuclear and other energy-related activities with particular attention to those is the United States. Research reports, conference proceedings, and various safety-related regulations are covered. It is issued semimonthly and is indexed annually. Therefore, it represents a convenient first step for a literature search. Since it is arranged by subject matter, with detailed subcategories, it is easy to browse through contributions in areas of interest. Sources for the literature cited are given. Another resource is the *Engineering Index*, which covers all areas of engineering.

Nonarchival Literature Sources

8.40. In addition to archival journals, some other literature sources are of particular interest to nuclear engineers.

Reports

8.41. It has been customary over the years to publish the results of work sponsored by the U.S. government in the form of both progress reports and final reports. Although some of this work eventually makes its way into the archival journals, the report literature provides a picture of ongoing activities. Various types of reports are also required as part of the reactor licensing process (§12.237). For example, *Safety Analysis Reports* are an excellent source of information regarding power reactor systems. The Electric Power Research Institute in Palo Alto, California also maintains a Research Report Center for its sponsored work. The *EPRI Journal*, a periodical that describes EPRI activities, lists newly issued reports.

Standards and codes

8.42. Standards and codes have played an important role in engineering practice for many years. The American Society of Mechanical Engineers Boiler Code had its beginnings in the nineteenth century, for example. Presently, a large body of standards developed under the auspices of the respective engineering professional societies, including the American Nuclear Society, are coordinated by the American National Standards Institute (ANSI). However, the American Nuclear Society maintains the Information Center on Nuclear Standards (ICONS) for all nuclear standards, including those developed by other societies.

Federal regulations, guides, and periodicals

8.43. A great deal of technical information is provided in the regulatory literature. Title 10 of the *Code of Federal Regulations* (10 CFR) is devoted to nuclear energy and consists of a number of parts. For example, Part 20 (10CFR20) deals with the standards for protection against radiation. The various parts of the *Code of Federal Regulations*, which have the force of law, are supplemented by *Nuclear Regulatory Guides*. These are advisory and are coordinated in some cases with engineering society standards and codes. There are several hundred published guides divided into ten categories. They are revised periodically.

8.44. The U.S. Department of Energy publishes several periodicals and newsletters. For example, *Nuclear Safety* is a technical progress review issued four times per year.

Data Centers

8.45. Once he or she has developed the methods needed to solve a given problem, the nuclear engineer frequently needs such specialized information as neutron cross sections, physical constants, material properties, etc. To meet this need, a number of centers have been established to maintain special collections termed *data bases*. In many cases, the information is stored in a computerized retrieval system that can be searched on-line by the end user. Another approach has been to collect formulations for properties and other information so that they can be made available on a suitable computer tape. Only several of the major data centers will be mentioned here. Other information sources are indicated elsewhere in the text, as appropriate.

8.46. The National Technical Information Service (NTIS) in Springfield, Virginia maintains a very large collection of reports and other materials resulting from activities of the federal government. Data bases, such as the Energy Data Base, are available for on-line searching.

8.47. The Technical Information Center at Oak Ridge, Tennessee, operated for the U.S. Department of Energy, is responsible for a wide variety of services, including abstracting of reports and publication of technical progress reviews. It prepares the Energy Data Base available from NTIS.

8.48. The earliest center, originally devoted primarily to nuclear cross sections, is the Brookhaven National Nuclear Data Center at Upton, New York. Data evaluation remains a major responsibility. Results are available through publications and computer tapes.

8.49. The Center for Information and Numerical Data Analysis and Synthesis (CINDAS) at West Lafayette, Indiana is an example of a data center providing engineering information. Published data describing thermophysical and other properties of many materials are collected and evaluated. A data base with on-line interactive capability is maintained.

Data Access Networks

8.50. The ability to access various data bases in information centers by computerized communications networks is a major convenience. In most cases, the computerized data base with interactive capabilities is maintained by the responsible center, while access is obtained by means of a commercial communications network such as SPRINTNET or BT TYMNET [9]. Generally, a commercial *service provider* serves as an intermediate link between the communication network, which is "dialed up" by the user, and the various data bases. Such providers, which offer numerous information and search services, generally provide catalogs of data bases that are available through them [10]. A given data base may be offered by more than one provider. In some cases, a provider may also offer communication network service.

8.51. On-line interactive access to some data bases may be available directly by telephone. For example, the U.S. Department of Energy provides an on-line search service known as FEDIX for universities and other research organizations. No registration or access fees are required. The Federal Telecommunications Service (FTS) also provides network access to many data bases to U.S. government contractors. RECON, developed by the Technical Information Service for Department of Energy contractors and other agencies, is an interactive search system for a large number of data bases of interest to the nuclear engineering community. Included are such areas as safety information, codes available at the National Energy Software Center (§8.33), and electric power data.

8.52. The NUCLEAR NETWORK, operated by the Institute of Nuclear Power Operations (INPO), provides direct communication between reactor licensees, reactor vendors, and other nuclear power–related groups such as the Nuclear Safety Analysis Center of the Electric Power Research Institute. Most use is devoted to communications rather than literature searches.

8.53. The INTERNET, when it began in 1969 as ARPANET, linked a few universities, government agencies, and defense-related laboratories. During recent years, it has grown at a rapid rate to accommodate commercial as well as institutional users and has become the world's largest computer network. Since most networks are connected to the INTERNET, it has become in a sense a worldwide network of computer networks. A large variety of services are available through the INTERNET, including electronic mail, access to data bases, and on-line discussion groups [11]. Public access is available through commercial providers (§8.50).

Information Age Challenges

8.54. Making effective use of all of the information resources available to help the engineer solve a given problem is somewhat of a challenge. The discussion above is intended only as an introduction to the types of services available, which are being continually expanded and updated. Therefore, guidance from an information service professional or technical librarian is desirable.

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CHAPTER 9

Energy Transport

INTRODUCTION

The Role of Energy Transport in Reactor Design

9.1. Most of the fission reaction energy deposited in fuel is immediately converted to heat. If the fuel is to remain at steady state (constant temperature), the heat must be transported away at the same rate as it is generated. Although we have seen that negative reactivity feedback, if present, tends to limit power increases, it is essential for the heat generation–heat removal rate balance to be maintained, to prevent temperatures that might result in the failure of fuel and structural materials. Thus, the design of the core depends just as much on adequate heat removal as on nuclear considerations.

9.2. Effective heat removal is a function of many design parameters, including the fuel geometry, coolant flow characteristics, and properties of materials, as well as related neutronic behavior. In many aspects of the design, conventional engineering principles of heat transfer and fluid mechanics are applicable. The term *thermal-hydraulics* is commonly used to describe the effort involving the integration of heat-transfer and fluid mechanics principles to accomplish the desired rate of heat removal from the core under both operating and accident conditions. The purpose of this chapter is to introduce this area of nuclear reactor engineering.

9.3. Various accidental causes of a mismatch between the fuel heat generation and heat removal capabilities of the coolant leading to reactor damage are carefully analyzed as part of the required safety analysis to be discussed in Chapter 12. Of particular concern are events that lead to a reduction in the coolant mass flow rate. Since each unit mass of coolant would then receive more energy, boiling is a probable consequence. Such a transition to a two-phase system would result in an increase in flow resistance, further aggravating the unstable situation. Therefore, the thermal-hydraulics of two-phase systems plays an important role in reactor design. Analysis is required for the design of safety features which are provided in a reactor plant to reduce the consequences of a serious accident in which the core might be damaged. Although the relevant thermal-hydraulics basics will be discussed in this chapter, treatment of the design applications will be deferred until Chapter 12.

Thermodynamic Viewpoint

9.4. As background for the thermal-hydraulics aspects of reactor design, it is helpful to consider the path of energy transport from the thermodynamic viewpoint. In a nuclear power plant, heat flows from the fuel elements to a coolant, then perhaps to a secondary coolant, as in a PWR, to form steam. Work is done by expanding the steam in a turbine. The turbine exhaust steam is then condensed and recycled. Heat extracted during the condensation process is rejected to the environment. Thus, we have a movement of energy from the heat generation at high temperature in the fuel elements progressively downward on the temperature scale until the environment is reached. Along the way, a portion of the original energy is converted to useful work by a steam expansion in the turbine that is almost thermodynamically reversible. However, at each heat flow step, an irreversible temperature "drop" is needed to accomplish the heat transfer at a practical rate. As we shall see, this temperature difference acts as the *driving force* for the thermal transport rate process.

9.5. The thermodynamic efficiency of the conversion to work is improved if the heat input from the steam to the turbine is at as high a temperature as possible and the heat rejection in the condenser is at as low a temperature as possible. Since the "high" temperature in a water-cooled reactor is usually limited by materials and pressure considerations, and the "sink" or rejection temperature by environmental and other factors, the irreversible temperature "drops" required for heat transfer between these two limits must be expended carefully by the thermal system designer.

Design Methods

9.6. Engineers have used design methods describing the transfer of heat and the movement of fluids for many years. In a typical reactor core, the transport of heat from the fuel to the moving coolant involves the traditional processes of conduction and convection. Further steps in the thermodynamic energy path described above also involve the behavior of fluids in motion, particularly the effects of a second phase, as is produced by boiling. Over the years, a substantial body of literature has been developed describing the principles relevant to thermal system design. Early, largely empirical design methods have been supplemented by models having some theoretical basis. Finally, the general availability of powerful computers has made a reasonably sophisticated description of transport processes practical for design purposes.

9.7. Since it is not our purpose here to treat the principles of conductive and convective heat transfer from the beginning, readers who have not been introduced to this area are advised first to consult an elementary text. Of importance is the concept of a rate process and the role of a mathematical representation of the conservation of mass, energy, and momentum in the description of such processes. With this background, we are then able to introduce the highlights of nuclear reactor thermal-hydraulics. Discussions of sophisticated analysis methods are beyond the scope of this book but are available in standard sources [1].

9.8. Our objective here is merely to provide a picture of energy transport considerations in a reactor system using simple analysis methods. Such methods may be useful for preliminary design or scoping, but appropriate computer codes are needed for subsequent more detailed design. However, it is important to recognize that every method has a level of confidence or range of error that should be identified by the user.

HEAT SOURCES IN REACTOR SYSTEMS

Fission Energy

9.9. The energy released in the core by fission appears in various forms, but mainly as the kinetic energy of the fission fragments, the fission neutrons, and the beta particles resulting from radioactive decay of the fission products (see Table 1.2). The fission fragments are usually stopped within the fuel elements themselves; the small fraction that escapes into the cladding penetrates only about 0.01 mm. The beta particles of high energy may travel up to 2 mm in a cladding material such as zircaloy (§7.117), and so a large fraction of these particles may escape from the fuel element into the moderator or coolant, but they will not get out of the reactor core.

The fission neutrons lose most of their energy in the first few collisions with moderator atoms, and they travel distances ranging from some centimeters to a few feet. It is seem, therefore, that most of the heat from the three sources under consideration, comprising about 90 percent or more of the total energy generated, will be released within the reactor core.

9.10. The remaining 10 percent, or less, of the energy produced in fission appears as gamma rays which are distributed throughout the reactor core and surrounding components in a manner dependent on the specific materials and geometries involved. Thermal stresses as a result of gamma heating of the reactor vessel and surrounding radiation shielding must be considered by the designer (§7.30).

9.11. As we have seen (§2.213), heat generation by radioactive decay of the fission fragments continues after the fission reaction ceases. Therefore, provision must be made for cooling the fuel elements after shutdown. Particularly important are emergency core cooling features to be effective in the case of an accidental loss of normal cooling capability (§12.33). Also, the role of gamma heating affects the after-shutdown spatial distribution somewhat. For example, 1 hour after shutdown, the heat generation rate in the fuel elements will be about 1.5 percent of the operating value, whereas in the reflector and shield it will be approximately 10 percent of the rate during operation.

9.12. It was seen in Chapter 1 that the total energy released in fission, which ultimately appears as heat, is made up of contributions from a number of sources. In general, the total energy, exclusive of the neutrino energy which is lost to the reactor system, may be expressed by

$$E \approx 191 + E_c \text{ (in MeV)},$$

where E_c is the energy liberated as a result of various parasitic neutron capture processes, e.g., nonfission capture in uranium-235 and uranium-238, and capture in moderator, coolant, structure, etc.; this includes the energy of the capture gamma radiations and the decay energies, i.e., the energies of the alpha and beta particles and gamma rays, of any radioactive species that are formed by parasitic neutron capture.* Since the value of E_c will obviously depend upon the nature of the materials present in the reactor core, it is evident that the total amount of heat produced by fission will vary, to some extent, from one type of reactor to another.

*Although E_c should include the decay energies of all radioactive species formed as a result of nonfission neutron reactions, only those of relatively short half-life contribute to the energy release in the state of pseudo-equilibrium attained by the reactor soon after startup.

Example 9.1. Determine the total energy release in the core of a pressurized-water reactor (PWR) having volume fractions of uranium oxide (UO_2), water, and iron of 0.32, 0.58, and 0.10, respectively. The oxide fuel (density = 10.2×10^3 kg/m^3) has an average enrichment of 2.8 percent ^{235}U, and the average cooling water density is 0.69×10^3 kg/m^3. The energy released per neutron captured in uranium, water, and iron may be taken as 6.8, 2.2, and 6.0 MeV, respectively.

The solution requires the calculation of the macroscopic neutron capture cross section for each component (except for the oxygen in UO_2 which is very small) as well as the ^{235}U macroscopic fission cross section. The final result depends on ratios of cross sections rather than absolute values; hence, the tabulated values for 0.0253-eV neutron microscopic cross sections may be used in the calculations. The capture-to-fission ratios are then as given in the following summary:

	^{238}U	^{235}U	H_2O	Fe
N (nuclei/m^3 $\times 10^{28}$)	0.71	0.020	1.34	0.85
σ_c (b)	2.7	98.6	0.664	2.55
σ_f (b)	—	582	—	—
Σ_c (m^{-1})	1.9	1.97	0.89	2.2
Σ_f (m^{-1})	—	11.6	—	—
$\Sigma_c/^{235}\Sigma_f$	0.16	0.17	0.076	0.19

The energy released as a result of nonfission captures is

$$E_c = (6.8 \times 0.16) + (6.8 \times 0.17) + (2.2 \times 0.076) + (6.0 \times 0.19)$$
$$= 3.6 \text{ MeV}$$

The total energy released per fission in this core is thus

$$E = 191 + 3.6 \approx 195 \text{ MeV} = 3.1 \times 10^{-11} \text{ J}.$$

Spatial Distribution of Energy Sources in Reactor Core

9.13. As already noted, a large proportion of the energy from fission is available in the form of heat within a very short distance of the fission event. The total rate of heat generation is proportional to the fission rate, i.e., to $\Sigma_f\phi$ or $N\sigma_f\phi$, where σ_f is the microscopic fission cross section, ϕ is the neutron flux, and N is the number of fissile nuclei per unit volume of fuel. If N remains uniform throughout a particular reactor or reactor region, as an initial approximation, the thermal source function, expressed as the power density, may be taken to be the same as the spatial distribution of the neutron flux.

9.14. The overall flux distribution in the reactor core can be calculated by the methods described in Chapter 3 and 4. For a reflected cylindrical reactor in which the fuel is distributed uniformly, the flux distribution in the core and reflector can be represented approximately by

$$\frac{\phi}{\phi_{\max}} = J_0\left(2.405\frac{r}{R'}\right)\cos\frac{\pi z}{H'}, \tag{9.1}$$

where $\phi_{\max}$ is the flux at the center of the core where it is assumed to have its maximum value; J_0 is the zero-order Bessel function of the first kind, r and z are the radial and vertical coordinates, as in Fig. 3.11, and R' and H' are the effective radius and height, respectively, of the reactor, including an allowance for the reflector.

9.15. The average neutron flux in the core is given by

$$\phi_{\text{av}} = \frac{1}{\pi R^2 H}\int_0^R \int_{-1/2H}^{1/2H} \phi 2\pi r\, dr\, dz, \tag{9.2}$$

where R and H are the actual radius and height of the fuel-containing region. Upon substituting the expression for ϕ from equation (9.1) and performing the integration, the result is

$$\frac{\phi_{\text{av}}}{\phi_{\max}} = \frac{2RR'J_1\left(2.405\dfrac{R}{R'}\right)}{2.405R^2}\cdot\frac{2H'\sin\left(\dfrac{\pi}{2}\cdot\dfrac{H}{H'}\right)}{\pi H}, \tag{9.3}$$

where J_1 is the first-order Bessel function.

9.16. If the concentration of fissile material is uniform throughout the reactor core, the power density may be taken to be proportional to the neutron flux.* It then follows that the reciprocal of equation (9.3) is equal to $P_{\max}/P_{\text{av}}$, i.e., the ratio of the maximum to the average power density. For a reflected cylindrical reactor, R/R' and H/H' may be taken as roughly 5/6, and then

$$\frac{P_{\max}}{P_{\text{av}}}\text{ (reflected)} \approx 2.4.$$

*Commercial power reactors have radial fuel regions with different enrichments in uranium-235; as a result, the power distribution differs from that for uniform enrichment assumed here. Also, the axial power density is likely to be skewed for reload core loadings as a result of previous operational history (see Chapter 10).

For a bare reactor, i.e., one without a reflector, R/R' and H/H' are both unity; then equation (9.3), after inversion, yields

$$\frac{P_{max}}{P_{av}}\text{(bare)} = \frac{2.405}{2J_1(2.405)} \cdot \frac{\pi}{2} = 3.64.$$

If in a bare cylindrical reactor the arrangement of fuel elements or control rods (or both) is such that the flux is uniform in the radial direction, the factor containing the Bessel function is unity, and P_{max}/P_{av} is $\pi/2$, i.e., 1.57. Similar calculations have been made for reactors of other shapes, and the results of some core geometries are given in Table 9.1. It may be noted that, because of the arguments presented in §3.54, based on the similarity of the curves in Fig. 3.12, a cosine distribution of the flux may be assumed in all cases, e.g., even in the radial direction of a cylindrical reactor.

Average and Maximum Power in Single Fuel Channel

9.17. In reactor cooling problems, it is often of interest to estimate the maximum heat-generation rate (or power density) at a point in a given fuel channel, rather than for the whole reactor core. In equation (9.1), the first (Bessel function) term gives the radial flux distribution, whereas the second (cosine) term represents the axial distribution in a cylindrical reactor. In any specified axial fuel channel, at a fixed radial distance r from the reactor center, the neutron flux distribution is

$$\phi_{axial} = (\phi_{max})_{axial} \cos\frac{\pi z}{H'}, \tag{9.4}$$

where $(\phi_{max})_{axial}$, the maximum flux at the center of the given channel, is equal to ;$\phi_{max}J_0(2.405r/R')$.

TABLE 9.1. Ratio of Maximum to Average Power Densities in Reactors With Uniform Fuel Distribution

Core Geometry	P_{max}/P_{av}
Sphere (bare)	3.29
Rectangular parallelepiped (bare)	3.87
Cylinder (bare)	3.64
Cylinder (bare, flat radial flux)	1.57
Cylinder (reflected)	2.4
Pool type (water reflected)	2.6

9.18. The average flux in an axial fuel channel in a cylindrical core is given by

$$(\phi_{av})_{axial} = (\phi_{max})_{axial} \frac{1}{H} \int_{-1/2H}^{1/2H} \cos \frac{\pi z}{H'} \, dz$$

$$= (\phi_{max})_{axial} \frac{2H' \sin\left(\frac{\pi}{2} \cdot \frac{H}{H'}\right)}{\pi H}.$$

If the fuel enrichment is uniform in the axial direction, as is sometimes the case, the axial power density will be proportional to the neutron flux; hence,

$$\left(\frac{P_{max}}{P_{av}}\right)_{axial} = \frac{\pi H}{2H' \sin\left(\frac{\pi}{2} \cdot \frac{H}{H'}\right)}. \tag{9.5}$$

9.19. The result expressed by equation (9.5), or the simplification for the core without end reflectors, is also applicable to fuel channels in a rectangular parallelepiped reactor. For this geometry, the flux distribution in any direction parallel to one of three principal axes is represented by an expression analogous to equation (9.4). The only change necessary is to replace H by the actual length of the reactor core in the given direction, and H' by the effective length including allowance for the reflector.

Power and Flux Flattening

9.20. In most reactor designs, the power output is limited by the maximum permissible temperature of the fuel elements, i.e., by the value of P_{max}. In some cases, it is possible to increase the power output by decreasing P_{max}/P_{av}. In other words, the power output of the reactor can be increased by a more uniform distribution of the power density. This is sometimes referred to as a "flattening" of the power distribution or of the neutron flux.

9.21. We have seen that the presence of the reflector provides some flux flattening (§3.136). In addition, reduction in the concentration of fissile atoms in fuel that would otherwise have a high neutron flux could provide a flattening effect. Such fuel management schemes, described in Chapter 10, may also provide improved utilization of the available neutrons and decrease the cost of the electricity produced by the reactor plant.

Other Heat Sources

9.22. About 3 percent of the heat generated in the reactor system is produced in the moderator as the result of the slowing down of fission neutrons, the stopping of beta particles from the fission products, and the absorption of gamma rays from various sources. In light-water reactors, this presents no special cooling problem. Cooling requirements in other systems may be a function of the design. The need for cooling should also be recognized in the design of control elements. Heat is generated in such components outside the core as the thermal shield and reactor vessel in water reactors and the radiation shield in all reactors. This heat is a result of the absorption of escaping neutrons and gamma radiation. The subject of heat generation in shields has been treated in Chapter 6.

HEAT-TRANSMISSION PRINCIPLES

Introduction

9.23. Three general mechanisms are distinguished whereby heat is transferred from one point to another, namely: (1) conduction, (2) convection, and (3) radiation. These three methods for heat removal will be reviewed and their general characteristics outlined. The subsequent discussion, however, will be devoted to the first two, since they are of major importance in existing reactor designs. The preliminary treatment will refer to heat transmission in systems without internal sources, and subsequently the problems associated with internal sources, such as exist in reactor fuel elements, will be taken up.

Conduction of Heat

9.24. The term *conduction* refers to the transfer of heat by molecular (and sometimes electronic) interaction without any accompanying macroscopic displacement of matter. The flow of heat by conduction is governed by the relationship known as the Fourier equation, i.e.,

$$q = -kA\frac{dt}{dx}, \tag{9.6}$$

where q is the rate (per unit time) at which heat is conducted in the x direction through a plane of area A normal to this direction, at a point where the temperature gradient is dt/dx.* The quantity k, defined by equa-

*The quantity q/A, having the dimensions of heat/(time)(area), i.e., W/m^2, is called the *heat flux*.

tion (9.6), is the *thermal conductivity*. In SI units, q is expressed in J/s (or W), A in m^2, and dt/dx in °C/m (or K/m); hence, the units of k are (W/m^2)(m/K), usually written in the form W/m · K. In English units, k will be in Btu/(hr)(ft^2)(°F/ft).

9.25. The thermal conductivity k is a physical property of the medium through which the heat conduction occurs. For anisotropic substances, the value of k is a function of direction; although methods are available for making allowance for such variations, they are ignored in most analytical solutions of conduction problems. The thermal conductivity is also temperature dependent and can generally be expressed in a power series; thus,

$$k = c_0 + c_1 t + c_2 t^2 + \cdots,$$

which, in many cases, may be approximated to the simple linear form

$$k = k_0(1 + at).$$

Where considerable accuracy is desirable (and possible), allowance must be made for the variation of thermal conductivity with temperature. But very frequently, especially when the temperature range is not great, k is taken to be constant. However, when uranium oxide is used as fuel, as it is in most power reactors, the temperature gradients are quite large, and allowance must be made for the variation of the thermal conductivity with temperature (§9.46). Some values of k of interest in reactor design are given in the Appendix.

9.26. Upon integration of equation (9.6) over the x direction, it is found, for unidirectional heat flow by conduction in a slab of constant cross section, with k independent of temperature, that

$$q = -kA\frac{t_1 - t_2}{x_1 - x_2}, \tag{9.7}$$

where t_1 and t_2 are the temperatures at two points whose coordinates are x_1 and x_2. The result means that the temperature gradient at a point, i.e., dt/dx, in the Fourier equation (9.6) may be replaced by the average gradient over any distance, i.e., by $(t_1 - t_2)/(x_1 - x_2)$.

9.27. If $t_1 - t_2$ is replaced by Δt, the temperature difference, and $x_2 - x_1$ by L, the length of the heat-flow path, equation (9.7) upon rearrangement takes the form

$$q = \frac{\Delta t}{L/kA}. \tag{9.8}$$

This expression is analogous to Ohm's law, $I = E/R$; hence the quantity L/kA is often called the *thermal resistance* for a slab conductor. The analogy between conduction of heat and electricity is the basis of the *thermal circuit* concept which is very useful in solving heat-transfer problems. In general, the rate of heat flow q is equivalent to the current I; the temperature difference Δt is the analogue of the potential difference (or EMF) E; and the thermal resistance replaces the electrical resistance.

Convection of Heat

9.28. Heat transmission by *convection*, which is usually concerned with the transfer of heat across a solid-fluid interface, involves macroscopic motion of the fluid. In *free convection*, the motion is a consequence of the buoyant forces generated in the fluid due to temperature differences within it. In *forced convection*, on the other hand, the fluid is moved by mechanical means, e.g., by a pump. When applied to heat removal in reactors, two aspects of convection heat transmission must be considered; there is, first, transfer of heat from the material which is being cooled, e.g., the fuel element, to the coolant; and, second, the transport of this heat, usually in sensible form, by flow of the coolant from one point in the system to another. Fluid flow is thus an important problem, as will be seen later, in the study of reactor cooling systems.

9.29. The fundamental equation of convection heat transfer, for both free and forced motion of the fluid, is the so-called Newton's law of cooling, which may be written as

$$q = hA_h\Delta t, \tag{9.9}$$

where q is the rate of convection heat transfer to or from a surface of area A_h when the temperature difference is Δt. The quantity h, defined by equation (9.6), is commonly called the *heat-transfer coefficient* and is expressed in units of $W/m^2 \cdot K$. It should be noted that equation (9.9) applies to convection heat flow in either direction, i.e., from solid to fluid or from fluid to solid; the actual direction of the flow depends, of course, upon the sign of Δt.

9.30. Although the value of h is dependent upon the physical properties of the fluid medium, it is also a function of the shape and dimensions of the interface, and of the nature, direction, and velocity of the fluid flow. Thus, the heat-transfer coefficient is a property of the particular system under consideration. Another factor which determines h is the exact definition of Δt, i.e., the temperature difference between the surface and the fluid. Whereas the surface temperature of the solid is uniquely defined, that of the fluid is subject to several arbitrary definitions. The latter is usually taken as the "mixed-mean (or bulk) fluid temperature" t_m given by

$$t_m \equiv \frac{\int_{Af} \rho c_p v t \, dA_f}{\int_{Af} \rho c_p v \, dA_f},$$

where ρ, c_p, and v are the density, specific heat, and flow velocity, respectively, of the fluid, and A_f is the cross-sectional (or transverse) flow area of the fluid. The value of the heat-transfer coefficient in any given circumstances can be determined experimentally, but for design purposes, the general practice is to use results predicted by means of various theoretical and semiempirical expressions (see §9.78 et seq.).

9.31. By slight rearrangement, equation (9.9) can be written in the Ohm's law form

$$q = \frac{\Delta t}{1/hA_h},$$

so that, as before, q and Δt are analogous to current and potential difference, respectively, but $1/hA_h$ now represents the thermal resistance to convection-heat transfer. Thermal circuits can thus be constructed with one or more stages of thermal conduction combined with convection for the solution of problems involving both types of heat transfer, as will be shown in the succeeding paragraphs.

Conduction With Convection Boundary Conditions

Infinite Slab

9.32. Although thermal problems in reactors generally involve internal sources, the first (and simplest) case to be considered is that of an infinite slab of finite thickness with *no internal heat source*. It is postulated that heat is transferred at both faces by convection and that the heat-transfer coefficients are h_a and h_b, respectively. The idealization of an infinite slab is a good approximation to real (finite) systems in which the transverse dimension is large compared with the thickness, e.g., the cladding of a parallel-plate type fuel element or the flat plate of a heat exchanger. In the following treatment, steady-state conditions with the thermal conductivity independent of temperature will be postulated.

9.33. The characteristics of the physical system are shown at the left in Fig. 9.1, and the equivalent thermal-circuit diagram is given at the right.

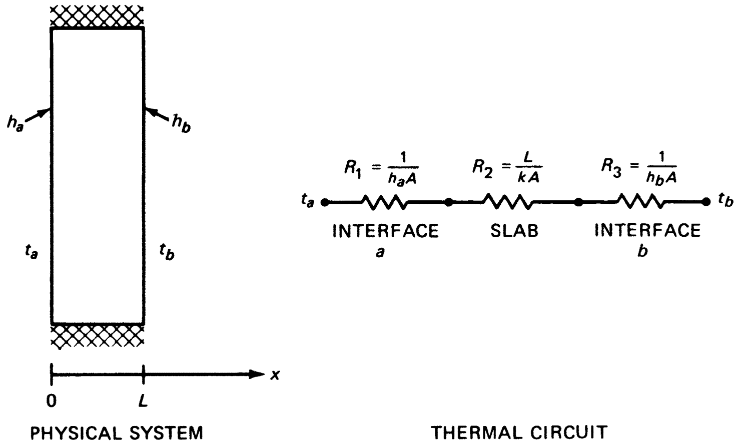


Fig. 9.1. Heat conduction in an infinite slab with convection boundaries.

The total thermal resistance is the sum of three terms, namely, two (R_1 and R_3) for the convection resistances and the third (R_2) for the conduction resistance of the slab; hence, if A is the heat-flow area, the total thermal resistance, R, is

$$R = \frac{1}{h_aA} + \frac{L}{kA} + \frac{1}{h_bA} = \frac{1}{UA}. \tag{9.10}$$

The relationship between the heat-flow rate and the temperature difference $\Delta t = t_a - t_b$, where t_a and t_b are the mixed-mean fluid temperatures at the two faces, is thus given by

$$q = \frac{t_a - t_b}{R} = UA(t_a - t_b)$$

or

$$t_a - t_b = \frac{q}{UA}.$$

The quantity U, defined by equation (9.10), is called the *overall coefficient of heat transfer*; it has the same dimensions as the heat-transfer coefficient.

9.34. These equations may be used to determine the steady-state difference in temperature $t_a - t_b$ between two fluids in a heat exchanger if the heat-flow rate q is given. Alternatively, if $t_a - t_b$ is fixed, the corresponding value of q can be calculated for specified conditions. The individual temperature differences, i.e., the two fluid-solid differences and that between the two faces of the slab, can be determined by the use of equations (9.9) and (9.7), respectively.

Hollow Cylinder

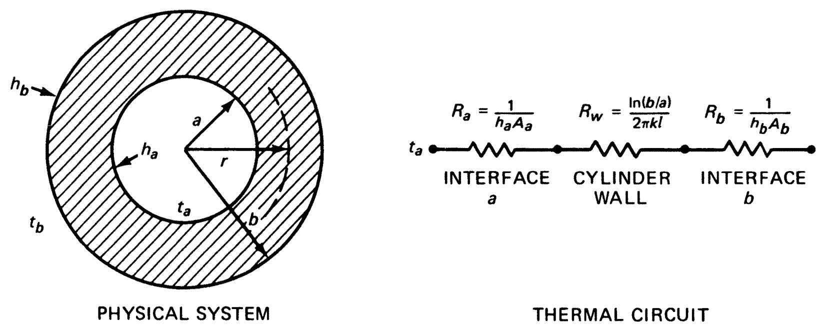
9.35. Tubular heat-transfer surfaces are used in many heat exchangers, and the performance of these can be analyzed on the basis of heat conduction through hollow circular cylinders with convection at both interior and exterior surfaces. The physical system in cross section and the thermal-circuit diagram are shown in Fig. 9.2; it is assumed that all the heat flow is in a radial direction. The inner and outer radii of the cylinder are a and b, respectively; h_a and h_b are the heat-transfer coefficients for the inner and outer fluids, and t_a and t_b are the mixed-mean temperatures of the inner and outer fluids, respectively.

9.36. The convection thermal resistances are equal to $1/h_aA_a$ and $1/h_bA_b$, where A_a and A_b are the areas of the inner and outer surfaces. The thermal conduction resistance of the cylinder wall remains to be determined. If l is the length of the cylinder, then the heat-flow rate, expressed at an arbitrary radius r, due to conduction within the wall is given by equation (9.6) as

$$q = -k(2\pi rl)\frac{dt}{dr},$$

where k is the thermal conductivity and $2\pi rl$ is equivalent to A, the area through which heat flow occurs. This expression may be rearranged and integrated over the wall thickness, i.e., from $r = a$ to $r = b$; the result

Fig. 9.2. Heat conduction in a hollow cylinder with convection boundaries.



is*

$$q = \frac{\Delta t}{\ln(b/a)/2\pi kl}, \tag{9.11}$$

where Δt is the difference in temperature across the cylinder wall.

9.37. It is evident from equation (9.11) that the thermal resistance R_w of the wall is given by

$$R_w = \frac{\ln(b/a)}{2\pi kl}. \tag{9.12}$$

The total thermal resistance of the system under consideration is consequently

$$R = \frac{1}{h_a A_a} + \frac{\ln(b/a)}{2\pi kl} + \frac{1}{h_b A_b} = \frac{1}{U_b A_b},$$

where U_b is the overall heat-transfer coefficient based on the outer surface of the cylinder. According to the thermal-circuit concept

$$q = \frac{t_a - t_b}{R} = U_b A_b (t_a - t_b)$$

or

$$t_a - t_b = \frac{q}{U_b A_b},$$

where

$$U_b = \frac{1}{\dfrac{b}{h_a a} + \dfrac{b \ln(b/a)}{k} + \dfrac{1}{h_b}}. \tag{9.13}$$

Radiation Heat Transfer

9.38. As a result of changes in the thermal motions of its constituent particles, which are a function of the temperature, every body emits energy

*For values of $b/a < 2$, the quantity $\ln(b/a)$, which appears here and in heat-transfer equations for clad fuel rods (§9.47 et seq.), is well approximated by $2(b - a)/(b + a)$.

in the form of electromagnetic radiations over a range of wave lengths; this is called *thermal radiation*. The amount of energy carried by the radiation is unchanged as it passes through a vacuum and is largely unaffected by dry air and many other gases, with the notable exceptions of carbon dioxide and water vapor. But when the radiation falls on a solid body or in its passage through the last-mentioned gases, part or all of the energy is absorbed. The fraction of the incident radiation that is absorbed is called the *absorptivity*. An ideal material which absorbs all the radiation falling upon it, and thus has an absorptivity of unity, is designated a *black body*. The emissive power or thermal radiation flux, i.e., the energy radiated in unit time per unit area of a black body, is given by the Stefan-Boltzmann law, namely,

$$\text{Thermal radiation flux} = \sigma T^4,$$

where the constant σ has the value of 5.68×10^{-8} W/m$^2 \cdot$ K^4. The ratio of the emissive power of an actual surface to that from a black body is referred to as the *emissivity*, and for a body in thermal equilibrium, the emissivity and absorptivity are equal (Kirchhoff's law).

9.39. The emissivity (and absorptivity) of surfaces vary with the nature of the material and its temperature as well as with its physical condition, e.g., roughness, cleanliness, etc. For metals, the emissivity is relatively low; the values generally range from about 0.05 or less for a highly polished surface to 0.2 or 0.3 for a roughened surface. If the metal is covered with an oxide film, the emissivity is greatly increased. Nonmetals have fairly high emissivities, although, in contrast to metals, the values decrease with increasing temperature. The emissivity of graphite, for example, approaches unity at temperatures up to about 800 K (530°C), so that it approximates a black body. It is therefore a good emitter and absorber of thermal radiation. When radiation falls on a body, the proportion which is not absorbed is partly transmitted, e.g., through a "transparent" material such as air, and partly reflected, e.g., by a polished metal surface.

9.40. If two surfaces at different temperatures are separated by a nonabsorbing medium, there is an interchange of radiation between them since they both act as absorbers and emitters. However, the net result is the transfer of energy from the hotter to the colder surface, the rate of energy transfer, e.g., in watts, being given by

$$q_r = A_1 \varepsilon_{1,2} \sigma (T_1^4 - T_2^4),$$

where T_1 and T_2 are the absolute temperatures of the hotter and colder bodies, respectively, A_1 is the surface area of the former, and $\varepsilon_{1,2}$ is an interchange factor which is related to the emissivities (or absorptivities) of the two surfaces; if both bodies were black, i.e., perfect absorbers and emitters, $\varepsilon_{1,2}$ would be unity. Certain geometrical factors should be included in this expression, but they may be disregarded here.

9.41. Even though radiation does not transfer any appreciable amount of energy directly to the coolant in a reactor, it may do so indirectly. In gas-cooled reactors operating at high temperatures, heat is transferred by convection from the fuel to the coolant, which is generally helium gas. The latter is transparent to thermal radiation and so does not absorb any radiant energy directly. However, radiation is transferred through the gas to the moderator (graphite), and this then loses energy by convection to the gaseous coolant. Thus, radiative transfer provides a means for conveying heat from the fuel element to the gas in an indirect manner. Radiation may also be significant in the transfer of heat from one reactor component, such as a fuel element, to another component, thus leading to thermal gradients which must be taken into account in stress analysis.

Systems With Internal Heat Sources

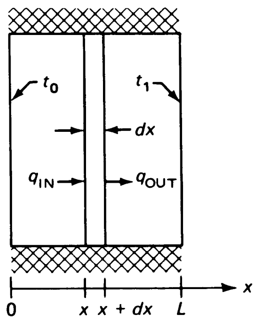
One-Dimensional Slab

9.42. The treatment of heat transmission in the steady state will now be extended to systems with internal sources; for this purpose, the simple one-dimensional case of an infinite slab will be examined (Fig. 9.3). Consider a thin section of the slab of thickness dx at x; then in the steady state the heat-rate balance requires that

Heat conducted out of $A\ dx$

= Heat generated in $A\ dx$ + Heat conducted into $A\ dx$,

Fig. 9.3. Heat transmission in an infinite slab with internal source.



where A is the heat-conduction area and $A\ dx$ is the volume of the thin section. If the thermal conductivity is taken to be independent of temperature (and hence of x), it follows from equation (9.6) that

$$q_{\text{in}} = -kA\left(\frac{dt}{dx}\right)_x$$

and

$$q_{\text{out}} = -kA\left(\frac{dt}{dx}\right)_{x+dx},$$

assuming the heat flow is from left to right. Upon taking the difference of these terms and expanding the expression for q_{out}, it is found that

$$\begin{aligned} q_{\text{out}} - q_{\text{in}} &= -kA\left\{\left[\left(\frac{dt}{dx}\right)_x + \left(\frac{d^2t}{dx^2}\right)_x dx + \cdots\right] - \left(\frac{dt}{dx}\right)_x\right\} \\ &= -kA\left(\frac{d^2t}{dx^2}\right) dx, \end{aligned} \tag{9.14}$$

higher-order terms being neglected.

9.43. If $Q(x)$ represents the volumetric thermal source strength, expressed as heat generated per unit time per unit volume at x, e.g., in W/m^3, then $Q(a)A\ dx$ is the heat generated per unit time in the section of the slab under consideration. It follows, therefore, from the heat-balance equation and from equation (9.14) that in the steady state

$$-kA\left(\frac{d^2t}{dx^2}\right) dx = Q(x)A\ dx$$

so that

$$-\frac{d^2t}{dx^2} = \frac{Q(x)}{k}, \tag{9.15}$$

where $Q(x)$ may or may not vary with x, but k is assumed to be constant.

General Geometry

9.44. It can be shown that equation (9.15) is a particular form of the general equation for the steady state in a system with an internal source, namely,

$$-\nabla^2 t = \frac{Q(r)}{k}, \tag{9.16}$$

where the thermal conductivity k is the same in all directions, i.e., for a truly isotropic medium, and ∇^2 is the Laplacian operator (§3.11). It is apparent that in the one-dimensional case in Cartesian coordinates, equation (9.16) reduces to equation (9.15). For conduction in a cylinder in the radial direction, equation (9.16) becomes

$$-\left(\frac{d^2t}{dr^2} + \frac{1}{r}\cdot\frac{dt}{dr}\right) = \frac{Q(r)}{k} \tag{9.17}$$

and for a sphere

$$-\left(\frac{d^2t}{dr^2} + \frac{2}{r}\cdot\frac{dt}{dr}\right) = \frac{Q(r)}{k}. \tag{9.18}$$

One-Dimensional Conduction in Cylindrical Fuel Rod

9.45. In most power reactors, the fuel consists of cylindrical pellets arranged end to end. This can be idealized as an infinitely long cylindrical rod with a uniformly distributed heat source. Because of the idealized infinite length, heat may be assumed to be conducted in the radial direction only; hence, equation (9.17) is applicable with $Q(r)$, represented by Q, independent of the radial coordinate r since the source is uniform. Provided that k is independent of temperature, this equation may be written in the form

$$-\frac{d}{dr}\left(r\frac{dt}{dr}\right) = \frac{Q}{k}r,$$

which, if integrated twice, leads to

$$t = -\frac{Qr^2}{4k} + C_1 \ln r + C_2, \qquad \text{for} \quad 0 \leq r \leq a,$$

where a is the outer radius of the cylinder (Fig. 9.4). The constants C_1 and C_2 can be evaluated from the boundary conditions which are

$$\frac{dt}{dr} = 0, \text{ for } r = 0 \text{ and } t = t_1, \qquad \text{for} \quad r = a.$$

The appropriate solution is therefore

$$t - t_1 = \frac{Q}{4k}(a^2 - r^2) \qquad \text{for} \quad 0 \leq r \leq a,$$

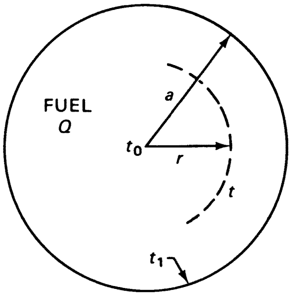


Fig. 9.4. Radial heat transmission in an infinitely long fuel rod.

where t is the temperature at the radial distance r and t_1 is that at the surface. If t_0 is the temperature along the central axis where $r = 0$, it follows that

$$t_0 - t_1 = \frac{Qa^2}{4k} \tag{9.19}$$

for the temperature drop across the fuel rod.

9.46. When the thermal conductivity is not independent of temperature, equation (9.19) may be written as

$$\int_{t_1}^{t_0} k(t)\, dt = \frac{Qa^2}{4},$$

where the quantity on the left side is characteristic of the fuel material for the given temperature range. A related useful parameter is the heat-generation rate per unit length of the cylindrical fuel element (or *linear heat rate*) q_L. Since $q_L = Q(\pi a^2)$, it is seen that

$$\int_{t_1}^{t_0} k(t)\, dt = \frac{Qa^2}{4} = \frac{q_L}{4\pi}. \tag{9.20}$$

Equation (9.20) shows that the temperature difference between the axis and surface of a long cylindrical fuel rod with uniform heat generation depends only on the linear heat rate and is independent of the rod diameter, even if the thermal conductivity varies with temperature. The linear heat rate is therefore a useful design parameter. For example, a linear heat rate that would lead to central melting of the fuel material is usually listed as a limiting core design specification.

Example 9.2. Determine the steady-state temperature difference between the center and the surface of a cylindrical uranium dioxide fuel pellet, enriched to 2.9 percent in ^{235}U, which forms part of a long PWR fuel rod (§1.57). What is the heat flux at the surface of the pellet? (At the specified low enrichment, weight and atomic percentages are essentially the same.)

The outer diameter of the pellet is 8.19 mm and its density is 10.2×10^3 kg/m^3. An average thermal-neutron flux of 4.1×10^{17} neutrons/m$^2 \cdot$ s is applied to the pellet and the conditions are such that the effective microscopic fission cross section of ^{235}U is 360 b.

The first step is to determine the thermal source strength Q; this is equal to $\Sigma_f \phi E_f$, where Σ_f is the effective macroscopic fission cross section in the fuel, ϕ is the thermal-neutron flux, and E_f is the energy released in the fuel per fission, which may be taken to be 200 MeV. Since ϕ and E_f are known, $\Sigma_f = N\sigma_f$ remains to be evaluated; N is the number of ^{235}U nuclei/m^3 of the fuel, and σ_f is the effective microscopic fission cross section given above. This leaves N to be determined.

Since the fuel contains 97.1 percent of uranium-238, it is a good approximation to take the average molecular weight of the uranium dioxide as 270 (0.270 kg/mole). The number of molecules in a mole is 6.02×10^{23} (the Avogadro number); hence,

$$\text{Number of } UO_2 \text{ molecules per m}^3 \text{ of fuel} = \frac{(10.2 \times 10^3)(6.02 \times 10^{23})}{0.270}$$

$$= 2.28 \times 10^{28},$$

where 10.2×10^3 kg/m^3 is the density of the uranium oxide. Each molecule contains one atom of uranium; hence, this is also the number of atoms of uranium per m^3 of fuel. Of this number, 2.9 percent are uranium-235 atoms; it follows, therefore, that

$$\Sigma_f = N\sigma_f = (0.029)(2.28 \times 10^{28})(360 \times 10^{-28}) = 24 \text{ m}^{-1}.$$

The fission rate is $\Sigma_f \phi$, and since fissions at the rate of 3.1×10^{10} s^{-1} produce 1 W of power (§1.48),

$$Q = \frac{(24)(4.1 \times 10^{17})}{3.1 \times 10^{10}} = 3.20 \times 10^8 \text{ W/m}^3.$$

The temperature difference between the center and surface of the pellet is determined from equation (9.19) where a, the pellet radius, is 8.19/2 mm (0.0041 m), and k may be taken to be 2.80 W/m · K for UO_2 in this case. Hence,

$$t_0 - t_1 = \frac{Qa^2}{4k} = \frac{(3.20 \times 10^8)(0.0041)^2}{(4)(2.80)} = 480°\text{C}.$$

Consider a unit of length of a fuel pellet; this has a volume of πa^2, and the heat generated within it is $\pi a^2 Q$. The heat is transmitted in a radial direction out of the surface area of unit length, i.e., $2\pi a$. Hence, the heat flux q/A is given by

$$\frac{q}{A} = \frac{\pi a^2 Q}{2\pi a} = \frac{aQ}{2} = \frac{(0.0041)(3.20 \times 10^8)}{2} = 656{,}000 \text{ W/m}^2.$$

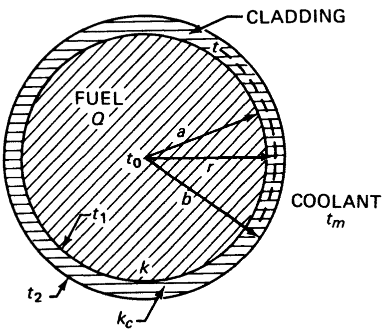
These values are typical for averages in a PWR core. However, as will be seen later, the maximum heat-generation rates, with corresponding temperature differences and thermal fluxes, may be several times the average values.

One-Dimensional Conduction in Clad Fuel Rod

9.47. The cylindrical fuel rods in power reactors are usually clad with a zirconium alloy or in some cases with stainless steel. The heat path is then from the inner fuel cylinder (pellets), through the interface between the fuel and cladding, by conduction through the cladding, and finally by convection from the outer surface of the cladding to the coolant (Fig. 9.5).

9.48. Consider first the fuel cylinder, radius a; if t_0 is the temperature along the central axis and t_1 is the temperature at the fuel-cladding inter-

Fig. 9.5. Radial heat transmission in a clad cylindrical fuel rod.



face, it follows from equation (9.19) that

$$t_0 - t_1 = \frac{Qa^2}{4k}. \tag{9.21}$$

If t_m is the mixed-mean (or bulk) temperature of the coolant surrounding the rod, the transmission of heat through the cladding to the coolant can be represented by the thermal-circuit method; thus,

$$t_1 - t_m = q(R_{\text{clad}} + R_{\text{conv}}),$$

where R_{clad}, the thermal resistance of the cladding, is identical with R_w in §9.37 and is given by equation (9.12); R_{conv}, the convection thermal resistance at the cladding-coolant interface (heat-transfer coefficient = h), is $1/hA$, where A is the outside area of the cylinder of length l, i.e., $2\pi bl$. It follows, therefore, that

$$t_1 - t_m = \frac{q}{2\pi l}\left(\frac{1}{k_c}\ln\frac{b}{a} + \frac{1}{hb}\right),$$

where k_c is the thermal conductivity of the cladding. At the interface between fuel and cladding, where $r = a$, $dt/dr = -Qa/2k$; since the surface area is here $2\pi al$, it follows from a heat-rate balance that

$$q = \pi a^2 l Q$$

and, hence,

$$t_1 - t_m = \frac{Qa^2}{2}\left(\frac{1}{k_c}\ln\frac{b}{a} + \frac{1}{hb}\right). \tag{9.22}$$

This gives the temperature difference between the fuel-cladding interface and the coolant. The first term on the right (including $Qa^2/2$) represents the temperature drop $t_1 - t_2$ across the cladding, and the second term (including $Qa^2/2$) is the drop $t_2 - t_m$ at the cladding–coolant interface.

9.49. Upon adding equations (9.21) and (9.22), the temperature drop from the center of the fuel element to the coolant is found to be

$$t_0 - t_m = \frac{Qa^2}{4k} + \frac{Qa^2}{2}\left(\frac{1}{k_c}\ln\frac{b}{a} + \frac{1}{hb}\right), \tag{9.23}$$

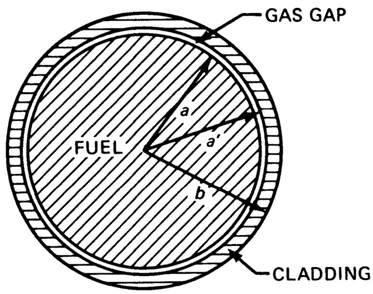
assuming negligible thermal resistance at the fuel–cladding interface. Equations for the temperature distribution in the cladding; or for the difference between the temperature at any point in the fuel or the cladding and the coolant can be readily derived, if required.

Gap Conductance

9.50. As shown in Fig. 9.6, fuel rod designs normally provide for a small annular gap between the uranium dioxide fuel pellet and the cladding. The gap is initially filled with helium, but during irradiation there is some dilution of the helium with fission-product gases. More important, however, is the swelling and cracking of the pellet (§7.172) which tends to close the gap in a nonuniform manner. The interface heat-transfer problem therefore progresses from one involving conduction through the gas to a combination of conduction through the gas and across the interface between two surfaces in partial contact. Also, the gap conductance changes with the power level and material effects related to fuel exposure (Chapter 7). Although analytical models have been developed based on the thermal conductance of a gas film of the thickness indicated by the roughness of the two surfaces, the results tend to be unreliable. Semiempirical computer models based on test data are therefore used for design purposes [2]. The gap conductance is also important in the calculation models required for safety evaluation (§12.128). For orientation purposes, the order of magnitude of the heat-transfer coefficient at the interface of irradiated oxide pellets and metallic cladding may be taken to be 6000 W/m^2 · K.

Example 9.3. A PWR fuel rod (see Example 9.2), with pellets 8.19 mm in diameter, is clad with zircaloy 0.57 mm thick; the outer diameter of the clad rod is 9.5 mm. If the bulk (mixed-mean) coolant temperature is 315°C and the volumetric heat generation rate (power density) in the fuel is 3.20 × 10^8 W/m^3, determine the temperature at the center of the fuel and at the outer surface of the cladding. The heat-transfer coefficient

Fig. 9.6. Section through fuel pellet with gas gap and cladding.



at the cladding-coolant interface may be taken to be 34 kW/m^2 · K. The thermal conductivity of the zircaloy is 13 W/m · K.

The temperature drop between the coolant and the cladding is obtained from the second term on the right of equation (9.22); thus,

$$t_2 - t_m = \frac{Qa^2}{2hb},$$

where

$$Q = 3.20 \times 10^8 \text{ W/m}^3$$
$$a = 1/2\ (8.19 \text{ mm}) = 4.095 \text{ mm}$$
$$b = 1/2\ (9.50 \text{ mm}) = 4.75 \text{ mm}$$

Hence,

$$t_2 - t_m = \frac{(3.20 \times 10^8)(0.004095)^2}{(2)(3.4 \times 10^4)(0.00475)} = 17°\text{C}.$$

Since t_m is 315°C, the cladding surface temperature is 315 + 17 ≈ 332°C.

The temperature drop across the cladding is obtained from the first term on the right of equation (9.22); that is,

$$t_1 - t_2 = \frac{Qa^2}{2k_c} \ln \frac{b}{a'},$$

where, as shown in Fig. 9.6, a' is the inner radius of the cladding, i.e., 1/2(9.50) − 0.57 = 4.18 mm. Hence,

$$t_1 - t_2 = \frac{(3.20 \times 10^8)(0.004095)^2}{(2)(13)} \ln \frac{0.00475}{0.00418} = 26°\text{C}.$$

(Use of the approximation for ln (b/a') in §9.36 footnote leads to essentially the same result.)

Next, consider the temperature drop Δt across the gas gap between the fuel and the cladding. This is given by an expression equivalent to the second term on the right of equation (9.22), except that b is replaced by a and h by the gap conductance. As stated in §9.50, the latter may be taken to be 6000 W/m^2 · K; hence,

$$\Delta t = \frac{Qa^2}{2ha} = \frac{(3.20 \times 10^8)(0.004095)}{(2)(6000)} \approx 109°\text{C}.$$

The temperature drop across the fuel itself has been determined in Example 6.2 as 480°C; hence, the central fuel temperature is given by

$$t_0 = 480 + 109 + 26 + 332 \approx 950°\text{C}.$$

This is an average temperature; it is expected that in some fuel rods the central temperature will be significantly higher. The results of the foregoing calculations are plotted in Fig. 9.7 with the central temperature value rounded off.

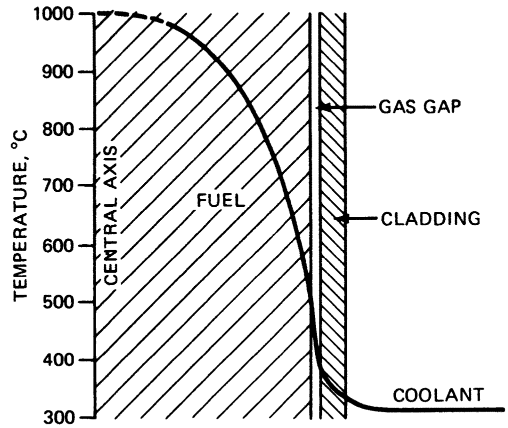
Exponential Heat Source

9.51. In all the preceding examples it has been postulated for simplicity, since it is a good approximation, that the heat source is uniformly distributed within a conductor. A situation will now be considered in which the heat source has an exponential distribution. This is sometimes the case in certain external reactor components, such as thermal shields (§6.90) and pressure vessels, where the heat generated is due to the absorption of gamma rays and the slowing down of neutrons. Although the heat-generation processes are actually quite complex, it may happen that, as the result of a combination of circumstances, the volumetric heat source can be represented approximately by

$$Q = Q_0 e^{-\mu x},$$

where μ is an effective linear attenuation coefficient (or macroscopic cross section) of the radiations (cf. §6.199).

Fig. 9.7. Approximate radial temperature distribution in a fuel rod of a water-cooled reactor.



9.52. If the shield or pressure vessel is a sphere or cylinder of large dimensions or is rectangular in shape, it may be treated as a slab; equation (9.15) then takes the form

$$-\frac{d^2t}{dx^2} = \frac{Q_0}{k}e^{-\mu x}.$$

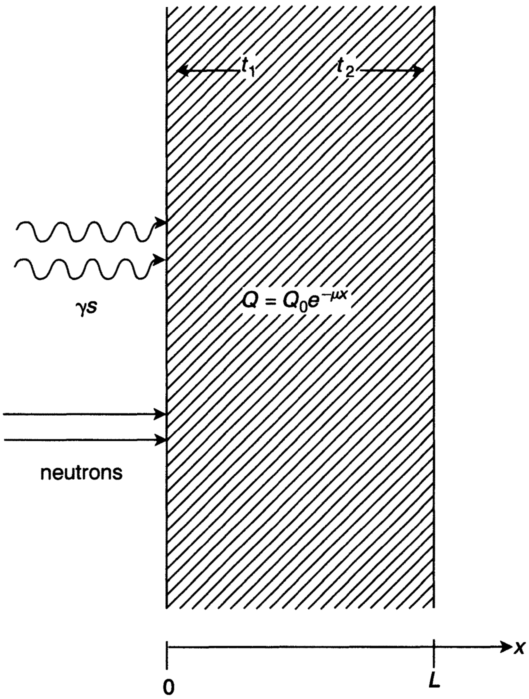
Provided k is constant, the general solution is

$$t = -\frac{Q_0}{k\mu^2}e^{-\mu x} + C_1x + C_2.$$

The boundary conditions are (see Fig. 9.8),

$$t = t_1 \text{ for } x = 0 \qquad \text{and} \qquad t = t_2 \text{ for } x = L,$$

Fig. 9.8. Heat transmission in a slab with exponential source.



where L is the thickness of the slab. These lead to the solution

$$t - t_1 = (t_2 - t_1)\frac{x}{L} + \frac{Q_0}{k\mu^2}\left[(e^{-\mu L} - 1)\frac{x}{L} - e^{-\mu x} + 1\right] \tag{9.24}$$

for the steady-state temperature distribution in the slab. It may be noted that the first term on the right represents the linear temperature distribution due only to the difference in temperature between the two faces of the slab, whereas the second gives the effect of the exponential heat source.

9.53. Under certain conditions, the combination of the two terms just mentioned leads to a temperature maximum within the slab. The point at which this occurs may be obtained by setting the derivative of equation (9.24) with respect to x equal to zero. The result is

$$(x)_{t\max} = -\frac{1}{\mu}\ln\left[(t_1 - t_2)\frac{k\mu}{Q_0 L} + \frac{1 - e^{-\mu L}}{\mu L}\right]. \tag{9.25}$$

The maximum temperature can be obtained by inserting this value for x into equation (9.24).

Example 9.4. A water-cooled and water-moderated power reactor is contained within a thick-walled pressure vessel. This vessel is protected from excessive irradiation (and, thus, excessive thermal stress) by steel, cylindrical thermal shields between the reactor core and the pressure vessel. One of these shields, 50 mm thick, whose surfaces are both maintained at 300°C, receives a gamma-ray energy flux of 10^{18} MeV/m^2 · s. Calculate the location and magnitude of the maximum temperature in this shield, which may be treated as a slab. The coefficient μ of the gamma radiation (§9.51) in the steel may be taken to be 27 m^{-1}, and the thermal conductivity as 40 W/m · K.

In this case, t_1 and t_2 are both 300°C, so that $t_1 - t_2$ is zero. Hence, by equation (9.25),

$$\begin{aligned}(x)_{t\max} &= -\frac{1}{27}\ln\frac{1 - e^{-(27)(0.05)}}{(27)(0.05)}\\ &= 0.022 \text{ m (22 mm)}.\end{aligned}$$

The maximum temperature is thus attained slightly inside from the mid-radius of the shield.

The value of the maximum temperature is calculated from equation (9.24) with $t_1 = 300$°C, $k = 40$ W/m · K, and $x = 0.022$ m. The value of

Q_0 is derived from §6.22 as $\phi_\gamma E_\gamma \mu_a$, where μ_a, the energy absorption coefficient, is 16.4 m^{-1} for steel. Since $\phi_\gamma E_\gamma$ is 10^{18} MeV/m^2 · s, it follows that

$$Q_0 = (10^{18})(16.4) = 1.64 \times 10^{19} \text{ MeV/m}^3 \cdot \text{s}$$
$$= 2.63 \text{ MJ/m}^3 \cdot \text{s}.$$

Hence, from equation (9.24),

$$t_{\max} - 300 = \frac{(2.63 \times 10^6)}{(40)(27)^2}\left[\{e^{-(27)(0.05)} - 1\}\frac{0.022}{0.05} - e^{-(27)(0.022)} + 1\right]$$
$$= 11°\text{C},$$

and $t_{\max}$ is consequently 311°C.

Conduction in Irregular Geometries

9.54. The combination of nonuniform heat-source distribution and irregular geometries, which occur in many reactor components, complicates the calculation of thermal-conduction rates and temperature distributions. The problem is an important one because high thermal stresses frequently result from large temperature gradients. A study of the temperature stresses in reactor components, which frequently have irregular shapes, depends upon a knowledge of the temperature pattern within the materials. A number of relatively simple codes suitable for microcomputers are available to solve multidimensional conduction problems under both steady-state and transient conditions. A stepwise, finite-difference approach, as described in the following sections for transient heat conduction, is effective. An example of such a situation is found in graphite-moderated, gas-cooled reactors. These generally consist of massive graphite blocks pierced by long holes with cylindrical fuel elements and passages for coolant. Pressure vessels containing the reactor and thermal and biological shields frequently present similar difficulties in the determination of temperature distribution and thermal-stress analysis. For such cases, a rigorous solution of the basic heat-conduction equation (9.16) is usually impossible. However, approximate methods have been developed in some instances; thus, a circular harmonics series solution has been proposed for the case of a massive graphite block treated as equivalent to a number of adjacent cylinders [3].

9.55. Temperature patterns for nonregular boundary conditions may be determined using numerical "relaxation" methods. The procedure is first to divide the irregular volume, so far as is possible, into a number of regular subvolumes. The heat-conduction equation may now be applied to these subvolumes in a systematic manner based on an assumed temperature distribution compatible with the boundary conditions. By carrying out a heat-balance calculation for each of the subvolumes, the inaccuracy of the assumed temperature pattern is determined and expressed as a so-called residual for each subvolume. On the basis of these residuals, the assumed temperatures are changed in a systematic manner to reduce (or "relax") the residuals from point to point. The heat-balance procedure and temperature adjustment are repeated until the residuals have been reduced to zero; the corresponding temperature distribution is then taken to be the correct one.

Transient Heat Conduction

9.56. For a reactor operating in the steady state, the spatial distribution of temperature in any component, e.g., a fuel rod, is given by equation (9.16), assuming the medium to be isotropic and the thermal conductivity independent of temperature. In a transient situation, however, such as when the reactor is being started up or shut down, the steady-state condition is not applicable. In deriving the steady-state equation (9.15) for one-dimensional heat flow, of which equation (9.16) is the general form, a heat (or energy) balance was obtained by equating the difference between the rates of heat flow into and out of a volume element to the rate of internal heat generation.

9.57. For the transient situation, the energy balance (or conservation) must include the time rate of change of internal energy in the volume element. This is given by $c_p\rho(A\ dx)(dt/d\theta)$, where c_p is the specific heat of the material and ρ is its density; $A\ dx$ is the volume of the element under consideration (§9.42) and $dt/d\theta$ is the rate of increase of temperature t with time θ. Hence, equation (9.15) becomes

$$-kA\left(\frac{d^2t}{dx^2}\right) dx = Q(x)A\ dx - c_p\rho A\ dx\,\frac{dt}{d\theta}$$

or

$$-\frac{d^2t}{dx^2} = \frac{Q(x)}{k} - \frac{1}{\alpha}\frac{dt}{d\theta}, \qquad (9.26)$$

where α, called the *thermal diffusivity* of the material, is defined by

$$\alpha \equiv \frac{k}{c_p\rho}.$$

The general form of equation (9.26) for the spatial distribution of temperature in a conducting medium with internal heat generation under transient conditions is

$$-\nabla^2 t = \frac{Q(r)}{k} - \frac{1}{\alpha}\frac{dt}{d\theta}, \tag{9.27}$$

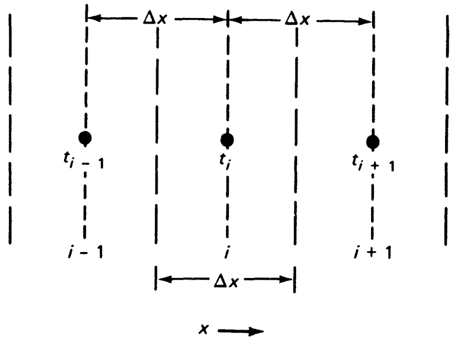
which may be compared with the steady-state equation (9.16).

9.58. In the analysis of transient behavior, the appropriate form of equation (9.27) is generally solved by numerical methods using a computer; for this purpose, the differential equation must be expressed in finite-difference form. To illustrate the general principles used in these (and many other) calculations, the simple case will be considered of conduction in one dimension in a component (or components) which may be regarded as being made up of several adjacent volume elements (or cells).

9.59. The system can be treated as a series of grid points or *nodes*, with the characteristic properties, e.g., the temperature, of each cell being represented by the value at its nodal point. Three such cells, designated $i - 1$, i, and $i + 1$, respectively, with the corresponding nodal points, assumed to be at the cell centers, are shown in Fig. 9.9; the nodal temperatures are t_{i-1}, t_i, and t_{i+1}. The dimension of each cell in the x direction is Δx, and the volume is $A\ \Delta x$, where A is the cross-sectional area of the cell.

9.60. Suppose, for the present, that there is no internal heat generation. Conservation of energy then requires that the difference between the rates of heat flow into and out of the cell i be equal to the rate of increase of internal energy $c_p\rho(A\ \Delta x)(dt_i/d\theta)$. If $q_{i-1,i}$ is the rate of heat flow in the x direction from cell $i - 1$ into cell i, and $q_{i,i+1}$ is the rate of heat flow out

Fig. 9.9. Representation of temperatures at three nodes in one (space) coordinate.



of cell i into $i + 1$, then

$$q_{i-1,i} - q_{i,i+1} = c_p \rho A \, \Delta x \frac{dt_i}{d\theta}. \tag{9.28}$$

From equation (9.7), it follows that

$$q_{i-1,i} = kA \frac{t_{i-1} - t_i}{\Delta x}$$

and

$$q_{i,i+1} = kA \frac{t_i - t_{i+1}}{\Delta x},$$

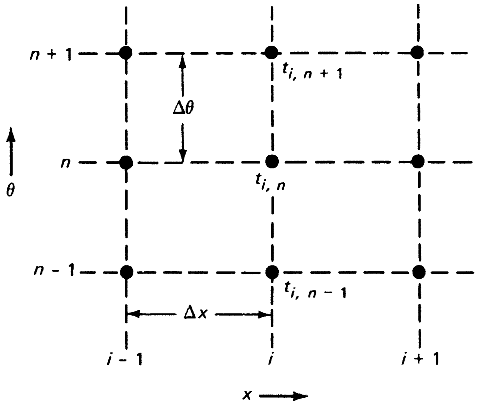
assuming k to be independent of temperature. If these values are inserted into equation (9.28), it is found that

$$\frac{t_{i-1} + t_{i+1} - 2t_i}{(\Delta x)^2} = \frac{1}{\alpha} \frac{dt_i}{d\theta}. \tag{9.29}$$

It can be readily shown that the left side is the linear-difference form of $+d^2t/dx^2$ so that equation (9.29) is the equivalent of equation (9.26) with $Q(x)$ equal to zero.

9.61. In order to express $dt_i/d\theta$ in linear-difference form, the nodal approach is extended into a time coordinate θ as in Fig. 9.10, where $n - 1$, n, and $n + 1$ indicate successive times. For a time step $\Delta\theta$ from n

Fig. 9.10. Representation of temperatures at nodes in space and time coordinates.



to $n + 1$, the temperature change Δt_i in the cell i is $t_{i,n+1} - t_{i,n}$, so that

$$\frac{dt_i}{d\theta} \approx \frac{\Delta t_i}{\Delta\theta} = \frac{t_{i,n+1} - t_{i,n}}{\Delta\theta}. \tag{9.30}$$

Upon insertion of this result into equation (9.29) for the time θ_n and rearranging, it is found that

$$t_{i,n+1} = \frac{\alpha\Delta\theta}{(\Delta x)^2}(t_{i-1,n} + t_{i+1,n}) + \left[1 - \frac{2\alpha\Delta\theta}{(\Delta x)^2}\right]t_{i,n}. \tag{9.31}$$

It is thus seen that if the temperatures t_n at the various nodal points are known at the time θ_n, the temperature t_{n+1} at the time θ_{n+1}, i.e., after a specified time increment $\Delta\theta$, can be calculated directly from equation (9.31). When the values of t at all the nodal points at the time θ_{n+1} are known, the calculation can proceed to the next time step, i.e., to time θ_{n+2}, and so on. This procedure is referred to as the *explicit* method for solving the differential equation (9.26) with $Q(x)$ equal to zero. Internal heat generation may be included, however, without affecting the general principles.

9.62. A simplification of equation (9.31) is possible if the time and distance increments $\Delta\theta$ and Δx are chosen so that the dimensionless quantity

$$\frac{\alpha\Delta\theta}{(\Delta x)^2} = 0.5.$$

It is apparent from equation (6.31) that, in these circumstances, the temperature at a node i after a time step is equal to the mean of the temperatures at the two adjacent nodes before the time step, i.e., $t_{i,n+1} = \frac{1}{2}(t_{i-1,n} + t_{i+1,n})$.

9.63. In any event, it has been shown that for a numerical solution of equation (9.31) to be possible, $\alpha\Delta\theta/(\Delta x)^2$ must be positive and less than 0.5. This limitation on $\alpha\Delta\theta/(\Delta x)^2$ places a restriction on the magnitude $\Delta\theta$ of the time step, relative to $(\Delta x)^2/\alpha$, that can be used in the computations. Consequently, when the calculations must be made for an extended time period, a large number of time intervals may have to be used. Furthermore, it is found that if the time intervals are small, instabilities arise in applying the numerical techniques.

9.64. When, for one reason or another, longer time intervals are required than are permitted by the explicit method, the *implicit* form of the difference equation is used. The spatial finite-difference temperature terms are expressed at the advanced time point $n + 1$ instead of at the point n as in the explicit procedure. That is to say, equation (9.30) is inserted into equation (9.29) in which the temperatures are now $t_{i-1,n+1}$, $t_{i+1,n+1}$, and

$t_{i,n+1}$. It is then found that

$$\left[1 + 2\frac{\alpha\Delta\theta}{(\Delta x)^2}\right]t_{i,n+1} - \frac{\alpha\Delta\theta}{(\Delta x)^2}(t_{i-1,n+1} + t_{i+1,n+1}) = t_{i,n}.$$

9.65. In order to solve this equation for the temperatures at the time θ_{n+1} from the known value $t_{i,n}$ at the preceding time θ_n, it is necessary to write a set of simultaneous equations for all the space points at each time step. These equations can be represented by a tridiagonal matrix, i.e., a matrix in which only the main diagonal and the two adjacent diagonals are nonzero. Such a matrix is readily solved on a computer by the Gauss elimination procedure [4]. The implicit method requires a greater calculational effort at each time step than does the explicit method, but larger time steps can be taken and stability problems are eased; as a result, the overall computer time is often less than for the explicit method.

9.66. In the foregoing discussion a simple situation has been treated to provide some insight into the general principles used in solving transient problems in heat transmission. In practice, boundary conditions, e.g., a convection boundary at the fuel cladding–coolant interface, must be included in the calculations. Allowance may also be required for internal heat generation, which is usually time dependent. In addition, in some cases it may be necessary to determine the spatial temperature distributions in more than one dimension. It should be noted, too, that α, which is a characteristic property of the material, may change from one nodal point to another. The calculations are thus often quite complex and are performed by means of appropriate computer codes.

Transient Heat Transfer

9.67. Transient (or unsteady state) heat transfer is important in reactor safety analysis where the thermal behavior as a function of time is of major interest. Generally, the same principles apply as for steady-state heat transfer with the addition of time and heat capacity as parameters. In particular, consideration must be given to the heat release after shutdown (§2.215). A typical problem might be to determine the maximum temperature attained by the fuel cladding in a water-cooled reactor if the coolant flow is reduced. Even after the reactor is tripped, heat input to the cladding continues; this results from sensible heat stored in the fuel rods which may cause an initial heat-release rate approaching 50 percent of the full power value in a pressurized-water reactor, from fissions by delayed neutrons, and from radioactive decay of the fission products.

9.68. If the coolant-flow rate is decreased, not only will the heat-transfer coefficient decrease, but the thermal transport capacity of the coolant will decrease since less mass will flow past a given point per unit time. Therefore, for the same heat flux (or heat-flow rate) from the cladding, the coolant will tend to attain a higher temperature. The temperature of the cladding as a function of time is then determined by a balance between the heat input from the fuel and the heat loss to the coolant. Since the temperature-difference driving force for each rate depends on the cladding temperature, the situation is complicated and numerical (computer) methods are required for solution of the problem. Subchannel analysis approaches (§9.135) may be incorporated in these codes to provide a detailed representation of core temperature behavior.

HEAT TRANSFER TO ORDINARY FLUIDS

Introduction

9.69. The heat flow from a fuel rod to the coolant proceeds through several resistances in series, the last of which is associated with the heat-transfer coefficient at the cladding–coolant interface. This resistance can vary over a wide range, depending on flow conditions and other parameters, and so it is an important consideration both in the initial reactor design and in the analysis of the consequences of various accident possibilities. For example, an increase in resistance, corresponding to a decrease in the heat-transfer coefficient, would result in an increase in the temperature difference between the cladding and the bulk of the coolant if the heat flux were to remain constant. If the coolant temperature is not to change, the net result would be an increase in the temperature at the outer surface of the cladding, and ultimately in the temperature of the fuel, as can be seen from equation (9.21) et seq. Thus, an examination of the considerations that affect the convection heat transfer is important in reactor design.

Laminar and Turbulent Flow

9.70. When a fluid flows through a straight pipe at low velocity, the particles of fluid move in straight lines parallel to the axis of the pipe, without any appreciable radial motion. This is described as laminar, streamline, or viscous flow, and the "velocity profile" is depicted diagrammatically in Fig. 9.11A; the lengths of the arrows indicate the relative magnitudes of the fluid velocity at various points across the pipe. The velocity distribution curve in a circular pipe is a parabola, and the average velocity is half the maximum at the center of the pipe.

9.71. One characteristic of laminar flow (in the x direction) is that it

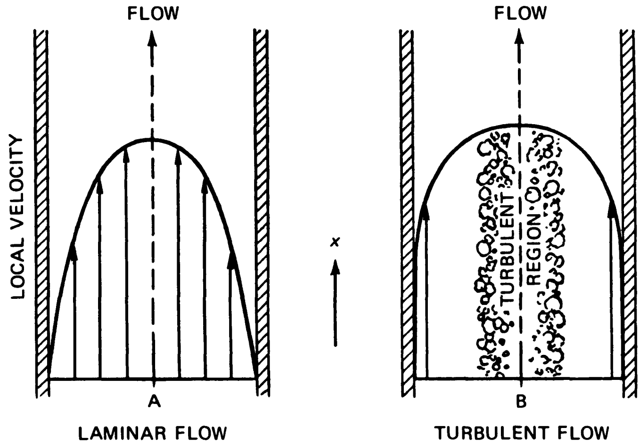


Fig. 9.11. A. Laminar flow. B. Turbulent flow.

obeys Newton's equation,*

$$F = \mu A \frac{du}{dy}, \tag{9.32}$$

where F is the shearing force (or fluid friction) over an area A between two parallel layers of fluid flowing with different velocities u in a region where the velocity gradient perpendicular to the flow direction is du/dy; the symbol μ represents the *absolute* (or *dynamic*) *viscosity* of the fluid and is defined by equation (9.32). The dimensions of viscosity are seen to be mass/(length)(time), i.e., kg/m · s (or Pa · s) in SI units. In the cgs system, the unit of viscosity, expressed in g/(cm)(s), is called the poise, and it is in terms of this unit (or its hundredth part, called a centipoise) that viscosity values have frequently been calculated in the past.

9.72. Laminar flow may be distinguished from other types of flow by means of a dimensionless quantity called the *Reynolds number* (or *modulus*); this is represented by Re and defined by

$$\mathrm{Re} \equiv \frac{Du\rho}{\mu}, \tag{9.33}$$

where D us the pipe diameter, u is the mean velocity of the fluid, ρ is its density, and μ is its viscosity. Experiments with many fluids have shown

*Certain fluids, e.g., suspensions, which do not obey the Newton equation, are referred to as *non-Newtonian fluids.*

that, in general, flow is laminar, and Newton's equation is obeyed in ducts of uniform cross section as long as Re is less than about 2100. The precise critical value of the Reynolds number depends, to some extent, on the flow conditions.

9.73. If the circumstances are such that Re exceeds about 4000 in such systems, the fluid motion is turbulent; this type of flow is characterized by the presence of numerous eddies which cause a radial motion of the fluid, i.e., motion across the stream, in addition to the flow parallel to the pipe axis. Newton's equation is then no longer valid.

9.74. The velocity profile for turbulent flow is shown in Fig. 9.11B. As indicated, three more-or-less distinct regions exist in a turbulent stream. First, there is a layer near the wall in which the flow is essentially laminar. This is followed by a transition (or buffer) zone in which some turbulence exists, and finally, around the pipe axis, there is the fully turbulent core. In the latter region, turbulence mixes the fluid so that the velocity in the axial direction changes less rapidly with radial distance than it does under laminar flow conditions. As a result, there is a marked flattening of the velocity profile; this flattening becomes more pronounced with increasing Reynolds number. The ratio of the mean flow velocity to the maximum ranges from about 0.75 to 0.81, as Re increases from 5000 to 100,000.

9.75. In the range between the Re values of 2100 and 4000, there is generally a transition from purely laminar flow to complete turbulence. Laminar behavior sometimes persists in this region or turbulence may increase, depending, among other factors, on entrance conditions, the occurrence of upstream turbulence, the roughness of the pipe, the presence of obstacles, and on pulsations in flow rate produced by the pump or some other component in the system.

9.76. The Reynolds number, as given by equation (9.33), is applicable only to pipes of circular cross section; for noncircular channels, such as the flow regions between fuel rods, long rectangular ducts, and annular spaces, a good approximation is obtained if D in equation (9.33) is replaced by the *equivalent* (or *hydraulic*) *diameter* D_e, defined by

$$D_e = 4 \times \frac{\text{Cross section of stream}}{\text{Wetted perimeter of duct}}, \tag{9.34}$$

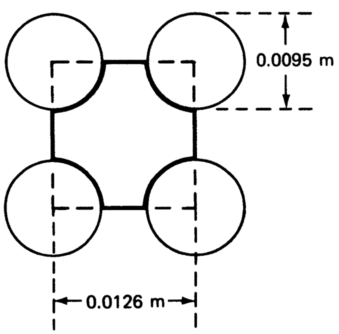
which becomes identical with the actual diameter for a circular pipe. The use of the equivalent diameter makes possible the application of relationships for circular channels to the prediction of heat-transfer coefficients, pressure drops, and burnout heat fluxes for noncircular channels, as will be seen in due course.

Example 9.5. Calculate the average Reynolds number in a PWR from the following data:

Average coolant temperature	311°C
Total coolant mass-flow rate	1.83×10^4 kg/s
Fuel rods, outer diameter	9.50 mm
Pitch (square)	12.6 mm
Rod array in assembly	17×17
Number of assemblies	193

The coolant *mass-flow rate* (in kg/s) is equal to the *fluid velocity u* (in m/s) multiplied by the fluid density (in kg/m^3) and the flow area (in m^2).

To determine D_e, consider a unit cell containing quadrants of four adjacent fuel rods, i.e., the cell is effectively associated with a single rod. The stream cross section and wetted perimeter of the coolant flow channel per rod may then be obtained from the accompanying illustration. Thus,



$$\text{Stream cross section} = (0.0126)^2 - \tfrac{1}{4}\pi(0.0095)^2 = 8.79 \times 10^{-5}\ \text{m}^2.$$

$$\text{Wetted perimeter} = \pi(0.0095) = 0.0298\ \text{m}.$$

Hence,

$$D_e = \frac{(4)(8.79 \times 10^{-5})}{0.0298} = 1.18 \times 10^{-2}\ \text{m}.$$

Next, it is useful to determine the *mass velocity G*, which is equal to $u\rho$; this is obtained by dividing the given mass-flow rate by the corresponding flow area (or stream cross section multiplied by the total number of fuel rods*). Hence,

$$G = \frac{1.83 \times 10^4}{(17)(17)(193)(8.79 \times 10^{-5})} = 3730\ \text{kg/m}^2 \cdot \text{s}.$$

*A small flow at the periphery of the core fuel assemblies is neglected here.

Then, taking the viscosity of water at 311°C to be 8.8×10^{-5} Pa · s (kg/m · s) as given in the Appendix, it follows from equation (9.33) that

$$\text{Re} = \frac{D_e u \rho}{\mu} = \frac{D_e G}{\mu} = \frac{(1.18 \times 10^{-2})(3730)}{8.8 \times 10^{-5}} = 500{,}000.$$

9.77. Although the use of the equivalent diameter is convenient for preliminary design calculations, certain limitations should be recognized. The theoretical validity of the equivalent diameter concept for generalized use has, in fact, been questioned. For example, for a duct having a cross section in the shape of an isosceles triangle, the turbulent flow friction for air was found to be 20 percent lower than the value calculated using the equivalent diameter [5]. Heat-transfer coefficients also varied considerably around the perimeter. Several correlation procedures have been proposed for ducts with noncircular cross sections, but additional experimental work is required for their verification.

Heat-Transfer Coefficients of Ordinary Fluids

9.78. In essentially all cases of reactor cooling where forced convection is used, the coolant is under turbulent flow conditions. Satisfactory predictions of heat-transfer coefficients in long, straight channels of uniform cross section can be made on the assumption that the only variables involved are the mean velocity of the fluid coolant, the diameter (or equivalent diameter) of the coolant channel, and the density, heat capacity, viscosity, and thermal conductivity of the coolant. From the fundamental differential equations or by the use of the methods of dimensional analysis,† it can be shown that heat-transfer coefficients for turbulent flow conditions can be expressed in terms of three dimensionless moduli; one of these is the Reynolds number, already defined, and the others are the *Nusselt number* (Nu) and the *Prandtl number* (Pr), defined by

$$Nu \equiv \frac{hD}{k} \quad \text{and} \quad \text{Pr} \equiv \frac{c_p \mu}{k}.$$

9.79. As a result of numerous experimental studies of heat transfer, various expressions relating the three moduli have been proposed; one of these, for an ordinary (nonmetal) fluid in a long, straight channel, is the Dittus-Boelter correlation,

$$\frac{hD}{k} = 0.023 \left(\frac{Du\rho}{\mu}\right)^{0.8} \left(\frac{c_p \mu}{k}\right)^{0.4} \tag{9.35}$$

†See standard texts on heat transfer, e.g., General References for this chapter.

or

$$\text{Nu} = 0.023\text{Re}^{0.8}\text{Pr}^{0.4}, \tag{9.36}$$

with all the physical properties evaluated at the bulk temperature of the fluid. In a modified form,

$$\text{Nu} = 0.023\text{Re}^{0.8}\text{Pr}^{0.33},$$

the physical properties, except the specific heat, are the values at the film temperature, i.e., in the laminar (film) layer of fluid adjacent to the surface. This is taken as the arithmetic mean of the wall (or surface) temperature and the bulk fluid temperature.

Example 9.6. Calculate the heat-transfer coefficient for the water coolant in Example 9.5.

From the data in the Appendix, k at 311°C is estimated to be 0.518 W/m · K and Pr about 1.06. Hence, from equation (9.36),

$$h = \frac{(0.023)(0.518)(500{,}000)^{0.8}(1.06)^{0.4}}{0.0118} = 37{,}500 \text{ W/m}^2 \cdot \text{K}.$$

9.80. It is seen from equation (9.35) that if the viscosity, thermal conductivity, density, and specific heat of the coolant are known, the heat-transfer coefficient can be estimated for turbulent flow of given velocity in a pipe or channel of specified diameter (or equivalent diameter). The results appear to be satisfactory for values of Re in excess of about 10,000, and for Pr values of from 0.7 to 120. This range of Prandtl numbers includes gases and essentially all liquids, except liquid metals; the latter have very low Prandtl numbers, primarily because of their high thermal conductivity but also often because of their low viscosity and heat capacity. The correlations for liquid metals will, therefore, be considered separately (§9.84 et seq.).

9.81. Where large differences exist between the temperatures of the solid and of the fluid, the associated variations in the physical properties of the coolant can influence the heat transfer. As indicated, such variation is most significant for the viscosity, and in order to make allowance for it the relationship

$$\text{Nu} = 0.027\text{Re}^{0.8}\text{Pr}^{0.33}\left(\frac{\mu}{\mu_w}\right)^{0.14}$$

has been proposed, where μ is the viscosity at the bulk fluid temperature, and μ_w is that at the wall temperature. This equation, used in conjunction with the equivalent diameter concept, has been found to be satisfactory for predicting local and average heat-transfer coefficients for ordinary fluids flowing through thin, rectangular channels [6].

Heat-Transfer Coefficients of Gases

9.82. Heat transfer to gases is treated in much the same way as for ordinary fluids such as water, and convection heat-transfer coefficients can be predicted by means of the correlation in equation (9.36). For many gases, including helium, carbon dioxide, and air, Pr is approximately 0.70, and so equation (9.36) reduces to

$$\mathrm{Nu} \approx 0.020\mathrm{Re}^{0.8}.$$

The marked differences in physical (especially thermal) properties result, however, in some important general differences in heat-transfer behavior between gases and liquids. For example, the thermal conductivity of helium gas at atmospheric pressure is only about a third that of water although it is enhanced by an increase in pressure. In considering the transport of heat away from the heated surface by the fluid, the specific heat of the latter is important. Here again, an increase in the gas pressure is advantageous; there is a density increase accompanied by a corresponding increase in the heat capacity per unit volume.

9.83. In so-called high-speed flow, where the gas velocity is greater than about 0.2 that of sound (§9.129), certain special effects occur that influence the temperature driving force for convection heat transfer. For example, "aerodynamic heating" results from the frictional effects in the boundary layer, and kinetic energy considerations (stagnation effects) lead to a difference between flowing and "at rest" conditions. For such matters, heat-transfer texts should be consulted [7].

HEAT TRANSFER TO LIQUID METALS

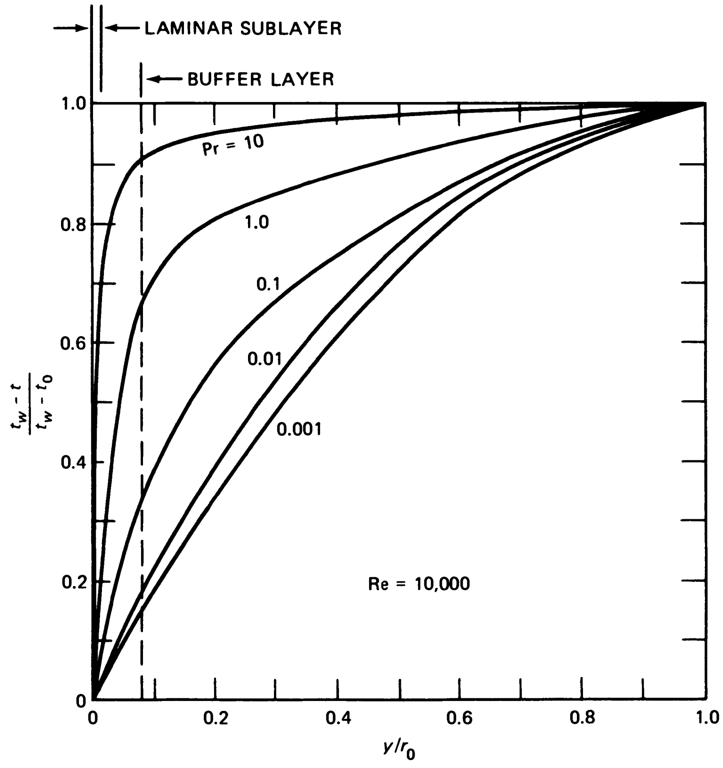
Introduction

9.84. For ordinary fluids, the principal mechanism of heat transport is by the effect of turbulence, as a result of which a "parcel" of fluid is rapidly moved from a region close to the hot wall into the main body of fluid. In liquid metals, however, thermal transport occurs mainly by molecular conduction. Whereas this mechanism may provide 70 percent of the heat transfer for a liquid metal, it contributes only about 0.2 percent to heat transfer in water. This means that the laminar boundary thickness, which is important for ordinary liquids, is not significant for liquid metals, and heat-transfer relationships applicable to gases and nonmetallic liquids cannot be used.

9.85. The essential difference between the heat-transfer properties of liquid metals and ordinary fluids is illustrated by the temperature profiles in a heated tube shown in Fig. 9.12 [8]. In these curves, the approach of the fluid temperature to the tube-wall temperature is represented by the dimensionless quantity $(t_w - t)/(t_w - t_0)$, where t_w and t_0 are the wall and centerline temperatures, respectively. The abscissa is the ratio y/r_0, where y is the distance from the wall at which the fluid temperature is t, and r_0 is the tube radius. The Prandtl numbers are the parameters for the various curves, and the Reynolds number is 10^4 in all cases.

9.86. For Pr = 1, the velocity and temperature profiles are identical; most of the resistance to heat transfer occurs in the laminar sublayer and

Fig. 9.12. Dependence of temperature profile on Prandtl number [8].



in the buffer layer, and there is little further change in temperature as the center of the tube ($y/r_0 = 1$) is approached. For liquid metals (Pr << 1), however, molecular conduction is so significant that there is a marked thermal gradient from the buffer layer boundary all the way to the center, much as would be observed for a solid rod (Pr = 0). The heat-transfer coefficient is normally based on a mixed-mean temperature obtained by integration of the thermal profile (§9.30). It can be seen that for fluids of low Prandtl number this temperature may be quite different from that at the centerline.

Heat Transfer in Reactor Rod Bundles

9.87. The contribution of high molecular conductivity results in heat transfer coefficients for a sodium-cooled core rod bundle that may be larger than those for a pressurized water reactor by a factor of 2 or 3. Uncertainty in prediction methods can be tolerated by the designer since such high values result in low temperature differences between the heat transfer surface and the bulk of the flowing fluid. A number of semiempirical correlations of experimental data are described in the literature [9].

9.88. Sodium-cooled fast reactors have a power density several times larger than that for a PWR. Also, sodium boiling in the core increases reactivity and must be avoided (§5.129). Hence, the designer needs to consider a number of heat-transfer problems that are only indirectly related to the heat-transfer coefficient based on average conditions. Attention must be given to local effects around a rod's periphery, local flow restrictions, and a variety of safety-related questions arising from the possibility of accidental loss of coolant flow.

BOILING HEAT TRANSFER

Pool Boiling

9.89. Boiling is of importance in nuclear reactor systems both as a means of achieving high heat-transfer rates from fuel to coolant and for generating steam in a heat exchanger. The mechanisms involved are complex and depend upon many factors, including surface conditions. For the preliminary treatment of boiling heat transfer from a solid fuel rod, it is convenient to consider a heated surface at temperature t_s, immersed in a pool of liquid. Suppose the temperature difference between the heated surface and the liquid saturation temperature, i.e., $t_s - t_{sat}$, is steadily increased;* the

*The saturation temperature is the temperature of the saturated vapor, i.e., saturated steam, at the existing pressure.

corresponding variation in the heat flux, i.e., q/A, across the surface is then as shown in Fig. 9.13, in which both scales are logarithmic. Although the data in this figure are representative of natural convection boiling from a heated surface in a pool of water at atmospheric pressure and a liquid temperature of 100°C, some of the same general characteristics apply to forced convection boiling and to other pressure and temperature conditions.

9.90. The curve for pool boiling can be divided into a number of regions, in each of which the mechanism of heat transfer is somewhat different from that in the others. Until the heated surface exceeds the saturation temperature by a small amount, heat is transferred by single-phase convection; this occurs in region I. The system is heated by slightly superheated liquid rising to the liquid pool surface where evaporation occurs. In region II, vapor bubbles form in crevices on the heated surface; this is the *nucleate boiling* range in which formation of bubbles occurs upon nuclei, such as solid particles or gas adsorbed on the surface, or gas dissolved in the liquid. Nucleate boiling is a common phenomenon, since it is encountered in standard power-plant steam generators.

9.91. The steep slope of the heat flux curve in region II is a result of the mixing of the liquid caused by the motion of the vapor bubbles. A maximum flux is attained when the bubbles become so dense that they coalesce and form a vapor film over the heated surface. The heat must then pass through the vapor film by a combined mechanism of conduction and radiation, neither of which is particularly effective in this temperature range. Consequently, beyond the maximum, the heat flux decreases appreciably despite an increase in temperature. The maximum flux, which is a design limitation, is referred to as the DNB (*departure from nucleate boiling*) value (§9.98). In region III, the film is unstable, it spreads over a part of the heated surface and then break down. Under these conditions, some areas of the surface exhibit violent nucleate boiling, while *film boiling*, due to heat transfer, occurs in other areas.

9.92. For sufficiently high values of $t_s - t_{sat}$, as in region IV, the film becomes stable, and the entire heated surface is covered by a thin layer of vapor; boiling is then exclusively of the film type. If attempts are made to attain large heat fluxes with film boiling, as high as those possible with nucleate boiling, for example, the temperature of the heated surface may become so high as to result in damage to the material being heated. This is called a *burnout* and is, of course, to be avoided. It may also be noted from Fig. 9.13 that if a system undergoing nucleate boiling is operating at conditions near the maximum of the curve, a slight increase in the heat flux will cause a sudden change to film boiling, which could result in burnout.

9.93. *Subcooled (or local) boiling* occurs when the bulk temperature of the liquid is below saturation but that of the heated surface is above sat-

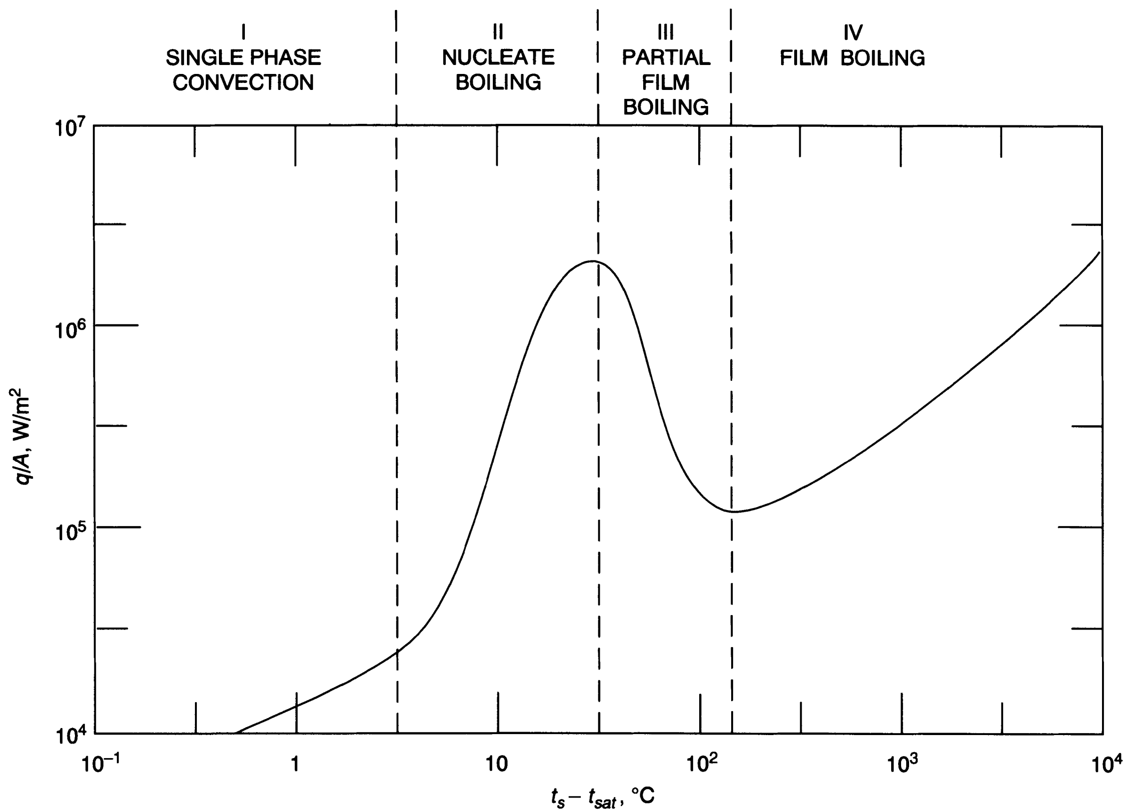


Fig. 9.13. Variation of heat flux with surface-liquid temperature difference in pool boiling.

uration. Vapor bubbles form at this surface but condense in the cold liquid, so that no net generation of vapor is realized. Very high heat fluxes can be obtained under these conditions; values as large as 4×10^7 W/m^2 have been reported in forced-convection heating of water, but 6×10^6 W/m^2 appears more realistic if the surface temperatures are to be kept low enough to avoid burnout. When the bulk temperature of the liquid reaches the saturation point, the vapor bubbles no longer collapse and then *bulk boiling occurs*.

Flow Boiling

9.94. In practical reactor systems the coolant is not stationary, and the boiling which takes place, called *flow boiling*, is hydrodynamically quite different from pool boiling. Flow boiling commonly occurs under forced-convection conditions, as in boiling-water reactors (BWRs) and to some extent in pressurized-water reactors (PWRs). It can also be experienced, however, when there is natural circulation in a loop configuration, such as may be present during the transient conditions that arise when a coolant circulation pump fails.

9.95. Suppose that water, below the saturation temperature, is forced through a channel between or around the solid fuel elements of a reactor; heat is then transferred from the solid surface (or wall) to the water. As long as the fuel-wall temperature, which increases along its length (§9.143 eq seq.), remains below the steam saturation temperature, single-phase heat transfer only will occur. In PWRs, the pressure on the cooling system is increased in order to raise the saturation temperature and thus prevent bulk boiling, but some local boiling is tolerated; PWRs and BWRs may therefore be regarded as having certain features in common.

9.96. The flow patterns in boiling two-phase flow are complex. Primarily for conceptual purposes, we show the classical Fig. 9.14 representation for flow boiling in a vertical hollow tube [10]. At first, the rising flowing water is merely heated by convective transfer with no boiling occurring. As the temperature of the flowing water increases, a point is reached where the temperature of the water in the slowly moving laminar layer next to the wall is above the saturation temperature although the temperature at the same level in the more rapidly moving core is still below saturation. Initial vaporization therefore occurs along the wall under *subcooled* (local boiling) conditions. Bubbles grow and are carried along in the superheated layer close to the wall, but they condense on being mixed with the subcooled liquid core.

9.97. After the central core reaches saturation conditions, the vapor bubbles being fed from the superheated layer near the surface no longer collapse but are carried along in the stream. As boiling progresses, the

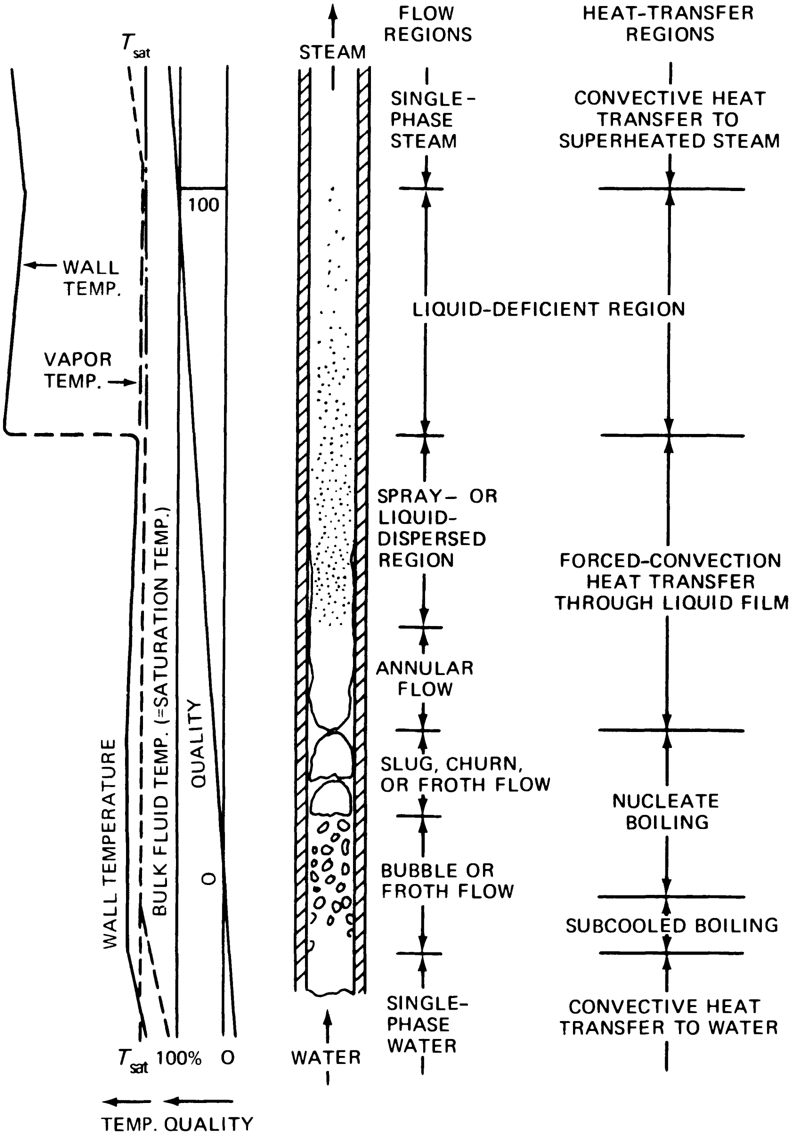


Fig. 9.14. Regimes of two-phase flow.

flow mechanism becomes quite complex and depends on how the vapor and liquid phases flowing in the same direction distribute themselves. As the vapor volume increases, the flow must accelerate. *Bubble flow* is characterized by a regular distribution of vapor bubbles in the continuous liquid phase. With increased vaporization, and a resulting higher void fraction or quality (§9.99), a transition to *annular flow* occurs. This is characterized by a continuous vapor phase in the central core in which some liquid drops may be dispersed.

Boiling Crisis

9.98. With increased vaporization in a coolant channel, the heated surface becomes intermittently exposed to patches of vapor. Since the heat-transfer coefficient decreases markedly when the surface is blanketed with vapor, the wall temperature rises correspondingly. Hence, the wall temperature may at first oscillate as the surface is alternately blanketed with vapor or liquid, but will then rise after the wall liquid is completely vaporized. Such behavior, characterized by a marked temperature rise of the heated surface during boiling, as a result of a change in the heat-transfer mechanism, is called a *boiling crisis*. The foregoing is actually an oversimplified representation of a complex phenomenon. For example, a wall temperature excursion can occur at a sufficiently high heat flux as a result of bubble-layer blanketing near the surface when the core of the channel is liquid and below the saturation temperature. The term *departure from nucleate boiling* (or DNB) (§9.91) is often applied to such a case as well as to surface overheating when the bulk of the fluid is at the saturation temperature. *Dryout* occurs in a channel with a vapor core and annular liquid film when the liquid is evaporated and the heat-transfer coefficient is reduced to that for a vapor in forced convection. *Critical heat flux* is often used as a generic term to cover all of these boiling-crisis possibilities.

9.99. A key critical heat flux parameter is the *quality* of the vapor-liquid system; it is defined as the mass fraction of vapor present in the mixture. The quality χ may be expressed in thermodynamic terms by

$$\chi = \frac{H - H_f}{H_{fg}},$$

where H is the enthalpy of the mixture at the point in the coolant channel of interest, H_f is the enthalpy of the saturated liquid at the applied pressure, and H_{fg} is the enthalpy of vaporization at this pressure. Application of this equation to the subcooled region yields a negative quality, which is used in some correlations for predicting the critical heat flux under such conditions. The local quality for use in correlations must be calculated by an enthalpy balance.

9.100. In PWRs, the water is primarily in the subcooled or low-quality region. Here, a boiling crisis is represented by the DNB condition. It occurs only at a relatively high heat flux. In addition to the heat flux, the onset of DNB depends on such parameters as flow rate, quality of the fluid, system pressure, and increase in enthalpy of the coolant as it passes through the core.

9.101. In BWRs, where the quality is higher and the coolant is saturated, the boiling crisis of concern is dryout. In this case, where a dry patch forms by disruption of the annular liquid film, the wall-temperature excursion tends to be modest and slow when compared with subcooled-system excursions. Since the vapor velocity is high, the heat-transfer coefficient from the surface to steam tends to be large enough to prevent the cladding from being damaged. However, failure could result from thermal cycling as the cladding surface is intermittently wetted and dried.

Prediction of Burnout Conditions

9.102. The critical heat flux, at which burnout is expected to occur, is an important design consideration in water-cooled (and moderated) reactors. A knowledge of burnout conditions is also necessary for possible accidental situations, such as might arise from loss of coolant flow or excessive fuel temperatures due to a power excursion. Consequently, methods for estimating the critical heat flux in a reactor are required. Much experimental work has been done to develop the many correlations reported in the literature. In addition, much effort has been devoted to the development of theoretical models which would support empirical correlations.

9.103. For PWRs, the so-called W-3 correlation for the DNB condition has been widely used.* This is a many-termed complex expression for the critical heat flux with pressure, mass velocity, steam quality, enthalpy, and system dimensions as parameters [11]. In practice, the equation is readily solved as part of a digital computer calculation, and some general conclusions are of interest. The DNB flux decreases as the quality increases and hence reaches a minimum at the coolant channel exit. An increase in coolant-mass velocity tends to increase the DNB flux. The pressure effect is complex but an increase in critical heat flux can occur under PWR conditions as the system pressure is increased. As part of the core design effort, the DNB condition is calculated at various axial locations to determine the closest approach to the design flux (see Fig. 9.22). Since the DNB flux and heat flux both decrease toward the exit, the closest approach is normally in the upper half of the channel.

9.104. The conditions in a BWR differ from those in a pressurized-water system largely because of the higher quality of the fluid in the former

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*The designation "W-3" arises from the fact that it is the third such correlation developed by the Westinghouse organization.

case. The approach generally used for determining the critical heat flux is based on numerous experiments with electrically heated rod bundles with fluid flow resembling that in an actual reactor. The results are incorporated into a thermal-hydraulic computational model, together with correlations that take into account the steam quality and coolant flow rate [12].

Boiling Heat-Transfer Coefficients

9.105. As for normal nonboiling heat transfer, it is possible to describe the heat-transfer rate for a flow-boiling system in terms of a heat-transfer coefficient and a temperature-difference driving force. However, from the reactor core design viewpoint, the boiling heat-transfer coefficient is not of great significance. With boiling present, heat is transferred from the heated (fuel rod) surface to the liquid coolant by several evaporative mechanisms resulting in vapor-bubble growth but not requiring much of a temperature-difference driving force between the surface and the bulk of the fluid. Since the temperature driving force is small compared with that in nonboiling systems, the designer is primarily concerned with fixing specifications to allow a sufficient margin based on the critical heat flux limitation. Therefore, the following paragraphs devoted to a discussion of the boiling heat-transfer coefficient are intended for background only.

9.106. It is apparent from Fig. 9.13 that $\log(q/A)$ is a linear function of $\log(t_s - t_{sat})$ over a considerable portion of the nucleate boiling range, so that the general expression

$$\frac{q}{A} = C(t_s - t_{sat})^n, \tag{9.37}$$

where C and n are constants, is applicable to a particular set of conditions. If the difference between the temperature of the heated surface and the saturation temperature of the coolant is represented by Δt_b, equation (9.37) may be written as

$$\frac{q}{A} = C(\Delta t_b)^n.$$

If h_b represents the boiling heat-transfer coefficient, it may be defined by

$$\frac{q}{A} = h_b \Delta t_b,$$

so that

$$h_b = C(\Delta t_b)^{n-1}.$$

9.107. For subcooled or local boiling, the relationship

$$\frac{q}{A} = \left(\frac{e^{p/6.2}}{0.79}\,\Delta t_b\right)^4 \qquad (9.38)$$

has been found to be applicable, where q/A is in W/m^2, p is the pressure in MPa, and Δt_b is in kelvin [13]. In the region of a PWR channel prior to the inception of local boiling, the heat-transfer coefficient is predicted using equation (9.35). Since equation (9.38) applies to the local-boiling region, designers often define the initiation of local boiling as the point where the surface temperatures predicted by the two equations become equal.

9.108. Although the heat-transfer coefficient is of little importance in the design of a BWR, it is often desirable to evaluate the surface temperature of the fuel elements corresponding to a desired average heat flux. This temperature may be important from the standpoint of corrosion resistance, and it may also serve as a reference for estimating the maximum internal temperature within the fuel element. According to equation (9.38), the temperature difference Δt_b at the surface is proportional to the 0.25 power of the heat flux. The surface temperature itself would thus appear to be relatively insensitive to changes in the flux.

Example 9.7. Determine the surface temperature of the fuel in a reactor core under subcooled-boiling conditions at a system pressure of 7.2 MPa when the average heat flux is (*a*) 0.5 MW/m^2 and (*b*) 5 MW/m^2. The saturation temperature of water at this pressure is 288°C.

(*a*) From equation (9.38),

$$\Delta t_b = \frac{0.79}{e^{p/6.2}}\left(\frac{q}{A}\right)^{0.25}$$

$$= \frac{0.79}{e^{7.2/6.2}}(5 \times 10^5)^{0.25} = 6.6 \text{ K } (6.6°\text{C})$$

The fuel surface temperature is consequently $288 + 6.6 \approx 295°\text{C}$.

(*b*) For an average heat flux of 5.0 MW/m^2,

$$\Delta t_b = (6.6)(10)^{0.25} = 12 \text{ K } (12°\text{C}),$$

so the fuel surface temperature is $288 + 12 = 300°\text{C}$. (As stated, the surface temperature is not very sensitive to changes in the heat flux; the temperature increases from 295°C to only 300°C for a tenfold increase in heat flux. However, the higher flux may be above the critical heat flux limit.)

CORE FLUID FLOW

Introduction

9.109. Since energy must be removed from the core by the flow of the coolant, some well-known flow fundamentals are reviewed here. We restrict our attention to semiempirical methods that have been used for many years to solve practical problems. In a fluid flowing in a pipe or "channel," shear forces between fluid particles and the wall and between the particles themselves result in friction that must be overcome by pumping. Hence, we will first examine the flow pressure drop.

Flow Pressure Drop

9.110. At this point it is useful to review the concept of the macroscopic, or overall energy balance based on the first law of thermodynamics. For a control volume, we write

(Rate at which quantity enters volume element)

= (Rate at which quantity leaves volume element)

+ (Rate of quantity accumulation in volume element).

For example, a macroscopic energy balance on a volume element can be classically expressed as

$$\Delta H + \Delta Z + \Delta u^2 = Q - W_s,$$

where H, Z, and u represent enthalpy, potential energy, and fluid velocity, respectively. Hence, u^2 is the kinetic energy, Q is the heat *added* to the volume element, and W_s is the mechanical work *done* by the fluid, such as by means of a turbine. It is sometimes called shaft work.

9.111. The pressure drop, or loss, in a flow system is evaluated as part of a mechanical energy balance over the system. This is often known as the *extended Bernoulli equation* or the *mechanical energy equation*. It can be derived from the equation in §9.110 as well as from a force balance on a volume element. In its simplified integral form for a *unit mass* in an incompressible fluid having a density ρ moving between two points, 1 and 2, at different heights Z with respect to a datum level, and with no shaft work or friction, the equation becomes

$$\frac{p_2 - p_1}{\rho} + \frac{u_2^2 - u_1^2}{2} + (z_2 - z_1)g = 0,$$

where p is the static pressure, u the velocity in the kinetic energy term, and g the acceleration of gravity. A term W_s may be added, expressing the work required per unit mass if a pump is included between the two points. Also, for a practical system, the term $\Delta p_f/\rho$ should be added describing the friction loss in energy units per unit mass of fluid flowing. The following equation results, which can also serve as the basis for determining pumping power requirements in the system:

$$\frac{p_2 - p_1}{\rho} + \frac{u_2^2 - u_1^2}{2} + (Z_2 - Z_1)g + \frac{\Delta p_f}{\rho} + W_s = 0.$$

Therefore, the total pressure drop is the sum of terms arising from the difference in level, acceleration of the fluid, friction, and pump work. Since the kinetic energy term is usually negligible for a liquid, our review will concentrate on frictional effects.

9.112. The pressure drop Δp_f accompanying isothermal *laminar* flow at a mean velocity u is a cyclindrical pipe of diameter D and length L is the familiar Poiseuille equation

$$\Delta p_f = \frac{32\mu u L}{D^2},$$

which may be written as

$$\Delta p_f = 64\frac{\mu}{Du\rho} \cdot \frac{L}{D} \cdot \frac{\rho u^2}{2}, \tag{9.39}$$

where μ is the viscosity.* The subscript f is used in p_f to indicate that the pressure loss described is due to fluid friction. The factor $\mu/Du\rho$ on the right of equation (9.39) is the reciprocal of the Reynolds number, and so

$$\Delta p_f = \frac{64}{\text{Re}} \cdot \frac{L}{D} \cdot \frac{\rho u^2}{2}. \tag{9.40}$$

This gives the pressure drop due to fluid friction for laminar flow in a straight pipe of circular cross section. It should be noted that replacement of D by the equivalent diameter D_e, as defined in §9.76, is not valid for laminar flow in channels which do not have circular cross sections, although it is a good approximation for turbulent flow (§9.115).

*If English units are used in equation (9.39) or in subsequent equations, g_c, the factor relating force and mass, must be used as appropriate.

Pressure Drop in Turbulent Flow

9.113. For turbulent flow at a mean velocity u, the relationship corresponding to equation (9.40) is

$$\Delta p_f = 4f\frac{L}{D}\cdot\frac{\rho u^2}{2}, \tag{9.41}$$

known as the Fanning equation, in which the dimensionless quantity f is called the *friction factor.** Comparison of equations (9.40) and (9.41) shows that for laminar flow $f = 16/\mathrm{Re}$, but the relationship between the friction factor and the Reynolds number in the case of turbulent flow is more complicated. Several empirical expressions have been developed; one of the simplest of these, which holds with a fair degree of accuracy for flow in smooth pipes at Reynolds numbers up to about 2×10^5, is the Blasius equation

$$f = 0.079\mathrm{Re}^{-0.25}. \tag{9.42}$$

This permits the evaluation of the turbulent-flow friction factor for a given Reynolds number.

9.114. For turbulent flow in a commercial rough pipe, the friction factor is larger than for a smooth pipe at the same Reynolds number, as shown in Fig. 9.15. The deviation from the ideal behavior of a smooth pipe increases with the roughness of the pipe, especially at high Re values. The variation of the friction factor with Reynolds number has been determined experimentally for different degrees of roughness, expressed in terms of a dimensionless quantity ε/D, where ε is a measure of the size of the roughness projections and D is the pipe diameter.

9.115. For turbulent flow in noncircular channels of relatively simple form, the pressure drop due to friction may be calculated by substituting the equivalent diameter for D in equation (9.41). The same value is used in determining the Reynolds number. For in-line flow along channels in rod bundles, the friction factor depends to some extent on the pitch-to-diameter ratio. For initial design purposes only, however, a correction factor of 1.3 is a reasonable approximation for the usual range of these ratios [14].

Example 9.8. Estimate the pressure drop required to overcome friction for the turbulent flow of water in the channel between the fuel rods in Example 9.5, along a length of 4.17 m.

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*In the equivalent Darcy-Weisbach equation a friction factor equal to $4f$ is used.

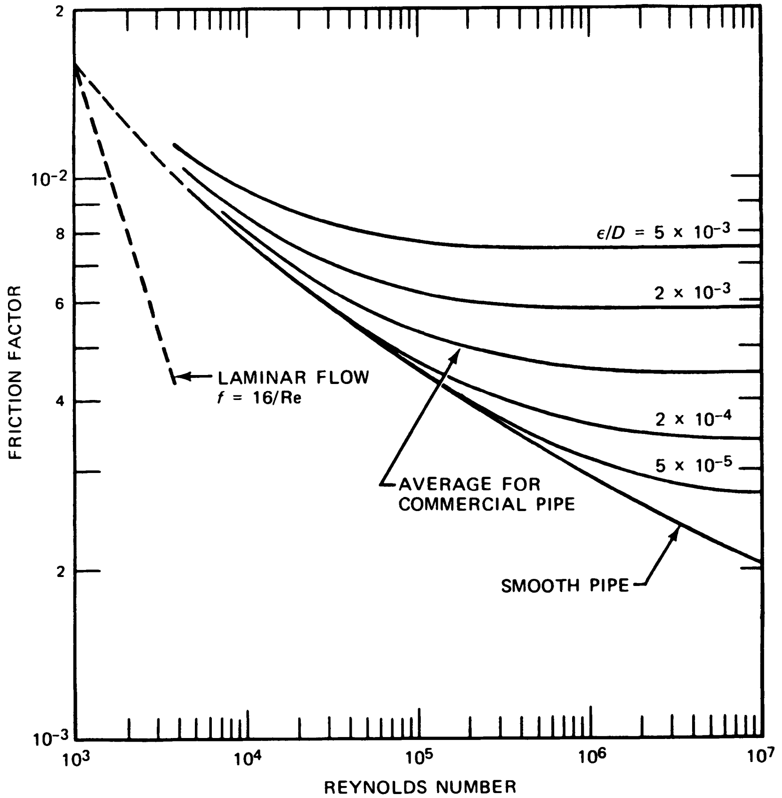


Fig. 9.15. Friction factor as a function of pipe roughness.

The value of Re was found to be 5.00×10^5; hence, assuming the fuel rods to be moderately smooth, the value of f is seen from Fig. 9.15 to be about 0.0032.

The mass velocity $G(=u\rho)$ in Example 9.5 was 3730 kg/m$^2 \cdot$ s, and since ρ for water at 311°C is 0.691×10^3 kg/m^3, u is 3730/691 = 5.40 m/s. Hence, with D_e equal to 0.0118 m, it follows from equation (6.49) that

$$\Delta p_f = (4)(0.0032)\frac{4.17}{0.0118} \cdot \frac{(691)(5.40)^2}{2} = 4.56 \times 10^4 \text{ Pa.}$$

Upon applying the 1.3 correction for flow in rod bundles (§9.115), the result is

$$\Delta p_f = 4.56 \times 10^4 \times 1.3 \approx 6 \times 10^4 \text{ Pa.}$$

This calculation does not taken into account the pressure losses in the flow channel resulting from the spacer grids (normally about six) provided at intervals to support the rods and enhance mixing (§9.119).

Velocity Head Losses

9.116. In a flowing fluid there will be changes in pressure due to changes in velocity resulting from gradual or abrupt changes in flow area. Such pressure changes are usually considered in terms of the *velocity head*, defined as $u^2/2$; the pressure corresponding to a head H is essentially equal to $H\rho$.* Since u^2 is generally high for turbulent flow, the pressure losses accompanying changes in cross-sectional area of fluid conduits may be very significant. In general, for abrupt expansion (Fig. 9.16A) or contraction (Fig. 9.16B), the pressure change Δp due to the loss of head ΔH, which occurs in either case, can be represented by

$$\Delta p = \rho \Delta H(\text{expansion}) = K_e \frac{\rho u_1^2}{2}$$

$$\Delta p = \rho \Delta H(\text{contraction}) = K_c \frac{\rho u_2^2}{2},$$

where u_1 and u_2 are the upstream and downstream (smallest pipe) velocities, respectively; the value of K, the loss coefficient, depends on the conditions.

9.117. For abrupt *expansion*,

$$K_e = \left(1 - \frac{D_1^2}{D_2^2}\right)^2,$$

where D_1 and D_2 are the pipe diameter upstream and downstream, respectively. For expansion into a large reservoir, D_2 is large, and K_e becomes virtually unity. The loss of head is then almost equal to $u_1^2/2$.

9.118. For abrupt *contraction*, the value of K_c varies with the ratio D_2/D_1 in the following manner:

D_2/D_1	0.8	0.6	0.4	0.2	0
K_c	0.13	0.28	0.38	0.45	0.50

*Since the velocity head is equal to the kinetic energy of a unit mass of flowing fluid, the term arises from the similarity with the potential energy of a height (or "head") of a unit fluid mass.

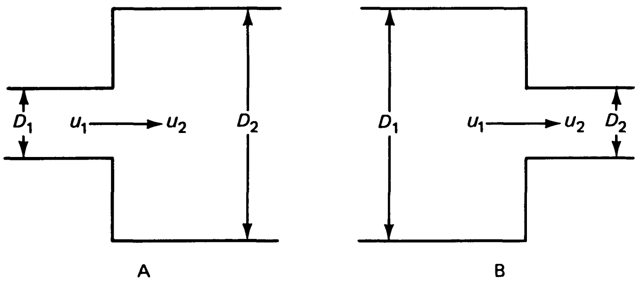


Fig. 9.16. A. Abrupt expansion in fluid flow. B. Abrupt contraction in fluid flow.

For the case of inlet from a very large reservoir, D_2/D_1 approaches zero, and K_c is then approximately 0.5; the corresponding loss of head is $u_2^2/4$. It should be understood that these results apply only to cases of abrupt expansion or contraction. The loss of head decreases if the fluid exit is rounded, making the change less abrupt. Where the exit is tapered so that the included angle is 7° or less, the losses will usually be negligible.

Example 9.9. Estimate the pressure loss due to contraction and expansion of coolant as it enters and then leaves a channel between the fuel rods considered in Example 9.8.

The coolant undergoes sudden contraction and expansion as it enters and leaves the channel, respectively, from or to a header or manifold of large cross section. The loss of head upon entering the channel, i.e., upon contraction, may then be taken as $0.5u_2^2/2$, where u_2 is the velocity of the coolant in the channel. Similarly, upon leaving the channel, the loss is $u_1^2/2$, where u_1 is also the velocity of the coolant in the channel. The total loss of head upon entry and exit is thus $1.5u^2/2$, where u is the velocity of the coolant in the channel, i.e., 5.40 m/s (Example 9.8). The pressure loss is equal to the loss of head multiplied by the density of the coolant; hence,

$$\text{Pressure loss} = \frac{(1.5)(5.40)^2(691)}{2} = 1.5 \times 10^4 \text{ Pa}.$$

9.119. Additional contraction and expansion losses are introduced in the flow channels between long fuel rods by spacer grid assemblies which support the fuel rods at intervals along their length. Loss coefficients for such a special geometry can best be determined by experiment. However, for an order of magnitude estimate of the effect, a loss of one velocity head for each spacer grid may be assumed to account for both contraction and expansion. The pressure loss is thus roughly $\rho u^2/2$ per grid. In Examples 9.8 and 9.9, the loss for six grids would be about 6×10^4 Pa.

9.120. Losses in pipe fittings, due to changes in direction, e.g., in elbows, curves, etc., or to contraction in valves, can also be expressed in terms of the velocity head, e.g., $K_1 u^2/2$. The factor K_1 may range from 0.25 or less for a gradual 90° curve or a fully opened gate valve, to 1.0 for a standard screwed 90° elbow, or as high as 10 for a fully opened globe valve.*

Two-Phase Flow

9.121. Our discussion of flow boiling (§9.94 et seq.) introduced the complexity of two-phase flow. As we shall see, the ability to predict flow rates is of major importance in evaluating the safety of water-cooled reactors. Therefore, developing a good understanding of two-phase flow phenomena has been the objective of much recent work [15]. However, space limits our presentation to a few introductory ideas.

9.122. In single component two-phase flow, we have a liquid flowing concurrently with its own vapor with evaporation or condensation from one phase to the other occurring in accordance with whatever heat transfer is taking place involving the system. The vapor phase, which is a gas, follows the laws of compressible flow as described in a significant branch of fluid mechanics. To develop a model for the system, a classical approach, as described in various texts [15], is to write equations for the conservation of mass, momentum, and energy over an element of the flow channel and to evaluate the pressure gradient. The necessary, rather complicated, set of relationships can best be managed by appropriate computer codes. For a reactor core, for which the picture is further complicated by interconnected flow regions, the calculational development is known as *subchannel analysis* (§9.135).

Two-Phase Pressure Drop

9.123. For preliminary design purposes, the pressure drop problem can be made more manageable by applying one of two simplifications. In *homogeneous* models, the two phases are assumed to flow as a single phase mixture possessing mean flow properties and a suitable single-phase friction factor is developed to represent the two-phase flow. *Separated-flow* models consider the phases to be segregated into two separate streams, with each moving at a different velocity. Empirical correlations are then used to

*For further information, standard reference books and manufacturers' catalogs should be consulted. The loss coefficients K_1 are often expressed in terms of an "equivalent" length of straight pipe.

determine a so-called friction multiplier (described in the next paragraph) which is a function of flow parameters. The different models vary in sophistication and may be specific to given flow regimes. Furthermore, models must be used with care since the behavior being modeled is inherently complex [15] with numerous simplifications necessary for a practical description.

Boiling-water reactors

9.124. The problem of the two-phase pressure drop in a vertical channel is of particular importance in BWR design. One treatment involves the use of a *friction-factor multiplier R*, defined as the ratio of the two-phase friction-pressure gradient $(dp/dL)_2$ to the single-phase value for the liquid phase alone $(dp/dL)_l$, i.e.,

$$R \equiv \frac{(dp/dL)_2}{(dp/dL)_l} \approx \frac{(\Delta p)_2}{(\Delta p)_l},$$

where $(\Delta p)_2$ is the two-phase pressure drop in a given channel in which $(\Delta p)_l$ would be the pressure drop for the liquid phase alone. Values of R, obtained by semiempirical correlation procedures and graphical integration of local values with respect to length, are given in Fig. 9.17 as a function of the system pressure for steam ranging in exit quality from 1 to 100 percent [16]. This figure is applicable when the axial heat flux distribution is uniform; other curves have been determined for the situation in which the axial heat flux has a sinusoidal distribution.

Pressurized-water reactors

9.125. In the design of the first PWRs, local (nucleate) boiling was considered to be undesirable, and conditions were chosen to prevent its occurrence. In later designs, however, local boiling during steady-state operation became a design requirement. Local boiling will affect the pressure drop and, in an open lattice, will result in flow redistribution with the possibility of burnout. An underestimating of the factors affecting the pressure drop is therefore essential.

9.126. Where the parallel flow channels have the same inlet and outlet headers, the pressure drop must be equal in all channels. If there is a tendency in any channel for the resistance per unit flow rate to be higher, because of local boiling or other factors, the flow rate in that channel will automatically decrease until the pressure drop is the same as that in the other channels. Such flow distribution may result in instabilities because the decreased flow will mean a lower thermal transport and, consequently, higher surface and fluid temperatures. This could lead to ultimate failure

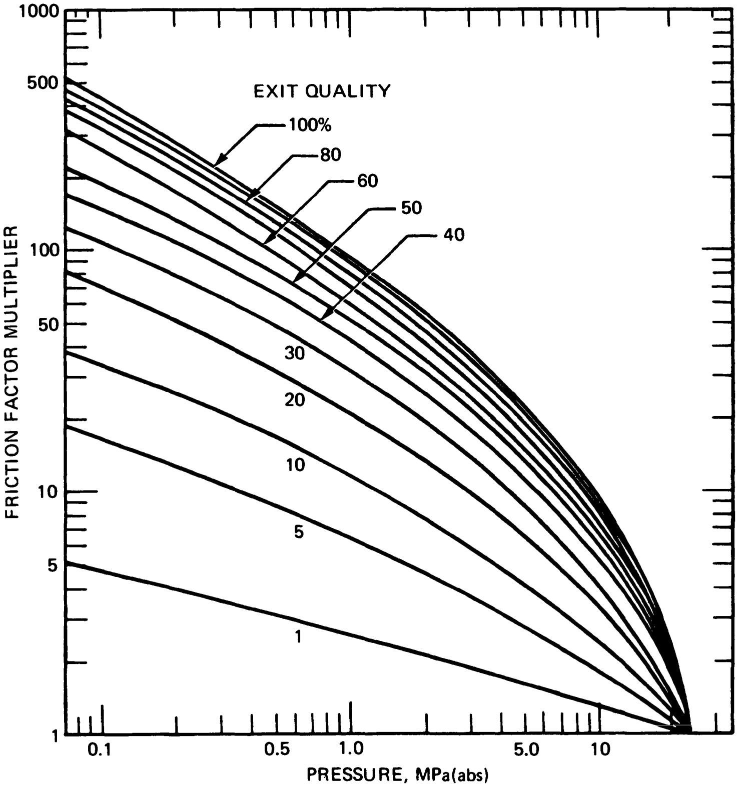


Fig. 9.17. Friction-factor multiplier for two-phase flow [16].

if the higher temperatures resulted in bulk boiling, accompanied by increased flow resistance, and so on (cf. §9.98).

9.127. It has been found [17] that the pressure drop for a nonboiling, open, parallel-tube system may be predicted by means of the standard Fanning friction-factor correlation based on an equivalent diameter. Under nonboiling conditions, there are only minor variations caused by differences in bulk-water temperatures. When local boiling occurs, however, there is an increase in momentum exchange and in heat transfer which may be due to the increase in hydrodynamic turbulence caused by repeated growth and collapse of steam bubbles. The bubbles may also be considered, in a broad sense, to contribute an increased surface roughness. Whatever the mechanism, there is an increase in pressure drop, and this may be represented by a friction factor applicable to the local-boiling conditions.

9.128. The friction factor f for local boiling at 13.8 MPa (2000 psi) may be expressed empirically in terms of the isothermal (nonboiling) factor f_{iso} as a function of the bulk temperature t_m alone; thus,

$$f = f_{iso}\left(1 + \frac{t_m - 293}{70}\right),$$

where the bulk temperature is between 293 and 335°C (saturation temperature of water) and above the local-boiling temperature. The pressure drop is equal to that for nonboiling until the water temperature reaches 335°C. Alternative procedures for local-boiling, pressure-drop calculations have been based on two-phase flow predictions.

Limiting Flow With Compressible Fluids

9.129. An important characteristic of gaseous coolants, particularly from the safety viewpoint, is that the maximum flow rate is limited by the velocity of sound in the gas. This flow limitation applies to all compressible fluids, e.g., steam that may have been vaporized during the course of a coolant blowdown (§12.79) of a water-cooled reactor. The velocity of sound c in an ideal gas, equal to the limiting flow rate, can be expressed as

$$c = \sqrt{\gamma \frac{RT}{M}},$$

where $\gamma = c_p/c_v$, the ratio of the specific heats, R is the ideal molar gas constant, T is the absolute temperature, and M is the molecular weight.*

9.130. In the flow of gas through a channel, the temperature changes as the pressure is reduced since adiabatic conditions can normally be assumed. Therefore, other, more complex relations must be used when considering limiting flow in terms of the upstream conditions or when friction must be allowed for. However, as a rough guide, the critical (or limiting) flow conditions can generally be expressed in terms of the pressure; thus,

$$\frac{P_c}{P_0} = \left(\frac{2}{\gamma + 1}\right)^{\gamma/(\gamma - 1)},$$

*In SI units, R = 8.314 J/K · mol, and the molecular weight (mass of one mole) is in kilograms (§1.12). Thus, the velocity of sound in helium at 1000 K is

$$c = \sqrt{\gamma \frac{RT}{M}} = \sqrt{\frac{(1.66)(8.314)(1000)}{4.0 \times 10^{-3}}} = 1.86 \times 10^3 \text{ m/s},$$

where γ for helium, a monatomic gas, is 1.66 (§9.130), and the molecular weight is 4.0×10^{-3} kg.

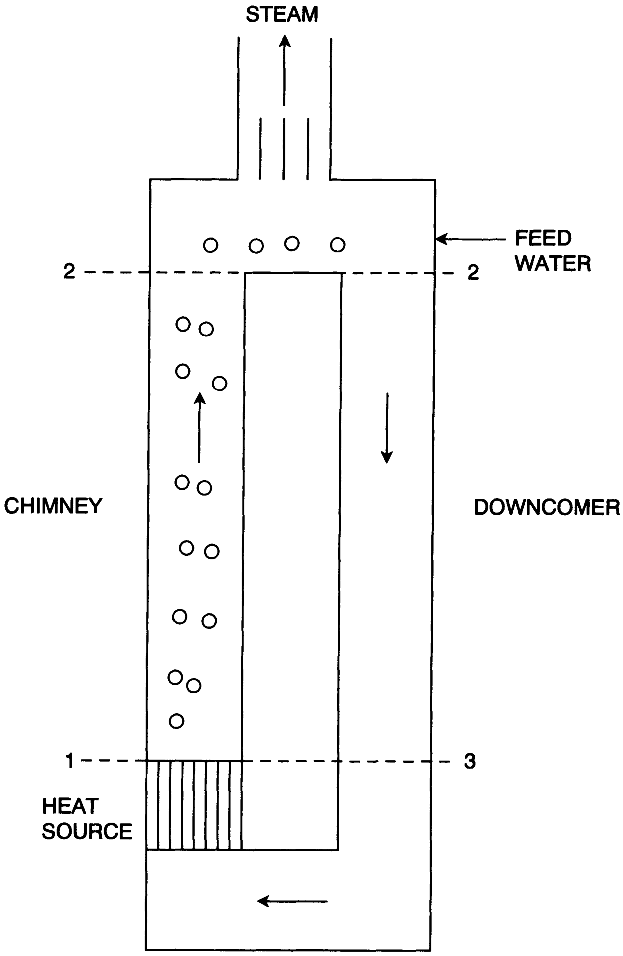
where P_c is the pressure at the critical flow condition and P_0 is the upstream pressure. For monatomic gases at all temperatures ($\gamma = 1.66$) and for diatomic gases up to about 500°C ($\gamma = 1.40$), P_c/P_0 is roughly 0.5. This means that when the downstream pressure of the gas is less than about half the upstream pressure, a limiting flow condition is likely to occur. In the case of compressible flow through a pipe, frictional effects reduce the limiting velocity, as to be expected.

9.131. Critical or "choked" flow also occurs in two-phase systems, but the analysis picture is complicated by the need for writing sets of defining conservation equations for each phase and expressing interfacial transport of mass, momentum, and energy. For example, in steam-water flow through a pipe, flashing of the water occurs toward the exit as the fluid encounters a decrease in pressure. Therefore, experimental measurements have an important role in developing predictive models useful for design. As in single-phase compressible flow, frictional effects reduce the maximum velocity.

Natural Circulation Cooling [18]

9.132. There is often confusion regarding what is meant by the terms *natural circulation* and *free convection*. If we have a fluid being heated in a vertical tube, its density will decrease. Now, if the system is closed, that is, consists of a "loop" and the fluid is cooled before returning to our heated tube, we would have flow induced by the so-called *chimney effect*, which causes a pressure differential, or "*driving pressure*," in the heated tube which would be resisted by frictional effects. Flow can be either laminar or turbulent, depending on the flow velocity achieved. This effect is known as *natural circulation* and can be analyzed by application of the mechanical energy equation (§9.111) to determine the flow rate. We would have the same effect, but less pronounced, if our heated tube was in a reservoir of fluid, as we have in a fireplace chimney.

Example 9.10. As an introduction to the role of the chimney in a natural circulation boiling-water reactor (§15.25), estimate the driving pressure resulting from a 10-m-high chimney above a boiling core that has an exit quality of 15 percent. At the dome pressure of 7.2 MPa, the specific volumes are 0.00136 m^3/kg for the saturated liquid and 0.0265 m^3/kg for the saturated vapor. Figure 9.18 shows the circulation path schematically. For purposes of this estimate, assume that the system is isothermal and that friction outside the core is negligible. Feedwater replaces the steam produced.



From the mechanical energy balance,

$$(p_2 - p_1) + (Z_2 - Z_1)(\rho_{2-1})g = 0$$

$$(p_3 - p_2) + (Z_3 - Z_2)(\rho_{2-3})g = 0$$

$$(p_3 - p_1) = (Z_1 - Z_2)(\rho_{2-1})g + (Z_2 - Z_3)(\rho_{2-3})g$$

$$Z_1 = Z_3 = 0; \qquad Z_2 = +10.$$

Specific volume of two-phase mixture

$= (0.85)(0.00136) + (0.15)(0.0265)$

$= 0.00513\ \text{m}^3/\text{kg}; \qquad \rho_{2-1} = 195\ \text{kg/m}^3;\ \rho_{2-3} = 735\ \text{kg/m}^3;$

then

$$p_3 - p_1 = 10(735 - 195)(9.8) = 53\ \text{kPa}.$$

This compares with a core pressure loss in the SBWR of 44 kPa.

9.133. The terms *free convection* or *natural convection* normally apply to fluid motion caused by buoyancy effects when we have a heated surface immersed in a surrounding fluid. Under such conditions we have a so-called boundary layer problem rather than a chimney effect.

9.134. In designing reactor core cooling systems, it is very desirable to make the geometry and pressure losses such that there will be significant natural circulation cooling after shutdown in the event that the circulating pumps fail. Natural circulation cooling considerations are also important in the design of water-filled storage pools for spent-fuel assemblies (§11.34).

SUBCHANNEL ANALYSIS AND SYSTEM CODES

9.135. In the preceding sections, the thermal-hydraulic behavior of a coolant channel associated with a single fuel rod was considered. For reactor core analysis, however, a more detailed treatment is required. For this purpose, *subchannel* analysis is commonly used. A subchannel may consist of the coolant flow "cell" associated with a single rod as considered in Example 9.5, or a cluster of such cells. A number of parallel interacting subchannels running the length of the fuel rods may be included in the treatment. In addition, each subchannel length is divided into a number of axial intervals or "control volumes." The result of this subdivision is a two-dimensional network of thermally and hydraulically connected nodes (§9.59) in the axial and radial directions. Heat transfer occurs primarily by conduction and convection in the radial direction from fuel to coolant. Allowance is also made for cross flow and turbulent mixing between coolant subchannels.

9.136. Numerous subchannel design codes have been written for solving the thermal-hydraulic equations based on the conservation of mass, energy, and momentum for each nodal volume [19]. Solutions yield such quantities as the radial and axial variations in the fluid enthalpy and mass velocity.

The approach to burnout conditions can then be determined by means of an appropriate critical heat flux correlation.

9.137. For safety analysis purposes, it is important to couple the core with the remainder of the coolant system and to consider transient conditions. System codes, which extend the subchannel approach, have been developed to meet this need. Typical such codes are RETRAN [20], which was developed under Electric Power Research Institute sponsorship and RELAP [21]. The development effort has been supported by an experimental program designed to simulate accident conditions, and the use of operating plant data for verification. Further consideration of system codes for safety analysis will be given in Chapter 12.

CORE DESIGN CONSTRAINTS

General Considerations

9.138. In order to provide a basis for reactor design, certain operating conditions, averaged over both the volume and lifetime of the core, must first be specified for the reactor to operate at the desired power level. In principle, the appropriate core specifications, such as number, dimensions, and arrangement of the fuel rods of a given type, coolant-flow rate, temperature distribution, etc., could then be computed to meet the requirements. In practice, however, the situation is complicated by the fact that the ideal specifications cannot be met precisely. For example, the dimensions, density, and enrichment of the uranium dioxide fuel pellets may vary slightly, and so also may the dimensions and spacing of the rods and the thickness of the cladding. Consequently, the actual heat flux, temperature, and other parameters at various locations in the core may differ considerably from the specified (or nominal) values.*

9.139. A fundamental requirement in reactor design is to ensure that, in spite of such unavoidable variations from average values which are inherent in reactor design, e.g., nonuniform neutron flux distribution, there shall be no point in the core where certain limiting parameters are exceeded during operation under normal conditions. Two important limiting operating parameters (or constraints) are the critical heat flux (§9.98) and the fuel temperature.† A critical heat flux, even locally, can result in a boiling

*For a fuel rod of given dimensions and fuel composition, the heat flux, i.e., the heat flow per unit area q/A, is proportional to the local heat-generation rate (or heat source) per unit volume Q; the latter is equivalent to the local power density which is determined by $\Sigma_f \phi$ [cf. equation (2.54)].

†Other parameters, e.g., fuel cladding temperatures, become limiting under emergency conditions (Chapter 12).

crisis accompanied by an increase in surface temperature, which could lead to cladding failure.

9.140. A principle that has been useful in reactor design is to relate the nominal (or average) performance to the maximum value that can be expected anywhere in the core. To this end, each of several relevant design specifications is assigned an adjustment or correction factor which represents the ratio of the maximum to average values of such parameters as heat flux, coolant-flow rate, and enthalpy rise that would result from the most probable variations in the given specifications. The various factors are then combined in a suitable manner to yield the overall ratio of the maximum to average values of a particular operating parameter. The constraints mentioned must then apply to the maxima determined in this manner.

Peaking and Hot-Channel Factors

9.141. Early in the course of reactor development, the term *hot-channel factors* was used to relate maximum to nominal design conditions. The hot-channel concept was based on the assumption that a reactor core, with solid fuel elements having coolant passages (or channels) between them, would have one channel in which a combination of variations in power density and dimensions result in a heat flux that is greater than that at any other point in the core. Although there is a continuing need to relate maximum to average conditions to develop safety margins, the concept of an actual hot-channel or hot-spot tends to be simplistic. Therefore, *peaking factors* are now generally used, although utilization of *hot-channel* continues. In fact, we will use the term *hot-channel* as a convenient designation for the hypothetical vulnerable location in the core. In any case, the design approach to limiting conditions through the use of peaking factors must be described in licensing documentation for every reactor initial core as well as for subsequent reload cores and approved by the U.S. Nuclear Regulatory Commission (§10.39). These peaking factors are included in the plant's Technical Specifications as limits that are not to be exceeded (§12.240).

9.142. For PWRs, the peaking factors of importance are the heat flux factor and the enthalpy rise factor. These are decomposed through the use of various subfactors. However, before considering such detail, let us consider the principles involved. A three-dimensional nodal nuclear calculation can yield a "map" of the core power distribution, a starting point for both the heat flux factor and the enthalpy rise factor. In fact, as we shall see, the *nuclear* enthalpy rise factor merely describes the *radial* power distribution. However, when it is combined with appropriate flow-related factors, it also yields a limiting coolant condition which can be associated with the maximum heat flux in boiling crisis predictions. The need to determine

DNB design margins is a major reason for PWR peaking factor development. Other limits depend primarily on the heat flux (see §9.163 et seq.).

Idealized Axial Temperature Distributions

9.143 As background for describing the temperature variations along the path of the reactor coolant, it is useful first to consider a classical idealized channel with uniform fuel enrichment and no axial reflector. A generalized representation of such a coolant channel, with its associated effective heat-removal area, is shown in Fig. 9.19, where the different variables are identified. The x coordinate represents the direction of coolant flow in a vertical channel. The rate $dq(x)$ at which heat is added to the stream in a differential length dx of any coolant passage is then given by

$$dq(x) = wc_p \, dt, \tag{9.43}$$

where w is the mass-flow rate of the coolant in the channel associated with one fuel rod, c_p is the specific heat, and dt is the differential temperature increase in the coolant across the length dx. For simplicity it will be postulated that all heat flow within the solid fuel is normal to the coolant stream, i.e., there is no heat conduction along the fuel rod parallel to the coolant channel.* Then $dq(x)$ is the rate of heat generation in a volume $A_c \, dx$, where A_c is the cross-sectional area of the fuel rod (in the y, z plane). Since the volumetric heat source is, in general, a point function $Q(x, y, z)$, it is possible to write

$$dq(x) = \left[\int Q(x, y, z) \, dy \, dz\right] dx,$$

the integration being carried over the fuel rod cross section.

9.144. If a local average heat source per unit volume $Q(x)$ is defined by

$$Q(x) = \frac{1}{A_c} \int Q(x, y, z) \, dy \, dz,$$

it follows that

$$dq(x) = A_c Q(x) \, dx. \tag{9.44}$$

*This postulate implies that the following analysis gives a good approximation provided $dt/dx << dt/dn$, where n is a coordinate normal to x.

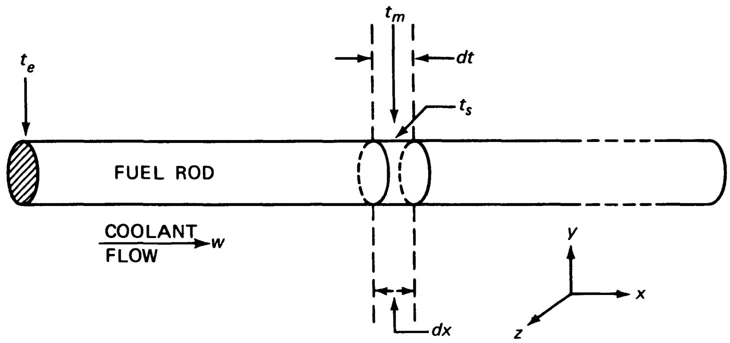


Fig. 9.19. Generalized representation of fuel rod and coolant flow channel. (Note that the rod and channel are actually vertical.)

Hence, from equations (9.43) and (9.44), the temperature distribution in the direction of coolant flow must satisfy the relation

$$\frac{dt}{dx} = \frac{A_c Q}{wc_p},$$

it being understood that Q is really $Q(x)$. If the quantity A_c/wc_p is independent of x, the distribution of the coolant mixed-mean (or bulk) temperature t_m along the channel will be given by

$$t_m - t_e = \frac{A_c}{wc_p} \int_0^x Q\, dx, \tag{9.45}$$

where the fluid entrance temperature, t_e at $x = 0$, is chosen as the datum. This expression gives the increase in the coolant temperature due to the heat added as it flows through the channel.

9.145. By equation (9.9), the local (solid) surface temperature t_s is related to the coolant bulk temperature t_m by

$$\frac{dq}{dA_h} = h(t_s - t_m), \tag{9.46}$$

where h is the heat-transfer coefficient for the given conditions and dA_h is a small element of heat-transfer surface, i.e., the fuel-rod surface (not its cross section). Although h will vary to some extent along the length of the channel, it will be assumed to be constant and independent of x, in order to simplify the treatment without affecting the general conclusions. If p is the circumference of the rod, which may be clad or unclad, then

$dA_h = p\ dx$, and equation (9.46) may be written as

$$dq = hp(t_s - t_m)\ dx. \tag{9.47}$$

From equations (9.44) and (9.47), it is seen that

$$t_s - t_m = \frac{QA_c}{hp}.$$

For a clad fuel rod, the heat-generating volume is based on the *pellet radius*, a, and the heat-transfer area is based on the outer clad surface. It is then best to write

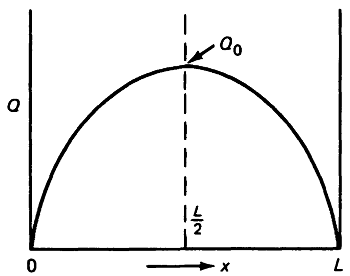
$$t_s - t_m = \frac{QV}{hA_h}, \tag{9.48}$$

where A_h and V are the heat-transfer area and heat-generating volume, respectively, associated with the coolant channel.

Sinusoidal axial source distribution

9.146. In a bare cylindrical or rectangular parallelepiped reactor core, the neutron flux has a cosine distribution in the axial direction (Table 3.2). The volumetric heat-source distribution in this (x) direction will thus be represented by a cosine function, provided the fuel enrichment is uniform along the length of each rod. Previously, the origin of the coordinates was chosen at the center of the core, but here, however, it is convenient to take the origin at the point where the coolant enters the channel (Fig. 9.20). The heat-source distribution along a particular channel, i.e., in the

Fig. 9.20. Sinusoidal axial distribution of heat source.



axial (or flow) direction, can then be expressed in sinusoidal form; thus,

$$Q = Q_0 \sin \frac{\pi x}{L}, \tag{9.49}$$

where Q is the volumetric heat-release rate at the point x, and Q_0 is the maximum value at the center of the channel; L is the total channel length, assuming no reflector. Upon inserting this expression for Q into equation (9.45) and performing the integration, the result is

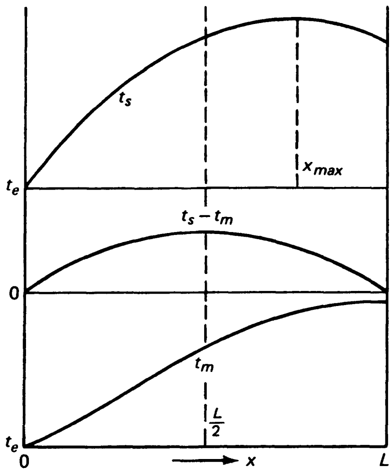
$$\begin{aligned} t_m - t_e &= \frac{Q_0 A_c L}{\pi w c_p}\left(1 - \cos \frac{\pi x}{L}\right) \\ &= \frac{Q_0 V}{\pi w c_p}\left(1 - \cos \frac{\pi x}{L}\right), \end{aligned} \tag{9.50}$$

which gives the rise in temperature of the coolant as it flows through the channel.

9.147. The temperature of the coolant increases continuously along the channel, as shown in the bottom curve of Fig. 9.21; the bulk coolant temperature rise, represented by Δt_c, is thus seen to be

$$\Delta t_c = (t_m - t_e)_{x=L} = \frac{2 Q_0 V}{\pi w c_p}, \tag{9.51}$$

Fig. 9.21. Axial variations of coolant and fuel-surface temperatures.



It is of interest to note that this result can be derived in another manner. Since Q_0 is the maximum value of the heat-release rate per unit volume, the average is $2Q_0/\pi$ (cf. §9.16), and the total heat-release rate for the given channel is then $2Q_0V/\pi$. In the steady state this must, of course, be equal to the rate at which heat is removed by the coolant, i.e., $wc_p\Delta t_c$, so that equation (9.51) follows immediately.

9.148. The temperature difference between the solid and the fluid at any point is obtained from equations (9.48) and (9.49) as

$$t_s - t_m = \frac{Q_0V}{hA_h}\sin\frac{\pi x}{L}, \tag{9.52}$$

which is seen to be analogous to equation (9.49) for the source distribution. The value of $t_s - t_m$, consequently, passes through a maximum when $x = \frac{1}{2}L$, i.e., in the middle of the channel, as shown in the second curve of Fig. 9.21. This result is, of course, to be expected since at any point the fluid-solid temperature difference will be proportional to the radial heat-flow rate at that point, and this is a maximum in the center of the fuel channel.

9.149. The value of $t_s - t_e$, i.e., the fuel surface temperature at any point with respect to the value at the channel entrance, is obtained by adding equations (9.50) and (9.52), so that

$$t_s - t_e = \frac{Q_0V}{hA_h}\sin\frac{\pi x}{L} + \frac{Q_0V}{\pi wc_p}\left(1 - \cos\frac{\pi x}{L}\right). \tag{9.53}$$

It is apparent that $t_s - t_e$ must pass through a maximum at some point beyond the middle of the channel, as shown in the top curve of Fig. 9.21. This maximum represents the highest surface temperature of the fuel element. The point at which the maximum is attained is found by differentiating equation (9.53) with respect to x and setting the result equal to zero; thus,

$$x_{\max} = \frac{L}{\pi}\tan^{-1}\left(-\frac{\pi wc_p}{hA_h}\right). \tag{9.54}$$

From this expression it is seen that the position of the maximum value of t_s approaches the end of the channel, i.e., $x_{\max} \to L$, as w and c_p decrease, and h and A_h increase.

9.150. The maximum surface temperature in the given channel can now be obtained by insertion of $x_{\max}$ into equation (9.53). Assuming the coolant-flow rate to be the same in all channels, the maximum fuel-element surface temperature occurs in the channel where the neutron flux is a maximum.

In this case, Q_0 is the volumetric heat-release rate at the center of the reactor, and the corresponding acceptable maximum value of t_s will affect the maximum reactor power.

9.51. It should be noted that the surface temperature t_s is less than the temperature in the middle of the fuel element. The difference, represented by the temperature drop through the fuel region and the cladding, can be calculated by the methods described earlier (§9.47 et seq.). Because Q varies along the length of the fuel element, the point at which the surface temperature is a maximum will generally not be the same as that at which the interior temperature of the fuel has its maximum value.

9.152. For purposes of computation, it is convenient to represent $\pi x/L$ by the symbol α, i.e.,

$$\alpha \equiv \frac{\pi x}{L}. \tag{9.55}$$

It can then be found from equation (9.54) that

$$\tan \alpha_{\max} = -\frac{\pi w c_p}{h A_h}, \tag{9.56}$$

where $\alpha_{\max} = \pi x_{\max}/L$. Utilizing this expression for $\tan \alpha_{\max}$, equation (9.53) can be reduced to

$$(t_s)_{\max} - t_e = \frac{Q_0 V}{\pi w c_p}(1 - \sec \alpha_{\max}). \tag{9.57}$$

With these two equations, the value of the maximum surface temperature and its location can be readily evaluated.

9.153. This treatment is strictly applicable to a bare reactor. For a reflected reactor, the flux (or power) distribution is flatter than in an equivalent bare core, and the ratio of maximum to average specific power (or volumetric heat source) is lower, as shown in §9.17. One method of treating the problem of a reactor with a thick reflector is to assume a cosine source distribution, as in Fig. 9.21, but to suppose that the value of Q goes to zero at some distance, e.g., about one or two reflector diffusion lengths, beyond the boundary of the core. The integration of sin $\pi x/L$, referred to in §9.146, then does not start from zero, but from x equal to roughly two diffusion lengths in the reflector. Of course, as a result of fuel burnup, the enrichment will no longer be axially uniform. The high-flux region will "flatten" and the axial variations described will change accordingly.

Heat-Flux-Related Limitations in Pressurized-Water Reactors

9.154. The first design limitation to be considered is related to the heat flux at which a boiling crisis could occur; for a PWR this is the DNB flux, which can be computed by means of a suitable correlation (§9.103). The DNB flux is expected to decrease as the coolant quality increases, since the larger the vapor content of the fluid, the closer it would be to the conditions at which DNB occurs.* The quality increases as the enthalpy of the fluid is increased; hence, the enthalpy rise has an important bearing on the value of the DNB flux. In computing this flux for design purposes, the maximum value of the enthalpy, i.e., the value in the hot channel, must therefore be used.

9.155. Since the enthalpy of the coolant increases as it flows upward through the core, the (computed) DNB flux decreases correspondingly, as depicted in Fig. 9.22. Assuming a sinusoidal axial distribution of the volumetric heat source, as in §9.146, the heat flux along the hypothetical hot channel will ideally have the same general distribution, as indicated in the figure. The maximum of this curve, representing the highest possible heat flux in the core for normal operating conditions, is equal to the average core heat flux multiplied by the overall heat flux hot-channel factor. In practice, the axial heat flux will not have a sinusoidal distribution, and this must be taken into consideration by the designer. For the present purpose, however, the ideal axial distribution may be assumed.

9.156. Comparison of the computed DNB heat flux with the hot-channel value at any point along the flow channel gives the *departure-from-nucleate-boiling ratio* (DNBR) or *critical heat flux ratio* (CHFR) at that point. The results for the whole channel are shown by the uppermost curve in Fig. 9.22. It is seen that the DNBR passes through a minimum. In the interest of reactor safety, an essential requirement of reactor design at present is that the minimum DNBR shall be greater than 1.3 for the hottest channel at the 118 percent overpower specified for a PWR. This is one of the important constraints in reactor core design. It should be noted that the minimum DNBR does not occur at the point of maximum heat flux but more toward the exit where the enthalpy of the coolant is higher and the DNBR flux is lower (§9.103). Factors that change the axial neutron (and heat) flux distribution, e.g., insertion of control rods, will affect the location of the minimum DNBR.

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*Since the average core outlet temperature in a PWR is below the saturation temperature, little or no vapor leaves the reactor vessel. However, there is substantial subcooled (local) boiling in the flow core to the heated surface, and bulk boiling may occur in the "hot channels."

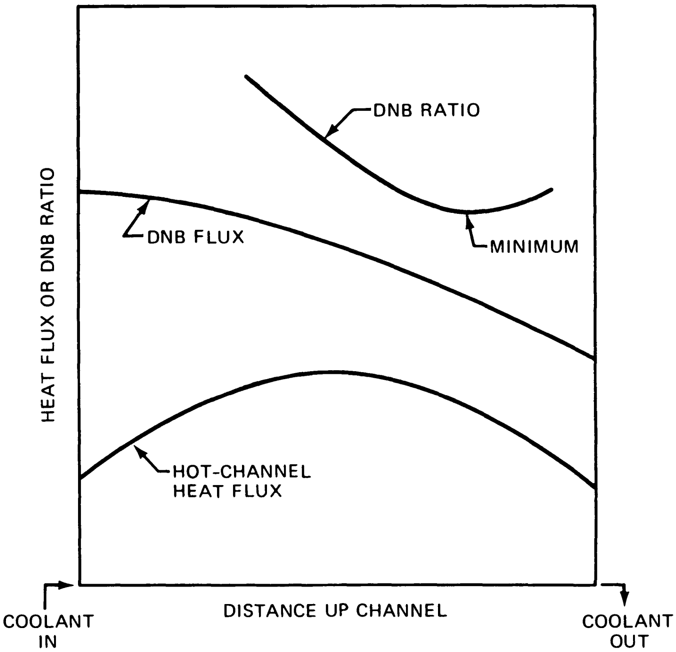

Fig. 9.22. Qualitative representation of heat flux and related conditions along the hot channel in a pressurized-water reactor. (A sinusoidal axial distribution of the heat flux is assumed for simplicity.)

9.157. The design constraint just considered should provide safety from the onset of DNB under normal operating conditions, allowing for a certain amount of overpower that could arise from the instrumental errors or minor reactivity transients. However, no allowance is made for abnormal situations that might result from excessive overpower or inadvertent decrease in the coolant-flow rate. These matters are aspects of reactor safety considered in Chapter 12.

9.158. A somewhat more sophisticated approach preferred by many PWR designers uses the *CHF power ratio* as a measure of overpower that can be tolerated before the critical heat flux (CHF) condition is reached at the hot channel. In this method, the CHF correlation curve is recalculated by computer, point by point, as the channel power levels under consideration are changed, thus reflecting shifts in mass-flow rate and other parameters. On a thermal flux, coolant enthalpy coordinate system, a family of curves are then plotted showing the proposed channel flux and corresponding variation of CHF, each at the same power level. The CHF power ratio is defined as the power at which the two curves would be tangent to one another, divided by the design or "rated" power. Since relative displacements of the curves with power level are considered, the margin is more physically meaningful than the DNBR [22].

9.159. The central fuel temperature of a cylindrical fuel rod depends on Qa^2, where a is the diameter of the fuel pellet, as may be seen from §9.45 et seq. and Example 9.3. By equation (9.20), this is related to the linear heat rate q_L, which is a characteristic property of the fuel material for the specified temperature range. Experiments with uranium dioxide, commonly used as fuel in water-cooled (and moderated) reactors, have shown that melting does not occur until the linear heat rate in the fuel exceeds about 70 kW/m. This is, therefore, the maximum permissible linear heat rate anywhere in the reactor core.

9.160. From equation (9.20), q_L is equal to $Q(\pi a^2)$ and the heat flux q/A is equal to $Qa^2/2b$, where b is the outer radius of the clad rod. It follows, therefore, that

$$q_L = 2\pi b\left(\frac{q}{A}\right).$$

If the maximum heat flux in a PWR is about 1.5×10^6 W/m², and the outer diameter of the clad fuel rod is 0.0095 m, i.e., $b = 0.00475$ m, the corresponding value of q_L is found to be about 45 kW/m. It is apparent that the maximum (hot-channel) heat flux in this reactor is moderate. The average linear heat rate, equal to the maximum divided by the heat flux hot-channel factor 2.5, is about 18 kW/m.

9.161. The zirconium alloy (zircaloy) fuel-rod cladding is subject to failure from a variety of causes, including formation of hydrides, stress corrosion, embrittlement, and interaction with the uranium oxide (§7.172). An increase in the cladding temperature appears to result in an enhanced failure rate. There is little danger in normal operation that the cladding will become hot enough to fail, but under accidental conditions, such as a decrease (or cessation) of the coolant flow, the cladding temperature becomes a critical parameter (Chapter 12).

9.162. Although the temperatures of the fuel and cladding are not likely to represent design constraints, it is nevertheless a common practice to compute the maximum cladding and central fuel temperatures for normal operation and to include them in the list of specifications. The procedure is based on the principles developed in §9.45 et seq. and §9.143 eq seq., with allowance for the variation of the heat conductivity of the fuel material with temperature. The coolant temperature required for the calculations is derived for the hot channel with the minimum flow rate. As seen in §9.149, the maximum cladding surface temperature occurs beyond the point of maximum heat flux (or input), and so also does the maximum central fuel temperature, although not at the same point as the maximum cladding temperature. Typical maximum values in a PWR are 350°C for the cladding and 1700 to 1900°C for the fuel. The melting point of the uranium dioxide in the fuel pellets is about 2760°C initially, but it drops to some 2650°C during reactor operation as a result of the accumulation of fission products.

Factors and Subfactors

9.163. We have mentioned that a *nuclear* heat flux factor is developed from appropriate nuclear calculations. It can be defined as the ratio of the maximum local fuel rod linear power density to the average core-wide linear power density. This assumes that there are no deviations from manufacturing specifications in fuel pellets and rods, densification effects, or uncertainties in calculations and the core monitoring system. Therefore, it is necessary to use subfactors to account for these considerations.

9.164. Most subfactors used in reactor core design tend to fall into two classes. First, there are factors arising from random-type uncertainties, such as those resulting from manufacturing operations, and second, factors introduced to account for inadequate knowledge. Random uncertainties lend themselves to statistical treatment, but those based on inadequate knowledge, e.g., flow distribution and intermixing of the coolant, are generally combined by multiplying them together. The multiplication of subfactors implies that the corresponding events will occur at the same time and at the same place in the core; this is a very conservative assumption. For example, subfactors accounting for flow and mixing uncertainties are design dependent and are not treated statistically. On the other hand, the statistical combination of subfactors is justified when there is a body of experimental or manufacturing data to provide meaningful information. However, it is sometimes possible to establish arbitrary confidence limits for a given uncertainty and to treat the resulting subfactor statistically, as will be seen.

9.165. Statistics theory provides procedures for the analysis and prediction of data which are widely used in the quality control of manufacturing processes. For the present purpose, a simple discussion is adequate. In considering a large number of measurements of a particular quantity, the dispersion of the measurements about a mean value is often expressed as the *standard deviation* σ; this is the square root of the arithmetic mean of the squares of the deviations of each value X_i from the arithmetic mean $\overline{X}$, i.e.,

$$\sigma = \left[\frac{1}{n}\sum_{i=1}^{n}(X_i - \overline{X})^2\right]^{0.5},$$

where n is the number of measurements. If the deviations are random in character, the measurements will exhibit a so-called normal (Gaussian) distribution which is symmetrical about the mean value. In such a distribution, it can be shown that 99.73 percent of the values will lie within the range of $\pm 3\sigma$. In other words, there are only 2.7 chances in 1000 that the actual value will be outside the range. In view of this high probability, the $\pm 3\sigma$ variation from the average is taken as the basis for calculating engineering hot-channel subfactors.

9.166. In order to combine standard deviations arising from variations in several different quantities, e.g., σ_1, σ_2, σ_3, etc., to obtain the overall standard deviation σ, the relationship

$$\sigma = [(\sigma_1)^2 + (\sigma_2)^2 + (\sigma_3)^2 + \cdots]^{0.5}$$

is used. A similar expression is, of course, applicable to the 3σ values employed in the determination of engineering hot-channel factors. An application of this procedure will be given in the next section.

9.167. In obtaining the nuclear heat flux factor for a PWR, it is usually helpful to assume that the radial and axial power distribution are separable. The radial factor $F^N_{\Delta H}$, also called the enthalpy rise hot-channel factor, is effectively the maximum-to-average ratio of the total power generated (or coolant enthalpy increase) in a (vertical) flow channel to that for all channels. The axial factor F^N_z is the ratio of the maximum to average power density in the axial (vertical) direction; it is derived from calculations utilizing the principles in §9.146 et seq. with a more realistic distribution—rather than the sinusoidal form—of the heat source. The product of the radial and axial factors gives the heat flux nuclear hot-spot (or peaking) factor, F^N_q; thus,

$$F^N_q = F^N_{\Delta H} \times F^N_z.$$

It is the ratio of the maximum heat flux, i.e., at the hot spot in the reactor, to the core average based on ideal design parameters. This must be multiplied by the engineering hot-channel factor to obtain the overall peaking factor for the heat flux.

9.168. The engineering hot-channel factor is a combination of a number of subfactors arising from variations in the density, diameter, and enrichment of the uranium dioxide fuel pellets contained in the fuel rod, and the diameter of the rod with its cladding. The engineering subfactors may be treated statistically since they are based on manufacturing quality control data. The combination of these subfactors, in terms of variations of $\pm 3\sigma$ from the average, then gives the heat flux engineering hot-channel factor

F_q^E. Some values of the engineering subfactors for a pressurized-water reactor and their statistical combination are quoted in Table 9.2; each subfactor represents the ratio of maximum-to-average heat flux associated with variations of $\pm 3\sigma$ in the indicated quantities.*

9.169. For the fuel rod considered in Table 9.2, a reasonable computed value for $F_{\Delta H}^N$ is 1.55 and F_z^N is also about 1.55; hence, the heat flux hot-channel nuclear factor is

$$\begin{aligned} F_q^N &= F_{\Delta H}^N \times F_z^N \\ &= 1.55 \times 1.55 = 2.40. \end{aligned}$$

The total heat flux hot-channel factor F_q is the product of the nuclear and engineering factors, so that

$$\begin{aligned} F_q &= F_q^N \times F_q^E \\ &= 2.40 \times 1.04 = 2.50. \end{aligned}$$

In a typical PWR, the average flux for the whole core is in the vicinity of 6×10^5 W/m^2; hence, the peak heat flux at the "hottest" spot in the hot channel is $(2.5)(6 \times 10^5) = 1.5 \times 10^6$ W/m^2.

Enthalpy Rise Hot-Channel Factor

9.170. In developing the design constraints referred to earlier, the hot-channel factor for the enthalpy rise is required in addition to that for the flux. The enthalpy rise hot-channel factor is the ratio of the maximum enthalpy increase, i.e., in the hot channel, to the average enthalpy increase per channel. The enthalpy rise in a channel is related to the total heat input (or power) to the channel and to the mass-flow rate of the coolant through that channel. Hence, the enthalpy rise hot-channel factor is the product of a factor for the heat input in the hot channel and one for the flow rate. The ratio of the heat input in the hot channel to the average value is equal to the product of $F_{\Delta H}^N$ and F_q^E. This must be multiplied by an engineering hot-channel factor which relates the flow rate in the hot channel to that in an average channel.

9.171. The flow-rate engineering factor is made up of several subfactors which affect the coolant flow; some characteristic values of these subfactors

*It can be readily shown that the heat flux at the outer surface of a clad fuel rod is $Qa^2/2b$, where Q is the average rate of heat generation (by fission) per unit volume of fuel, a is the radius of the fuel pellet, and b is the outer radius of the clad rod. These three quantities are affected by the parameters in Table 9.2.

TABLE 9.2. Typical Engineering Heat Flux Hot-Channel Factors

	Subfactor (Based on $\pm 3\sigma$)
Pellet density	1.024
Pellet diameter	1.0027
Pellet enrichment	1.020
Clad rod diameter	1.012

Statistical combination:

$$[(0.024)^2 + (0.0027)^2 + (0.020)^2 + (0.012)^2]^{0.5} = 0.034$$

Combined heat flux engineering factor, $F_q^E = 1.034$
(rounded off to 1.04)

in a PWR are listed in Table 9.3. Variations in fuel-rod pitch and bowing have a statistical basis, as do the variations in the diameter of the clad fuel rod; these factors may then be combined statistically. Other factors, which involve uncertainties and cannot be treated statistically, account for possible maldistribution of coolant from the inlet plenum, flow redistribution resulting from differences in hydraulic resistance from some local boiling, and interchannel mixing of the coolant. The latter effect tends to reduce the enthalpy rise and hence the corresponding subfactor is less than unity. In order to obtain the overall hot-channel flow factor, the combined sta-

TABLE 9.3. Typical Engineering Subfactors For Flow Rate

Statistical subfactors	Based on $\pm 3\sigma$
Rod pitch and bowing	1.045
Clad rod diameter	1.042
Statistical combination	1.062
Nonstatistical subfactors	
Inlet flow maldistribution	1.03
Internal mixing, and boiling flow redistribution and diversion	1.10
Flow mixing	0.90

tistical factor is multiplied by the product of the nonstatistical subfactors; thus,

$$\begin{aligned}\text{Hot-channel flow factor} &= 1.062 \times 1.03 \times 1.10 \times 0.90 \\ &= 1.08.\end{aligned}$$

The hot-channel factor for the enthalpy rise $F_{\Delta H}$ is then given by

$$\begin{aligned}F_{\Delta H} &= F^N_{\Delta H} \times F^E_q \times \text{Flow factor} \\ &= 1.55 \times 1.04 \times 1.08 = 1.74.\end{aligned}$$

The hot-channel enthalpy rise is thus 1.74 times the enthalpy rise computed from the ideal (or nominal) reactor specifications.

Statistical Core Design Techniques

9.172. With the increasing availability of digital computers for engineering calculations, methods have been developed for the treatment of core design limits in PWRs that are more realistic than the basic hot-channel concept described in previous sections. Rather than identify a single hot channel as representing the "worst case," it is possible to analyze each channel individually and to determine the corresponding minimum DNBR and other limiting parameters. Statistical methods are then used to evaluate the number of fuel rods that may have a given small probability of failure, as a function of design and operating conditions.

9.173. In one example of the statistical core design technique [23], a radial power factor (ratio of actual to average), analogous to $F^N_{\Delta H}$, is computed for each rod in the core, and so also is an axial factor, equivalent to F^N_z for each rod. Engineering factors, which are sometimes called "hot-channel" factors although they do not refer to a specific hot channel, are used to describe deviations from nominal (or design) values in physical characteristics of the fuel material, fuel pellet and cladding dimensions, and flow area. These factors are derived from several subfactors based on measurements made on the actual core under construction, on similar cores previously constructed, or on specified manufacturing tolerances. The subfactors are then combined statistically to yield a heat-input factor F_Q, a local hot-channel heat flux factor $F_{Q''}$, and a flow-area reduction factor F_A. Typical values of these engineering factors in a large PWR are as follows:

Heat input (or power peaking) factor F_Q	1.011
Local heat flux factor $F_{Q''}$	1.014
Flow-area reduction factor F_A	0.98 (interior)
	0.97 (periphery)

9.174. The computations are made for several different operating conditions but two are of special interest, namely, maximum and average (or most probable) design conditions. For the maximum design condition, the maximum values of the radial and axial power factors are assumed to be applicable to each fuel rod in the core; the computed heat input (or heat flux) for each rod is then multiplied by the factors F_Q and $F_{Q''}$, both of which are greater than unity. The coolant-flow rate is determined using the Factor F_A, which is less than unity. In a particular case, the maximum radial factor was 1.78 and the maximum axial factor was 1.70; hence, with the engineering factors given above, the maximum-to-average heat flux ratio for each coolant channel is

$$\text{Heat flux factor} = 1.78 \times 1.70 \times 1.011 \times 1.014 = 3.10.$$

In evaluating the coolant enthalpy rise, the factor includes an allowance for the reduction in the flow area which applies to all the channels; thus,

$$\text{Enthalpy factor} = 1.78 \times 1.011 \times 1.014 \times 0.98 = 1.79.$$

By means of these data, it is possible to compute the DNB flux (and DNBR) along each fuel rod (or channel) instead of a single hot channel in the earlier treatment.

9.175. For the average (or most probable) design condition, the heat input (or power) distribution for each fuel rod is derived from neutronic calculations, and the coolant-flow rate is compatible with the location of each channel in the reactor core. The DNB data are then evaluated for each fuel rod (or coolant channel) in the usual manner.

9.176. The results obtained from the foregoing computations are treated statistically to derive the number of possible burnouts at a specified confidence level for various operating conditions. Table 9.4 gives the values obtained at a 99 percent confidence level for both maximum and most probable design conditions of a PWR with 36,816 rods operating at 100 percent of design power and at 112 percent overpower. It is seen that, under the worst conditions that might arise in normal operation, i.e., at maximum design conditions and 112 percent overpower, there is a 99 percent confidence that almost 99.9 percent of the fuel rods will be protected from burnout. The minimum DNBR under these conditions is 1.39 at the same confidence level.

Boiling-Water Reactors

9.177. The design approach based on hot-channel and peaking factors used for BWRs is similar in principle to that described earlier for a PWR

TABLE 9.4. Statistical Results For Burnout in a Pressurized-Water Reactor at the 99 Percent Confidence Level

Percent of Full Power	Number of Possible DNB (Out of 36,816 Rods)	Percentage of Rods Protected	Minimum DNBR
	Maximum Design Condition		
100	3	99.992	1.75
112	41	99.889	1.39
	Most Probable Design Condition		
110	1	99.997	1.93
112	15	99.959	1.55

system, although there are differences in details. In general, engineering factors are not treated separately but are included in the peaking factors. For example, the hot-channel factor for the heat flux is taken as the product of three factors: the relative assembly power factor, the local peaking factor, and the axial peaking factor. The relative assembly power factor is the total power generated in a fuel assembly divided by the average assembly power over the whole core. The local peaking factor is the ratio of the maximum heat flux in a fuel assembly to the assembly average at the same level in the reactor core. The product of these two factors is essentially equivalent to the radial factor $F^N_{\Delta H}$ defined in §9.167. Finally, the axial factor, equivalent to F^N_z, is defined as the ratio of the maximum heat flux along the length of a fuel rod, i.e., the peak flux in the hot channel, to the average for that rod (or channel).

9.178. Typical values of the three factors for a boiling-water reactor are as follows:

Relative assembly power factor	1.40
Local peaking factor	1.13
Axial peaking factor	1.40

The overall hot-channel (peaking) factor for the heat flux is thus

$$F_q = 1.40 \times 1.13 \times 1.40 = 2.21.$$

The hot-channel factor for the enthalpy rise does not include the axial factor and it is consequently

$$F_{\Delta H} = 1.40 \times 1.13 = 1.58.$$

The average heat flux for a typical BWR is about 5×10^5 W/m^2; the maximum (hot-channel) flux is consequently $(2.2)(5 \times 10^5) = 1.1 \times 10^6$ W/m^2. The outer diameter of a clad fuel rod is 0.0125 m; the linear heat rates are, consequently, 20 kW/m average and 44 kW/m maximum.

9.179. As for a PWR, an important limiting design condition for a BWR is related to the critical heat flux (or CHF) at which a boiling crises would be expected to occur. This flux, as computed on the basis of an acceptable procedure (§9.104), is a function of the local steam quality (and hence of the enthalpy rise), the mass-flow rate of the coolant, the flow-area geometry, and the pressure. Like the DNB flux, the CHF decreases with increasing enthalpy of the coolant, and the ratio of the CHF at any point in the flow channel to the actual flux at that point, i.e., the CHF ratio (or CHFR), passes through a minimum. The design limitation is that the minimum CHFR in the hot channel must exceed 1.9 at 120 percent overpower.

9.180. In determining the most vulnerable location in a BWR core, as indicated by the minimum CHFR, consideration must be given to the change in the neutron flux shape (or power density distribution) that may result from the control elements, which enter from the bottom of the core (§5.195), and from the presence of steam voids in the coolant. Furthermore, except toward the end of the core life when the control rods are withdrawn, these rods may be used to shape the neutron flux. The accompanying spatial changes in the heat-generation rate must also be borne in mind in connection with the limitation on the linear heat rate in order to avoid central fuel melting.

9.181. A figure of merit used for BWR design and operation is the *critical power ratio* (CPR). Correlations of experimental data yield the steam quality, or "critical quality," at which the transition from nucleate boiling occurs as a function of mass-flow rate, power, and other parameters. The *critical power* is defined as that fuel bundle power which would produce the critical quality at some point within the bundle. The *critical power ratio* is then defined as the ratio of the critical power to the operating bundle power, at the reactor condition of interest. Design and operating limits are then established in terms of the CPR [24].

Gas-Cooled Reactors [25]

9.182. Gas-cooled reactors such as the HTGR have the advantage of not having the boiling crisis design limit characteristic of water-cooled reactors. The core thermal design is based on classical methods describing conduction and convection transport to a gas with the aid of suitable computer codes. A design objective is to achieve a core power distribution so that the coolant exit temperature will be radially uniform. Power peaking factors developed by computer codes are used primarily as a design aid in modeling the core power distribution.

Fast Liquid-Metal-Cooled Reactors

9.183. The peaking factor design approach for liquid-metal-cooled fast breeder reactors, i.e., LMFBRs (§15.51), is inherently similar to that used for water-cooled reactors, but differs in some respects [26]. In PWRs and BWRs, hot-channel factors are used primarily as part of the determination of design margins in relation to a boiling crisis. In LMFBRs, however, the factors and subfactors are used together with computer codes to identify operating margins in relation to a variety of other thermal and hydraulic design limits, e.g., those affecting cladding temperature and the structural integrity of the core internals. Fuel failure propagation is also of concern. A statistical treatment is generally used to analyze the entire core rather than to identify a single hot channel. An objective is to determine the number of channels that approach design limits within specified confidence ranges in a manner analogous to the PWR statistical approach described in §9.172 et seq.

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PROBLEMS

1. The heat flux at the outer surface of a 12.5-mm-diameter BWR fuel rod is 1.1 MW/m^2. (*a*) Determine the corresponding thermal neutron flux if the UO_2 pellet has a diameter of 10.6 mm and is enriched to 1.9 percent in ^{235}U. (*b*) What is the linear heat rate and temperature difference between the pellet center and surface?
2. A PWR operating at 3250 MWt is tripped by the step insertion of 9.0 dollars of negative reactivity in response to an accident situation in which the coolant heat transfer coefficient is degraded to 2 percent of its former value after 60 s. Neglecting only stored heat, make an estimated plot showing the temperature profile from the fuel center to the coolant, which is at 340°C. Be sure to give temperature values at the interfaces. In example 9.3 the temperature profile of a similar PWR fuel rod at rated power was shown to include drops of 17°C, 26°C, 109°C, and 480°C across the fluid film, clad, gas gap, and pellet, respectively.
3. (*a*) An evaluation of the effect of a film of scale or "crud" on the temperature of the fuel rod clad in a pressurized water reactor is desired. The 9.5-mm-o.d. rod has an internal heat generation rate of 43.8 kW per meter of length, which is the same for both the clean and scaled cases. If the scale is estimated to have a thickness of 5×10^{-5} m and $k = 2$ W/m · K, how much higher in temperature is the outer surface of the clad for the scale case than in the clean case? (*b*) What is the surface heat flux? (*c*) If the surface temperature is fixed at 370°C and the zircaloy ($k = 14$ W/m · K) clad has a thickness of 0.57 mm, what is the inner wall temperature of the clad for the scale case?
4. At one time, a 19 × 19 assembly design for a PWR core was considered as an alternate to the 17 × 17 design described in Table 13.1. The fuel rods in such a design would have an outside diameter of 8.51 mm, but the clad thickness would be unchanged. The assembly power would be the

same for both designs and the ratio of maximum to average rod surface flux would be 2.5. With reference to a bulk coolant temperature of 320°C, determine the temperature at the center of the fuel and at the outer surface of the clad for both the maximum and average heat flux in the 19 × 19 design. You may assume that for uranium dioxide, k = 2.8 W/m · K.

5. Water at a temperature of 200°C and a pressure of 7.0 MPa flows through a 10-mm-i.d. tube 4.0 m long at a rate of 0.1 kg/s. Heat at a rate of 10 kW is applied uniformly to the tube. Estimate the pressure drop.
6. Coolant enters a PWR core at 293°C and leaves at 329°C at a rate of 18.3 Mg/s. There are 193 fuel assemblies, with each assembly containing 9.5-mm-o.d. rods on a 17 × 17 square pitch of 12.6 mm. Estimate the pressure loss through the 4.17-m-high core neglecting the effect of rod spacers but considering inlet and outlet losses. It has been determined that the friction factor for "tube bundles" such as these are 30% higher than those determined from the smooth pipe correlation.
7. In a heat exchanger consisting of steel tubes, with k = 43 W/m · K, having an internal diameter of 28 mm and external diameter of 33 mm, heat is transmitted from a primary coolant, inside the tubes, to a secondary coolant, outside. The respective heat-transfer coefficients are 48 and 36 kW/m^2 · K. Determine the temperature difference between the primary and secondary coolants and the temperature drop across the tube wall when heat is being transferred at a rate of 5 kW/m of tube.
8. As a result of a circulating pump problem, the coolant flow rate drops to 20 percent of the rate given in Example 9.6. What is the corresponding heat-transfer coefficient?
9. Cooling water, containing 5 ppm of sodium by weight, is circulated through a reactor so that in each cycle the water spends 5 s in the reactor and 30 minutes in the external heat exchanger, holdup tanks, etc. If the average thermal flux of the reactor is 10^{18} neutrons/m^2 · s, calculate the equilibrium value, attained after extended operation, of the activity of the water leaving the reactor, expressed in Bq/m^3, due to sodium-24 (half-life 15 h).
10. Assume that the hot-channel enthalpy rise factor $F_{\Delta H}$, as described in §9.171, is 1.74 for the reactor described in Table 13.1. (*a*) What is the maximum enthalpy of the coolant? (*b*) What is the corresponding coolant temperature? (*c*) Assume that the other hot-channel factors given in §§9.169 and 9.170 apply. If the rod-coolant heat-transfer coefficient is 37,500 W/m^2 · K, estimate (show work) the maximum clad outside temperature.
11. Consideration is being given to replacing a *few* of the 4700 tubes in a PWR steam generator, which are 16.6 mm i.d. and 17 m long, with 13-mm-i.d. tubes of the same length. The primary coolant rate in the steam generator is 4370 kg/s. Estimate the mass flow rate through one of the new tubes.
12. A 3800-MW (thermal) reactor has a coolant flow rate of 18.3 Mg/s. The average core linear heat rate is 17.5 kW/m. The fuel thermal conductivity is 3.2 W/m · K and the HZP coolant temperature is 293°C. Estimate the power defect.

$$\text{Doppler coefficient} = \frac{-3 \times 10^{-4}}{T^{1/2}} \quad (T \text{ is in K})$$

$$\text{Moderator coefficient} = -4 \times 10^{-4}\ \text{K}^{-1}$$

$$\text{Coolant } C_p = 5.4 \times 10^{3}\ \text{J/kg} \cdot \text{K}$$

13. It has been pointed out that for a core with constant channel cross section, specific power, peaking factor, and core temperature rise, the pressure drop for a PWR is approximately proportional to the cube of the core height. Do you agree with this statement? If so, justify it.

CHAPTER 10

Reactor Fuel Management and Energy Cost Considerations

INTRODUCTION

10.1. *Reactor fuel management* is concerned with those activities involved in the planning for and design of the fuel loading for a nuclear power plant. Planning activities include fuel procurement over a period of years as required for future reload batches and consideration of utility operational strategy that may affect the energy production of the given generating unit. Fuel loading design requires not only the development of specifications for a new fuel batch, but also the determination of the core location for fuel assemblies from previous batches which will be reinserted. The nature of these activities will become clearer to the reader as the relevant topics are presented.

10.2. The *nuclear fuel cycle* is the path followed by the fuel in its various states, from mining the ore to the disposal of the final wastes. At one time, the fuel "cycle" envisioned the recycling of recovered fissile and fertile material from spent fuel. However, in the United States, the prospect of this being implemented in the near future is remote. Therefore, from the viewpoint of an electric utility operating a nuclear power plant, major interest is in the fuel burnup stage and associated fuel management. In fact, without plutonium recycle, as practiced in Europe, the objective in the United States is to obtain as much energy as possible at minimum cost using a once-through cycle. We will therefore cover the pre-reactor fuel cycle steps in only enough detail so that fuel procurement is understandable. Post-reactor operations and waste management are treated in Chapter 11.

10.3. Economic considerations play a significant role in fuel management. Although a comprehensive discussion of nuclear power economics is beyond our scope, some background is essential for an understanding not only of core management but of many aspects of plant system design and operation. Nuclear energy-generating expenses normally include costs associated with the capital investment required, the cost of fuel, and operation and maintenance charges. Later in this chapter, a brief introduction to these cost categories will be given. In Chapter 14 we will show how the relative cost contributions affect plant operation strategy.

PRE-REACTOR FUEL OPERATIONS

Production

10.4. Uranium ores are found in many parts of the world, with the contained percentage of uranium varying widely. Rich ores contain about 2 percent uranium, chiefly in the form of oxide minerals. However, most commercially mined ore is of medium grade, in the range of 0.1 to 0.5 percent uranium in the form of minerals of complex composition. The practically recoverable grade depends on numerous economic factors, as is true for most ores.

10.5. Since medium-grade ores contain such a large fraction of inert material, a concentration operation is needed in a mill located close to the mining area. Typical operations utilize either chemical leaching or solvent extraction procedures. The product is *yellow cake*, containing about 80 percent of U_3O_8. The yellow cake is then shipped to a central purification facility where further chemical operations are used to remove impurities and produce either uranium dioxide or uranium hexafluoride feed for subsequent isotopic enrichment processes.

Isotopic Enrichment [1]

10.6. Since natural uranium contains only about 0.7 percent of the uranium-235 isotope and commercial LWRs require fuel containing about 3 percent uranium-235, an enrichment step is necessary. Enrichment is difficult because the effectiveness of separation using possible physical methods depends on the difference in molecular weights of isotopic UF_6 compounds, which is slight. During World War II, the development of processes yielding uranium enriched to the weapons-grade level of about 93 percent uranium-235 was one of the major challenges of the overall effort. From this activity, a complex of plants using the gaseous diffusion process evolved which were used for many years for reactor fuel enrichment as well as to meet weapons requirements. More recently, gas centrifuge processes have become practical to meet reactor needs. From the viewpoint of the fuel manager, process details are secondary in importance to the materials and work requirements for the separation which are common to any process.

Isotopic Feed Material and Separative Work Requirements

10.7. From the economic standpoint, the user of fuel needs to know how much uranium feed is required to produce a certain amount of enriched material and the cost of enrichment. To show how these quantities are determined, general relationships will first be derived between the amounts and assays of feed, enriched product, and tails. The following treatment is applicable to both gaseous-diffusion and gas-centrifuge processes although it has hitherto been employed primarily for the former.

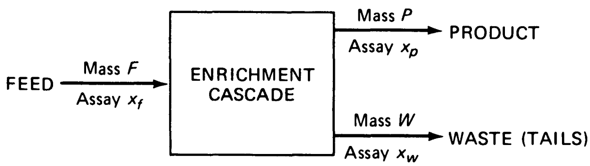
10.8. If F is the mass of uranium (regardless of its isotopic composition) in the feed material supplied to a separation cascade, P is the mass of product withdrawn, and W is the mass of the waste, a uranium mass balance (Fig. 10.1) requires that

$$F = P + W, \tag{10.1}$$

assuming, as is generally true, that there is no appreciable loss of uranium during the operation. A similar balance can be applied to the uranium-235 only; thus,

$$F(x_f) = P(x_p) + W(x_w), \tag{10.2}$$

Fig. 10.1. Feed, product, and waste (tails) streams in an isotope enrichment cascade.



where x_f, x_p, and x_w are the assays (expressed as mass fractions of uranium-235) in feed, product, and waste, respectively. By eliminating W from these two equations, the result is

$$\frac{F}{P} = \frac{x_p - x_w}{x_f - x_w}. \tag{10.3}$$

This equation gives the mass of uranium feed of assay x_f required per unit mass of uranium product of assay x_p, assuming a tails (waste) assay of x_w.

10.9. The cost of enrichment is determined by the amount of work that has to be done to achieve the enrichment. A so-called *value function* has been developed on the basis of the theory of the gaseous-diffusion cascade. It is represented by $V(x)$ and is defined for any material of assay x by

$$V(x) = (1 - 2x) \ln \frac{1 - x}{x}. \tag{10.4}$$

Because x is a fraction, the value function, which is characteristic of a given assay (uranium-235 content), is a fraction and has no units. It is used to determine the work required to yield a product of a desired assay from a given feed with a specified waste.

10.10 The effort expended in separating a mass F of feed of assay x_f into a mass P of product of assay x_p and waste of mass W and assay x_w is expressed in terms of the number of *separative work units* (SWU) needed. This is given in terms of the respective value functions by

$$\mathrm{SWU} = WV(x_w) + PV(x_p) - FV(x_f). \tag{10.5}$$

Since the value functions have no units, the SWU will have the same units as the masses W, P, and F. The general practice is to state the number of separative work units in terms of *kilograms* of uranium. Upon dividing equation (10.5) through by P, the result

$$\frac{\mathrm{SWU}}{P} = \frac{W}{P} V(x_w) + V(x_p) - \frac{F}{P} V(x_f) \tag{10.6}$$

gives the number of separative work units received per unit mass of product.

Example 10.1. The owner of a nuclear power plant requires 100,000 kg of 3.0 percent enriched uranium. Determine the amount of natural uranium feed and the number of separative work units needed, assuming a tails assay of (*a*) 0.200 percent and (*b*) 0.300 percent.

First, the necessary value functions are determined from equation (10.4); the results are as follows:

	x	$V(x)$
Product	0.03	3.27
Feed	0.0071	4.87
Tails	(*a*) 0.002	6.19
	(*b*) 0.003	5.77

(*a*) For $x_w = 0.002$, equation (10.3) gives

$$\frac{F}{P} = \frac{0.0300 - 0.0020}{0.0071 - 0.0020} = 5.48$$

and from equation (10.1)

$$\frac{W}{P} = \frac{F}{P} - 1 = 4.48.$$

Since F/P is 5.48, the natural uranium feed requirement is (5.48) (100,000) = 548,000 kg.

From equation (10.6),

$$\frac{\text{SWU}}{P} = (4.48)(6.19) + 3.27 - (5.48)(4.87) = 4.31.$$

Hence, the SWU requirement is 431,000 kg.

(*b*) For $x_w = 0.003$, $F/P = 6.57$ and $W/P = 5.57$; hence, the natural uranium feed requirement is 657,000 kg.

The separative work is obtained from

$$\frac{\text{SWU}}{P} = (5.57)(5.77) + 3.27 - (6.57)(4.87) = 3.42.$$

The SWU requirement is thus 342,000 kg.

10.11. The cost per separative work unit is determined from the enrichment plant operating costs and the cost of the capital invested in the plant. If C_s is the cost of a separative work unit, e.g., in dollars per kilogram, then

$$P(C_p) = (\text{SWU})C_s + FC_f - WC_w \tag{10.7}$$

and

$$C_p = \frac{\text{SWU}}{P} C_s + \frac{F}{P} C_f - \frac{W}{P} C_w, \tag{10.8}$$

where C_p, C_f, and C_w are the costs per kilogram of product, feed, and waste, respectively. According to equation (10.7), the cost of the product is equal to the cost of the enrichment operation (separative work cost) plus the cost of the feed, less credit for the value of the waste (tails). In practice, the credit for the tails is usually neglected. The unit cost of the product then depends on the ratios SWU/P and F/P, which are determined by the assays of the product, feed, and tails, and on the costs per kilogram of the separative work and the feed.

Fabrication of Fuel Assemblies

10.12. Following enrichment, the uranium hexafluoride is shipped to the plant of a fuel assembly vendor, where it is converted on-site to oxide. Fuel fabrication is a highly automated process with the need for close quality control to meet exacting specifications. Uranium dioxide powder having the proper characteristics is first prepared, then formed into pellets using ceramic techniques. Adherence to close dimensional tolerances is obtained by grinding. They are then inserted into clad tubing which had been previously prepared. The various components making up the finished fuel assembly (§§7.168, 13.23) are then brought together using standard fabrication procedures.

IN-CORE MANAGEMENT

Introduction

10.13. A commercial operating reactor requires refueling every year or so in accordance with plans made a year or more in advance. In-core fuel management is concerned with the design of such fuel reload packages. Generally, the objective is to minimize the utility's energy generation cost while staying within a number of safety-related design constraints and materials limitations. However, we will see that the task is challenging, not only because of the large number of design parameters that require attention, but also as a result of changes in constraints made from year to year as the technology improves. Thus, earlier designs are useful only as general guides.

10.14. For the reload core design, a neutronic model is needed of each assembly covering the operating cycle from fuel loading to removal. Contributing to the model are core thermal-hydraulics and reactivity effects. Also relevant is the operating economics of the electric utility generating system. Hence, the picture is complicated. Our purpose here is to provide an initial understanding of both the design challenge and current practice with emphasis given to light-water reactors, particularly the pressurized-water reactor concept.

Fuel Burnup

10.15. The useful lifetime of fuel in a reactor is measured by the burnup, expressed as the total amount of *thermal* energy generated per unit quantity of heavy element charged to the core. In LWRs, the freshly loaded fuel consists of plutonium-free uranium oxide. Hence, the burnup is expressed as the thermal energy produced during the lifetime of the fuel per unit mass of *uranium* initially charged to the reactor even though significant plutonium is produced during this time. Alternative methods for expressing burnup, especially in irradiation effects studies, are (1) number of fissions per m^3 of fuel and (2) percentage of heavy-element atoms that have undergone fission. As seen in Example 1.2, for a reactor with uranium dioxide fuel, e.g., a LWR, a burnup of 2.6 TJ/kg U (or 30,000 MW · d/t) is equivalent to about 7.4×10^{26} fissions/m^3 of fuel and to about 3 percent of atoms fissioned as uranium-235 plus plutonium-239 and plutonium-241.

10.16. There are several reasons why reactor fuel must be discharged long before the fissile and fertile materials are consumed. In the first place, accumulation of fission products and of the isotopes of heavy elements, which act as neutron poisons, and depletion of the fissile species, e.g., uranium-235 and plutonium-239, can decrease the reactivity to such an extent that the operational requirements of the reactor will no longer be satisfied. Furthermore, materials performance levels may deteriorate with continued exposure so that they become unacceptable. For example, leakage of fission products into the coolant is not tolerated. The relevant fuel performance considerations are described in §7.170 et seq.

10.17. Generally, a high burnup is desirable so that unit costs associated with the preparation of the fuel could be spread over more units of energy generated. When fuel reprocessing was considered, cost studies showed that the increased burnup benefits leveled at a burnup in the vicinity of 2.6 TJ/kg as a result of the influence of enrichment charges [2]. However, with present "once-through" (no reprocessing) practice in the United States there is an incentive to extend the burnup in LWRs until the fuel materials performance levels are no longer acceptable. On the other hand, as burnup is extended, the higher enrichment needed makes the in-core fuel management design more difficult (§10.26 et seq.). In addition to more efficient resource utilization, extended burnup reduces the number of discharged fuel assemblies that must be stored per amount of generated energy. Experimental burnups on the order of 4.3 TJ/kg have been obtained in some LWRs [3].

Staggered Refueling

10.18. Should the reactor be initially loaded with a complete core of a single enrichment, the power distribution will be similar to the flux distribution shown in the reflected reactor case of Fig. 3.18. This means that the fuel located in the region of high power will "burn" faster than the fuel in regions of lower power. Although the power distribution will tend to "flatten" with time as the fuel in the higher power regions becomes depleted of fissile atoms, the fuel in the lower flux regions will be underutilized when the core reactivity is reduced to the level requiring shutdown for refueling.

10.19. Furthermore, safety limitations are based on the highest power fuel rod, which normally will be operating at the highest temperature. Thus, if the maximum power of the core is close to the average power, the reactor could be operated at a higher level than would be permissible if the peak-to-average power ratio would be higher. As we will see, the fuel can be better utilized and the reactor operated at a higher power level if reloading is done on a staggered basis. Also, if there is good neutron coupling between fresh fuel and fuel about to be discharged, additional burnup can be obtained, since the older fuel can be used to control reactivity. Otherwise, parasitic absorbers would be required.

10.20. Further explanation of staggered refueling may be helpful here. As an example, let us consider a typical PWR core containing 193 fuel assemblies. For our present orientation purposes, assume that the planned fuel burnup is about 2.6 TJ/kg U (or 30,000 MW · d/t), which corresponds to about three calendar years of operation. During annual shutdowns for refueling and plant maintenance, a batch of about 64 assemblies *averaging* the planned burnup of 2.6 TJ/kg U would be removed from the core and replaced with fresh fuel. We would have remaining in the core two batches, one exposed to 1 year of irradiation, the other to 2 years. The assemblies in these three batches would then be rearranged or "shuffled" in a carefully designed pattern that includes the fresh fuel to provide good neutron coupling. After one year of operation, the process is repeated, but the loading pattern may be slightly different to accommodate possible differences in individual assembly burnup histories. We can see that from material balance considerations, the number of batches resident in the core must be equal to the number of burnup periods experienced by a given discharged batch. It should be noted that most present PWRs now operate on 18- or 24-month cycles, with burnup in the range 40,000 to 60,000 MW · d/t.

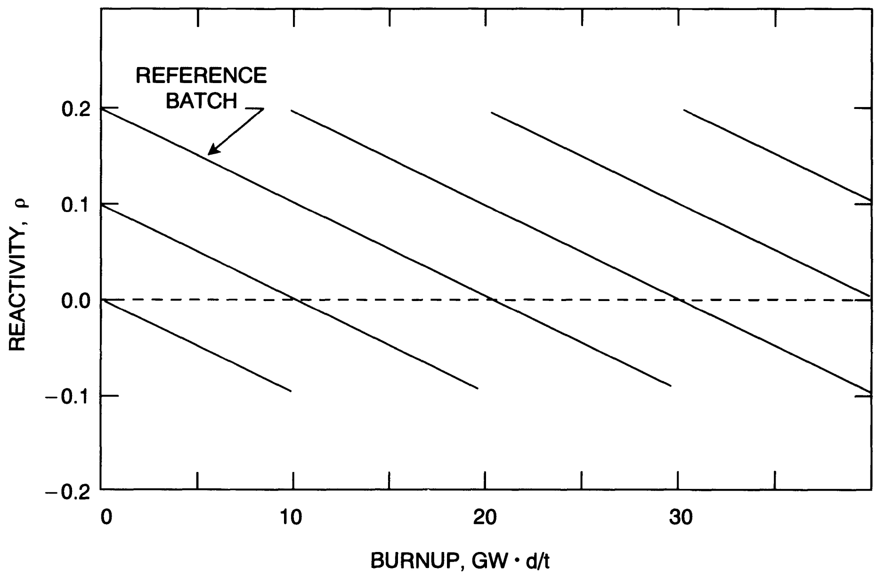
However, the staggered refueling principles are the same as for the example given here.

10.21. This fuel batch reactivity sharing behavior, which is inherent in staggered reloading, is shown in Fig. 10.2. The batch reactivity, ρ, is plotted versus burnup. Note that the ordinate is reactivity, defined as the excess of neutron production per neutron produced, not the multiplication factor, k, i.e.,

$$\rho \equiv \frac{k - 1}{k}$$

It has been shown that ρ is a linear function of burnup for LWR fuel for a practical burnup range [4]. When the reference assembly batch is inserted in the core at zero burnup, two previous batches are shown in the figure to be also in the core, with reactivities corresponding to burnups of 10 and 20 GW · d/t, respectively. We see that as a result of reactivity sharing, the oldest batch burnup is extended well into the negative reactivity range. The figure shows an idealized equilibrium loading scheme in which each identical one-third core batch is replaced after a three-operating-cycle burnup of 30 GW · d/t on a staggered basis.

Fig. 10.2. Fuel batch reactivity sharing.



Fuel Management Terminology

10.22. The terms *nuclear fuel management* and *in-core fuel management* are often used interchangeably. However, it is useful to refer to those decisions that relate to multioperating cycle planning as nuclear fuel management. This might also include strategic decisions associated with pre-reactor fuel operations. For example, planned cycle length is dependent on utility requirements, while feed enrichment and feed batch size are cycle dependent. Thus, there is a necessary coupling between the larger multicycle, and often multigenerating unit picture and the smaller single-cycle design requirements. For our purposes, the term *in-core fuel management* refers to the design decisions for a single burnup cycle, which are concerned primarily with specifying the loading pattern and control strategies within the constraints given by plant design limits and technical specifications.

PRESSURIZED-WATER REACTOR CORE MANAGEMENT

The Initial PWR Core and Subsequent Reload Patterns

10.23. The fuel loading pattern for the first core only for a typical Westinghouse four-cooling-loop plant containing 193 fuel assemblies (§13.9) is shown in Fig. 10.3. By means of this pattern, we can see some of the challenges associated with reload batch design. In this case, fuel assemblies of three different enrichments are used. The low and intermediate enrichment assembles are arranged in a checkerboard pattern in the central portion of the core while the high enrichment assemblies are placed about the periphery. This approach utilizes the reactivity differences between adjacent fuel assembles in the central region to flatten the power density distribution. Since the fresh, highly enriched fuel is placed in the region of lowest neutron importance, where the leakage is highest, its tendency to cause local power peaking is mitigated. However, fast-neutron leakage from peripheral fresh fuel assemblies leads to vessel embrittlement (§10.26).

10.24. The general modified scatter-loading pattern may then be continued for subsequent burnup cycles by discharging after one burnup period the fuel of lowest enrichment and reinserting the remaining fuel in the central region in a modified checkboard pattern in accordance with design considerations to be discussed further in §10.43. Fresh fuel assemblies would be placed about the periphery and after the next burnup period, the fuel having the initial intermediate enrichment would be discharged and the procedure continued in a manner similar to that described above. Such an approach is also known as a modified out–in type of loading. Although in this idealized scheme, the discharged fuel will be that having

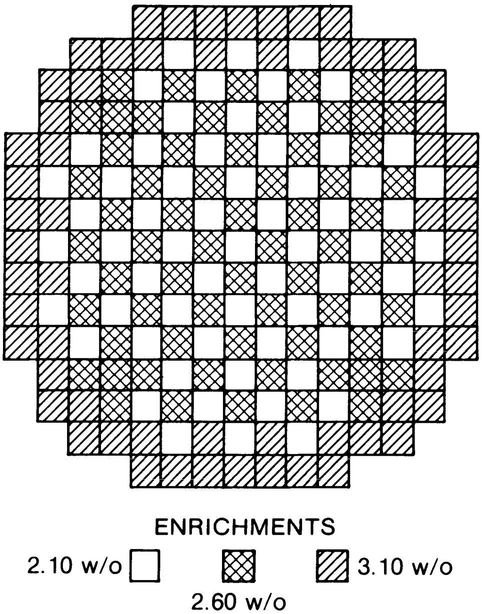


Fig. 10.3. Initial fuel loading pattern (modified scatter loading) for a PWR (Westinghouse Electric Corp.).

maximum burnup, this might not be so in actual practice when various feed enrichments may be used for special purposes.

10.25. After a number of burnup cycles, a pattern such as that shown in Fig. 10.4 for a one-eighth core might evolve. We can see that there have been substantial deviations from the initial core interior checkboard pattern as a result of changes in batch size and feed enrichment in earlier burnup cycles. In the figure which is based on calculations using a two-dimensional nodal model (§10.42), we show "region number," which corresponds to batch number. Since three regions of different enrichment were loaded for the initial burnup cycle, the figure shows the reload pattern for Cycle 7. Also of interest is the beginning-of-cycle (BOC) relative assembly power and the BOC exposure in GW · d/t, where the average cycle burnup has been 10 GW · d/t. We see that there are significant differences in burnup among assemblies of the same region as a result of neutron flux variations during previous cycles. Therefore, the reload core designer searches for an arrangement that will yield a reasonably uniform burnup for assemblies that are eventually discharged from the core while staying within assembly power peaking constraints [5].

LEGEND

XX	REGION NUMBER
Y	POWER PEAKING
ZZ	EXPOSURE (GWD/T)

1 0.42 18.72					
1 0.56 18.72	7 0.80 20.37				
7 0.94 19.93	8 1.18 7.38	7 1.03 22.39			
8 1.17 10.04	7 1.06 20.37	8 1.20 9.64	7 1.02 22.17		
7 1.00 22.17	8 1.16 10.43	7 1.02 21.72	8 1.14 7.38	7 0.79 22.17	
8 1.10 10.26	7 0.96 22.84	8 1.16 7.38	7 0.86 24.86	1 0.65 18.72	8 0.79 11.81
7 0.89 23.96	8 1.05 11.81	7 0.91 24.64	9 1.24 0.00	9 1.11 0.00	9 0.78 0.00
9 1.08 0.00	9 1.04 0.00	9 1.05 0.00	9 0.94 0.00		

Fig. 10.4. One-eighth PWR core loading pattern for cycle 7 [23].

10.26. During recent years, several trends have affected PWR reload core design. As reactor vessels approach the end of their planned 40-year operating life and consideration is being given to extended service, it is desirable to minimize the fast-neutron fluence reaching the interior surface to reduce the increase in the brittle-to-ductile transition temperature (§7.12). This can be accomplished by so-called "low-leakage" fuel loading, in which fresh fuel is loaded in the central region and older fuel about the periphery. A loading that is partially of this nature is shown in Fig. 10.5. In this case, LEGEND

XX	TIMES BURNED
Y	BOC EXPOSURE
ZZ	EOC EXPOSURE

3 25.49 36.37					
1 9.40 21.71	3 29.76 39.83				
2 21.68 32.55	0 0.00 11.77	3 29.76 39.75			
2 21.63 31.32	3 29.41 39.10	1 10.77 22.93	3 26.16 36.82		
3 31.75 41.10	0 0.00 12.36	3 26.70 37.34	0 0.00 12.08	3 31.61 40.90	
1 9.91 21.79	3 26.29 36.90	1 11.37 23.53	2 19.41 30.70	1 11.60 22.46	0 0.00 9.89
3 26.16 36.12	0 0.00 11.93	2 18.48 29.02	0 0.00 10.71	2 22.43 28.55	2 22.89 26.70
0 0.00 9.56	1 11.62 19.63	1 11.08 17.97	2 23.07 27.17		

Fig. 10.5. Partial "low-leakage" PWR loading pattern [23].

extending burnup through a fourth cycle is also an objective. Some twice-burned assemblies are loaded toward the outside. Since the neutron flux is intentionally lower toward the outside, we see that such assemblies experience relatively low burnup during the operating cycle. It is not easy to meet the low-leakage objective since the power peaking introduced by the centrally loaded fresh fuel must be carefully controlled by the use of burnable solid absorbers, which will be discussed shortly. Another advantage of the low-leakage concept is improved neutron economy which tends to lower the feed enrichment and, in turn, fuel cycle costs. Advanced fuel rod designs, such as those using axial blanket pellets of natural uranium oxide, have also been found to improve neutron economy.

10.27. A second trend is to extend the burnup cycle from approximately 12 months to 18 months, a step that yields operating economies by increasing plant availability (the fraction of calendar time that the plant is available for energy generation). However, to provide enough initial reactivity to operate for this longer period, it is necessary to raise the enrichment of the fresh fuel and use burnable solid absorbers to control power peaking.

10.28. Another trend is to extend the fuel assembly total burnup from perhaps three annual burnup periods to four. Fuel utilization is thereby improved and the number of discharged assemblies that must be managed is reduced. Again, in this case, some increase in feed enrichment is needed requiring burnable absorber use for power peaking control. These trends apply only to once-through cycles, as practiced in the United States, not when plutonium recycle is applied, as in Europe.

Burnable Absorber Rods

10.29. The use of burnable absorber rods or lumped burnable poison (LBP) plays an important role in PWR core management. In Westinghouse cores, these may take the form of aluminum oxide–boron carbide rods placed in those positions in a fuel assembly normally left open for the insertion of rod cluster control elements (§5.185). This means that LBP rods may only be used in assemblies not assigned to rod cluster control positions.

10.30. Since the first core initially contains no fission-product poisons, it is necessary to use burnable absorber rods in addition to soluble boron to compensate for the excess reactivity. It is not possible to use additional soluble boron as an alternative since the need to maintain a negative moderator temperature coefficient limits the boron concentration in the coolant. However, since first core use is limited to new reactors, most LBP use is for tailoring the individual assembly reactivity to avoid local power peaking in designing reload core patterns (§10.46). However, it is emphasized that neutrons are lost unproductively in LBP rods. Therefore, only sufficient rods should be used to control the power distribution so that it is within licensing peaking factor limits. These peaking factors are normally determined by loss-of-coolant accident criteria (§12.90).

10.31. During the burnup cycle, burnable poison rods are depleted, fission product poisons are formed, and the fissile atom concentration is reduced. Although this generally results in a loss of fuel assembly reactivity, attention must be given to the relative reaction rates so that the fixed absorber is not depleted so rapidly that an unacceptable power peak would appear in midcycle. Another concern with some absorbers is the residual poisoning effect remaining after the initial burnup cycle. This leads to a reactivity penalty when the fuel assembly is reinserted in subsequent cycles (see Fig. 13.3). Therefore, a general design objective is to minimize the use of lumped burnable poison rods.

PWR Fuel Assembly Design Trends

10.32. Design flexibility has been the most significant trend as operating experience has accumulated during the post decade and assemblies have been available from several vendors. As the use of lumped burnable poisons has increased, more flexible designs have been developed that do not require separate rods in control cluster positions. For example, absorber material can be fabricated with the fuel, either as a coating or integral with the fuel pellet [6].

10.33. Gadolinia bearing rods, which have been used in BWR assemblies for some time, are now being used in PWR assemblies. Such use of gadolinia as a solid solution with the UO_2 pellets permits a variation of absorber concentration as well as a selection of the number and location of absorber rods within the assembly. The capability to shape the power distribution is thereby enhanced. Similar advantages may be obtained by using a thin coating of ZrB_2 as a burnable poison on the surface of selected fuel pellets. Boron is somewhat easier than gadolinium to model neutronically, although it has the disadvantage of not being combinable with uranium oxide. On the other hand, for gadolinium rods, the buildup during exposure of the high neutron capture isotope, gadolinium-157, results in a slightly higher residual poison penalty than for boron-coated rods.

10.34. Increasing the water/fuel ratio in PWR lattice design improves uranium utilization and reduces fuel costs. This could be done by reducing the rod diameter. The resulting slight decrease in the negative moderator temperature coefficient can be compensated for by decreasing the soluble boron and, in turn, increasing the solid absorber in the assembly. Uranium utilization is also improved by using some natural uranium oxide pellets at each end of the fuel rod to serve as an axial blanket. However, some increase in axial power peaking results. Thus, the use of partially enriched axial blankets is another option.

10.35. Hardware improvements are also receiving continued attention. For example, the use of zircaloy-4 instead of Inconel in spacer grids improves neutron economy. Assembly components, such as the upper tie plate, have been redesigned to ease the replacement of damaged rods between burnup cycles, and if desirable, to reconstitute (rebuild with some different rods) the assembly. Another significant PWR improvement is the use of debris filters in the assembly bottom nozzles to prevent foreign particles from entering the core. In the past, such foreign material would be trapped in the rod spacer grids and cause vibration-induced wear (fretting) of the rods.

The Fuel Reload Design Process [7]

10.36. The design procedure for a reload batch of fuel assemblies should accomplish the general objective of achieving minimum energy-generating costs within licensing and utility planning constraints. In practice, the responsibility for the several aspects of the design process may be divided among several groups. For example, in a typical utility, one group may have the primary responsibility for plant scheduling, including refueling outages, and establishing energy generation requirements. Fuel procurement and economic analysis are likely to be the responsibility of a second group, while in-core nuclear analysis and reload assembly design would be accomplished by a third group.

10.37. Initially, a preliminary core reload design is prepared by the third group to meet cycle energy needs that have been specified by the first group and economic guidelines provided by the second group. In this preliminary step, which is likely to involve all three groups, the batch size and enrichments are selected. After agreement, subsequent analyses and design are pursued primarily by the third group. We will concentrate on the activities of this third group.

Levels of Core Modeling

10.38. The modeling in reload core design has normally been carried out at different levels of computational sophistication depending on the accuracy required as well as the dimensional representation and geometric detail that are appropriate. However, as computing capability advances, it is likely that levels can be combined with desirable sophistication applied earlier. In the first level, in which feed enrichment, new assembly number (batch size), older fuel reinsertion, and new cycle length are to be determined, a *multicycle* point reactor model is appropriate. At the next level, a more sophisticated model is needed to develop, by iteration, the detailed loading pattern that is likely to meet licensing requirements. For this purpose, a three-dimensional *nodal* simulator using course-mesh representation (§4.86) is normally used. A number of trial calculations may be needed to assess the power peaking effects of candidate loading patterns. Therefore, some compromise between accuracy and calculational efficiency is appropriate. Core symmetry may be assumed and only a one-fourth or one-eighth core need be considered.

10.39. Finally, to assure compliance with design limits, a detailed fine-mesh model that has been "qualified" by the U.S. Nuclear Regulatory

Commission is used to validate the final reload design. Two-dimensional diffusion theory codes have been used for this purpose for some years, but more recently, comparable results have been obtained by sophisticated nodal codes such as SIMULATE [8], which incorporate three-dimensional fine-mesh pin power reconstruction techniques.

Neutronic Analysis Methods

10.40. A "scoping" analysis is carried out in the first phase of the preliminary design effort to develop long-term strategy considerations and to estimate average characteristics of a fuel batch. For such purposes, a point reactor model that provides a minimum level of geometric representation is usually adequate.

10.41. An example of a modern, fast-running, but sophisticated scoping code is BRACC, which can be run on a microcomputer [9]. Fuel batch reactivity is approximated as a linear function of burnup. Although zero dimensionality (point reactor model) is used, the code accounts for neutron leakage, burnable poison effects, and coupling between assemblies of different batches, by various functions and submodels.

10.42. Methods used in conjunction with reload assembly pattern development are generally based on two- or three-dimensional nodal models. For some years, the Electric Power Research Institute (EPRI) has supported the development of a family of codes to provide a standard core analysis capability for electric utility in-house core management applications. This effort has been known as the Advanced Recycle Methodology Program (ARMP). EPRI-NODE-P was a preliminary design code developed for PWRs. More recently, codes of the SIMULATE family have been developed under the auspices of EPRI for both preliminary and final design [8].

10.43. A two-dimensional version for preliminary design purposes uses the cross-section code CASMO [10] and a four-node per assembly representation in the model. A number of features are provided, including the ability to analyze core response to control rod insertion. The ARMP package includes SIMULATE-E, a three-dimensional version suitable for final design for which the two-dimensional diffusion theory code, PDQ-7 [11], has traditionally been used. Such three-dimensional analysis is also useful for the evaluation of certain accident possibilities as control rod ejection (§12.75).

Reload Core Pattern Design Considerations

10.44. A number of considerations are clear to the experienced reload core designer. Although some of these have previously been mentioned, a summary here may be helpful. The feed batch size, cycle length, and discharge burnups are related by material balance requirements as illustrated in Example 10.1.

Example 10.1. A PWR rated at 3000 MW(th) has a core consisting of 193 assemblies, each containing 450 kg of uranium (as oxide). If a plant capacity factor* of 0.8 is assumed, approximately what fraction of the core must be discharged per year if a burnup of 30 GW · d/t is achieved? Staggered batch loading is used.

$$\text{Energy generated per year} = (3\ \text{GW})(365)(0.8) = 876\ \text{GW} \cdot \text{d}$$

$$\text{Reactor inventory} = (193)(0.45) = 87\ \text{tonnes U}$$

Energy generated per reactor inventory before discharge

$$= (30)(87) = 2610\ \text{GW} \cdot \text{d}$$

$$\text{Core lifetime} = \frac{2610}{876} = 3\ \text{years}$$

If an annual refueling is assumed, one-third of the core would be discharged each year. For a 12-month operating cycle yielding a burnup of 10,000 MW · d/t (864 GJ/kg), should the *discharge* burnup be increased from a nominal 30,000 MW · d/t to 40,000 MW · d/t, the feed batch size would be reduced from one-third to one-fourth of the total core assemblies.

10.45. The feed enrichment must be high enough to provide the reactivity needed for the planned burnup cycle length. However, when designing a loading pattern, careful management of the fresh fuel assemblies is needed to avoid local power peaks. Power peaking may be looked upon as a result of the reactivity contributions of adjacent assemblies. Thus, it is helpful to balance the reactivity of fresh or once-burned fuel in interior or "inboard" positions with neighboring depleted fuel assemblies.

10.46. In practice, compromises are likely to be needed in the adjacent assembly reactivity balancing procedure as a result of limitations on positions available. The use of solid burnable absorbers is then necessary to suppress the local power peaks selectively. However, such use of burnable poisons should be minimized. One approach that provides additional flex-

*The plant capacity factor is the ratio of the actual electrical energy (in kW · h) generated in a given period (usually one year) to the amount that could be produced if the plant operated continuously at its rated power during the whole period.

ibility in locating fresh assemblies is to divide the fresh assemblies into two subbatches, each with a different enrichment. Since the use of a split batch permits assigning a lower enrichment to those assemblies that will eventually be discharged after several operating cycles with lower-than-average exposure for the batch, there is some saving in fuel cycle costs.

10.47. The selection of assembly positions and determination of burnable poison loading are carried out in conjunction with core neutronic modeling calculations to determine the acceptability of candidate patterns. However, an alternative promising approach is to develop an extensive data base of loading patterns which include appropriate specific design parameters. A computer search isolated from neutronics methods can then select a desirable pattern [12].

Automation and Optimization [7]

10.48. Advances in both computing methods and core modeling during recent years have led to automation of many of the iterative steps in the design process. Although the development of optimization procedures has received much attention at universities, the parameters that affect the "best" strategy tend to change over the long planning periods required. Therefore, electric utilities and fuel vendors tend to concentrate on the development of fast and efficient procedures that allow a wide range of loading scenarios to be explored to take advantage of changing economic and regulatory conditions.

The Haling Principle

10.49. In connection with both automation and optimization procedures, the Haling principle has proven to be a useful concept [13]. It was shown that if a power shape (power distribution) could be maintained constant throughout the operating cycle by proper management of soluble poison, solid absorbers, and movable control rods, the peak-to-average power value will be at a minimum. Although in practice it may be difficult to achieve such a goal, which would require matching control rod withdrawal with burnup effects, the concept is useful as a reference.

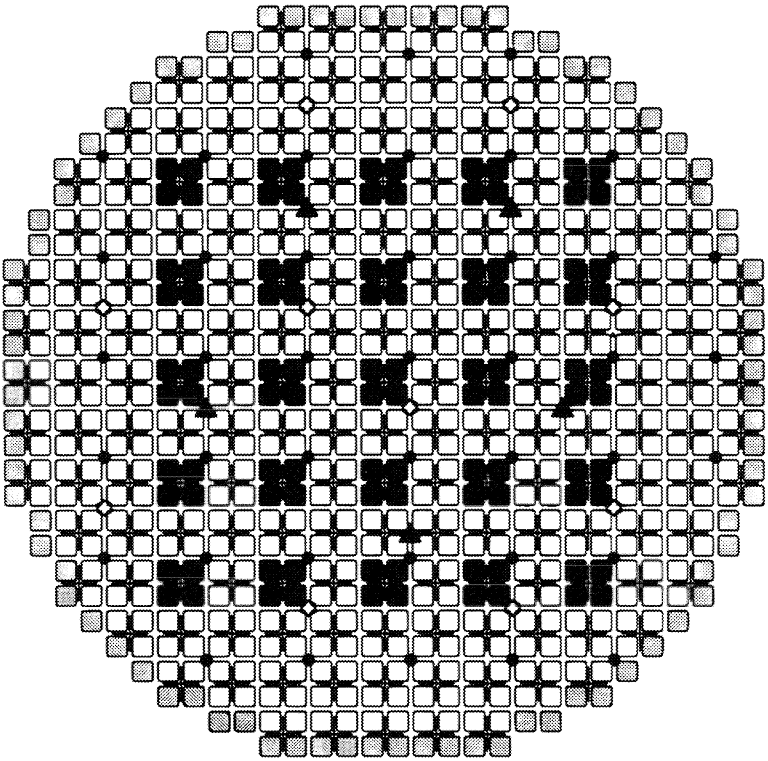
10.50. The Haling principle has been used in some fuel management schemes to separate the multicycle management problem from that of an individual cycle. However, the 1988 revision of regulatory emergency core cooling system modeling methods had the effect of increasing allowable core peaking factors (§12.132). Therefore, there has been a shift in design practice from attempting to achieve minimum core peaking to an emphasis on minimum fuel cycle costs. Such methods do not make use of the Haling principle.

BOILING-WATER REACTOR CORE MANAGEMENT

Introduction

10.51. Although the general principles of core management for BWRs are similar to those for PWRs, the differences in core design have an important effect on the specific approaches used. Hence, a description of the core arrangement from the fuel assembly management viewpoint is helpful. An important feature is the placement of a group of four assemblies with each cruciform control element as shown in Fig. 10.6. Since BWR assemblies are about 40 percent smaller than PWR assemblies and the core

Fig. 10.6. Typical BWR control cell core configuration (General Electric Co.).



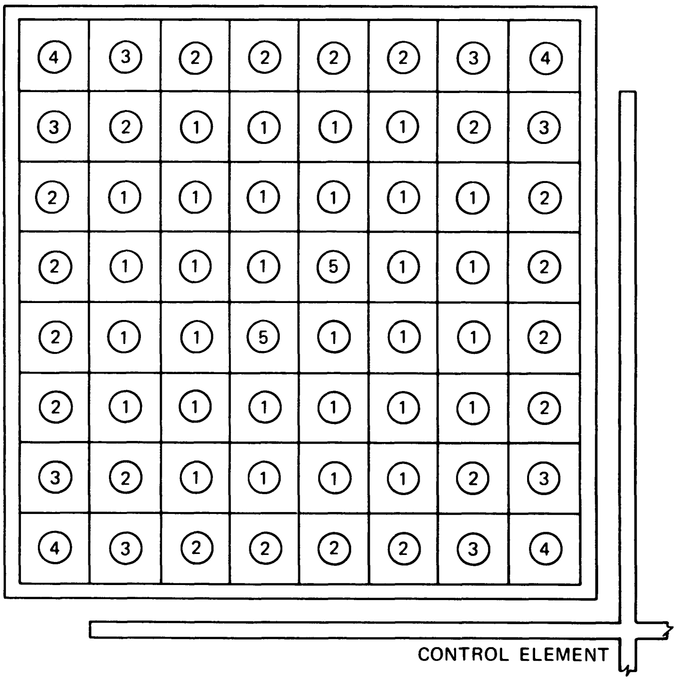
Number of Bundles = 732
Number of Control Blades = 177

– 548 Standard Bundles
– 84 Peripheral Orifice Bundles
– 100 Potential Control Cell Bundles

is larger, there are many more of them that require management, i.e., about 750 compared with about 190 for a PWR.

10.52. In PWRs, all the fuel rods (about 260) in an assembly have the same enrichment, but this is not the case in BWRs in which the assemblies are smaller. In order to compensate for the water gap effect when the control blades are withdrawn, the fuel rods in the corners of each assembly have a low enrichment. In fact, there are four levels of enrichment in each assembly, as indicated in Fig. 10.7. The numbers 1, 2, 3, and 4 represent decreasing proportions of uranium-235 designed to even out the power density distribution. In the initial core, the enrichments range from about 1.7 to 2.1 percent, but in subsequent cores they are approximately from 2.5 to 3 percent. The number 5 shows the locations of two "water rods," which contain water but no fuel. The introduction of extra moderator in this manner increases the neutron flux near the center of the assembly; the result is an increased and more uniform burnup. From two to five of the most highly enriched fuel rods include a few percent of gadolinium oxide (Gd_2O_3) to serve as a burnable poison (§5.197).

Fig. 10.7. Arrangement of fuel rods in a 64-rod bundle in a BWR (General Electric Co.). The numbers 1, 2, 3, and 4 represent decreasing enrichments; the number 5 indicates a rod without fuel ("water rod").



10.53. Soluble boron cannot be used in a BWR. Therefore, shim control is provided by partial insertion of control elements with resulting perturbation of the neutron flux around neighboring assemblies. Such reactivity control elements, in selected patterns, are generally initially inserted deep in the core. Therefore, control management is closely related to fuel management. Since operation of the recirculating pumps can be used to adjust the reactor power within limits (§5.199), some movement by other sets of control elements is available for power shaping. The Haling principle (§10.49) provides a reference for planning.

10.54. BWR loading patterns follow low-leakage strategies, where either blanket or lowest reactivity fuel is located at the core periphery. According to one general strategy, fresh and once-burned fuel would then be mixed in interior positions except for the so-called control cell locations (Fig. 10.6). However, because of multienrichment fuel assembly loadings and the influence of control rod movement strategy, specific practical patterns tend to be complex and will not be represented here. Also, as in the case of PWR management, deviations from a given configuration are likely after a number of burnup cycles.

Control Cell Core

10.55. In the control cell core (CCC) concept, shim control rod movement is limited to a fixed group of control elements that are surrounded by low power fuel. Each of these cruciform elements and its four surrounding fuel assemblies comprise a control cell. The design was developed to reduce power peaking, increase thermal margins, reduce pellet–cladding interaction tendencies (§7.172), and provide operational advantages. Since fresh fuel containing underdepleted burnable absorber is never adjacent to inserted control elements, spatial distortion of burnable poison absorption is avoided.

10.56. With the many assemblies in a BWR core, a reduction in the fuel shuffling is a desirable goal. Therefore, an alternative scheme is to provide backup control cell locations that would serve on the next cycle without moving fuel. The control cell assemblies would then be those of low reactivity but not necessarily from the same batch. It is emphasized that advances in BWR fuel reloading practice are continually being made.

BWR Fuel Assembly Design Trends

10.57. BWR fuel assemblies using an 8×8 rod lattice, as described in Chapter 13, have been standard for some years. Barrier cladding (§7.172) is now generally used to avoid pellet–clad interaction problems. Other lattice options have become available. For example, a 9×9 rod lattice design is currently being used in some cores. Since the assembly overall dimensions are unchanged, the rods are of smaller diameter. One version features an interior water channel that displaces a 3 × 3 rod configuration. The increased neutron moderation in the assembly interior improves fuel utilization, particularly should long operating cycles be desired. Since there are more fueled rods in an assembly, the linear heat rate can be reduced, thus increasing operational flexibility [14]. Other design variations used in Europe also provide for a softer spectrum in the lattice interior. The use of axial natural uranium oxide blankets has also become relatively common.

BWR Spectral Shift Operation

10.58. Boiling-water reactors are operated so that the fuel economy is enhanced by shifting the neutron spectrum to control reactivity. During the first half, or so, of the operating cycle, the coolant flow rate is reduced so that the core coolant void fraction is relatively high. Thus, the core is somewhat undermoderated. The resulting increased resonance region neutron flux results in the increased conversion of uranium-238 to fissile plutonium. Since this undermoderation acts in the direction of reducing the reactivity at the beginning of the operating cycle, when compensation for the reactivity introduced by fresh fuel is needed, the neutrons used for conversion would otherwise be lost by absorption in control poisons.

10.59. As reactor operation proceeds, fuel is consumed and fission product poisons accumulate. The coolant flow is then increased, thus reducing the core coolant void fraction and softening the neutron spectrum. This acts in the direction of increasing the reactivity, as necessary to maintain operation. The fissile atoms formed by the previous conversion process now contribute to energy production.

BWR Core Modeling Methods

10.60. As a result of the boiling process, the axial power shape is much more of an important parameter than is the case for PWRs. Control rod sequencing during burnup also requires attention. Therefore, three-dimensional calculations are necessary, even at the preliminary core design level. In any fuel design method, a key requirement is to evaluate safety margins for a candidate reloaded core. For a BWR, power peaking effects require evaluation for a variety of operating conditions, including various control element insertion scenarios. By contrast, an initial PWR evaluation need be made only at the hot, full-power, control elements out, condition. In many cases the cold shutdown margin is a limiting criterion for a BWR. Therefore, a methodology that would accurately assess a design in terms of all of these requirements at reasonable computing costs is needed.

10.61. Most modeling methods have been developed in-house by reactor or fuel vendors. For example, the General Electric Company, the major BWR vendor, uses a three-dimensional BWR nodal simulator code called PANACEA, which also describes two-phase flow behavior. This serves as the basis for an interactive method for reload and control element pattern design using a coupled mainframe and minicomputer [15].

10.62. Among other nodal codes useful for BWR reload core design, various versions of SIMULATE developed for the Electric Power Research Institute are noteworthy. In Sweden, an integrated system of codes for both static analysis and optimization was developed by the ASEA-ATOM group called CORE MASTER. This system is based on POLCA, a three-dimensional, two-group, nodal model which includes a thermal-hydraulic description [16].

NUCLEAR FUEL UTILIZATION

Introduction

10.63. During the 1970s, concern about the long-term adequacy of nuclear fuel resources led to substantial attention being given to ways of improving the utilization of fissile and fertile materials, independent of economic considerations. Associated with this effort, the matter of proliferation risk, which involved the diversion of fissile material for weapons purposes, was considered. As a result of the slowing of worldwide reactor construction, there is now less concern about resources and more emphasis given to enhancing safety, improving public acceptance, and minimizing investment risks. This change has affected the design objectives for fast reactors and other fuel-efficient systems. Nevertheless, we will briefly review here the principles of conversion and breeding since they may again become important.

10.64. The efficiency with which fuel is being utilized in a reactor may be expressed by the ratio of the number of fissile nuclei (of all kinds) formed to the number destroyed. This ratio is called the *conversion ratio* (CR) when it is less than unity, as it is in thermal reactors based on uranium fuels, and the *breeding ratio* (BR) when it is larger than unity and breeding occurs. There are several different ways in which the breeding ratio is defined, but for the present purpose both the conversion and breeding ratio may be written as

$$\text{CR or BR} = \frac{\text{Number of fissile nuclei produced}}{\text{Number of fissile nuclei destroyed}}, \tag{10.9}$$

where the numbers of fissile nuclei formed and destroyed are usually integrated over an operating period of the reactor between fuel loadings. The fissile nuclei include uranium-235, plutonium-239 and plutonium-241 in a uranium-fueled reactor, and uranium-233 and uranium-235 when the fertile species is thorium-232.

10.65. In principle, the conversion or breeding ratio could be calculated in the same manner as fuel depletion (or burnup), along the lines described in §4.73 et seq., provided the reactor composition and neutron energy spectrum were known as functions of space and time. In practice, such calculations are very difficult, and a number of approximate approaches to the problem have been used. For example, the neutron spectrum may be simplified by considering three energy groups, namely, fast, resonance, and thermal. Furthermore, the local conversion or breeding ratio may be determined by dividing the reactor system into a number of regions of given composition. Allowance for changes with time can then be made as indicated in §4.75. The numbers of fissile nuclei formed and destroyed in each interval are then summed over the operating period to obtain the net conversion or breeding ratio for the given reactor system.

10.66. In the subsequent treatment, no attempt will be made to calculate the overall (or net) conversion or breeding ratio. Some useful semiquantitative results will be obtained, however, by considering the conversion or breeding ratio *at a given time* (or short time interval). For this purpose, equation (10.9) is modified by writing

$$\text{CR or BR (at a given time)} = \frac{\text{Rate of formation of fissile nuclei}}{\text{Rate of destruction of fissile nuclei}}, \qquad (10.10)$$

where the rates of formation and destruction refer to a given time. The ratios calculated from this expression will generally vary with time during reactor operation; this point should be borne in mind in the following discussion. The net breeding or conversion ratio would be represented by integrating both numerator and denominator of equation (10.10) over the operating time of the reactor.

The Conversion Ratio

10.67. The conversion ratio is applicable to thermal reactors in which the fuel is natural or slightly enriched uranium. In deriving an expression for the conversion ratio, the formation (and consumption) of plutonium-241 will be neglected since it is relatively small, at least initially, except in reactors employing recycled plutonium.

10.68. In a thermal reactor, plutonium-239 is produced as a result of the capture by uranium-238 of thermal and resonance neutrons, formed by the slowing down of fast neutrons generated in fission. The rate of production of fission neutrons from a given fissile species in a thermal flux ϕ is $\phi N\sigma_a\eta\varepsilon$, where N is the concentration of fissile nuclei, σ_a is the thermal-neutron absorption cross section, η is the number of fast neutrons produced per neutron absorbed in fissile nuclei, and ε is the fast-fission factor (§3.143).

10.69. Of the fast neutrons produced, a fraction of P_1 reaches the resonance energy region, where P_1 is the nonleakage probability in slowing down into the resonance region. If p is the resonance escape probability (§3.110), the fraction $1 - p$ of the neutrons in the resonance region is captured by uranium-238 to form plutonium-239. Hence,

$$\text{Rate of formation of Pu-239} = \phi\{N^{238}\sigma_c^{238} + [\varepsilon P_1(1 - p)\Sigma(N\sigma_a\eta)]\}, \qquad (10.11)$$

where σ_c^{238} is the capture cross section of uranium-238 for thermal neutrons. The two terms on the right give the rates of production of plutonium-239 from uranium-238 by capture of thermal and resonance neutrons, respectively; the summation in the second term includes both fissile species, uranium-235 and plutonium-239. Soon after reactor startup with uranium (not recycled) fuel, the plutonium-239 concentration is zero; the rate of destruction of fissile nuclei is then $\phi N^{235}\sigma_a^{235}$. The *initial conversion ratio*, as defined by equation (10.10), can thus be represented by

$$\begin{aligned} \text{CR(initial)} &= \frac{N^{238}\sigma_c^{238} + \varepsilon P_1(1 - p)N^{235}\sigma_a^{235}\eta^{235}}{N^{235}\sigma_a^{235}} \\ &= \frac{\sigma_c^{238}}{N^{235}\sigma_a^{235}/N^{238}} + \varepsilon P_1(1 - p)\eta^{235}. \end{aligned} \qquad (10.12)$$

10.70. As the reactor operates and plutonium is generated, fissions (and captures) occur in plutonium-239 as well as in uranium-235; this tends to decrease the conversion ratio, mainly because of the large capture cross section of plutonium-239 for thermal neutrons (Table 2.8). Furthermore, fission products and heavy nuclides, including plutonium-239, compete with uranium-238 for resonance neutrons; this also results in a decrease in the conversion ratio. However, a large value of the initial ratio generally indicates a high conversion efficiency throughout the operating period of the reactor. A large conversion ratio is desirable because the fissile plutonium-239 formed slows the decrease in the overall reactivity as the uranium-235 is consumed. As a result, the fuel burnup is extended.

10.71. It is evident from equation (10.12) that the conversion ratio can be increased by decreasing the resonance escape probability, i.e., by increasing the resonance capture. One way whereby this may be achieved is by hardening the neutron energy spectrum (§2.102) so that the flux in the resonance region is increased. In a water-cooled reactor, a decrease in the moderator-to-fuel ratio, e.g., by closer fuel spacing, causes the neutron spectrum to be hardened. On the other hand, neutron leakage is increased, and fuel of a higher enrichment is needed to maintain the burnup, i.e., N^{235}/N^{238} is increased. These two consequences of spectral hardening have opposite effects on the conversion ratio.

10.72. In commercial water-cooled reactors, the fuel spacing is such that the initial conversion ratio is approximately 0.6. Of the plutonium formed, somewhat more than half undergoes fission during the normal fuel lifetime, and roughly one-sixth is lost by neutron capture. About one-third of the heat produced in a commercial LWR arises from the fission of plutonium-239 and plutonium-241.

The Breeding Ratio

10.73. The conditions under which breeding is feasible will be developed shortly, but for the present it is sufficient to state that a breeder will usually consist of a core, containing both fissile and fertile species, surrounded by a blanket initially containing only the fertile species. Fissile nuclei are then produced from the fertile nuclei in the core and also in the blanket by neutrons which have escaped from the core.

10.74. Bearing the foregoing in mind, the breeding ratio defined by equation (10.10) can be represented by

$$\text{BR (at a given time)} = \frac{\int_{\text{core}} \phi \Sigma_c^{\text{fertile}}\, dV + \int_{\text{blanket}} \phi \Sigma_c^{\text{fertile}}\, dV}{\int_{\text{core}} \phi (\Sigma_f + \Sigma_c)^{\text{fissile}}\, dV}, \qquad (10.13)$$

where Σ_f and Σ_c are the macroscopic fission and capture cross sections, respectively, for the indicated species, and dV is a volume element. (The destruction of fissile nuclei in the blanket is neglected.) By integrating over the core and blanket volumes, the breeding ratio is an average over the whole reactor system. It is to be understood that the neutron spectra are different in the core and blanket systems. As noted earlier, equation (10.13) refers to a particular time, at which the values of ϕ and Σ are applicable.

10.75. It will now be assumed that all the neutrons leaking from the core will eventually be utilized for breeding by capture in the blanket. This assumption is fairly reasonable since some neutrons are formed by fission in the blanket to compensate for neutrons lost in parasitic (nonbreeding) processes and by escape. If the total breeding ratio is divided into two parts, namely, external (or blanket) and internal (or core), it follows that

$$\text{External breeding ratio} = \frac{\text{Core leakage}}{\int_{\text{core}} \phi(\Sigma_f + \Sigma_c)^{\text{fissile}}\, dV}. \tag{10.14}$$

From neutron balance considerations in the core, it is found that

$$\text{Core leakage} = \int_{\text{core}} \phi(\nu\Sigma_f - \Sigma_f - \Sigma_c)^{\text{core}}\, dV,$$

where the first term in parentheses determines the rate of neutron production by fission and the other two represent losses by neutron absorption; ν is the number of neutrons produced per fission (§2.183). In this expression for core leakage, the macroscopic cross sections include the contributions of all the core materials, not merely the fuel. If now the spatial variations are neglected, equation (10.14) reduces to

$$\begin{aligned}\text{External breeding ratio} &\approx \frac{(\nu\Sigma_f - \Sigma_f - \Sigma_c)^{\text{core}}}{(\Sigma_f + \Sigma_c)^{\text{fissile}}} \\ &= \frac{\Sigma_f^{\text{core}}}{\Sigma_f^{\text{fissile}}} \cdot \frac{\nu - (1 + \alpha^*)}{1 + \alpha},\end{aligned}$$

where, by equation (2.52),

$$\alpha^* = \frac{\Sigma_c}{\Sigma_f} \text{ for the core}$$

and

$$\alpha = \frac{\Sigma_c}{\Sigma_f} \text{ for the fissile species.}$$

If α^* and α are assumed to be not very different, it follows that

$$\text{External breeding ratio} \approx \frac{\Sigma_f^{\text{core}}}{\Sigma_f^{\text{fissile}}}(\eta - 1),$$

since η is equal to $\nu/(1 + \alpha)$ by equation (2.57).

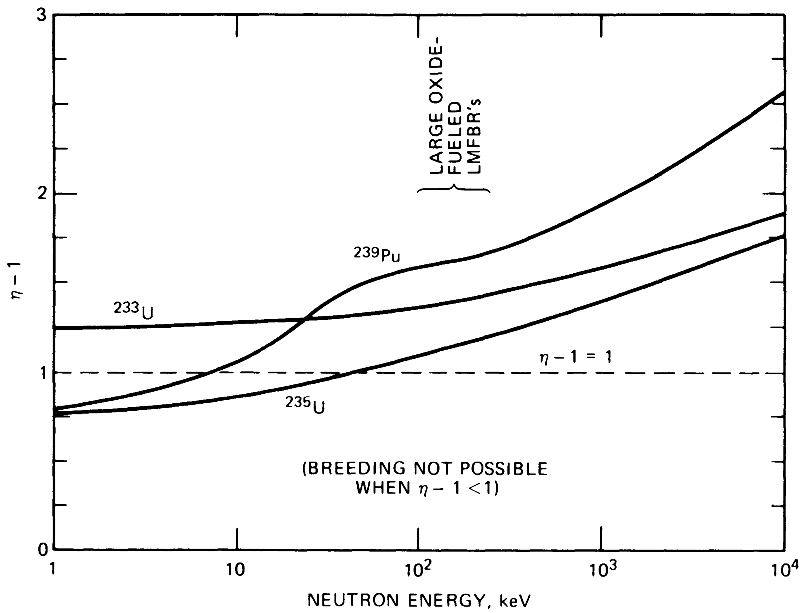
10.76. The fission cross section ratio does not vary significantly with neutron energy; hence, an indication of the effect of neutron energy on the potential for external breeding (or conversion) is given by the corre-

TABLE 10.1. Breeding (or Conversion) Potential

Nuclide	Neutron Energy				
	Thermal	1 to 3000 eV	3 to 10 keV	0.1 to 0.4 MeV	0.4 to 1 MeV
Plutonium-239	1.09	0.75	0.9	1.6	1.9
Uranium-235	1.07	0.75	0.8	1.2	1.3
Uranium-233	1.20	1.25	1.3	1.4	1.5

sponding value of $\eta - 1$. Approximate data for several energy ranges are quoted in Table 10.1, and general trends (omitting fine structure) are shown in Fig. 10.8. It is seen that only in a comparatively fast neutron spectrum (>0.1 MeV) is $\eta - 1$ sufficiently greater than unity to make plutonium-239 breeding practical. Breeding of uranium-233 from thorium-232 should be theoretically possible, although with a smaller efficiency, in a thermal spectrum as well as in a fast spectrum.

Fig. 10.8. Breeding potential as a function of neutron energy. (The curves show general trends but not the fine structure in the resonance regions.)



10.77. The internal (or core) breeding ratio is given by

$$\text{Internal breeding ratio} = \frac{\displaystyle\int_{\text{core}} \phi \Sigma_c^{\text{fertile}}\, dV}{\displaystyle\int_{\text{core}} \phi (\Sigma_f + \Sigma_c)^{\text{fissile}}\, dV} \approx \frac{\Sigma_c^{\text{fertile}}}{\Sigma_f^{\text{fissile}}} \cdot \frac{1}{1 + \alpha}, \tag{10.15}$$

where $1/(1 + \alpha) = \eta/\nu$ by equation (2.57). It is seen from Table 2.9 that the highest values of η/ν arise in a fast-neutron spectrum for all three fissile species. Hence, the internal breeding ratio is largest in such a spectrum. Apart from its contribution to the total breeding ratio, internal breeding has an important bearing on fuel-cycle costs since fissile material produced in the core can be utilized directly without the necessity of going through reprocessing and fabrication stages. It is apparent from equation (10.15) that the internal breeding ratio for a given fertile-fissile nuclide system depends on the core composition, since this determines the macroscopic cross-section ratio. For example, an increase in the ratio of fertile to fissile atoms in the core would lead to an increase in the internal breeding ratio. There is a general decrease in breeding ratio as the core is diluted with inert material because of spectral softening which results in a greater parasitic capture of neutrons in inert diluents.

Thorium Utilization

10.78. Fertile thorium-232, which can be converted to fissile uranium-233 (§1.35), is an important energy resource as an alternative or supplement to natural uranium. Thorium mineral reserves are believed to be of the same order as those of uranium, but there has been little demand for thorium since reactor fuel cycles presently use uranium and are likely to continue to do so for some years. This preference is based primarily on economic considerations for LWRs [25]. However, as uranium supplies are depleted its price will increase, and there will be a greater incentive to utilize thorium instead. An additional incentive for thorium utilization is to reduce the risk of nuclear weapons proliferation (§10.80). Thorium resources appear ample for many years of projected use.

10.79. As shown in Table 10.1, the conversion (or breeding) potential $(\eta - 1)$ for uranium-233 at thermal-neutron energies is significantly higher than for either uranium-235 or plutonium-239. In fact, a system in which the uranium-233, formed from thorium-232 in a low neutron loss reactor, would be recycled has the potential for breeding or self sufficiency. For reactors with poorer neutron economy, fissile atom makeup, proportional in amount to 1 − CR, would be required, but they would still have a favorable resource utilization. With the thorium–uranium-233 cycle, conversion ratios greater than in LWRs are readily achievable.

10.80. Some complications arise in the thorium–uranium-233 cycle that do not occur in the uranium–plutonium-239 cycle. For example, an intermediate stage in the production of uranium-233 from thorium-232 is protactinium-233 with a half-life of 27 days; the latter has relatively large cross sections for neutron absorption in both the thermal and resonance energy ranges. This absorption, which depends on the power density and the neutron spectrum, can have an important effect on the reactivity, although only a minor one on the thorium-to-uranium-233 conversion.

10.81. Another problem results from the presence of uranium-232, formed by $(n, 2n)$ reactions with uranium-233 and with thorium-232 by way of thorium-231 and protactinium-231 and protactinium-232 (see Fig. 11.3). Some of the members of the uranium-232 decay chain are strong gamma-ray emitters, and hence remote handling and shielding are required for fabrication of uranium-233 fuels.

Plutonium Utilization

10.82. Although spent LWR fuel is not reprocessed in the United States to recover the plutonium contained therein, extensive reprocessing is carried out in Europe. Also, the possibility exists that there may be a desire to make use of the energy available from plutonium recovered from dismantled nuclear weapons. Therefore, we will briefly examine a few of the design considerations in using some plutonium dioxide in LWRs.

10.83. Plutonium formed in present LWRs from slightly enriched uranium is fissioned in place during the latter part of the operating cycle. Therefore, one should anticipate no major problems in adding some plutonium to a fresh fuel batch. However, since fabrication costs would be high if the addition would be made uniformly to all rods, most plans call for using mixed uranium—plutonium oxide in selected rods. The large fission cross section of the fissile plutonium isotopes could result in power peaking in such rods. The neutron absorption in plutonium could also affect the reactivity worth of control rods as well as reactivity temperature coefficients. Therefore, we see that the use of plutonium to replace some of the uranium-235 in fresh fuel can affect the core safety characteristics and must be accounted for in the design. The use of recycled plutonium was extensively studied during the 1970s and significant literature developed [17]. Currently, recycled plutonium is being used routinely in French PWR fuels [24].

Proliferation Risk

10.84. It is assumed that the diversion of weapons-usable material by isotopic separation would be difficult, whereas chemical separation of plutonium from uranium (or uranium from thorium) would be relatively simple in comparison, provided the radioactivity level is low. On the other hand, chemical separation from highly radioactive spent fuel is considered quite difficult because of the need for remotely controlled equipment. Hence, spent LWR fuel, with its high radioactivity and low fissile content (about 1.5 percent), is regarded as presenting little proliferation risk. Similarly, should fresh fuel be radioactive from some remaining recycled fission (or other) products, separation would be difficult, with the level of difficulty related to the gamma activity. Another approach to reducing the accessibility of weapons-usable material would be to confine to special security areas, perhaps under international control, fuel-cycle operations in which such materials are produced.

10.85. Since one of the products of the Purex process (§11.74 et seq.) consists largely of fissile plutonium isotopes, the decision was made in 1977 to defer commercial reprocessing of spent fuel in the United States. Furthermore, concern over the plutonium produced by fast breeder reactors led to a de-emphasis of efforts to commercialize such reactors. The unavailability of plutonium that might have been recovered from spent fuel for reuse in LWRs and from fast breeder reactors would limit the energy that could be obtained from natural uranium. Consequently, attention has been paid to alternative fuel cycles which would improve resource utilization and also have an acceptable proliferation risk.

NUCLEAR ENERGY COSTS [18]

Introduction

10.86. The energy derived from the fuel in a nuclear reactor, which is generally sold in the form of electricity, represents a commercial product. The cost of generating this product is made up of several components, one of which is the cost of the fuel. In order to relate fuel costs to other costs, a brief outline will be given of the various contributions to nuclear energy costs. It is customary to apportion the costs of generating electricity, from either nuclear or fossil fuels, to three categories: capital (or construction) costs, expenditures for operation and maintenance, and fuel costs. These will be considered in turn. However, to provide desirable background, we will first review the engineering economics concept of the time value of money.

Time Value of Money [19]

10.87. The time value of money in any economic analysis can be most easily expressed in terms of interest. This is true even for equity funds where a "rate of return" is really meant. In the broad sense, therefore, interest can be considered money paid for the use of other money, either borrowed or part of the equity of a company. The interest concept must be applied whenever time is a variable whether or not interest is actually paid. This concept can be applied in various ways. For example, capital investments in equipment may not be made at the same time when the equipment is put into service or when "revenue" is produced as a result of the equipment purchased. Therefore interest charges must be added to the investment cost.

Interest relations and terms

10.88. The concept of interest is very simple; it is the application of a fixed rate applied to a "principal" sum of money over a period of time, but a number of equations and special terms are useful, particularly in dealing with compound interest. If a sum of money, P, is invested at an interest rate, i, for a year, the interest earned at the end of the period is Pi. In compounding, this amount is added to the principal so that the sum invested for the second year is now $P + iP = S_1$ and the new interest earned at the end of the second year is $(P + iP)i$, or more systematically,

$$\begin{aligned} S_0 &= P \\ S_1 &= P + iP = P(1 + i) \\ S_2 &= P(1 + i) + Pi(1 + i) \\ S_2 &= P(1 + i)^2. \end{aligned} \tag{10.16}$$

At the end of n years, therefore, the total sum accumulated is $S_n = P(1 + i)^n$.

10.89. The quantity $(1 + i)^n$ is known as the *single-payment compound amount factor*. The reciprocal, $1/(1 + i)^n$, the *single present-worth factor*, is useful in finding a principal, P, that will give a required total amount, S, in n years:

$$P = S\left[\frac{1}{(1 + i)^n}\right]. \tag{10.17}$$

P may then be considered as the *present worth* of the total, S, anticipated after n years.

10.90. Another type of accumulation results from the investment of a fixed sum, R, at the end of each year for n years. The total is the summation of the individual subtotals compounded over the years applicable for each payment.

$$S = R + R(1 + i) + R(1 + i)^2 \cdots R(1 + i)^{n-1}, \qquad (10.18)$$

where the amount $R(1 + i)^{n-1}$ is the accumulation from the payment made at the end of the first year and the first term, R, is the final payment, which earns no interest.

Multiplication by $(1 + i)$ and then subtraction of equation (10.18) yields

$$iS = R[(1 + i)^n - 1]$$

or

$$S = R\left[\frac{(1 + i)^n - 1}{i}\right]. \qquad (10.19)$$

This expression is known as the *uniform-annual-series compound factor*. Its reciprocal, $i/[(1 + i)^n - 1]$, is called the *sinking-fund-deposit factor*. This factor is used to determine the regular payment required to produce a desired amount at the end of a given period of time:

$$R = S\left[\frac{i}{(1 + i)^n - 1}\right]. \qquad (10.20)$$

The sinking-fund concept is used in one type of depreciation accounting to provide for the replacement of an asset at the end of its useful life.

10.91. A variation of this principle is the determination of the uniform end-of-the-year payment needed to repay a debt. From the viewpoint of the lender, this is the same as making a single present investment, P, which is returned, with interest, as a series of end-of-the-year payments. The payment scheme is the same as that for the sinking-fund deposit except that the lump-sum payment at the beginning is related to the alternate accumulation, S, by

$$S = P(1 + i)^n.$$

Then substituting in equation (10.20) yields

$$R = S\left[\frac{i}{(1 + i)^n - 1}\right] = P(1 + i)^n\left[\frac{i}{(1 + i)^n - 1}\right] = P\left[\frac{i(1 + i)^n}{(1 + i)^n - 1}\right]$$

(10.21)

The final expression in brackets in equation (10.21) is known as the *capital-recovery factor*. It is equal to the sinking-fund factor plus the interest rate. The reciprocal, which would be useful if P were expressed in terms of R, is known as the *uniform-series present-worth factor*.

Present-worth concept

10.92. In the evaluation of alternate engineering projects involving the expenditure of funds, incurring of costs, and receipt of revenue, all at different times, a systematic treatment of the effect of the time variable of money is useful. The value of money can be considered to change as it is moved through time. In the present, for example, money has a greater value than it would have at some time in the future because it can be put to a useful purpose in the interim. Some of the terms discussed in the previous section are useful in design comparisons involving expenditures.

10.93. The *present-worth concept*, for example, provides for the shifting of money from one time level to another with a corresponding shift in value. If r is the effective earning rate or interest rate, then the present worth of 1 dollar *due* 1 year in the future is $1/(1 + r)$. This corresponds to shifting backward in time. Similarly, the present worth of 1 dollar invested a year previously would be $(1 + r)$, corresponding to a shift forward. In electric utility economics, it is useful to know the present worth of revenue requirements for future years of plant operation. This corresponds to the case of the backward shift. In the case of simple interest (single payment), the present value, P, can be expressed as

$$P = \frac{1}{(1 + r)^n},$$

where n is the number of years involved.

Capital Costs

10.94. The capital costs (or charges) are those associated with the construction of the plant. For engineering economic estimates, as opposed to corporate accounting, it is usually adequate to consider charges on an annual basis during the plant lifetime. These charges are primarily the cost of the required investment needed to construct the plant, allowance for depreciation, and some taxes. A charge for decommissioning may be included in the depreciation allowance or carried as a separate contribution to the generating cost. We have followed the latter approach in Table 10.2.

10.95. Construction costs consist of direct and indirect charges. The direct costs are those for equipment, materials, and labor required to build the plant. An allowance for contingencies is often included with the direct costs. Indirect costs are essentially all other costs incurred in building and testing the plant, so that it can be turned over by the manufacturer to an electric utility ready for continuous operation at its rated capacity.

10.96. Direct construction costs include costs of such items as land, structure and site facilities, reactor plant equipment (exclusive of fuel), turbine plant (generating) equipment, electric (power transmission) plant equipment, miscellaneous equipment, and contingency allowance. (The latter is sometimes included with the indirect costs.) The direct costs also include the cost of materials and of labor; in some categories, e.g., structures and site facilities and electric plant equipment, labor costs constitute more than half the total.

10.97. Indirect construction costs cover several broad areas. One is professional services for engineering, design, licensing activities (see Chapter 12), and supervision of construction. A second area includes temporary facilities, equipment, and services during construction, the plant owner's general and administrative expenses, and reactor operator training and plant startup. A major indirect cost is the interest on funds expended during the design and construction of the plant. The interest charge on each type of expenditure is based on the amount spent, on the elapsed time between the expenditure and the conclusion of the construction period, and the interest rate. The construction period, i.e., from the award of initial contracts for reactor plant construction to the time when the plant is ready for commercial operation, may be as long as ten years or more. But the first few years are devoted primarily to design and licensing efforts, and expenditures are relatively small. Subsequently, after a construction permit is issued by the U.S. Nuclear Regulatory Commission, expenditures for equipment and labor rise markedly. Hence, most of the interest charges are incurred during the later stages of the construction period.

10.98. An escalation cost, which is applicable to both direct and indirect construction costs, arises from increases in cost due to inflation between the time the costs are first estimated and the time a firm purchasing commitment is made. The escalation allowance depends on the estimated cost of the item, the time when the item is purchased after the estimate is made, and an annual percentage to reflect the probable consequences of inflation. Since the inflation rate varies from year to year and is difficult to predict, there is a large uncertainty associated with the escalation allowance.

10.99. Now that we have determined the total construction cost, we need to develop an annual fixed-charge contribution to the generating cost that is based on it. After the plant is placed in full-power operation, dividends on stocks and interest on bonds must be paid each year. Such stocks and bonds were issued to obtain the funds to build the plant. As mentioned in §10.87, other annual "fixed" charges include some taxes and an allowance for depreciation. Although depreciation of present plants is based on an estimated 40-year lifetime, it should be noted that advanced plants are designed for a 60-year lifetime (Chapter 15). All of these fixed annual charges may be *estimated* by applying a percentage rate to the total construction cost. Since this percentage rate depends on the interest rates prevailing during the years of construction and other parameters, any value cited here is likely to become dated quickly. However, an annual rate of the order of 14 percent for an investor-owned utility may be used for orientation purposes.

Operation and Maintenance Costs

10.100. Operation and maintenance charges include the payroll for the plant, clerical, engineering, and administrative staffs, and the costs of maintenance materials and supplies, including chemicals for the treatment of reactor and condenser water and radioactive wastes produced in the reactor installation. In the same category are the costs of such miscellaneous items as liability insurance, public relations, training of new workers, travel, etc. In general, however, operation and maintenance costs contribute a relatively small proportion to the overall energy costs.

Fuel Costs

10.101. The exact accounting of fuel costs is complex, but it is helpful to consider each batch of fuel in the reactor core as representing the end product of several manufacturing operations. The fuel, therefore, has a certain value determined by the cost of the materials and the operations. This fuel "investment" would be reduced should the spent fuel have any salvage value, e.g., for its plutonium content, after allowing for the cost of the operations necessary to recover useful materials and the charges for waste management (§11.44 et seq.).

10.102. The fuel within the reactor can be considered as an investment; hence, it is necessary to apply carrying charges to its value while it is in so-called commercial use, i.e., while it is used to generate electricity which produces revenue to cover all the plant costs. These carrying charges are an added contribution to the fuel costs. Carrying charges are also applicable to the time spent on the various operations performed on the fuel prior to its use in the reactor.

10.103. Rather than express fuel costs on an annual basis, as is done for fixed costs, it is more meaningful to apportion the cost of each fuel batch among the kilowatt-hours of electricity generated while the batch is in use. Major contributions to the cost of a batch are the cost of the amount of uranium to be fed to the enrichment plant, the cost of separative work (§10.11), fuel fabrication, and waste management. Changes in each of these contributions, such as a shift in the market price for uranium ore, would have a corresponding effect on the batch cost. Similarly, changes in the fixed charge rate would affect carrying charges for the fuel in and out of the reactor. However, nonreactor carrying charges for each batch are normally included as part of the cost of the operations involved. If desired, the fuel batch cost may be expressed on an annual basis by determining the batches used per year.

10.104. Since each reload batch represents a certain investment, it is desirable to obtain as much energy from the assemblies in a given batch as possible by "suitable" in-core fuel management. However, if the fresh fuel enrichment is increased to extend the burnup, power peaking and vessel neutron exposure design constraints require consideration. Therefore, both economic and engineering parameters are important in developing a suitable strategy. It is true that capital costs tend to be larger than fuel costs. Hence, generating costs on a unit energy basis are not very sensitive to fuel cost savings. On the other hand, on an annual generating cost basis, capital costs, by their nature, are fixed, whereas fuel costs are not. Since fuel costs may amount to on the order of $80 million per year per typical large generating unit, there is a significant dollar incentive for optimum fuel management.

Electric Power Generation Costs

10.105. The annual cost of operating a nuclear power plant is obtained by summing the contributions from the three main categories considered: capital (or plant investment), operation and maintenance, and fuel. Table 10.2 lists some values that are given here primarily for orientation and to indicate relative contributions. A construction cost of $2000/kW(el) has been assumed based on the $1500/kW(el) goal for advanced reactors (§15.22), with $500/kW(el) added for conservatism. Since these contributions have

TABLE 10.2. Generating Cost, Relative Contributions: 1000 MW(el) Pressurized Water Reactor (80 Percent Plant Factor)

Cost Item	Annual Cost ($ millions)	Unit Cost (mills/kWh)
Fixed charges (total construction cost, $2 billion, 14% fixed charge rate assumed)	280	40
Fuel cycle	70	10
Operation and maintenance	50	7
Allowance for decommissioning	5	0.7
Total	405	58

changed significantly in the past from year to year, the current literature should be consulted.

10.106. We see that capital costs constitute the major contribution to the annual cost of generating electricity. Therefore, such fixed-cost parameters as construction time and the cost of money greatly affect the cost of the generated electricity over the life of the plant. Furthermore, once the plant is built, the annual fixed-cost charge cannot be reduced by operating economies. However, the plant capacity factor, the fraction of rated capacity experienced (§10.44), has an important bearing on the fixed costs *per unit of energy produced*. Therefore, there is a major incentive to minimize shutdown periods. Although fuel costs make a smaller contribution, they lend themselves to savings by efficient fuel management during the plant lifetime. Although less significant on a relative basis, operation and maintenance costs depend on efficient plant management.

Role of Rate Regulation

10.107. In considering the various factors affecting nuclear energy costs, the role of rate regulation deserves mention. Investor-owned public utilities are regulated by state public utility commissions since they have a franchise to serve a specific territory. Their responsibility is to establish rates that provide the utility with a fair rate of return on the capital invested. However, when the construction cost of new nuclear power plants escalated significantly during the late 1970s and early 1980s, opposition developed to burdening customers with corresponding major rate increases. These disputes were resolved in various ways. At any rate, as a result of the need to insulate utility rates from potentially high costs for new nuclear power plants, the Public Utilities Regulatory Policy Act was enacted in 1978. This provides for a modified free-market energy economy under control of the Federal Energy Regulatory Commission (FERC). The National Energy Policy Act (NEPA) of 1992 provided further deregulation, particularly in the opening of transmission grids for the transport of electricity so that competitive marketing is now possible. Also, now permitted are non-rate-based independent power producers who can take advantage of this new access to markets.

10.108. We can anticipate some restructuring of the electric utility industry in the United States with other than traditional forms of ownership being permitted. For example, independent power producers, not subject to rate structure–based regulation, might market energy on a competitive "wholesale" basis to traditional electric utilities, which would then, in turn, distribute the energy to their customers [20]. The role of nuclear power production under this marketing arrangement is likely to depend primarily on its economic competitiveness.

NUCLEAR MATERIAL SAFEGUARDS [21]

10.109. An extensive program of safeguards has been established in the United States to prevent the theft of "special nuclear material" defined as plutonium, uranium-233, or uranium enriched in uranium-235, and particularly of "strategic" special nuclear material, consisting of plutonium, uranium-233, or uranium enriched to more than 20 percent in an amount equivalent to more than 5 kg of uranium-235. The importance of the latter lies in the fact that it could be used for the illicit production of a nuclear explosive. The safeguard regulations apply to all stages of the fuel cycle, including fuel fabrication, power generation, fuel reprocessing, storage, and transportation. Broadly speaking, however, two situations are considered: material at fixed sites and material during transportation.

10.110. At fixed sites, special nuclear material must be held within a "protected area" with multiple physical barriers. Access to the area is controlled and unauthorized entry is prevented by monitoring systems with alarms. Continuously operating communication systems provide contact with local law-enforcement authorities.

10.111. During transportation, special nuclear material is probably more vulnerable to diversion than at any other time. All shipments, which may be by road or rail, must be accompanied by armed guards, and regular communication must be maintained with the shipper of the material. Transportation of special nuclear material could be minimized if nuclear reactors, fuel fabrication plants, and spent-fuel reprocessing facilities were in close proximity. However, single fuel fabrication and reprocessing plants would meet the requirements of more reactors than are likely to be constructed in a particular area. Consequently, even under the best conditions, some transportation of special nuclear material could not be avoided.

10.112. Additional protection against the diversion of special nuclear material is provided by a comprehensive program of internal material control and accounting. This program is designed to detect any loss and to take timely action should such a loss occur. In the United States, the Nuclear Regulatory Commission operates an inspection and enforcement program which includes supervision of material control and accounting. Although the overall costs of the safeguards program are substantial, they are expected to increase the cost of nuclear power by no more than a few percent.

10.113. The International Atomic Energy Agency (IAEA) conducts a broad safeguards program to enforce adherence to the Treaty of the Nonproliferation of Nuclear Weapons (NPT) which has been signed by over 100 nations. An inspection program features accountancy, containment, and surveillance. Material accountancy utilizes various sophisticated instruments to assay the amount of fissile material present to supplement ordinary "bookkeeping." "Containment" includes various measures such as locks and seals to assure that material has not been tampered with between inspections. By "surveillance" is meant an inspection program to keep track of material flows.

NUCLEAR CRITICALITY SAFETY [22]

Introduction

10.114. Whenever fissile material is handled, be it in the manufacture of fuel, loading of fuel assemblies, or storage of spent assemblies, nuclear criticality safety deserves attention. This topic normally refers to out-of-core operations. In-core reactivity safety considerations will be covered in Chapter 12.

10.115. All of the factors that influence the neutron balance affect criticality safety when fissile isotopes are being handled. The important considerations are the mass of the fissile isotopes, geometry, moderation, and reflection.

Design Approaches and Analysis

10.116. In designing a specific operation, attention is given to the above considerations. For example, a high-neutron-leakage geometry can be provided. Neutron absorbers, such as cadmium or boron, either in solution or in a geometric array, are useful. Accumulations of fissile material that might cause problems can be avoided by careful administrative control.

10.117. Design and analysis go hand-in-hand. A prediction of criticality is necessary to confirm the adequacy of design options. This is normally done by machine calculations using suitable codes supplemented by data from critical experiments. Since a nuclear criticality safety calculation should predict a noncritical condition with high reliability, conservative values of constants and inputs should be selected when uncertainty exists. However, since safety margins are provided in the design, the calculation methods can be relatively unsophisticated, provided that the confidence level of the results is known.

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PROBLEMS

1. A PWR fuel batch consists of 33,000 kg of UO_2 enriched to an assay of 3.2 percent uranium-235. (*a*) Determine the equivalent amount of "normal" (natural) uranium for a tails assay of 0.2, 0.25, and 0.30 percent uranium-235, respectively. (*b*) What is the separative work requirement for each of the cases above?
2. A PWR uses a three-batch fuel loading pattern, with the fresh fuel having an enrichment of 3.3 percent ^{235}U. Assuming that the average conversion factor is 0.6, make a crude calculation to estimate the isotopic consumption of the fuel at burnups of 0.86, 1.73, and 2.6 TJ/kgU.
3. Assuming a fuel enrichment of 3.3 percent, a burnup of 2.6 TJ/kg U, and a LWR conversion factor of 0.6, what fraction of the uranium in the original ore is converted to energy through fission? The isotope separation tails assay is 0.002 ^{235}U.
4. A PWR with a fuel loading of 115×10^3 kg of UO_2 has the following core arrangement:

Region	Enrichment	Assemblies
1	2.1%	65
2	2.6%	68
3	3.1%	60

   (*a*) If the projected burnup is 33,000 MW · d/t and the reactor power is 3800 MW(th), what is the core lifetime of the batch in Region 3? (*b*) At discharge, what is the ratio of total fissions to the original number of fissile atoms? Explain.
5. A PWR uses a modified scatter-loaded fuel management scheme that requires annual reloading of one-third of the assemblies having an enrichment of 3.0 percent uranium-235. (*a*) Describe the guidelines to be followed for the placement of retained and fresh assemblies in design of the reload core. (*b*) What are the requirements for low-leakage management?
6. (*a*) The total construction costs for a 1000-MW(el) PWR completed in the 1980s was $2 billion. If fuel costs are $40 million per year and operation and maintenance costs are $30 million per year, estimate the generating costs in dollars per kWh. Assume a capacity factor of 70 percent and a fixed-charge rate of 14 percent per year. (*b*) The plant above required a construction period of 12 years with an IDC (interest during construction) rate of 12 percent per year. Estimate the generating cost for a plant of similar design if the construction period could be reduced to 6 years and the IDC rate was 10 percent per year. Neglect escalation.

CHAPTER 11

Environmental Effects of Nuclear Power and Waste Management

INTRODUCTION

Environmental Concerns [1]

11.1. During recent years, an assessment of environmental effects has been a necessary part of every major construction activity in the United States. In many cases, the achievement of a balance between the benefits of the activity and the impact on the environment has resulted in controversy. Therefore, before considering nuclear power effects, it is helpful to examine what is meant by environmental "contamination" and the effects of fossil-fueled energy sources.

11.2. Although the terms *contamination* and *pollution* are synonymous in many dictionaries, there is a distinction from the environmental viewpoint. By contamination, we merely mean the introduction of a foreign or "unnatural" substance, while pollution is a stronger term describing a level of contamination that is harmful, such as smog from automobile exhaust systems. However, the picture becomes complicated when we consider the introduction of natural substances, which can be harmful. For example, volcanic eruptions have released large quantities of particulate matter and radioactivity into the atmosphere. Natural springs sometimes contain unpleasant and harmful sulfur compounds. Natural background radiation levels vary from place to place as a result of the influence of cosmic radiation at higher altitudes and the proximity to minerals such as granite containing radioisotopes. Therefore, one must view the introduction of foreign substances into the environment as undesirable only if harmful effects result compared with the impact of "everyday" natural events.

11.3. Environmental considerations apply to many sectors of our industrialized society, particularly where materials are processed. Effluents from metal smelters, chemical plants, and petroleum refineries must be controlled, for example. The reduction of pollutants from motor vehicles is a continuing challenge. Similarly, energy production is a vital component of this society. Nuclear and fossil fuels are our primary energy resources. Therefore, it is only fair to view the environmental impact of nuclear power plants in comparison with the impact of the other option, fossil-fueled plants.

Emissions from Fossil-Fueled Power Plants [1]

11.4. The most important contaminants from fossil-fueled power plant operation are carbon dioxide, sulfur dioxide, nitrogen oxides, and particulate matter. Combustion processes contribute about 10 percent of the carbon dioxide emitted to the atmosphere, the remainder being primarily the result of natural decay. The carbon dioxide concentration in the atmosphere is the result of the balance between generation and removal processes such as photosynthesis. However, as will be discussed in the next section, increasing carbon dioxide concentration levels could affect the earth's climate as a result of the *greenhouse effect.*

11.5. Power plants contribute the major proportion of sulfur dioxide emitted to the atmosphere, but industrial sources can be very significant in some areas. Local concentrations vary widely and tend to be highest in urban industrial regions. High concentrations on the order of 10 parts per million (ppm) can cause breathing problems.

11.6. Only about 4 percent of the total emission of nitrogen oxides is caused by power plants. However, when carried along by wind currents, sulfur compounds and nitrogen oxides combine with the water vapor in the atmosphere to form sulfuric and nitric acids as aerosols which precipitate back to earth with rain or snow long distances from the point of origin. This phenomenon, known as *acid rain*, has become a serious environmental problem. Therefore, flue-gas desulfurization systems are now required for new fossil-fueled power plants to minimize sulfur emission. These systems are expensive, generate a great deal of solid waste requiring disposal, and require significant maintenance.

The Greenhouse Effect [2]

11.7. The atmospheric carbon dioxide concentration has been increasing at a greater rate during recent years than during the middle of this century, raising concerns that the balance between generation and removal rates may now be upset so that more than half of the carbon dioxide produced may be retained. In addition, some other gases, such as Freon, which result from human activity, tend to "block the atmospheric window," similar to carbon dioxide.

11.8. Energy from the sun is primarily in the shortwave and visible portions of the spectrum, only a small fraction of which is absorbed or scattered back to space by the atmosphere. However, heat reradiated from the earth's surface is mostly in the infrared portion of the spectrum, in which carbon dioxide, methane, nitrous oxide, ozone, Freon, and water have some absorption bands. Thus, the various complicated rate processes that establish a thermal equilibrium may tend to shift to a position favoring a warmer earth surface temperature. On the other hand, an increase in particulate matter in the atmosphere favors a cooling trend as a result of greater reflection of the incident radiation back into space. Green plants absorb carbon dioxide by photosynthesis but discharge it by respiration. Although modeling of these rate processes on a global scale is very difficult, it does appear that warm climatic periods favor carbon dioxide emission, thus introducing a positive feedback effect.

11.9. Since warming predictions regarding the rate of possible long-term global warming depend on the sophisticated modeling of many world-wide climatic and other processes, the results remain somewhat uncertain. However, a conservative approach would be to accept the possibility and to limit carbon dioxide emissions as much as possible and discourage further reductions of green plant growth areas.

Overview of Nuclear Power Effects

11.10. The environmental effects associated with the generation of nuclear power are the result of a number of processes and operations. First, we have the various steps, which start with mining the ore required to manufacture the fuel assemblies loaded into the reactor (§10.4 et seq.). Although there are low-level radioactive wastes associated with milling and fuel manufacturing operations, their treatment is relatively routine and has not attracted public concern.

11.11. Nuclear power plants during normal operation release various gaseous and liquid effluents that affect the environment. The extent of the impact is determined by the transport of radionuclides through so-called "pathways" from their point of release to where people would be affected.

Releases must be "as low as is reasonably achievable" as a licensing requirement (§6.63). We will treat this topic in the next section but will defer a discussion of plant waste treatment systems and the management of low-level wastes until later in this chapter. The impact of accidents on the environment is treated in Chapter 12.

11.12. The management of spent fuel and high-level radioactive wastes has attracted considerable controversy. This is the third category to be discussed. Here again, the pathway concept is useful since harm is done only if the radioisotopes of concern manage to migrate to people in amounts above permissible levels.

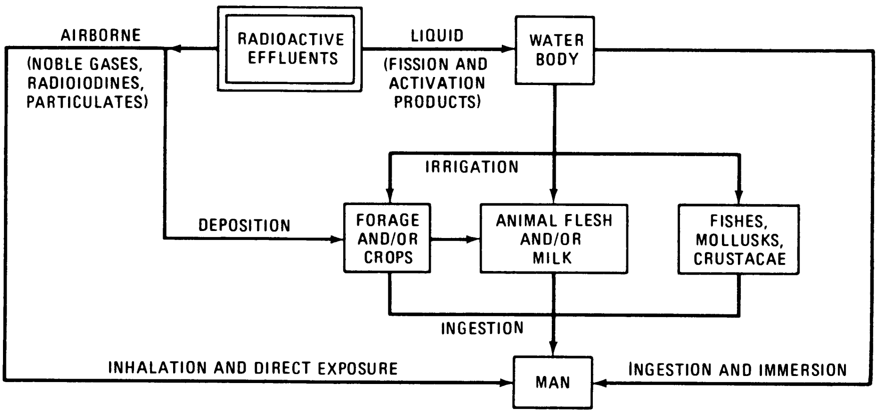
RADIATION EXPOSURE PATHWAYS [3]

Introduction

11.13. In Chapter 6 we considered the biological effects of radiation and exposure limits. However, the environmental impact of a nuclear power plant depends on how the sources of radiation exposure, i.e., the decaying isotopes, are transported to people. Similarly, in considering the long-term storage of radioactive wastes, we must evaluate the probability of some of the stored isotopes migrating from the storage site and leading to harmful exposure.

11.14. By *radiation exposure pathways* is meant the various ways in which sources of radiation are transported to cause exposure. As indicated in Fig. 11.1, the transport pathways from radioactive effluents can be fairly

Fig. 11.1. Pathways of radiation exposure to man from nuclear facility effluents.



complex when the food chain is considered. Should leaking waste containers be the pathway source, the usual concern is transport through groundwater and then to sources of potable water or through the food chain to people. However, the pathway term is usually applied to radionuclides from reactor effluents rather than those from waste containers.

Regulatory Bases for Exposure Pathways

11.15. Various regulations limit allowable exposure from reactor effluents and from waste packages. For example, in 10 CFR 50, Appendix I, to meet ALARA program (§6.63) objectives, the annual above-background dose or dose commitment from liquid effluents in unrestricted areas from *all pathways of exposure* is limited to 3 mrem to the total body or 10 mrem to any organ. The gaseous effluent total body annual dose limit is 5 mrem. Radioactive iodine is particularly important with the annual dose or dose commitment from it or particular deposits limited to 15 mrem to any organ of a person in an unrestricted areas. Note that ALARA *guidelines* are more restructive than exposure *standards*, as given in §6.67.

11.16. Radioactive exposure regulations also limit the concentration of radionuclides in gaseous and liquid effluents at the outer boundary of the plant site, that is, the limit of public access. These are listed in 10 CFR 20, Appendix B, and may be considered as the *maximum permissible concentrations* (MPCs). They take into account the manner of entry of each radionuclide into the body, the biological and radioactive half-lives, and the dose effects. The concentrations have been calculated to be consistent with the dose limit of 0.1 rem/yr for members of the public that was established in 1991.

11.17. Rules for controlling radioactive materials in effluents are established both by the NRC and the Environmental Protection Agency (EPA). EPA regulations are found in Title 40 of CFR. For example, 40 CFR 190 limits the dose from the uranium fuel cycle to the public to 25 mrem/yr. Licensing requirements for land disposal of radioactive wastes are given in 10 CFR 61. However, it should be recognized that such values change with time. Therefore, current references should be consulted.

Radiation Exposure Pathways

11.18. The applicant for a permit to construct (and later to operate) a nuclear power plant must include with his application calculations showing that the radiation doses to the general public from all pathways of exposure satisfy the criteria in 10 CFR Part 50, Appendix I. A general outline of these pathways is presented in Fig. 11.1; in addition, there may be a small exposure resulting from the transportation of spent fuel and solid radioactive wastes from the plant site.

11.19. External exposure arises mainly from immersion in the gaseous effluent (from a stack or roof vent) as diluted by the atmosphere and carried by the wind. Smaller exposures may result from swimming in a body of water into which the liquid effluent is discharged and from gamma rays emitted from land (or water) surfaces contaminated by the deposition of airborne radioactivity. The chief potential sources of internal exposure are inhalation of airborne radioactivity, including noble gases, radioiodines, and particulate matter, drinking water into which the effluent is discharged, eating fish caught in this water and, probably most important of all, drinking milk from cows pastured in areas close enough to the plant to be significantly contaminated by the deposition of radioiodines. Some iodines may also enter the human body directly by way of contaminated vegetation and fruits and in other ways, but the amounts are likely to be small.

11.20. Of the airborne iodine, from 10 to 70 percent is in the form of an organic compound, chiefly methyl iodine. This is produced by the interaction of elemental iodine and water (or steam) with carbon (in steel, oil, grease, etc.) in the presence of radiation. Methyl iodide is less easily removed than elemental iodine by the charcoal absorbers used to decrease the iodine content of the gaseous effluent; however, it is less readily deposited on vegetation and so tends to reduce the thyroid dose from radioiodines in milk. As a conservative measure, however, in calculating the thyroid dose, current practice is to assume that all the iodine released from the LWR plant is present in the elemental form.

11.21. In estimating the radiation dose from eating fish (including shellfish), allowance must be made for the fact that certain nuclides may have a higher concentration in the fish than in the water in which they live. The difference is expressed by the *biological accumulation* (or *bioaccumulation*) *factor*, also called the *concentration factor*, which is the ratio of the activity of a given nuclide (in Bq or Ci) per kilogram of fish to the activity per 10^{-3} m^3 of water. These factors are dependent on several variables, such as the species of fish, the elements normally present in the water, and the food habits of the fish, which vary with the season of the year. Furthermore, the bioaccumulation factor is not uniform throughout the fish, and the proper value to use for the present purpose is that of the edible portion. Approximate, conservative factors have been published for use in radiation dose calculations; they refer to fishes, crustaceans, and mullusks in fresh and salt water [4].

11.22. In addition to estimating the expected radiation dose from the nuclear plant to people, the applicant for a permit to build the plant must evaluate the radiological impact of the facility on biota other than man. The organisms to be included are those species of local flora and local and migratory fauna that are important in the plant vicinity. These species will, of course, depend on the locality. In making the dose estimates, the bioaccumulation factors for fishes are now those for the whole fish rather than for the edible portion. Appropriate factors must also be used for vegetation to determine the dose received by the plants themselves as well as by the animals that may eat them. Provided the radioactivity levels in the effluents from a nuclear facility can meet the criteria for maintaining radiation doses to the general population as low as is reasonably achievable, the radiological impact on biota other than man should be of little significance.

Radioactivity Levels "as Low as is Reasonably Achievable"

11.23. The radioactivity concentrations in the 10 CFR Part 20 regulations are intended to provide guidance as to the maximum permissible discharges of radioactive effluents to the environment. Licensees of nuclear power plants are required, however, to control the effluents in such a manner that radioactivity levels be kept "as low as is reasonably achievable." This requirement is interpreted as meaning that the levels should be kept "as low as is reasonably achievable taking into account the state of technology and the economics of improvements in relation to the benefits to the public health and safety. . . ."

11.24. In principle it should be possible, by increasing the complexity and hence the cost of constructing and operating the equipment of the "radwaste" system for the treatment of radioactive waste liquids and gases (§11.93 et seq.), to decrease the level of the radioactivity in effluents almost without limit. However, any such decrease means an increase in the radioactive material that must be stored, at least temporarily, in the plant, thereby increasing the potential risk to people working there. Furthermore, the greater the complexity of the radwaste system, the greater the chances of failures which might interfere with plant operation. The extent of the treatment of radioactive wastes in a given plant is thus based on an analysis of the benefits that would be expected to accrue, balanced against all the costs (including risks).

11.25. In expressing the numerical guides, the effluents from a plant are considered in three categories: liquid effluents, gaseous effluents, and radioiodines and radioactive material in particulate form. The liquid effluent generally contains some tritium (as tritiated water) and dissolved and suspended fission products and corrosion and erosion products that have become radioactive (§11.91). The gaseous activity consists mainly of radioisotopes of the noble gases krypton and xenon, most of which emit both gamma rays and beta particles. Because the gaseous effluent from PWRs is held up for a longer time before discharge than from BWRs, the shorter-lived isotopes have largely decayed leaving mainly krypton-85, which is almost exclusively a beta emitter, and xenon-133. The radioiodines and particulate matter are present, together with the noble gases, in the airborne effluents from the LWR installation. Some of the solid particles are formed by radioactive decay of the noble gases.

11.26. It should be understood that the purpose of the guidelines is to assess the adequacy of a proposed nuclear plant design, especially the design of the radwaste treatment system, before construction. When the plant is operating, the radioactivity concentrations in the effluent at the plant boundary must satisfy the requirements of 10 CFR Part 20. Furthermore the actual radiation doses to the public must comply with the EPA standards for the fuel cycle (40 CFR 190) [5].

THE SPENT-FUEL MANAGEMENT CHALLENGE

11.27. Chapter 10 was devoted to the management of fuel loaded into the core. We determined that the proper design of reload cores is essential for economical reactor operation. However, for every fuel assembly loaded into the core, one must be discharged and subsequently properly managed. Although discharged fuel no longer affects operation, its high radioactivity necessitates special provisions for temporary storage and shipping to a central facility. In many countries, the fuel assemblies are "reprocessed" to separate the fission products from useful fuel nuclides that can be recycled. However, in the United States, this option is not being pursued. Therefore, the assemblies must be stored either temporarily or permanently in a suitable facility. Since the assemblies are highly radioactive, they must be shipped in special shielded containers and if stored permanently, suitably packaged to preserve their integrity for thousands of years.

11.28. The technology of packaging discharged fuel so that it can safely be stored permanently has been satisfactorily developed. However, in the United States, political and legal issues have delayed the implementation of permanent repository arrangements. Unfortunately, the delay has led to some public misconception that the technical challenges have not been met. A related public perception problem has evolved as a result of unsatisfactory provisions made for storage of weapons program waste during World War II and for a decade thereafter. In the late 1980s, the need for cleanup at great expense became evident and received a great deal of publicity. Public confidence in the ability of the government to manage commercial reactor wastes safely was thereby diminished. A related complication has been the "NIMBY" (not in my backyard) syndrome. This refers to a general public unwillingness to accept the siting of many types of essential facilities, particularly those involving any form of waste, within their own geographical unit, which may be a city, county, or state.

11.29. An important consequence of the delays is the need to provide temporary storage of spent-fuel assemblies either at the reactor site or elsewhere. Although there are technical solutions to permanent storage problems, the challenge is to implement them. Our objective is to concentrate on the technical principles involved and to describe some of the options under consideration.

ON-SITE SPENT-FUEL STORAGE [6]

Introduction

11.30. After about a 3-year residence in the core, LWR spent-fuel assemblies are discharged and stored in an on-site water pool. When most present reactors were built, about a 5-month cooling period was planned, after which the fuel was to be shipped to an off-site interim storage facility or to a reprocessing plant. However, since neither option has become available, spent assembly storage has presented a logistic problem that we will examine.

11.31. Consideration of the options available for managing spent fuel assemblies should begin with a study of such characteristics as their isotopic mix, contributions to radioactivity, heat generation, and physical features.

Spent-Fuel Logistics

11.32. The number of assemblies discharged from a plant per year can be estimated from the average burnup, the mass of heavy metal per assembly, and the energy generated, as indicated in Example 11.1.

Example 11.1. Consider a typical 1000-MW(el) PWR which discharges fuel assemblies with an average discharge burnup of 33,000 MW · d/t. If the thermodynamic efficiency is 32 percent, and each assembly contains 450 kg of total uranium, how many assemblies would be discharged annually? A plant factor of 0.7 may be assumed.

$$\text{Thermal energy produced per year} = \frac{[1000 \text{ MW(el)}](365)(0.7)}{0.32}$$

$$= 7.98 \times 10^5 \text{ MW} \cdot \text{d}.$$

Then

$$\text{Assemblies per year} = \frac{(7.98 \times 10^5)(1000)}{(33{,}000)(450)} = 53.7,$$

which may be rounded off to 54 assemblies per year.

Actually, the number discharged would depend on the core management scheme followed, but the approach is useful for estimating storage requirements. For example, corresponding to the U.S. nuclear generating capacity of approximately 100,000 MW(el), about 5400 assemblies would be discharged per year before correcting for the BWR fraction of the assumed generating capacity. Of course, the adoption of longer operating cycles, which normally result in increased burnup, will reduce this estimate significantly.

11.33. On-site storage pool capacity prior to 1977, when reprocessing was deferred indefinitely in the United States, was designed to accommodate approximately the number of assemblies that would be discharged in 5 years. Also, room in the pool must always be available to hold a full core should it become necessary to defuel the reactor. Even with the introduction of enhanced capacity measures, about one-half of the presently operating nuclear units are expected to fill their existing storage capacity by 2000.

Pool Storage Capacity Enhancement

11.34. As a result of delays in the off-site management of spent fuel assemblies, most utilities proceeded to expand their on-site storage capacity. Originally, spent assemblies were stored in open lattice racks submerged in the water pool. A spacing of about 0.15 m (6 in.) between assemblies was provided to assure that the reactivity was about 0.9.

11.35. To conserve space, in all plants open racks were replaced with high-density racks that feature assembly compartments enclosed with a neutron-absorbing boron carbide–aluminum matrix. Using such racks, fuel can be stored in about one-half the volume required for storage in standard racks.

11.36. Additional space savings can be accomplished by removing the fuel rods from the assembly and storing them in a submerged grid which allows closer spacing than that in the original assembly. Such a procedure is known as spent-fuel-rod consolidation. Various designs have also been proposed for dry cask storage of either assemblies or consolidated fuel on site, but separate from the reactor building. Properly shielded canisters, which are generally air-cooled, hold from 24 to 31 assemblies.

11.37. In evaluating the reactivity of spent-fuel storage facilities, it has been customary to use the properties of fresh fuel. This is necessary when providing for a core defueling. However, for independent storage facilities, *burnup credit* for the reactivity loss during reactor exposure may be considered. Designs must satisfy the criteria in 10 CFR 72, Subpart F.

11.38. Another option is transshipment, which involves the shipment of spent fuel from one wet pool unit to another, generally within the same utility system, to relieve storage congestion. Since each utility can predict its spent fuel assembly storage needs many years in advance, appropriate management involving the various storage capacity enhancement measures as well as the expansion of on-site storage facilities is generally being implemented. Thus, a utility can store spent fuel on-site for the entire operating life of the plant, should it become necessary provided that licensing specifications permit. [7].

CHARACTERISTICS OF SPENT FUEL

11.39. After removal from the reactor and insertion in the storage pool, all of the radionuclides contained in the fuel continue the decay process. Short and moderately short half-life fission products become insignificant after a few months. Therefore, a cooling period of 150 days is a useful point of reference, which just happens to be the intended cooling period when reprocessing was planned.

11.40. The major contributions to the radioactivity of spent reactor fuel after 150 days of cooling are given in Table 11.1. The values, in curies and becquerels per metric ton (1000 kg) of uranium (initially free from plutonium) charged to the reactor, were calculated for a hypothetical LWR having a thermal power of 3300 MW and a fuel burnup (§10.15) of 2.85 $\times$ 10^{12} J (thermal)(2.85 TJ) per kilogram of uranium in the original fuel.* Somewhat different activities would be applicable to other operating conditions, but those in the table are fairly typical. An indication of the respective activities at times after 150 days can be obtained from the half-lives (see also Fig. 11.6). It is seen that for a cooling period of 150 days (or more) a relatively few fission products, namely, strontium, zirconium, niobium, ruthenium, cesium, and some rare-earth elements, are responsible for nearly all of the radioactivity. These are the most important elements from which the uranium and plutonium would be separated in spent fuel reprocessing.

11.41. The manner in which buildup of isotopes of the heavy elements occurs during reactor operation in a fuel consisting of uranium-235 and uranium-238 is illustrated in Fig. 11.2. Horizontal arrows pointing to the right represent (n, γ) reactions and those pointing to the left are for $(n, 2n)$ reactions with fast neutrons. Vertical arrows indicate beta decay. Where vertical arrows are absent, the nuclides are alpha-particle emitters. The

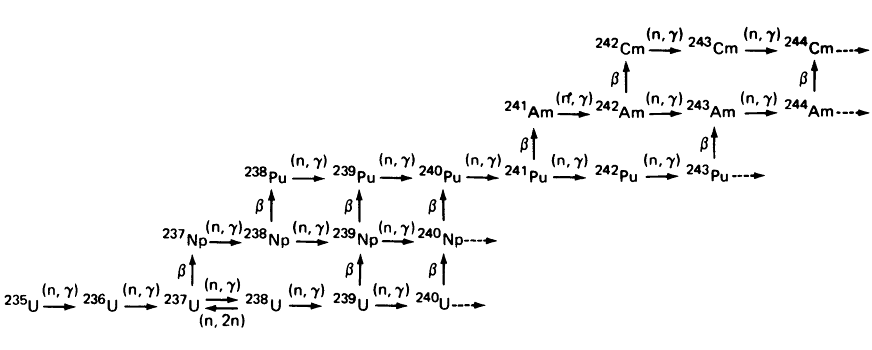
*A burnup of 2.85 TJ/kg U is equivalent to 33,000 megawatt-days/metric ton U (see Example 1.2).

TABLE 11.1. Major Contributions to Radioactivity of Spent LWR Fuel After 150 Days Cooling

Nuclide	Half-Life (years)	Main Decay Mode	Activity Ci/1000 kg U	Activity Bq/1000 kg U
Fission products				
Strontium-89	0.14	β	9.6×10^4	3.6×10^{15}
Strontium-90	29	β	7.7×10^4	2.8×10^{15}
Zirconium-95	0.18	β, γ	2.8×10^5	1.0×10^{16}
Niobium-95	0.095	β, γ	5.2×10^5	1.9×10^{16}
Ruthenium-106	1.0	β	4.1×10^5	1.5×10^{16}
Cesium-134	2.05	β, γ	2.1×10^5	7.7×10^{15}
Cesium-137	30	β, γ	1.1×10^5	4.1×10^{15}
Cerium-144	0.78	β, γ	7.7×10^5	2.8×10^{16}
Promethium-147	2.6	β	9.9×10^4	3.7×10^{15}
Heavy-element isotopes				
Plutonium-238	88	α	2.8×10^2	1.0×10^{13}
Plutonium-239	24,400	α	3.3×10^2	1.2×10^{13}
Plutonium-240	6,540	α	4.8×10^2	1.8×10^{13}
Plutonium-241	14	β	1.1×10^5	4.1×10^{15}
Plutonium-242	387,000	α	1.36	5.0×10^{10}
Americium-241	433	α, γ	2.0×10^2	7.4×10^{12}
Americium-243	7,370	α, γ	17.4	6.4×10^{11}
Curium-242	0.45	α, sf*	1.5×10^4	5.5×10^{14}
Curium-244	18	α, sf*	2.5×10^3	9.3×10^{13}

*Significant spontaneous fission accompanied by neutron emission.

Fig. 11.2. Heavy-isotope buildup in uranium. (Unless otherwise indicated, the nuclides are alpha-particle emitters.)



decay product is then a nuclide with an atomic number two units less and a mass number of four units less than the parent. Alpha-particle decays are of minor importance in the cooling period, but they affect the buildup of heavy isotopes at a later stage (§11.87).

11.42. Of immediate interest is uranium-237 (half-life 6.75 days) which is formed by two (n, γ) stages starting with uranium-235 or by the $(n, 2n)$ reaction with uranium-238. Any uranium-237 remaining in the spent fuel will be associated with the recovered uranium. Since uranium-237 is a gamma-ray emitter with a fairly short half-life, its presence makes the product difficult to handle. After a 150-day cooling period, however, the uranium-237 will have decayed to such an extent that the gamma activity is small enough to be tolerable. Moreover, the beta decay product, neptunium-237, will be separated from the uranium during the reprocessing operation.

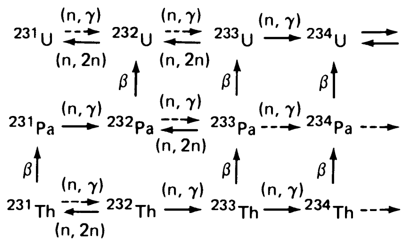
11.43. A chart similar to Fig. 11.2 showing the buildup of heavy isotopes when thorium-232 is included in the fuel is shown in Fig. 11.3. With thorium-232 as fertile material, a long cooling time would be required to permit decay of the intermediate protactinium-233 (half-life 27 days) to the fissile uranium-233. During this period, the activity of thorium-234 (half-life 24 days) would also decrease to a permissible level. However, at the same time, thorium-238 (half-life 1.9 years) would accumulate as a result of the alpha decay of uranium-232 (Fig. 11.3). Since thorium-228 has strong gamma-ray emitters among its daughter products, remote handling of spent fuel would appear to be necessary regardless of the cooling time.

STORAGE AND DISPOSAL OPTIONS [8]

Introduction

11.44. A systematic approach to the management of spent fuel was provided in the Nuclear Waste Policy Act (NWPA), which became effective

Fig. 11.3. Heavy-isotope buildup in thorium. (Unless otherwise indicated, the nuclides are alpha-particle emitters.)



in 1983 and was amended in 1987. Provisions were made for "characterizing" a permanent repository, and the need for interim storage was recognized. The Act provided for the establishment of the Office of Civilian Radioactive Waste Management (OCRWM) to implement the program and a Nuclear Waste Fund to cover all storage and disposal costs. A charge of 1.0 mill/kWh of nuclear electricity is paid into the fund. This amounts to approximately $300 million per year.

11.45. A detailed timetable was specified which has not been met. For example, the law directs the Department of Energy (DOE) to begin accepting spent fuel from utilities in 1998 to be placed in a monitored retrievable storage (MRS) facility until an underground repository is available in 2010. Should the MRS facility not be ready to accept fuel in 1998, as appears likely, an alternative plan provides for the storage of spent-fuel assemblies at federal sites in standardized dry canisters. Since many aspects of the program remain uncertain, we will limit our discussion to the basic approaches planned. Let us first make the distinction between *interim storage*, *monitored retrievable storage*, and *disposal*. An interim storage facility provides short-term temporary storage to accommodate fuel assemblies until more permanent facilities are available. A monitored retrievable storage facility is intended as an alternative to a geological repository for long-term storage. However, the assemblies could be retrieved for disposal elsewhere or perhaps for future reprocessing. In the disposal option, storage would be permanent with no possibility of retrieval.

Retrievable Storage

11.46. Approval for constructing an aboveground monitored retrievable storage (MRS) facility has been complicated by an unwillingness of state governments to accept it. Also, the 1987 amendment to the NWPA links the MRS to progress toward opening a permanent underground geological site, which has had numerous delays.

11.47. Over the years, various designs for a MRS facility have been proposed, varying from a scaled-up pool storage arrangement to a concrete storage cask array. There are two major considerations in the design. Decay heat must be managed and neutron and gamma-ray shielding provided. Heat generation and activity can be predicted accurately by codes such as ORIGIN (§2.213). However, as a rule of thumb, it is helpful to remember that during about 10 years of temporary storage the assemblies will have lost about two orders of magnitude in heat generation rate and activity from the discharge levels. Subsequent loss is fairly slow, being roughly proportional to the 0.2 power of time (§2.217).

11.48. An indication of retrievable storage cask designs is provided by those submitted by utilities for licensing by the NRC for air-cooled on-site storage to supplement pool capacity. A common design arrangement for a vertical cask accommodates 24 assemblies in a basket rack inside a concrete shield. A typical cask is about 3.66 m (12 ft) in diameter and 6.1 m (20 ft) high.

Permanent Disposal

11.49. In the United States, plans for a geological repository are centered on Yucca Mountain in Nevada. According to the NRPA, a comprehensive 10-year program to establish site suitability is required, the start of which has been delayed as a result of objections to the site by the state of Nevada. Should the site be acceptable, an additional 8 years would be required for licensing and construction activities.

11.50. In a conceptual design for the repository, a storage region would be at a depth of 600 to 1200 m from the surface. Suitability "packaged" spent fuel or reprocessed vitrified high-level waste (§11.84) would be stored either in rooms or in holes drilled in the rock. Following a test period of about 50 years when the packages could be retrieved should problems arise, the rooms would be backfilled with excavated rock, which would provide an additional barrier. Since spent fuel is an energy resource should it be reprocessed, there is an argument for reviewing the disposal option during the 50-year retrievable period.

11.51. Containment of the waste is based on a multiple barrier approach. The waste container itself is designed to have several barriers, including an outer stainless steel or copper jacket. If high-level reprocessed waste would be stored, it would be in the form of a solid, glasslike, inert material packaged in a metal container. The final, but extremely important barrier to radioisotope migration is the geologic medium itself, which will be considered further in the next section.

MIGRATION OF WASTE RADIONUCLIDES [9]

11.52. The hazards from deep geologic disposal of high-level radioactive wastes can best be evaluated by modeling the transport of radionuclides of concern through the barriers. In this model, consideration must be given to the decay taking place during the time required for transport. In other words, a risk model might assume that ground water would intrude upon the buried waste package; then dissolution of the various radionuclides into the water would be described. Finally, the various pathways by which the decaying radionuclides can reach people would be modeled.

11.53. One measure* of relative hazard from a given radionuclide is based on the concept of *water-dilution rate*. The calculated rate of future discharge of the radionuclide into the biosphere ("people environment"), after decay, when divided by the maximum permissible concentration of the specific radionuclide in drinking water, yields the water-dilution rate.

11.54. The starting point for the transport model is the dissolving of the radionuclides into the groundwater. However, ceramic fuel pellets discharged from water-cooled reactors are not only insoluble in water, but are encased in clad intended to resist dissolving. Such stored assemblies are then encased in a metal canister. As another option, high-level reprocessed waste would be incorporated into a solid matrix, such as glass, then encased in a metal canister. For modeling purposes, it is customary to assume a constant level of solution over a period of 10,000 years, although some specific elements are so insoluble that little of such components will actually dissolve.

11.55. The rate at which the dissolved elements are transported through the geologic medium is considerably reduced by "sorption," a combination of physical and chemical adsorption and absorption, with the soil or rock. However, the sorption tendency varies from element to element, resulting in a different mix leaving the medium than that entering. In general, the heavy elements, such as uranium and plutonium, are highly sorped, while lighter elements are not strongly affected. The very low migration of uranium and plutonium has been confirmed by actual experience with the debris from weapons tests and leaked waste from DOE facilities. Thus, the model to determine water-dilution rates for individual radionuclides must take into consideration the input rate from dissolution, the isolation distance, the groundwater velocity, a sorption equilibrium constant, the half-life, and the maximum permissible concentration in drinking water.

11.56. Groundwater is expected to move at a very low velocity in the neighborhood of a geological repository. Also, physical chemical solubility and mass transfer principles apply to the determination of the concentration of the radionuclide in the liquid. A lower concentration value results than if such principles are not considered.

11.57. Since a realistic model is complex and involves many assumptions, it is premature to cite results. However, a "feeling" for relative risks has been obtained by comparing water dilution rates from buried high-level wastes with those from uranium ore, uranium mill tailings, and coal

*Another measure of relative biological hazard is the *water* (or *air*) *dilution volume*, sometimes called the *toxicity*, defined as the volume of water (or air) required to dilute a given amount of radioisotope to a concentration for drinking (or continuous exposure). However, this measure does not take into account solubility or ingestion processes.

ash, using consistent modeling assumptions. The buried waste proved to be two orders of magnitude safer than the ore and mill tailings and slightly safer than the coal ash over a period of 1 million years. Of course, another comparison could be made with the lethal doses from various poisonous common chemicals of commerce (i.e., cyanides, arsenides, pesticide compounds, etc.) which are not subject to radioactive decay.

Lessons from the Oklo Reactor Waste [10]

11.58. At Oklo, in Gabon, near the equator in West Africa, a nuclear fission reactor located in part of a rich uranium ore vein started spontaneously about 1.8 billion years ago and ran for several hundred thousand years before shutting down by itself. It is estimated that about 15,000 MW · y of fission energy was produced, yielding about 6 tons of fission products and 2.5 tons of plutonium. Since both uranium-235 and uranium-238 are radioactive with different half-lives, the relative decay tends to shift the isotopic ratio. At the time of the reactor operation, natural uranium contained about 3 percent uranium-235. The existence of groundwater in the ore veins of this tropical area probably led to just the correct conditions for criticality. The reactor probably operated at "slow simmer" and was self-regulating since too much power would tend to boil off the moderating groundwater. Presumably, operation finally ceased when the ore became depleted.

11.59. The most interesting lesson from the Oklo reactor operation is that most of the radioactive waste has remained at the reactor site for over 1 billion years despite the abundance of groundwater. Metals having a valence of 1 or 2 are soluble in water, and were leached away. However, isotopes of these elements have short half-lives and thus would soon be harmless. The elements of concern, particularly plutonium, were retained in the geologic formation and effectively immobilized. Therefore, it would seem that a repository with additional barriers should perform at least as well.

THE REPROCESSING OPTION [1]

Introduction

11.60. Prior to 1976, it was generally assumed that spent fuel would be reprocessed for the recovery of uranium and plutonium for reuse as reactor fuel. It was felt that economic supplies of uranium were limited relative to future needs for the then-projected growth of the nuclear power industry. Also, planned fast reactors would require recovered plutonium for fuel.

11.61. In 1976, public concerns that the availability of chemically pure plutonium would encourage the proliferation of nuclear weapons led to a ban on reprocessing. Although the ban was lifted in 1980, reprocessing was no longer commercially attractive, for several reasons. First, processing plant operating costs had increased significantly. Next, many new nuclear power plants were canceled, depressing the price of uranium. Since new uranium reserves were also identified, the future salvage value of recycled fuel seemed uncertain, thus reducing the economic incentive for reprocessing. In addition, there was no need for significant fast reactor fuel since a U.S. demonstration plant at Clinch River, Tennessee was canceled and there were no plans for other plants. However, without the need for a significant level of commercial justification, fuel reprocessing continues to be carried out routinely in other countries. One incentive for such operations is the reduction in the volume of high-level radioactive waste that requires management (§11.63).

11.62. It appears likely that around the beginning of the next century, the combination of increased worldwide energy requirements, diminishing oil and gas supplies, and environmental concerns will lead to pressure to increase nuclear capacity substantially in the United States. With the conservation of nuclear fuels then becoming of interest again, fast reactors are likely to play a role in this new capacity.

11.63. The disposal of reprocessed high-level waste solidified as glass in canisters requires less volume than the disposal of spent fuel and utilizes a more straightforward leakproof package system. Thus, easier disposal provides some incentive for reprocessing. As a result of all these factors, it is probable that reprocessing of stored spent fuel will be resumed in the United States at some time in the future. Therefore, the highlights of the subject will be discussed here.

Head-End Treatment

11.64. After the cooling period, the spent-fuel assemblies would be shipped in strong metal casks to a reprocessing plant where the fission products are removed and the uranium and plutonium are recovered. The method commonly used for this purpose is based on extraction by an organic solvent, and this requires the fuel material to be dissolved in nitric acid to form a solution of nitrates. A simplified flow sheet of the reprocessing operations is given in Fig. 11.4. Each step is quite complex, but for the present purpose a brief overview will be adequate.

11.65. In the first (or "head-end") stage, the fuel rod assemblies, either with or without removal of hardware, are chopped into sections from which the spent material is leached with hot nitric acid. This process is commonly referred to as *chop-leach*. The hulls of zircaloy (or other) cladding and

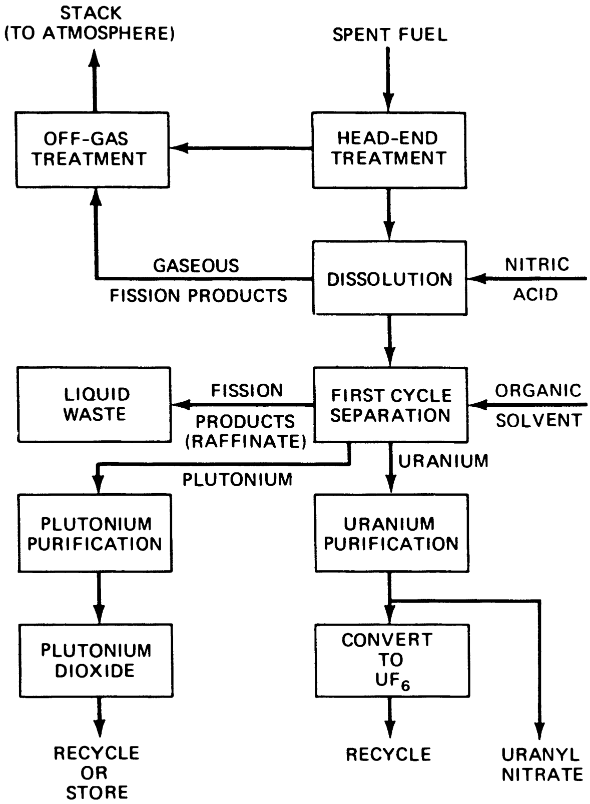


Fig. 11.4. Spent-fuel reprocessing flow diagram.

hardware that remain are subjected to a hot nitric acid soak to remove essentially all of the uranium and transuranic elements, i.e., elements of higher atomic number than uranium. The hardware and hulls form what are called TRU (for transuranium) wastes or alpha wastes, because they contain traces of alpha-emitting transuranium elements. These wastes have been buried temporarily in the past, but more permanent underground disposal is planned for the future.

Solvent-Extraction Separation Processes

11.66. The solvent-extraction method for separating the constituents of an aqueous solution can be used when one or more of these constituents are appreciably soluble, whereas the others are much less soluble, in an organic solvent which is essentially immiscible with water. When the organic liquid is brought into intimate contact with the aqueous solution, the substances present will distribute themselves between the organic and aqueous phases. The constituent (or constituents) with the greatest solubility in the organic medium will tend to pass into that phase whereas the others will tend to remain in the aqueous solution. Thus, a partial separation of the constituents of the solution will have been achieved.

11.67. In choosing the organic liquid for a particular solvent-extraction procedure, an important property is its selectivity, that is its ability to extract a particular component (or components) of a solution in preference to all others that are present. The selectivity is expressed by the separation factor, i.e., the ratio of the distribution coefficients of the wanted and unwanted species when equilibrium is attained between the two phases. The distribution coefficient or distribution ratio D is defined as

$$D = \frac{\text{Conc. of component in organic phase}}{\text{Conc. of component in aqueous phase}}$$

at equilibrium, and the separation factor α is given by

$$\alpha = \frac{D(\text{product})}{D(\text{impurity})}.$$

A good solvent for extraction is one for which the distribution coefficient for one component is large and the separation factor is either large or small. In other words, it is desirable that D(product) shall be large, whereas D(impurity) should be small, or vice versa.

11.68. The extraction of an inorganic compound, such as a nitrate, from an aqueous solution by means of an organic solvent is influenced by a number of circumstances. Of particular importance are the presence of (1) salting agents, (2) complex-forming anions, and (3) oxidizing or reducing agents.

11.69. A *salting agent* is either a salt or an acid, having the same anion as the inorganic compound to be extracted, the presence of which in the aqueous solution increases the distribution ratio. In the extraction of uranyl nitrate, for example, either nitric acid or one if its salts, such as sodium, potassium, calcium, or aluminum nitrate, can serve as a salting agent. These substances are soluble in the aqueous phase but not in the organic solvent.

11.70. The extraction of a specified element from aqueous solution by an organic medium is dependent upon the particular form in which the element is present in the solution. For example, uranyl nitrate hexahydrate can be extracted by certain organic solvents, but the corresponding sulfate is not extractable by these solvents. The addition of a sulfate or other salt of a complex-forming anion to an aqueous solution of uranyl nitrate will thus decrease the extractability of the uranium, since a proportion of the element will be in some form other than the nitrate. The complex-forming anions decrease the distribution coefficient between the organic and aqueous phases, and so their effect is opposite to that of the common-ion salting agents.

11.71. The solvent-extraction processes for separating uranium and plutonium from the fission products and then from each other depend on the somewhat unusual chemical behavior of the heavy elements. Starting with actinium (atomic number 89), there is a series of 15 elements, called the *actinide series*, in the sixth period of the periodic system which resemble the rare-earth (or *lanthanide*) elements in the fifth period. The lanthanide elements all have similar chemical properties, based on a positive valence of 3, resulting from the presence of three relatively loosely bound outer electrons in each atom. In the analogous actinide series, there are also marked resemblances among the elements, especially in the formation of a tripositive (III) valence state. However, because some of the actinide elements have inner electrons which are not very tightly bound, it is possible to realize tetrapositive (IV), pentapositive (V), and hexapositive (VI) states.

11.72. Although in a given valence state the various actinide elements have similar chemical properties, these properties often are very different in the different oxidation states. For example, the nitrates of the (IV) and (VI) states are appreciably soluble in certain organic liquids, but the nitrates of the (III) states are virtually insoluble in these liquids. The relative stability of the different oxidation states varies with the atomic number of the element. Hence, by the use of appropriate reagents, it is possible to shift the oxidation and reduction states so that two (or more) elements in a given solution will be in different states with differing solubilities in an organic liquid. Separation of the elements can then be accomplished by solvent extraction.

11.73. It is evident from the foregoing discussion that the nitrates of the (IV) and (VI) states of the actinide elements will have large distribution coefficients and hence will be extractable by an organic liquid, but the lower oxidation (III) state will have a smaller distribution coefficient and be less extractable. One consequence of this difference in the distribution coefficients is that after an element has been extracted into an organic medium it can be back-extracted into an aqueous solution if the (IV) or (VI) state is reduced to the (III) state. Suppose two actinide elements have been extracted into an organic solvent in the (IV) or (VI) state. If one of the elements is reduced to the (III) state, it can be separated from the other element by back-extraction into an aqueous solution. The uranium and plutonium in spent reactor fuel are separated from one another in this way.

The purex process

11.74. The "Purex" process, using *n*-tributyl phosphate (TBP) as the extractant, is typical of solvent-extraction procedures employed in the treatment of spent fuel. In the form of nitrates, uranium (VI) and plutonium (IV) can be readily extracted from aqueous solution by TBP, whereas the fission products are taken up to a much smaller extent. TBP is relatively stable in the presence of fairly high concentrations of nitric acid, hence, the latter is used as the salting agent.

11.75. In the first cycle of the Purex process, of which an outline flow sheet is shown in Fig. 11.5, the feed consists of an aqueous solution containing uranium (VI), plutonium (IV), and fission product (FP) nitrates plus an excess of nitric acid. Sodium nitrite is added to make sure that the plutonium is entirely in the (IV) state, since this form is best extracted by TBP. The feed solution enters at the middle of the first (extraction) column, while the less dense organic extractant (TBP in a kerosene-type solvent) entering from the bottom flows upward. The uranium (VI) and plutonium (IV) nitrates are thereby extracted from the aqueous solution and pass into the organic medium. In the upper part of the column the organic phase is scrubbed with concentrated nitric acid. Most of the fission products that may have entered the organic solvent are now back-extracted into the aqueous phase, but the nitric acid, which acts as a salting agent, largely prevents back-extraction of the uranium and plutonium. The aqueous effluent (raffinate) from the extraction column contains essentially all the fission products with little or no uranium or plutonium.

11.76. The organic phase containing the uranium and plutonium next passes into the second (partitioning) column where it flows upward and meets the downflowing aqueous strip solution containing a reducing agent to reduce plutonium (IV) to plutonium (III). In a modified Purex process, the reduction is performed electrolytically. The plutonium (III) nitrate is not soluble in the organic medium and so it is back-extracted into the aqueous phase. As this flows downward it is scrubbed with fresh TBP moving upward from the bottom of the column. Any uranium (VI) that has passed into the aqueous solution is thereby returned to the organic phase. The aqueous medium, containing plutonium (III) nitrate, leaves at the bottom of the partition column.

11.77. The organic solution of uranium (VI) nitrate, from which the plutonium and fission products have been almost completely separated, is now transferred to the bottom of the third (stripping) column where it flows upward and is stripped by dilute nitric acid flowing downward. In the absence of a salting agent, the uranium is back-extracted into the aqueous phase and then flows out of the bottom of the column. The spent solvent, leaving at the top, is sent to a recovery plant for purification and subsequent reuse in extraction.

11.78. For further purification, both the aqueous uranium (VI) and plutonium (III) nitrate solutions are submitted to a second and third cycle. The uranium purification in each cycle is essentially identical with the last two stages of the first cycle shown in Fig. 11.5. The aqueous uranium solution is first extracted into the TBP phase and scrubbed with a reducing solution; the organic phase is then stripped by dilute nitric acid in a second column. For each plutonium cycle, the plutonium (III) solution is converted into the (IV) state by means of sodium nitrite and nitric acid, extracted into the TBP medium, and scrubbed with nitric acid in the same column. The organic solution then passes to a stripping column where the plutonium is back-extracted into the aqueous phase by dilute nitric acid. Ion exchange has been employed for the final purification of plutonium following the first cycle of recovery by solvement extraction. The procedure is particularly valuable for the separation of zirconium and ruthenium, and their daughter products niobium and rhodium, respectively. The common practice, however, is to use two additional stages of solvent extraction to remove residual fission products, as stated above.

Other Separation Processes

11.79. Separation of uranium from the fission products can be accomplished by *volatilization* since nearly all of the fluorides of the latter elements vaporize at higher temperatures than uranium hexafluoride. Uranium may therefore be recovered from a mixture of fluorides by distillation at a relatively low temperature. If plutonium is present, difficulties arise because of the chemical reactivity of plutonium hexafluroide, but separation of plutonium from uranium can be accomplished by a more complicated volatilization process.

11.80. In reprocessing fast-reactor fuels, the complete removal of fission products may not be necessary since their effect on the neutron economy is much less in a fast-neutron spectrum than it is in a thermal spectrum. In these circumstances *pyrochemical* (or *pyrometallurgical*) processes are of interest. Among the procedures which has been proposed are high-temperature chemical methods involving molten salts and molten-metal refining by oxidation and volatilization. None of these processes have yet achieved commercial status and so will not be considered further. However, pyrochemical processes offer advantages over the Purex process from the weapons proliferation viewpoint since there is only partial removal of the fission products.

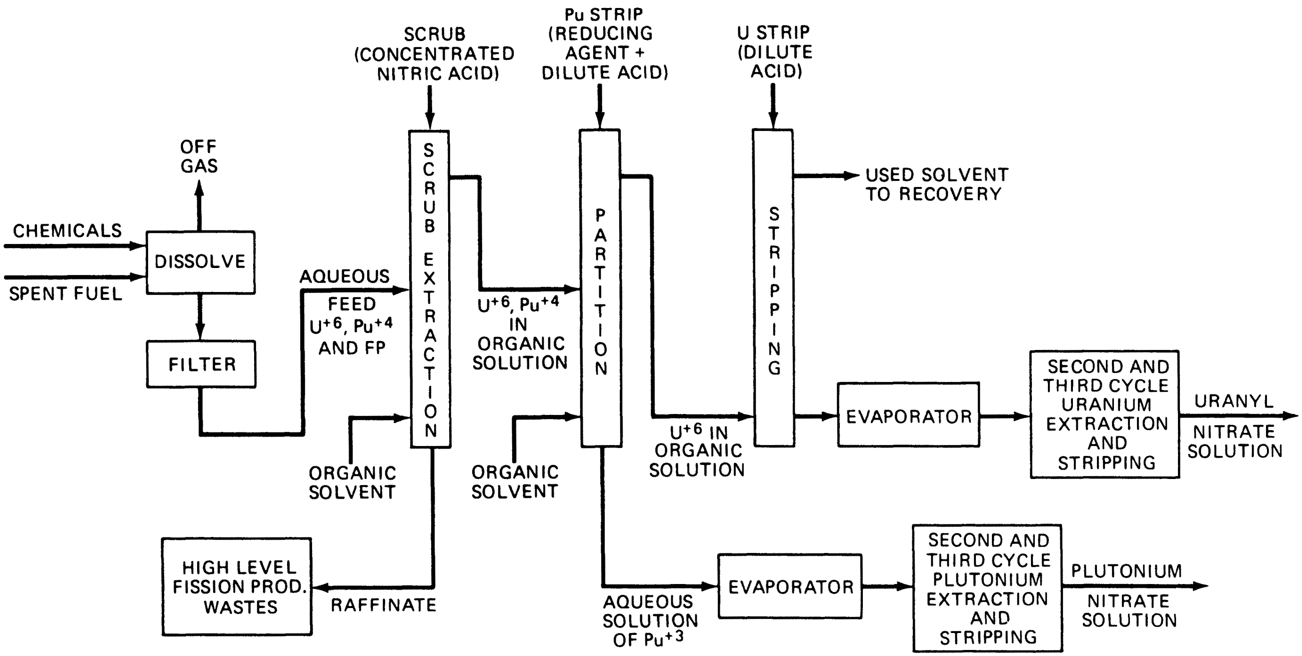


Fig. 11.5. Outline of Purex process for spent-fuel reprocessing.

Fuel Reprocessing Waste Management

11.81. Various wastes are generated by a reprocessing plant using the Purex or a similar process. Low-radioactivity wastes are handled by standard methods without problems. However, gaseous effluents that contain some radioactivity require special attention. The most highly radioactive liquid waste from the reprocessing of spent fuel by solvent extraction—and, in fact, from any stage of the nuclear fuel cycle—originates from the first-cycle raffinate (see Fig. 11.5). This *high-level waste*, as it is called, contains about 99.9 percent of the nongaseous fission products originally present in the spent fuel; it also contains some uranium and other actinide elements, including about 0.5 percent of the initial plutonium content. Because of its intense radioactivity, the high-level liquid waste presents a special problem.

11.82. The raffinate from the Purex process is evaporated to recover much of the nitric acid still remaining from the fuel dissolution and also to reduce the volume. The resulting solution, which contains nitrates of the metallic fission products and the actinides together with some free nitric acid, is stored, temporarily at least, in water-cooled, underground steel tanks.

11.83. Several processes have been investigated for converting high-level liquid wastes into solid form. Two of the more successful are spray calcination and fluidized-bed calcination. In the former, the liquid is sprayed into the top of a cylindrical tower heater in a furnace; the solid *calcine*, consisting of a mixture of fission product and heavy element oxides, is collected at the bottom. In the other process, which has been used since 1963 for solidification of liquid wastes from the reprocessing of special highly-enriched fuels, the water is evaporated in a heated, fluidized bed of particles made from previously dried waste.

11.84. For long-term storage, the calcine (mixed oxides) may be heated with a glass-forming frit containing borax and silica. The product is a borosilicate glass which is less leachable by water and has a higher thermal conductivity than the calcine. These changes are desirable from the standpoint of the ultimate disposal of the solidified high-level wastes. However, it appears that a sufficient increase in temperature, e.g., by radiation heating, may cause the glass to devitrify; the resulting microcrystaline material would then be more readily leachable than the original glass. This drawback could be overcome by reducing the fission-product content of the glass or by increasing the distance between the containers in storage, thereby decreasing the heating rate. Nevertheless, efforts are being made to develop alternative solid forms for high-level wastes; these include crystalline products similar to stable minerals and cermets consisting of ceramic (calcined) waste particles in a metal matrix.

11.85. A 1000-MW (electric) LWR plant will, on the average, discharge some 30 to 35 metric tons of spent fuel annually, assuming operation at full capacity.* This would result in the production of roughly 3.2 m^3 of high-level solid (borosilicate) waste. If this waste is packaged in steel cylinders 0.3 m in diameter and 3 m in length, about 15 such containers would be filled each year. Operation below the rated-capacity of the nuclear power plant, as will always be the case, will result in a smaller quantity of waste.

Characteristics of Solidified High-Level Waste

11.86. The radioactivity of the solid reprocessing wastes, as a function of time after removal from the reactor, can be inferred from Fig. 11.6, the data for which were calculated for LWR spent fuel, assuming 99.5 percent removal of plutonium (and uranium) [12]. Solid wastes would differ in the respect that the volatile (or gaseous) species, namely, iodine, krypton, tritium, and xenon, would not be present, since they are either removed or released at the reprocessing plant or during calcination of liquid wastes. The data in Fig. 11.6 refer specifically to 1000 MW(el)-year of LWR operation with a burnup of 2.85 TJ/kg of fuel, initially uranium free from plutonium; 0.5 percent of the plutonium (and all the americium and curium) formed as well as 0.5 percent of the uranium are assumed to remain.

11.87. It is seen that at 1 year after discharge from the reactor, the total activity would be about 10^8 curies or more than 10^{18} Bq; it decreases by a factor of roughly ten after 10 years and another factor of about ten after 100 years. Subsequently, the activity decreases more rapidly until some 600 years have elapsed, when the decrease becomes very slow. The radioactivity then arises mainly from the long-lived, alpha-emitting transuranium nuclides, especially isotopes of plutonium and americium. There is also a small contribution from the beta-emitting fission product, technetium-99 (half-life 2.1×10^5 years). Since iodine-129 is volatile, it is not present in the solid wastes.

11.88. Because the curves in Fig. 11.6 are for the total activity of all the isotopes of a given element, they do not show some interesting variations in heavy-element buildup. For example, the amount of plutonium-238, produced in the wastes by the alpha decay of the short-lived curium-242, increases to a maximum after some 10 years, and plutonium-240, from the decay of curium-244, attains a maximum after about 100 years. Plutonium-239, which is formed by alpha decay of americium-243 to neptunium-239 followed by beta decay of the latter, reaches a maximum activity after approximately 20,000 years. The sharp decrease in the total activity

*For brevity, this will be referred to as 1000 MW(el)-year of operation.

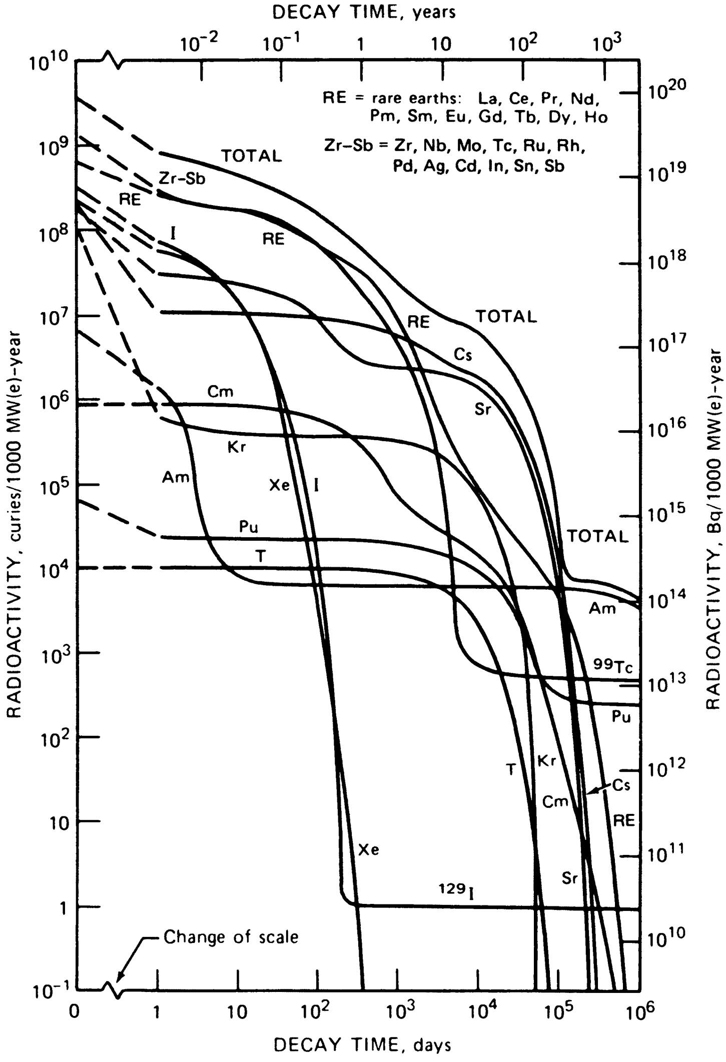


Fig. 11.6. Radioactivity of waste from uranium fuel reprocessing, based on a burnup of 2.85 TJ/kg U (33,000 MW · d/1000 kg) and 99.5 percent plutonium removal [12].

of plutonium between about 10 and 100 years, as seen in Fig. 11.6, results from the decay of plutonium-238 (half-life 88 years) and plutonium-241 (half-life 14 years) which have made the major contribution in the early stages.

11.89. The rate of heat generation (or thermal power) in the wastes, due to radioactive decay, varies in the same general manner as the total activity. For the 1000 MW(el)-year operation of an LWR referred to, the thermal power of the solid high-level wastes would be about 350 kW after 1 year and roughly 35 kW after 10 years; at the latter time, almost 90 percent of the heat would arise from strontium-90 (and its short-lived decay product yttrium-90) and cesium-137. After 100 years, the thermal power is approximately 4 kW, and after 600 years (or more) it is less than 0.05 kW.

REACTOR RADWASTE MANAGEMENT [13]

Sources of Radioactivity

11.90. For reactors cooled (and moderated) by light water, essentially all the radioactivity present in the effluent from a nuclear power plant originates in the reactor vessel from two sources: (1) the escape of fission products from the fuel into the coolant water and (2) activation by neutrons of corrosion and erosion products, e.g., iron, chromium, nickel, and cobalt, and other substances, e.g., boron (in boric acid), dissolved or suspended in the water. Gaseous radioactive isotopes of oxygen, nitrogen, and argon, in particular, are also formed by neutron activation of these elements in the water molecule and in dissolved air. Most of these isotopes have fairly short half-lives, and they will have largely decayed to stable nuclides before being discharged to the atmosphere. However, although these radioactive species are a negligible environmental hazard, they must not be overlooked in connection with the protection of plant workers.

11.91. Fuel rods are clad with zircaloy to prevent escape of fission products, but in a small fraction of the rods, typically below 0.1 percent, toward the end of their operating life small holes and cracks may develop in the cladding as a result of welding defects or localized corrosion. Consequently, some fission products will escape into the coolant. These consist mainly of the gaseous radioisotopes of krypton and xenon and of the readily volatile element iodine. In addition, solid fission products are extracted to a small extent by the high-temperature water. Hence, radioactive fission products can be found in both gaseous and liquid effluents from a reactor installation. Corrosion and erosion products from the reactor vessel, coolant pumps, steam generator, and piping are inevitably present in the water and these are activated by neutrons from the reactor core (§7.78).

11.92. Another important source of radioactivity in the water coolant is tritium, the radioactive isotope of hydrogen. About one fission in 10^4 of uranium-235 is a ternary fission in which tritium is one of the products; the other two products are heavier nuclides similar to ordinary (binary) fission products. Some tritium is also formed by the interaction of neutrons with boron in the control elements of BWRs. (The control rods of PWRs do not contain boron.) Thus, tritium can escape into the coolant through cladding defects in both fuel rods and control elements. Tritium is also formed directly in the coolant by the capture of neutrons by deuterium, i.e., heavy hydrogen, nuclei that are normally present in water. In PWRs, the major sources of tritium are the reactions of neutrons with boron (in the boric acid) used as a chemical shim (§5.187) and with lithium (in lithium hydroxide) used to control the pH of the water. Regardless of its origin, most of the tritium in the reactor coolant is found in the form of HTO (§6.14); a small proportion occurs as HT gas but this is of little significance.

Reactor Radwaste Systems

11.93. The basic objective of the reactor radwaste system is to reduce the radioactivity in liquid and gaseous wastes to such levels that they can be safely discharged to the environment. The general procedures used for this purpose are natural decay and decontamination, either by evaporation or by demineralization. Natural decay is commonly used to decrease the activity of short-lived gaseous radionuclides, such as nitrogen-16 and several isotopes of krypton and xenon. Gaseous wastes are thus held up for a period of time, from 30 minutes to several days, before discharge.

11.94. Liquid wastes contain such a wide range of fission (and other) products that natural decay is of minor consequence. These wastes are therefore treated by a decontamination process. The aqueous waste solution may be evaporated and the vapor condensed. The condensate is essentially pure water which can be reused in the plant or discharged. In the alternative demineralization process, dissolved radioactive substances are extracted by passage through an ion-exchange resin. The evaporator residues and spent resins contain the removed radioactive material and are disposed of in a safe manner.

11.95. The choice between evaporation and demineralization depends on circumstances and may vary from one plant to another. The decontamination factor, i.e., the ratio of the activity concentration before to that after treatment, is 10^3 to 10^4 for evaporation but is only 10 to 10^2 for demineralization. On the other hand, installation and operation is less expensive for a demineralizer. If the liquid waste contains significant amounts of dissolved material (other than fission products), decontamination by evaporation is preferred to avoid rapid saturation of the ion-exchange resin.

11.96. Radwaste treatment systems differ to some extent for pressurized-water and boiling-water reactors. There also are some variations among reactor systems of the same type fabricated by different vendors. The descriptions that follow are thus intended to indicate the general principles of operation of radwaste systems.

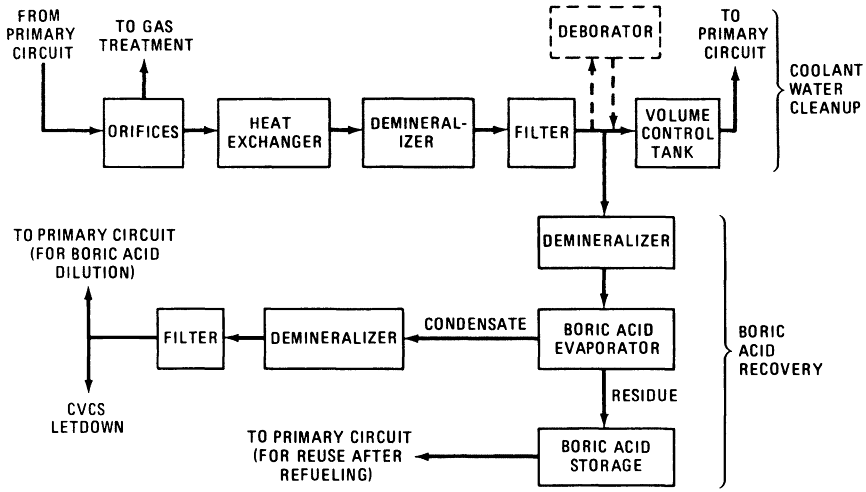
Pressurized-Water Reactors

11.97. The actual radwaste system in a PWR is preceded by a *chemical and volume control system* (or CVCS). As its name implies, the main purpose of the CVCS is to maintain the volume of water in the primary coolant system and to adjust the concentration of boric acid that serves as the chemical shim. At the same time, the water, which contains most of the radionuclides formed in the reactor vessel, is treated for the partial removal of dissolved and suspended radioactive material. In this respect, the CVCS is part of the radwaste treatment system.

11.98. In the CVCS, water is withdrawn ("bled") from the pressurized primary circuit, passed through orifices to reduce the pressure, and then cooled in a heat exchanger. Gases liberated are transferred to the gas treatment system. The water passes to an ion-exchange "demineralizer" for the removal of fission products and is then filtered. The clean (borated) water is stored in the volume control tank for use as make-up in the reactor vessel as required (Fig. 11.7). Tritium is not affected by the demineralizer and accumulates in the water.

11.99. As seen in §5.187, the concentration of boric acid in the coolant must be decreased as the reactor operates. In the early stages of operation, some of the filtered water from the cleanup subsystem is withdrawn for boric acid recovery. The water is passed through ion-exchange beds and then evaporated. The vapor is condensed and the condensate may be demineralized, filtered, and used to decrease the concentration of boric acid in the reactor coolant. The concentrated boric acid solution remaining

Fig. 11.7. Chemical and volume control system (CVCS) of a PWR.



in the evaporator is filtered and stored in a holding tank for reuse when the core is refueled. In the later stages of operation, when the boric acid concentration in the primary system is fairly low, it is more economical to decrease the boric acid still further by bypassing part of the coolant through a deborating demineralizer in the water cleanup system, as indicated by the dashed lines in Fig. 11.7.

11.100. Liquid wastes are classified into four categories: "clean" wastes, "dirty" wastes, laundry wastes, and turbine building drain wastes, where the terms *clean* and *dirty* refer to the chemical purity and not to the amount of fission products that may be present. Clean wastes are liquids arising from the primary coolant system, which generally have some radioactivity. Such liquids are held in tanks to allow decay of short half-life radionuclides, then filtered, possibly demineralized, and evaporated. Dirty wastes, arising from various floor and other drains, have very low radioactivity. They are generally filtered and evaporated.

11.101. Gaseous wastes from a PWR plant fall into three main categories: primary system gases, secondary system gases, and building ventilation gases. The primary system gases carry most of the volatile radioactive products in the reactor wastes. They include gases vented from the CVCS, and from the clean liquid waste holdup tank, and reactor cover gas. The major radioactive constituents are radioisotopes of krypton, xenon, and iodine as well tritiated water vapor. In addition, the waste gases contain hydrogen, which is used in PWRs to suppress radiolytic decomposition of the water (§7.149 et seq.), and nitrogen which is commonly used as a purge to remove air and prevent the formation of an explosive hydrogen–oxygen mixture.

11.102. In many PWRs, the primary gases are collected in a tank and compressed into one of several decay (or storage) tanks where the gases are held for a period of 60 days or so. The only radionuclides remaining in appreciable amount are then krypton-85 (half-life 10.8 years), xenon-133 (half-life 5.27 days), iodine-131 (half-life 8.05 days), and tritium (half-life 12.3 years). The residual gases are passed through a HEPA (high-efficiency particulate air) filter, which removes small ($>0.3\ \mu$) suspended particles, and is mixed with large volumes of filtered ventilation air before discharge through a roof vent or a stack up to about 65 m in height. In some later PWR plants, the volume of gas to be stored is reduced by mixing the gases with oxygen and passing through a catalytic recombiner to convert the hydrogen gas into water which is condensed and removed.

11.103. Tritiated water, formed in the primary coolant from neutron interaction with boron, is generally allowed to build up until the concentration reaches an unacceptable level. Then part of the liquid is withdrawn, solidified with cement, and buried in a low-level waste repository.

Boiling-Water Reactors

11.104. In a BWR, the nonvolatile fission products that escape from the fuel elements and those formed by neutron activation of materials dissolved or suspended in the coolant water remain in the reactor vessel and recirculation pump systems. The noble gases and iodine pass into the steam and hence to the turbine. Because neither boron nor lithium is present in the coolant, tritium does not present a problem in BWR waste disposal. Part of the tritiated water is discharged with the gaseous effluent and the remainder with the liquid effluent, but the amount is small in each case and well within acceptable limits.

11.105. Liquid radioactive wastes are categorized as high-purity wastes, low-purity wastes, chemical wastes, and laundry wastes. The first category refers to water of high chemical purity and some radioactivity that can be recycled to the reactor. It arises from primary system leakage and is treated by filtration and demineralization. Other wastes are treated by filtration and evaporated, as necessary.

11.106. Three types of gaseous wastes are associated with BWR operation: steam-jet air-ejector wastes, gland-seal effluent, and building ventilation air. The *air-ejector gases* carry off most of the radioactivity since they are derived from the steam formed by water boiling in the reactor core. For the same reason, the radioactivity level is much higher than in the ejector gases from a PWR system. The BWR gases contain large proportions of hydrogen and oxygen produced by radiolysis of the reactor water, so that the total volume is greater than for a PWR of the same thermal power. As a general rule, the gaseous wastes from a BWR also have larger amounts of the noble gases.

11.107. BWR air-ejector gases are passed through a catalytic recombiner in which the hydrogen and oxygen are recombined to form water; after condensation of the water vapor the gases have about one-fifth of their original volume. The residual gases are held up in the 30-minute delay line and then pass on to charcoal beds which serve a dual purpose. The adsorption and subsequent desorption of the noble gases provide a substantial delay in the release of these gases, especially xenon isotopes. Furthermore, the charcoal adsorbs and retains a large proportion of the iodine.

WASTE HEAT MANAGEMENT [14]

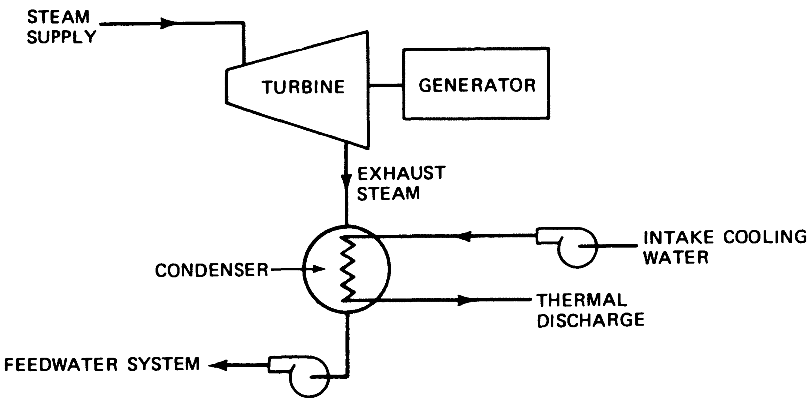
Condenser Cooling Requirements

11.108. In a steam-electric generating plant, only a fraction of the heat supplied, e.g., by burning a fossil fuel or by nuclear fission, is converted into electrical energy. The waste heat remaining in the exhaust steam from the turbine is removed by cooling water in a condenser and, at the same time, the steam is condensed to provide the feed water for the steam generator which may be a heat exchanger (in a PWR) or a boiler (in a BWR or fossil-fuel plant). A simplified representation of the turbine-condenser system of a steam-electric plant is shown in Fig. 11.8. The cooling water carrying the heat removed in the condenser represents the *thermal discharge* (or *thermal effluent*). The additional heat present in the thermal discharge must be dispersed in such a manner as to cause the least possible disturbance of the environment.

11.109. The thermal efficiency of a steam-electric plant is based on the fraction of the heat supplied by the fuel (or heat input) that is converted into electrical energy. A distinction must be made, however, between the gross and the net efficiency. The *gross* efficiency is based on the total generator output and includes the electricity used for operating auxiliary equipment within the plant. The *net* thermal efficiency, on the other hand, is based on the net (or busbar) output of the plant, i.e., the electrical energy available to a utility for sale. For LWRs of current design, the gross efficiency is about 34 percent and the net efficiency roughly 32.5 percent. Energy losses within the plant amount to only a few percent, and so most of the difference between the heat input and the heat equivalent of the total energy generated must be removed by the condenser water.

11.110. Modern coal-fired plants have gross thermal efficiencies of about 42 percent, although there are substantial variations among different units. In such plants, generally about 10 percent of the heat input escapes through the furnace stack, and some 10 percent of the gross output may be required

Fig. 11.8. Simplified schematic representation of turbine exhaust-steam condensing system.



to operate plant auxiliary equipment. As summarized in Table 11.2, the heat removed by the condenser water in a coal-fired plant is approximately two-thirds of that in a typical LWR of the same net electrical capacity. This is why the thermal-discharge problem is of special importance for nuclear power generation. Fast breeder reactors and high-temperature gas-cooled reactors would produce steam at higher temperatures than do LWRs; the thermal discharge would then be comparable to those from modern coal-fired plants.

11.111. In most nuclear power plants, the difference between the inlet and outlet temperatures of the condenser cooling water ranges from about 8 to 20°C, with an average near 12°C. Assuming a temperature increase of 12°C, and bearing in mind that the specific heat of water is about 4.2×10^3 J/kg · K, it follows that for a typical 1000-MW(el) LWR plant, the removal of 2030 MW (or 2030×10^6 J/s) of heat requires a condenser cooling-water flow rate of $(2030 \times 10^6)/(4.2 \times 10^3)(12) \approx 40{,}000$ kg/s or 40 m^3/s (630,000 gal/min).

11.112. The simplest and most economical treatment of the thermal effluent is the "once-through" or open-cycle procedure. The cooling water is drawn from an adjacent natural water body, e.g., a river, lake, ocean, or estuary, passed through the condenser, and then discharged into the same water body. As a result, the temperature of the water is raised, especially in the vicinity of the discharge point. The possibility of using once-through cooling depends on the availability of an adequate water supply and the effect of the temperature increase on the ecological balance of the water body. If the disturbance of the ecosystem is not tolerable, other methods for treating the thermal discharge are necessary; these include the use of cooling ponds or canals and cooling towers.

TABLE 11.2. Typical Thermal Economy For 1000-MW(el) LWR and New Coal-Fired Plants*

	LWR	Coal-Fired Plant
Thermal power input from fuel (MW)	3080	2610
(Btu/h)	10.5×10^9	8.9×10^9
Stack thermal power loss (MW)	—	260
Gross electric power generated (MW)	1050	1100
Plant auxiliary power (MW)	50	100
Net electric power generated (MW)	1000	1000
Thermal discharge rate to condenser (MW)	2030	1250
Thermal discharge rate/net generation rate	2.03	1.25

*Plant internal losses are neglected.

Regulation of Thermal Discharges

11.113. The Federal Water Pollution Control Act (FWPCA) Amendments of 1972 state that all sources of pollution, including condenser coolant water, "shall require application of the best available technology economically achievable." In order to meet this objective the U.S. Environmental Protection Agency (EPA) has proposed regulations for several effluents from steam-electric plants. As far as condenser water is concerned, the EPA concluded that "the best available technology" would be the use of cooling towers or other recirculation systems, e.g., ponds or canals. In effect, the EPA regulations preclude the use of one-through cooling for power plants of more than 500 MW(el) capacity.

11.114. The FWPCA Amendments, however, provide for exemptions from the regulations in special cases. If state requirements can be satisfied, once-through cooling may be permitted when it can be shown that the ecological disturbance would be minimal. Exemptions might also be granted when sufficient land is not available for cooling towers or where salt drift or water vapor plumes from the towers (§11.119 et seq.) would be a serious problem. Nevertheless, it is probable that most future nuclear power plants will have to employ some form of closed-cycle cooling for the condenser water.

11.115. In addition to complying with EPA regulations, as authorized by the FWPCA Amendments, thermal discharges must meet the standards set by individual states in accordance with the Water Quality Act of 1965. These standards, which vary from one state to another and often within the same state, are determined by the aquatic life forms to be protected and the normal seasonal temperatures within a water body. As a general rule, the state specifies maximum temperature increases and maximum permissible temperatures. Some states also specify maximum rates of temperature change to minimize the danger of heat shock or cold shock.

Treatment of the Thermal Discharge

11.116. When it is possible to meet the applicable water quality regulations and standards, once-through cooling would be the preferred method for disposing of the waste heat from the turbine condenser of a power plant. The condenser water is then discharged in such a way as to minimize the thermal impact on the receiving water body. In some situations, a slow discharge at or near the surface of the receiving body may be preferred; the warm discharge water then spreads over a large area. At the other extreme, the water may be forced through jets or diffusers located near the bottom of a flowing stream, so that rapid mixing occurs. Several variations between these extremes are possible. If an adjacent water body is an ocean or a large lake, the intake water is usually drawn from a depth, where the temperature is lower, and is discharged near the surface.

11.117. If the conditions do not permit once-through cooling, some form of closed (or partly closed) cooling cycle must be adopted to deal with the thermal discharge. Cooling ponds (or canals) or cooling towers are used for this purpose. In a closed-cycle system, all the condenser water is discharged and cooled in the pond or tower and the cooled water is withdrawn for reuse. Makeup water is added as required. In a partially closed system, part of the water would be discharged directly to an adjacent water body whereas the remainder would be cooled and reused. In some circumstances, a variable-cycle cooling system may be satisfactory: once-through (direct-discharge) cooling is used in the winter and early spring and a closed (or partially closed) system at other times.

11.118. In a cooling pond or cooling canal, the condenser water is discharged at one end of a large pond (or small lake) or a long canal and is withdrawn at the other end of a flow path. In the course of its passage through the pond or canal, which may take several days, the condenser water is cooled mainly by evaporation and also to some extent by convection and radiation. The cooling efficiency of a pond or canal can be greatly enhanced by pumping the water through nozzles to form sprays. The increase in water surface produced by the sprays increases the rate of heat loss. Depending on the local meteorological conditions, e.g., temperature, humidity, and wind, a pond or canal without sprays will have an area of from 4 to 12 $\times$ 10^6 m^2 (1000 to 3000 acres) for a 1000-MW(el) nuclear plant; with sprays, a much smaller area (one-tenth or less) would be sufficient.

11.119. Cooling towers are of two main types: wet and dry. In wet towers, the condenser discharge water flows down over a packing or "fill" and is broken up into droplets. A current of air is drawn through the fill, either by a powerful fan in mechanical-draft towers, or by a tall (up to 165 m high) chimney-like structure, with a hyperbolic profile, in natural-draft towers. Most of the cooling results from vaporization of the water, but there is also some cooling by convection and radiation. The lowest temperature attainable is approximately the wet-bulb temperature under the existing conditions; it is thus dependent on both the actual air temperature and the relative humidity. The temperature of the water collected at the bottom of the cooling tower is generally 4° to 6°C above the wet-bulb temperature.

11.120. One of the problems of wet cooling towers is the occurrence of "drift," that is, the entrainment of small droplets of water in the air leaving the tower. Drift eliminators, which cause the air to make abrupt turns, help to reduce the amount of drift but do not prevent it completely. The drift is worse for mechanical-draft than for natural-draft towers, partly because of the higher air velocities in the former. The presence of chemicals used to prevent biological fouling, corrosion, and structural deterioration in the tower may make the drift a hazard to plant and animal life on the ground in the downwind direction where the water droplets tend to settle. Drift elimination has been improved to such an extent in recent years, however, that sea water can be used without causing significant environmental damage.

11.121. Continuous evaporation of the water in the tower causes the chemicals, as well as the minerals normally present, to become more and more concentrated. Ultimately, the concentration reaches a point at which scale may deposit on the condenser tubes. In order to prevent this, some of the water is removed either continuously or periodically and discarded as "blowdown" and replaced with fresh water. In a large power plant, the blowdown rate is about 12 to 15 m^3 (3000 to 4000 gal) per minute. The blowdown water may have to be treated for the removal of various chemicals before it can be discharged to a nearby water body.

11.122. The main loss of water from a wet cooling tower is by evaporation of the water and this must be replaced continuously. Including the much smaller losses from drift and blowdown, makeup water is approximately 2.5 to 3 percent or so of the water passing through the condenser; hence, on the basis of the data in §11.111, the average makeup rate, even for a completely closed system, would be roughly 60 to 70 m^3 (15,000 to 17,000 gal) per minute for a 1000-MW(el) nuclear plant.

11.123. Most of the water lost from a wet cooling tower enters the atmosphere as water vapor; under certain conditions this could result in the formation of fog, leading to reduced visibility, and to the deposition of ice on roads and power lines in the vicinity in the winter. The available evidence indicates, however, that such occurrences are limited to a relatively few days of high humidity in cold weather. Natural-draft cooling towers discharge the moist air at much higher elevations than do mechanical-draft towers, and hence they are less likely to produce fog and ice near the ground.

11.124. Dry cooling towers used in steam electric plants are generally described as *air-cooled condensers*. They operate by using ambient air as coolant to condense the exhaust steam either directly or indirectly. In the *direct* condensing cycle, the steam leaving the turbine is passed through a system of finned pipes over which air is drawn by mechanical or natural draft. The air removes heat from the steam so that it condenses, and the condensate is returned as feedwater to the steam generator.

11.125. The most efficient dry tower utilizes an *indirect* condensing cycle known as the Heller system. The steam is condensed by direct contact with jets or sprays of water from previously cooled condensate. Part of the resulting warmed water is returned as feedwater to the steam generator, while the remainder is cooled by passage through a tower containing finned pipes over which air is drawn. The cooled water leaving the tower is recirculated to the condenser sprays.

11.126. In dry cooling towers, heat is removed mainly by convection to the ambient air, and there is no loss of water by evaporation. There is consequently no blowdown (or drift) and, except for leakage, no requirement for makeup water. Hence, steam-electric plants with dry towers can be located in areas where water is scarce. On the other hand, the absence of vaporization means that the lowest temperature attainable theoretically in a dry tower is the actual air temperature, which is higher than the wet-bulb temperature. As a general rule, therefore, the temperatures of the condensate would be higher for a dry tower than for a wet tower; the turbine back-pressure would also be higher. The overall effect would be a decrease in efficiency. In addition, construction and operation costs are expected to be greater for a dry tower than for a wet tower of the same cooling capacity.

11.127. Despite the greater cost of generating electricity with dry cooling towers, estimated to be about 20 percent more than for wet towers and about 27 percent more than for once-through cooling, there may be circumstances in which no other form of cooling is practical. So far, relatively few dry (Heller) towers have been used (or designed) for power plants, the largest with a capacity of 350 MW(el). However, studies are being made of the possibility of designing dry towers suitable for large nuclear power plants.

11.128. Wet-dry (or parallel-path) cooling towers, which use both wet and dry cooling, are attracting interest. The mechanical-draft tower consists of two regions: the upper part is a dry tower whereas the lower part is a wet tower. The condenser discharge water is first cooled by passage through finned tubes in the upper (dry) region and then flows down over the fill in the lower (wet) region for further cooling. Air is drawn by a fan in parallel paths through both dry and wet regions, and the streams are mixed before discharge to the atmosphere. The air flow between dry and wet regions can be adjusted to give preference to one or the other. In winter, when the air is cold, advantage is taken of dry cooling, with a marked decrease in fogging and icing as well as a reduction in water consumption, but in summer preference might be given to wet cooling.

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PROBLEMS

1. Coal contains such carcinogens as cadmium, arsenic, beryllium, nickel, and chromium. When coal is burned, these are released and, if not removed from the combustion products, would eventually be deposited on the ground.

Then, through various pathways, a certain fraction would be ingested by humans. Thus, the hazard is the product of the toxicity and the probability of ingestion. Making reasonable assumptions, Cohen ("The Nuclear Energy Option," Plenum Press, 1990) showed that the carcinogens produced from a year's operation of a large coal-burning plant could result in about 70 excess deaths.

Now, if all of the plutonium-239 in waste discharged from one large nuclear plant was ingested by people, 2300 cancers would result from the alpha-decay activity according to Cohen's model. We are neglecting here any chemical poisoning effects. However, this waste is not deposited on the ground, as waste from coal plants, but buried deep underground in a suitable package. By making some pathway assumptions, estimate the likely excess cancers from the buried plutonium. As a guide, it is known that in one year, about 3 parts per billion of uranium are leached from uranium contained in rock by groundwater in a typical aquifer.

2. Modeling of the Greenhouse effect on global climate usually involves both positive and negative feedbacks. An example of a positive feedback is that initiated by slight melting of polar ice as a result of some global warming. Since the ice is a better reflector than the open sea, more sunlight is absorbed by the sea, thus melting more polar ice, and so on. Now consider the following:
   (*a*) A little heating of the oceans generates more clouds, which can lead to both positive and negative feedback effects
   (*b*) An increase in atmospheric carbon dioxide increases plant growth.
   (*c*) The effect of gas absorption in surface waters.
   (*d*) The effect of climate changes on river runoff.

   Discuss the feedback effects associated with each of the above.

3. Ten years after discharge from a reactor, major heat-producing isotopes in spent fuel are cesium-137 and strontium-90. With the aid of the following data, determine the power generation from each isotope in kW/tonne:

Isotope	Activity at Discharge (Ci/tonne)	Effective Beta Energy per Decay (MeV)	Half-life (yr)
Sr-90	77,600	0.221	27.7
Cs-137	108,000	0.276	30

   In addition to beta emission, a 0.66-MeV gamma photon is emitted in 85 percent of the cesium-137 disintegrations.

4. Consider a single 1000-MW(el) PWR beginning full-power operation in January 1985. If an average discharge burnup of 33000 MW(th)d/tonne is assumed, how many tonnes of spent fuel will have been discharged by 2025?

5. One option for increasing the capacity of spent fuel storage pools at reactor sites is to install a second tier of racks on top of existing racks. Briefly describe the design considerations and possible problems that require analysis for such a project.

6. The Okla natural reactor in Gabon is estimated to have operated 1.9×10^9 years ago. Estimate (by calculations) the likely enrichment of natural uranium at that time. The radioactive half-life of uranium-235 is 0.7×10^9 years while that of uranium-238 is 4.5×10^9 years.
7. One possible PWR effluent of concern is tritium generated from boron-10 that is used in the primary coolant for chemical shim. Consider a natural boron concentration of 1000 ppm exposed to an average neutron flux of 10^{18} neutrons/s $\cdot$ m^2. Estimate the tritium activity after both 1 month and 1 year of operation. The boron-10 fraction in natural boron is 19.8 percent. Tritium has a half-life of 12.3 years.
8. (*a*) Wet cooling towers are used for a 900-MW(el) (net busbar) PWR for which the gross heat rate is 10,500 Btu/h-kW and the plant auxiliary power is 45 MW. Determine the makeup water rate if the tower blowdown requirement is 25 percent of the evaporative load and 15 percent of the cooling is nonevaporative. (*b*) If the cooling water rise through the condensers is 18°F, what flow rate would be required for once-through cooling? (*c*) Assume that water quality standards for an adjacent river permit a 4°F temperature rise and a 10 percent flow diversion for once-through condenser cooling. What is the minimum river flow rate that would meet these standards?

CHAPTER 12

Nuclear Reactor Safety and Regulation

INTRODUCTION

Technological Risk and Public Perception [1]

12.1. We introduce this chapter, which is devoted to those aspects of nuclear reactor engineering concerned with system safety, with a few thoughts on some nontechnical considerations that are relevant. First, there is the matter of terminology. The word *safety* is generally defined as "freedom from danger or hazard," while *risk* means "the chance of injury or loss." We will examine these terms more carefully later when we discuss the subject of risk analysis. However, mathematically based quantitative definitions of risk are beyond our scope. A problem is that in a technological society, no activity has *zero* risk. Thus, there cannot be absolute safety and the two terms are related to one another, with *low risk* meaning the same as *high level of safety*. However, psychologically, we tend to be more comfortable with the term *safety* than with the term *risk*, whether we are discussing nuclear reactors, automobiles, airplanes, or chemicals.

12.2. Unfortunately, in considering public perception, there is an element of fear which is often not at all related to the true risk of a technological activity as developed from statistical data. In general, self-imposed, or *voluntary* risks, tend to be more acceptable than *involuntary* risks. For example, the significant risks of highway travel are normally accepted, whereas isolated cases of chemical residues on some fruits with negligible statistical risks have led to national market withdrawals of such fruits. Intangible factors also affect the public perception of risk. For example, many members of the public fear ionizing radiation as such, even at harmless levels. Another example is the fear by some of using commercial aviation. For a planned technological activity, it is sensible to balance the benefits obtained with an "appropriate" level of risk that can be adjusted during the design process, taking into consideration the costs required. This balance is often known as the "how safe is safe enough?" question. However, public perception (fear) and sometimes associated political pressure can play a role in the "appropriate" balance.

Public Acceptance of Nuclear Power Plants

12.3. The prevention of the release of harmful amounts of radioactivity beyond the site boundaries has become the most important objective in the design and operation of nuclear power plants. Although our industrialized society has become accustomed to other types of hazards, a public sensitivity to the effects of radiation has played a strong role in the acceptability of nuclear power plants. This has been so although the actual risks to public health have been substantially less than those from the burning of fossil fuels [1].

12.4. The present generation of nuclear power plants in the United States and many other countries now have numerous design features that make the risk of harmful accidental release of radioactivity to the public extraordinarily low. These features have evolved not only through conservative design, but also by continued analysis and research on the nature of postulated accidents. The design and operation of nuclear power plants are also strictly regulated to assure public safety. In fact, regulatory requirements have an important role in establishing the *level of safety* or *risk* that is acceptable. Therefore, throughout this chapter, we will mention relevant regulatory considerations, as appropriate.

12.5. In the United States during the 1970s, the ordering of new nuclear power plants ceased, primarily because of economic factors and uncertainty regarding long-time regulatory requirements. A contributing economic factor was the added cost of many new safety-related features. In addition, public acceptance was eroded by the Three-Mile Island accident in 1979, and further reduced by the Chernobyl accident in 1986. Since there will be a need to construct new plants to meet energy requirements, substantial efforts have been made during recent years to develop advanced "next-generation" designs that will have passive safety features and be economical to construct. Improvements in regulatory procedures have likewise received attention.

12.6. Our purpose in this chapter is to describe the many aspects of reactor safety, including accident analysis. Design features can best be understood in light of postulated accident scenarios. Also, with this background, the advantage of future plant passive features can be examined.

Defense in Depth

12.7. *Defense in depth* is the key design principle in nuclear reactor safety. The approach is to provide a series of philosophical and physical layers of protection against the release of radioactivity to the public. In other words, we have one barrier which, if breached, is "backed up" by a second barrier. The second barrier is backed up by a third barrier, and so on.

12.8. The first barrier, or goal, is philosophical, i.e., to prevent accidents from happening. Should an accident happen, the next layer of defense is to provide various countermeasures, in a sequential manner, to control the accident. In our defense, we then provide a series of physical barriers and further countermeasures to confine fission products that might be released. Finally, a containment structure which encloses the nuclear portion of the plant is provided "just in case" there is need for further backup. We will examine these principles in the following sections.

ACCIDENT PREVENTION

Introduction

12.9. The first goal in reactor safety is to prevent accidents from occurring. This goal has two aspects. First, the reactor system needs to be designed, constructed, and operated so that the chances of a malfunction or operational error are very small. Since some equipment failures and operational mistakes are inevitable during the service lifetime of such a complex system as a nuclear power plant, the second aspect of the prevention goal is to provide "self-healing" features that will cope with such incidents.

12.10. The attainment of a reliable system by conservative design requires anticipating possible modes of failure and making provision for them, meeting demanding quality standards, and adhering to applicable regulatory requirements. Details of system design are beyond the scope of this treatment. However, we will examine some accident scenarios, the prevention of which must be demonstrated as part of the licensing procedure. The need for quality assurance is also enforced as a regulatory requirement.

Quality Assurance: Codes and Standards [2]

12.11. In a broad sense, the term *quality assurance* includes all actions necessary to provide adequate confidence that a component, structure, or system will perform satisfactorily in service. Engineering codes and standards are an important aspect of quality assurance, since they represent the recognized practice for assuring acceptable levels of quality and performance in materials and components. A number of preexisting codes, such as the ASME Boiler and Pressure Vessel Code (§7.26), have been improved, supplemented, or completely revised to satisfy requirements of the nuclear industry.

12.12. Standards have played a vital role in the practice of all branches of engineering for many years. These are generally developed by professional society committees and represent a consensus for acceptable procedures, quality requirements, or performance criteria. Often, uniform testing and evaluation methods are prescribed. Such development is coordinated by the American National Standards Institute (ANSI) (§8.42).

12.13. Hundreds of standards are relevant to the practice of nuclear engineering. These cover all aspects of nuclear power plant design, construction, equipment performance, and instrumentation, as well as the manufacturing of nuclear fuels. Many standards also deal with computer codes and information transfer. Since standards document accepted practice, it is important for a nuclear reactor engineer to become familiar with those standards that are relevant to a given professional assignment. These are generally available in technical libraries or through ANSI.

12.14. Special code requirements are established for *nuclear safety grade* components, which are considered vital to plant safety. When such components such as valves are manufactured to meet these stringent requirements, they may be labeled with an "N" stamp. Categorization into classes, depending upon safety significance, with differing requirements for quality assurance and in-service inspections is as follows:

Safety Class 1. This, the most vital category applies to components of the primary coolant system, whose failure would cause a major coolant loss.

Safety Class 2. In this category, we have structures and components that are required to fulfill a safety function such as shutting down the reactor, cooling other safety systems, and controlling the release of radioactivity.

Safety Class 3. This applies to systems whose failure would allow release to the environment of gaseous radioactivity that would normally be held for decay within the plant.

Nonsafety grade components must meet "high-quality industrial standards." Since a significant additional expense is associated with safety grade components, there has been in the past a design incentive to specify such components only for essential safety functions.

12.15. Many standards have been incorporated into Federal Regulations and Regulatory Guides issued by the U.S. Nuclear Regulatory Commission (NRC). General quality assurance criteria are specified in Title 10 of the Code of Federal Regulations, Part 50, Appendix B (10 CFR 50). Specific quality assurance requirements appear in the Regulatory Guide dealing with the particular topic considered.

12.16. There are five major aspects of a quality assurance program:

1. The actual formulation of the program itself includes specific detailed policies and procedures.
2. Should design be required, quality control of the necessary practices is needed.
3. Component and material procurement activities require the enforcement of quality assurance requirements.
4. Inspection and test procedures are prescribed to assure that all specifications are met.
5. Auditing and record-keeping procedures are provided to assure adherence to requirements.

Redundancy and Diversity

12.17. The term *redundancy* refers to the use of two or more similar systems in parallel, so that the failure of one will not affect the plant operation. Redundant components and systems are commonly employed in nuclear power plants. They are of special importance in systems, such as instrumentation, shutdown controls, and emergency cooling, upon which safety depends.

12.18. The components of a redundant system could all be rendered inoperative by a *common-mode failure*, that is, when one failure leads to another or a number of failures result from a single cause. For example, duplicate components of the same design could conceivably fail simultaneously when subjected to the same stress. Many potential common-mode failures can be foreseen and appropriate steps taken to circumvent them. In some cases, however, they are unpredictable and are revealed only after they have occurred. One way to minimize common-mode failure is by *diversity*, that is, by the use of two or more independent and different methods for achieving the same result, e.g., reactor shutdown in an emergency.

12.19. The electric power supply system for a nuclear plant provides an illustration of redundancy and diversity. Instruments are operated by direct current which is available from two independent storage batteries. The alternating current required to operate pumps, valves, and air blowers is normally supplied by the plant's generator connected to two separate bus-bar sets. In addition, two independent offsite power sources are available for use when the plant is shut down or the generator is not operating. If all the onsite and offsite electrical power should fail ("station blackout"), alternating current would be supplied by onsite diesel generators. The plant might then be without power for half a minute; this fact is taken into consideration in determining the conditions for safe operation.

Inherent Reactor Stability

12.20. In selecting a safe reactor design, an essential requirement is that the concept should provide inherent stability against an increase in reactivity. This can be realized if the reactor has a quick-acting negative temperature (or power) coefficient of reactivity. As seen in Chapter 5, there is then a self-limiting effect on disturbances that lead to an increase in temperature (or power level). This self-limiting feature is generally the result of the fuel Doppler coefficient, although in water-cooled reactors the expansion of the coolant-moderator contributes to the overall negative coefficient. However, the moderator contribution is delayed because of the time required for heat transfer from the fuel to the moderator (§5.92).

12.21. It should be understood that whereas a negative temperature (or power) coefficient is a requirement for reactor safety, it does not guarantee safety. If there were a sudden increase in reactivity, as discussed in §5.149 et seq., the power excursion would be terminated automatically after a finite time because of the negative temperature coefficient. But during this time the thermal power, and hence the fuel temperature, may have risen to such a high level that the fuel rods would suffer damage. This aspect of safety is taken into consideration in reactor system design. Also, should the system have a positive coolant void coefficient, as was the case in the Chernobyl 4 reactor (§12.187), the danger would be exacerbated.

Reactor Protection System

12.22. Nuclear power reactors are designed to produce heat to satisfy the demand for steam by a turbine-generator, up to a specified limit. The reactor control system, with its automatic and manual controls, serves to maintain safe operating conditions as the demand is varied (§5.185 et seq.). Because excess cooling capability is provided in the design of the reactor system, an overpower equal to about 118 percent (in a PWR) or 120 percent (in a BWR) of the rated (or design) power can be tolerated without causing damage to the fuel rods. If the thermal power should exceed the limiting value or if other abnormal conditions which might endanger the system should arise, the reactor protection system would cause reactor trip (or "scram"), as described in Chapter 5.

12.23. In reactor operations, the term *transient* describes, in general, any significant deviation from the normal value of one or more of the important operating parameters, e.g., system temperatures and pressures, thermal power level, coolant flow rate, turbine trip, equipment failure, etc. If the transient is a minor one, within the permissible operating limits of the system, the controls will be adjusted automatically to compensate for the deviation. A severe transient, however, will activate the reactor protection system.

12.24. The purpose of the protection system is to shut the reactor down and maintain it in a safe condition in the event of a system transient or malfunction that might cause damage to the core, most likely from overheating. The protection system includes a wide variety of instruments for measuring operating variables and other characteristics of the overall nuclear plant system. If the instruments indicate a transient that cannot be corrected immediately by the control system, the reactor is shut down automatically by the protection system. In addition, the reactor operator can cause an independent (manual) trip if there are indications that an unsafe condition may be developing.

12.25. When a reactor trip signal is received in a PWR, the electromagnetic clutches holding up the control rods are deenergized by an automatic cutoff of electric power. The rods then drop into the reactor core. Borated water (boric acid solution) can be injected from the chemical and volume control system or CVCS (§11.99) by manual action to provide a backup to the control rods if required. In a BWR, a rapid shutdown is achieved by forcing the control rods up into the core by hydrostatic pressure; at the same time, power to the recirculation pumps is cut off. The reactivity in a BWR can also be decreased by injection of an aqueous solution of sodium pentaborate.

12.26. An essential requirement of the reactor protection system is that it must not fail when needed; on the other hand, an error in the instrumentation or other malfunction with the system should not cause an unnecessary ("false") reactor trip. In order to avoid such false trips, three or more redundant channels, consisting of detector and actuator, are used to monitor operating variables. A reactor trip will occur only when two or more channels call for action simultaneously. The availability of several independent channels permits regular testings of the channels, one at a time, without impairing the effectiveness of the protection system.

Reactor Trip Signals

12.27. Some of the signals that would cause actuation of the protection system were mentioned in Chapter 5. A more complete listing is given here for water-cooled reactors; unless otherwise indicated, the trip signals apply to both PWRs and BWRs.

1. Rapid increase in the neutron flux during startup, resulting in a too rapid rise in the thermal power
2. High neutron flux during power operation, indicating an overpower above the permissible level
3. Abnormal reactor system temperature or pressure
4. Loss (or decrease) of coolant flow, e.g., from a pump failure
5. High steam flow, e.g., from a break in a steam line
6. Closure of a main steam isolation valve, especially in a BWR (see item 12)
7. Turbine-generator trip, e.g., from a loss of load
8. Loss of power supply for instruments (dc) or for pumps, valves, etc. (ac)
9. High water level in the pressurizer (in a PWR)
10. Low water level in the reactor vessel (in a BWR)
11. Low feedwater flow or low water level in a PWR steam generator
12. High radioactivity in the steam from a BWR

Shutdown Cooling

12.28. Although the reactor shutdown cooling system is not generally regarded as a component of the protection system, shutdown cooling is nevertheless an essential aspect of reactor protection. When a reactor is shut down, either deliberately or in response to a severe transient, the self-sustaining fission chain reaction is terminated but a considerable amount of sensible (or stored) heat is still present in the fuel rods. Furthermore, heat continues to be generated by decay of the fission products and (for a short time) by fissions caused by delayed neutrons. Hence, cooling of the reactor core must be maintained for many days after shutdown. The sensible heat and the delayed fission heat are removed within about half a minute and then only the decay heat determines the cooling requirements. The thermal power from this source is initially about 7 percent of the operating power of the reactor at shutdown, assuming the reactor has been operating for a substantial time. The decay power decreases to about 1.3 percent after an hour, 0.4 percent after a day, and 0.2 percent at the end of a week (see Fig. 2.33).

12.29. If the normal heat removal system is still operative when the reactor is tripped, cooling will, of course, be adequate. Steam, produced in the steam generator of a PWR or in the reactor vessel of a BWR, bypasses the turbine and goes directly to the condenser. The condensate then returns to the steam generator (in a PWR) or to the reactor vessel (in a BWR) in the usual way. When the system temperature and pressure have decreased to a sufficient extent, the cooling function is transferred to the residual-heat removal (or shutdown cooling) system which circulates primary system coolant through the reactor vessel and independent heat exchangers cooled by a separate service water supply system. This service water is obtained from the so-called ultimate heat sink which usually also provides the condenser cooling water. Since the heat sink is required for safe emergency shutdown of the reactor and subsequent dissipation of the residual heat, the system must be capable of operating for at least 30 days even if the most severe natural phenomenon expected at the plant site should occur.*

12.30. In some situations, the reactor would be tripped and isolation valves in the steam supply would close automatically to prevent the escape of possibly radioactive steam to the environment. The normal heat removal and condenser system would then not be available for cooling the fuel. In a PWR, a large condensate tank can provide an auxiliary supply of feed water to the steam generators to permit reactor cooldown for about 8 hours. By this time, the conditions would be suitable for the residual-heat removal system to function. In a BWR, a pressure-relief valve permits controlled release of steam to the pressure-suppression pool where it would be condensed (§12.51). The level of the water in the reactor vessel is then maintained by the reactor core isolation system which is supplied by water from a condensate storage tank. Finally, the residual-heat removal system can be actuated.

12.31. If electric power is lost when the reactor is tripped, so that the pumps cannot operate, overheating of the fuel can be prevented by releasing steam from the safety valves. This can be continued as long as feed water is available (from the auxiliary system) to the steam generators in a PWR or to the primary system in a BWR. In the meantime, startup of the emergency (diesel) generators will permit operation of the essential pumps, pending the restoration of offsite power.

ENGINEERED SAFETY FEATURES [3]

Introduction

12.32. The purpose of the engineered safety features is to prevent or limit the escape of radioactivity to the environment in cases of a highly

*The NRC recommends that, unless a guaranteed and reliable source of water is available, e.g., an ocean, large river, or large lake, the ultimate heat sink should consist of two separate sources, such as two independent cooling towers or spray ponds, or a tower and a spray pond, or one of these together with some other water source.

unlikely transient or accident that is too severe to be accommodated by the reactor protection system alone. The major engineered safety features are: (1) the emergency core-cooling system to supply water to the reactor core in the event of a loss-of-coolant accident, (2) the containment vessel (or structure) to provide a barrier to the escape to the environment of radioactivity that might be released from the reactor core, (3) the cleanup system for removing part of the radioactivity and heat that may be present in the containment atmosphere, and (4) hydrogen control to prevent formation of an explosive hydrogen-oxygen (air) mixture in the containment.

The Emergency Core-Cooling System

12.33. If a substantial break should occur in the primary coolant circuits of a water-cooled reactor, the system pressure would drop and the emergency core-cooling system (ECCS) would become operative. This system consists of a number of independent subsystems which would be actuated in sequence as depressurization proceeds. The subsystems introduce water into the core to provide cooling when the flow of primary coolant is significantly decreased or lost. The water is supplied from different sources, with larger volumes becoming available to flood the core during the later stages of depressurization. Redundancy of equipment and flow paths is provided to assure reliability of operation. Since the primary cooling systems are different in PWRs an BWRs, so also are the ECCS subsystems. There are also variations among PWRs fabricated by different vendors and also among BWRs of different designs. The following descriptions may be regarded, however, as being fairly typical of current practice.

Pressurized-water reactors

12.34. The *high-pressure injection system* (HPIS) comes into operation when the reactor system pressure suffers a moderate drop, from the normal operating pressure of 15.5 MPa to about 11 MPa. Such a decrease could occur as the result of a small break in the primary coolant circuit of a PWR or if, following an overpressure condition, the pressurizer relief valve failed to close after the normal pressure had been restored (§13.16). The HPIS would then rapidly compensate for the loss of coolant (water or steam). This ECCS subsystem, which utilizes the same pumps as the CVCS, injects borated water into the reactor coolant inlet lines (cold legs). A slow-acting backup system may be included to inject borated water into the reactor vessel outlets (hot legs).

12.35. In the event of a large break, the system pressure would drop rapidly and then another ECCS subsystem, the *accumulator injection system*, would be actuated. The accumulators are two or more independent tanks containing cool borated water stored under nitrogen gas at a pressure of about 1.4 to 4.1 MPa, according to the system design. The tanks are connected through check valves to the reactor cold legs or sometimes directly into the upper part of the downcomer system of the reactor vessel (see Fig. 1.4). As the primary system pressure drops below the pressure in the accumulator, roughly 20 to 25 s after a large break, the check valves would open automatically and borated water would be rapidly injected into the reactor vessel. These accumulator tanks, one for each coolant loop, constitute a "passive" system since it can function without pumps which require electric power.

12.36. As the pressure is reduced further, a *low-pressure injection system* (LPIS), which utilizes the pumps and heat exchangers of the auxiliary cooling (residual-heat removal) system, is actuated. This (like the HPIS) is an "active" system, which requires electrically driven pumps. If the offsite power should fail, there would be ample time (30 s from the time of the major break in the coolant line) for the emergency diesel generators to start. Details of the systems vary among reactor vendors, but generally the design provides for the water to be drawn initially from the refueling storage water tank* followed, if necessary, by water from a sump in the containment structure.

12.37. A typical flow scheme for the ECCS components of a PWR is shown in Fig. 12.1. In this particular case, the HPIS circuit includes a tank of concentrated boric acid solution. The contents of this tank are injected into the reactor vessel as the solution is displaced by water forced in by the CVCS pumps. The safety injection pump is a component of the HPIS backup system.

12.38. The most severe design basis accident (§12.73) for a PWR is taken to be a major break in one of the cold legs of the primary coolant system. In order to establish a conservative design for the ECCS, it is usually assumed that all the water injected into the broken loop will be lost out of the break. Only water injected into the other loops is regarded as being available for cooling the reactor core.

Boiling-water reactors

12.39. The design basis loss-of-coolant accident postulated for a BWR is a large break in the intake line to one of the recirculation pumps, i.e., in the line between the reactor vessel and the pump (see Fig. 1.5). Water and steam would escape from the break, and the system pressure would drop. However, the depressurization rate is less than in a PWR (cf. §12.87).

12.40. In more recent BWRs, the ECCS consists of three subsystems designed to be effective over the full range of potential pipe rupture sizes.

*This tank of borated water is normally used for temporary storage of spent fuel elements during refueling operations.

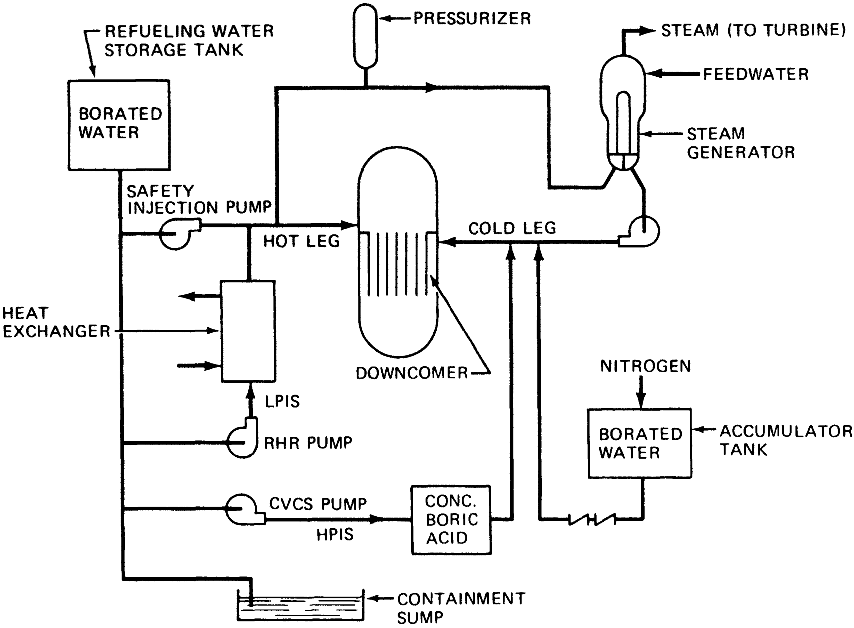

Fig. 12.1. Emergency core-cooling system of a PWR.

Redundancy of components assures a high level of operational reliability. A high-pressure core-spray system (HPCS), using a single pump, forces water through a sprayer above and around the periphery of the reactor core through nozzles. This system can function from the full operating pressure of the reactor (7.24 MPa) downward. A low-pressure core-spray system (LPCS), also with a single pump, provides protection against large breaks that would cause more rapid depressurization of the reactor vessel. Finally, a low-pressure injection system (LPIS), consisting of three separate pumping subsystems, can deliver a large volume of water to refill the vessel once it is depressurized.

12.41. The various ECCS subsystems would be actuated upon receipt of system pressure signals. An essentially continuous supply of water is available for all the subsystems from the pressure-suppression pool (§12.52). This pool would also collect water and steam lost from the reactor vessel through a break. An additional safeguard is an automatic relief valve to vent steam to the suppression pool. This would reduce the system pressure in case of small breaks so that the LPCS and LPIS could become effective. A schematic flow scheme of the ECCS subsystems for a BWR is given in Fig. 12.2.

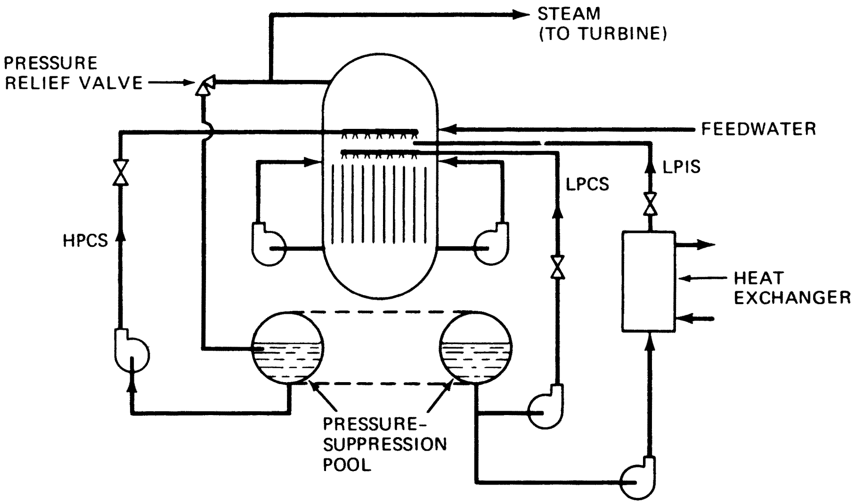

Fig. 12.2. Emergency core-cooling system of a BWR.

Passive ECCS

12.42. So-called "next-generation" reactor plants, to be described in Chapter 15, feature emergency cooling provided by passive means. For example, in a gravity-fed system used to flood the core, cooling will be available even in the event of loss of station power, thus providing some additional safety backup.

Containment Systems [4]

12.43. A containment structure enclosing the reactor primary system acts as the final backup barrier to fission product release to the environment in the defense-in-depth design of the reactor safety systems. Although containment design in LWRs is traditionally based on holding the pressure resulting from the release of the primary coolant in a loss-of coolant accident (LOCA) (§12.37) and to withstand the impact of internally generated missiles, margins are such that substantially higher pressures can be accommodated before failure. As we shall see later (§12.95), additional challenges are provided by so-called severe accidents in which the core is degraded. Fission product transport into the containment can be modeled analytically with various uncertainties involved. Therefore, it is useful to view the containment as the essential physical envelope which will indeed prevent the fission products from entering the environment despite the modeling uncertainties that may be present.*

12.44. Containments for PWRs are large cylindrical or spherical pressure vessels designed for pressures on the order of 345 kPa(g) (50 psig). Since the PWR containment must be large enough anyway to enclose all of the components of the entire primary system, which may include four steam generators and a pressurizer in addition to the reactor pressure vessel itself, it is practical to achieve the necessary pressure reduction merely by allowing steam formed by coolant flashing to vent into the large volume. On the other hand, the BWR primary system is contained essentially within the pressure vessel with much less building volume needed for housing it. Therefore, BWRs use means to condense the released steam in pressure suppression pools so that the containment building need not be much larger than that required to hold the system.

Pressurized water reactors

12.45. Containment structures for PWRs vary to some extent from plant to plant, but they are commonly cylindrical (roughly 37 m diameter) with a domed top (overall height some 61 m). They are usually made of reinforced concrete, about 1.07 m thick, with an internal steel liner, roughly 38 mm thick. As shown in Fig. 12.3, the entire primary coolant system is enclosed as well as elevated injection tanks.

12.46. A spherical design is shown in Fig. 12.4. Compared with a cylindrical design of equivalent free volume, the spherical configuration provides additional operating floor area and efficient placement of auxiliary and maintenance activities. An in-containment refueling water storage tank (IRWST) shown in the figure provides water for both safety injection and severe accident core debris cooling. Sphere diameters vary from about 40 m for a 2600-MW(t) plant to about 60 m for a plant rated at 3800 MW(t). Corresponding free volumes are about 57,000 and 96,000 m^3, respectively.

12.47. If there were to be a complete loss of coolant, nearly all the heat content of the coolant and fuel prior to the accident would be released to the containment atmosphere. The volume and strength of the structure are such that it can withstand the maximum containment temperature and pressure that would be expected from the steam produced by the flashing of all the water in the primary circuit and from the effects of the ECCS. Typically, the calculated maximum pressure would be about 280 kPa(g); the containment structure is thus designed to withstand 310 kPa(g) and is tested at 350 kPa(g). At the design pressure, the leakage rate should not

*Some severe accident scenarios lead to containment failure (§12.99). However, containment designs for future reactors are likely to include severe accident mitigation features.

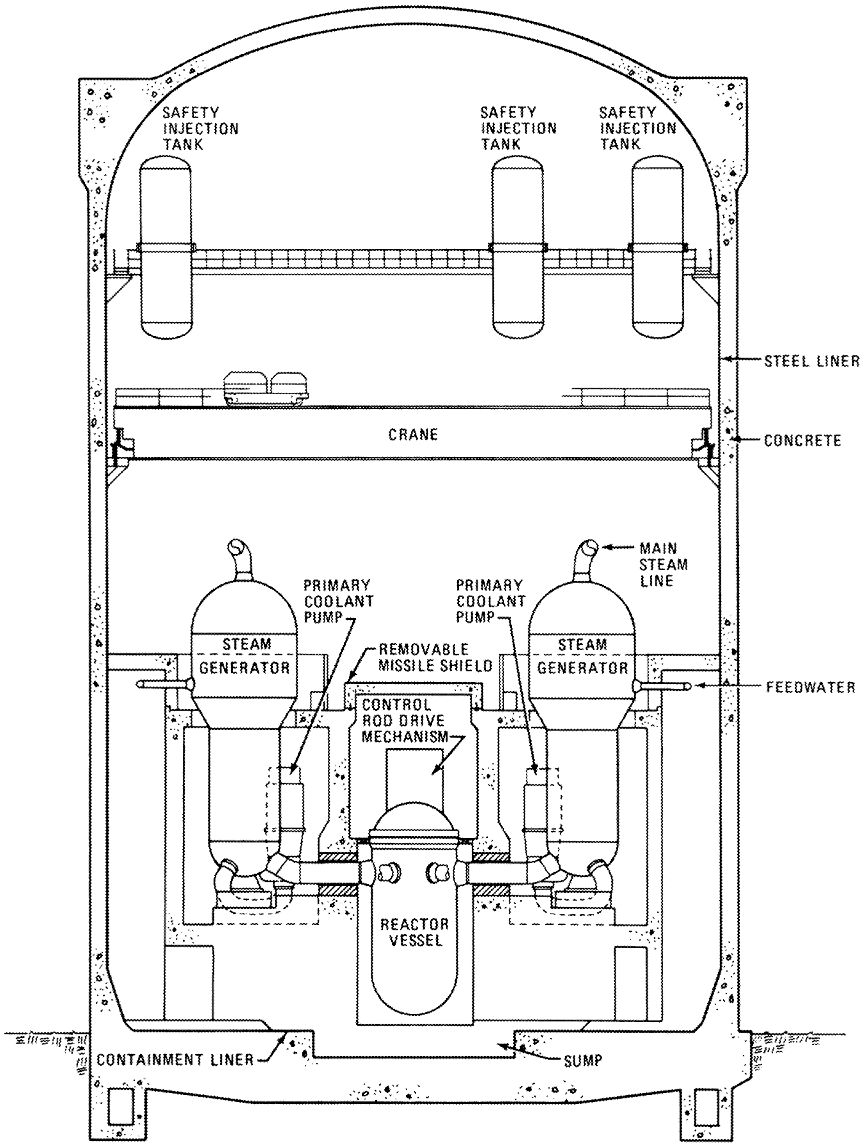

Fig. 12.3. Typical PWR containment structure.

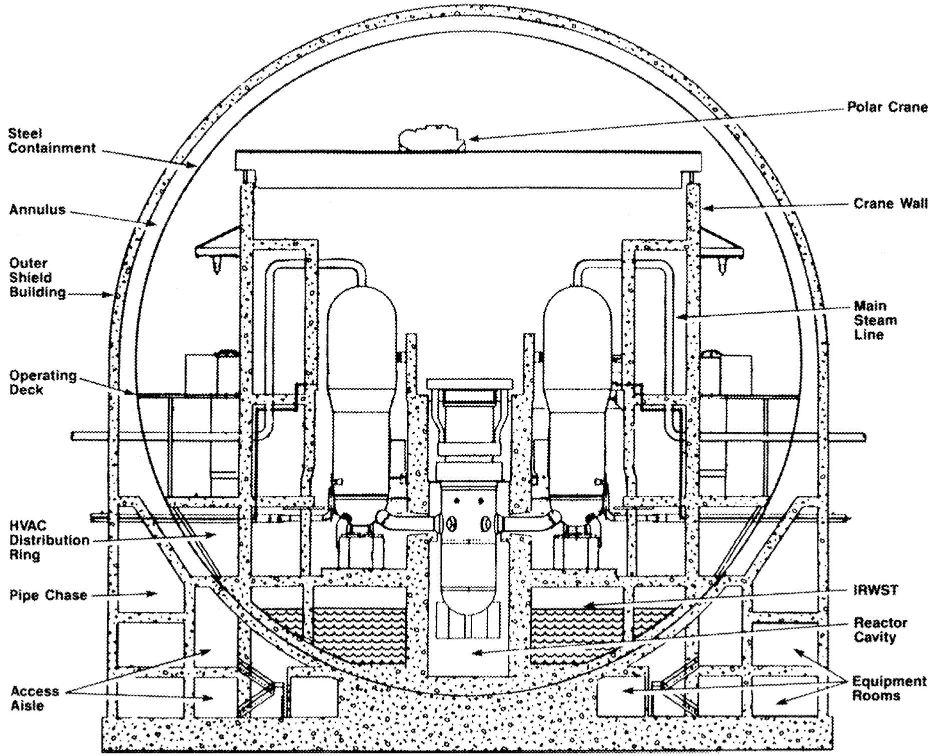

Fig. 12.4. Elevation view of System 80+™ spherical containment (© 1989 Combustion Engineering, Inc.).

exceed 0.1 percent of the containment volume per day. Spherical containments may be designed for pressures as high as 500 kPa(g).

Example 12.1. Consider a large PWR with a total primary system coolant inventory of 350,000 kg at 15.5 MPa(a) and 320°C. If the containment free volume is 57,000 m^3, make an initial *approximation* of the magnitude of the pressure load on the containment following a loss-of-coolant accident.

As a first step, let us determine the amount of steam produced if the final pressure is "guessed" as 200 kPa(a) (about 2 atm). From the steam tables, the specific enthalpy of the subcooled coolant water prior to blowdown is 1462 kJ/kg. At 200 kPa, the specific enthalpy of the saturated liquid is 505 kJ/kg and the latent heat of vaporization is 2202 kJ/kg.

The steam produced would then be approximately

$$\frac{350{,}000(1462 - 505)}{2202} = 152{,}000 \text{ kg}.$$

The corresponding specific volume is

$$\frac{57{,}000}{152{,}000} = 0.375 \text{ m}^3/\text{kg}.$$

This corresponds to a pressure of 500 kPa(a), which we use as a second trial. Then, using new properties, the steam produced would be

$$\frac{350{,}000(1462 - 640)}{1922} = 150{,}000 \text{ kg}.$$

The new specific volume is then 0.38 m^3/kg, which is relatively unchanged. Considering that the total pressure is made up of the partial pressures of air and steam, the *gage* pressure is *roughly* 5 atm. However, other effects must be considered in an actual calculation. Thermal energy absorbed by the components, building, and injected water must be considered as well as the heat input by fission product decay. A pressure in the neighborhood of 3 atm is then likely to result.

12.48. In order to cool the containment atmosphere and reduce the pressure by condensing part of the steam after a loss-of-coolant accident, water would be sprayed through nozzles near the top of the structure. The water, which collects in the containment sump, can be recirculated through the heat exchangers of the residual-heat removal system (§12.29) to provide continuous cooling of the containment atmosphere.

12.49. The containment sprays also serve to remove some of the radioactivity from the atmosphere. Sodium hydroxide or alkaline sodium thiosulfate in the water facilitates the removal of radioiodines which are generally the determining factor in the environmental hazard that would result from a large radioactive release (§12.160 et seq.). In some PWR installations, the radioactivity level in the containment atmosphere would be reduced by using blowers to circulate the air through iodine absorbers and particulate filters.

12.50. A special type of PWR containment system makes use of an ice condenser to reduce the interior pressure. Instead of the usual steel-lined concrete structure, the steel liner (or shell) is separated from the outer concrete (or shield) building. The annular space between the liner and the concrete, above the level of the reactor vessel, contains cells filled with refrigerated borated ice. In the event of a loss-of-coolant accident, condensation of the released steam by the ice would limit the pressure in the containment. Consequently, the structure may be designed for a pressure of only 69 kPa(g) and it can have a smaller volume than a conventional PWR containment building. Some fission products would also be removed in the condenser. Furthermore, the borated water formed by the melting ice would collect in the containment sump and would be available for core cooling.

Boiling-water reactors

12.51. Containment structure designs for BWRs have been modified somewhat over the years. However, all designs use *pressure suppression*, in which the steam released by the flashing of the coolant in a LOCA is forced into a pool of water and condensed. Generally, designs call for the condensation of 80 to 90 percent of the steam to keep the pressure from exceeding building specifications.

12.52. As shown in Fig. 12.5, an inner concrete cylindrical *drywell*, which encloses the reactor vessel and the recirculation system, acts as a pressure envelope and supports an upper refueling pool. In the event of a LOCA, steam passes through horizontal vents near the bottom of the drywell to an annular *suppression pool*, which also acts as a water source for emergency cooling. The building arrangement is shown in Fig. 12.6. For refueling, the center portion of the pool is drained so that the reactor vessel head will be accessible.

12.53. As a result of the action of the suppression pool, the system can be designed for modest pressures, about 175 kPa(g) for the drywell, and 100 kPa(g) for the suppression chamber. Leakage must be limited to 0.1

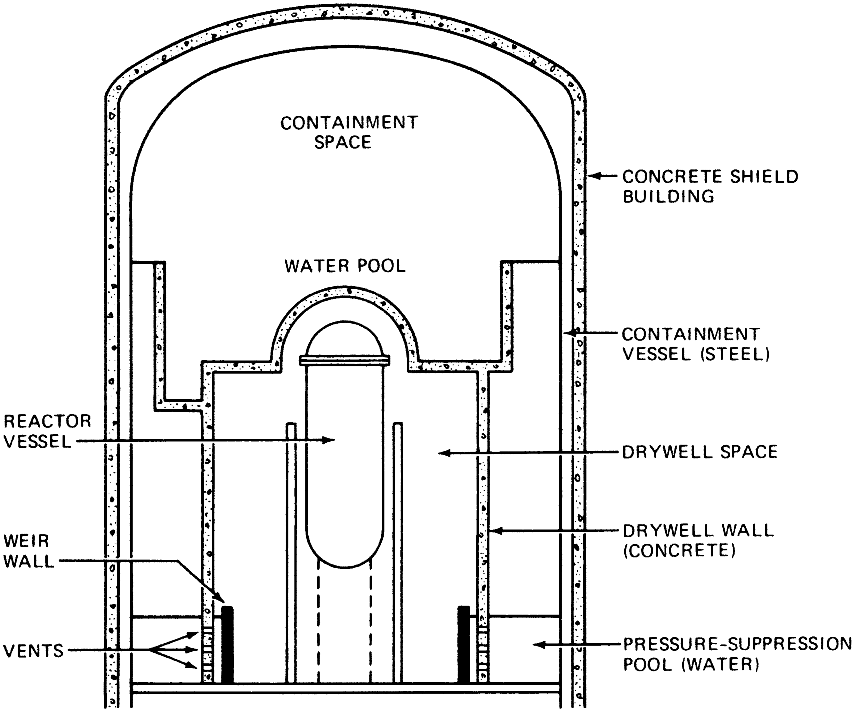

Fig. 12.5. Outline of a BWR containment structure.

percent per day. Typical volumes for the drywell and pool are 8000 m^3 and 4300 m^3, respectively. The suppression chamber volume is about 33,000 m^3.

12.54. A containment building for an advanced large BWR (§13.48) being marketed in the 1990s is shown in Fig. 12.7. The compartmentalized pool system shown in Fig. 12.5 is retained. In all BWRs, the suppression pool removes part of the radioisotopes that may be present in the blow-down, particularly radioiodine. Also, the suppression chamber is provided with various absorbers and particulate filters to remove isotopes that may be released from the pool.

ABNORMAL EVENT ANALYSIS

Categories of Abnormal Events

12.55. In part of the documentation required for reactor licensing, an applicant must show that the plant is designed in such a manner that a

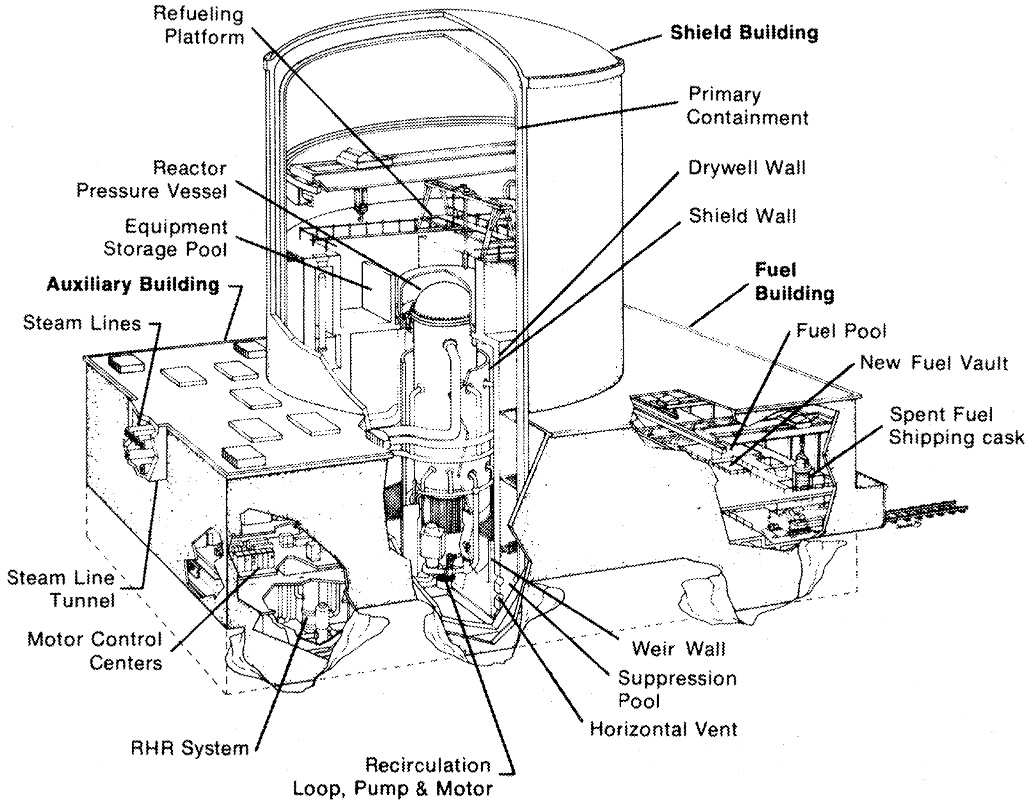

Fig. 12.6. Mark III BWR reactor building, fuel building, and auxiliary building (General Electric Co.).

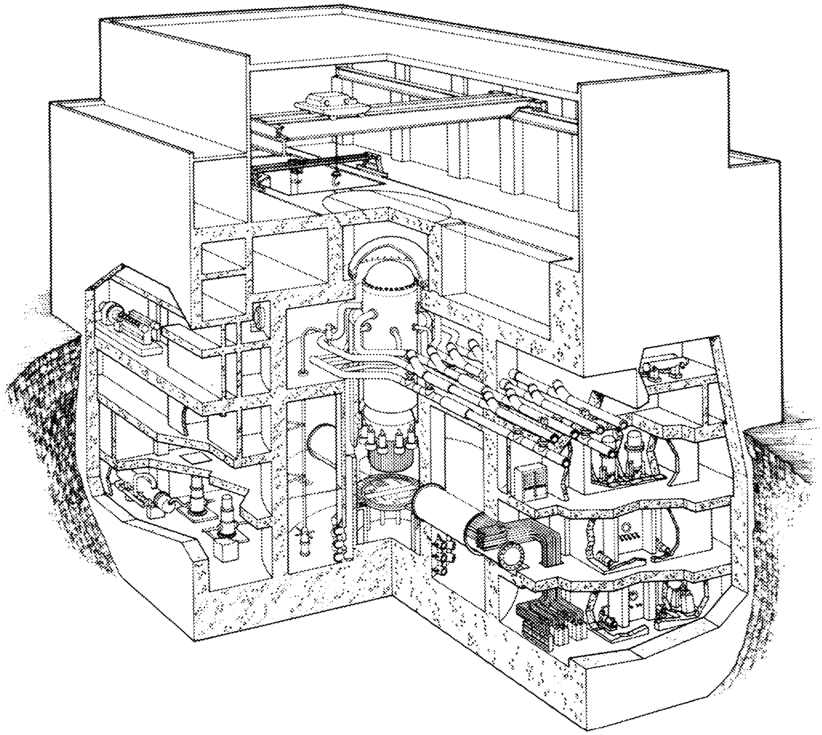

Fig. 12.7. A BWR reactor building (General Electric Co.).

wide range of conceivable abnormal events can be controlled, and that their consequences will not lead to undue risk to the public. For convenience, these events are usually grouped according to likely frequency of occurrence and severity. For example, transients may occur with moderate frequency as a result of operational faults, e.g., human error or minor malfunction of instruments or other equipment. These can be dealt with by the reactor control and protection system. For our purposes, we need consider only events of moderate frequency and events of low probability as examples of those considered.

Events of Moderate Frequency

12.56. Transients caused by operational (and other) occurrences can cause an imbalance between the rate of heat generation in the fuel (or thermal power) and the rate of heat removal by the coolant. If the former exceeds the latter, there is danger of overheating and hence of damage to the fuel-rod cladding. In many cases, the imbalance can be rectified automatically by the reactor control system without interruption of normal operation. The reactor protection system is designed to trip the reactor if the imbalance is too severe to be handled by the control system. The redundancy and diversity available in the control and protection systems make their total failure highly improbable.

12.57. The relationship between heat generation and heat removal rates may be considered in terms of the maximum heat flux (or heat generation rate) in a coolant channel and the critical heat flux (§9.98) at which some fuel cladding damage may be expected to occur. In order to prevent such damage, the critical flux must be well above the maximum (actual) flux, as depicted in Fig. 9.22, for example. An increase in the thermal power of a reactor will increase the (actual) flux, whereas a decrease in the coolant effectiveness will decrease the critical heat flux. Thus, transients capable of causing either (or both) of these effects will increase the possibility of cladding damage.

Increase in thermal power

12.58. *Coolant Temperature Decrease.* A decrease in the temperature of the coolant entering the reactor can cause an increase in reactivity, and hence in the neutron flux and thermal power, as a consequence of the negative temperature coefficient. In a PWR, this could occur as a result of malfunction in one of several systems. For example, a faulty valve might cause the feedwater rate to a steam generator to increase, or failure of a water preheater could result in a decrease in the feedwater temperature. However, such steam-generator effects in a PWR would tend to produce only a slow change in the primary coolant temperature. Since the feedwater (turbine condensate) in a BWR is pumped directly to the reactor vessel, the response from similar situations would tend to be faster. In each case damage to the fuel cladding would be prevented by a reactor trip actuated by the increase in the neutron flux.

12.59. Inadvertent startup of a pump in an inactive PWR coolant loop could also lead to an increase in reactivity and in the thermal power. Similarly, an accidental speedup in a BWR recirculation pump would result in a power increase. Such transients would be corrected automatically by the control system or the reactor would be tripped.

12.60. *Control Material Removal.* An inadvertent reduction in the control (poison) material in the core is another cause of an unplanned increase in the thermal power. Although the reactor design limits the rate at which control elements can be withdrawn, the consequences of an uncontrolled withdrawal resulting from a malfunction must be considered in the safety analysis. The effects depend on the withdrawal rate, the reactivity worth of the control element (§5.168), and the operating state of the reactor, e.g., startup or full power. During startup, special precautions are taken to prevent a too rapid increase in the reactivity, since this could lead to a potentially unsafe condition (§5.219). Here again, reactor trip would be initiated by the protection system should design limits be exceeded, regardless of the operating state of the reactor.

12.61. In a chemically-shimmed PWR (§5.187), an accidental decrease in the boric acid concentration would lead to a power increase. However, the response of the reactor power to boron dilution is slow, and numerous alarms and other indications would alert the reactor operator to take corrective action should an error be made in the manual dilution operation.

12.62. *System Pressure Increase.* The system pressure in a BWR would increase if a valve in the main steam line should close automatically, e.g., as a result of a sudden decrease in the turbine load (see also §12.65). An increase in pressure would decrease the steam-void volume, thereby producing an increase in reactivity. The thermal power would consequently rise, thus tending to cause a further increase in the system pressure. The reactor pressure-relief valve would then release excess steam to the suppression pool, and the reactor would be tripped before any damage could occur. Radioactivity present in the steam would be removed by the water in the pool and by the air circulation system in the drywell.

Decrease in cooling effectiveness

12.63. A decrease in the effectiveness of the coolant, and a decrease in the critical heat flux, could occur from a variety of causes, including a decrease in the mean flow rate of the coolant through the core, a mismatch between feedwater and steam flows, a decrease in steam demand, or a decrease in the primary system pressure in a PWR.

12.64. Slowdown of one of the primary coolant pumps in a PWR would result in a core temperature rise; the reactor power would then tend to decrease as a result of the negative temperature coefficient. The control elements would be withdrawn automatically in order to restore the power level, and this would aggravate the situation by causing a further increase in the coolant temperature. Hence, should the coolant flow rate fall below a specified value or the coolant temperature become excessive, the protection system would trip the reactor. On the other hand, a decrease in the recirculation rate in a BWR would cause an increase in the steam-void volume and a reduction in the thermal power and steam production rate. With the aid of the pressure-regulating system, the power would then stabilize at a new lower level.

12.65. The control system must regulate the feedwater flow to the steam generators in a PWR and to the reactor vessel in a BWR in accordance with the steam demand (or turbine load). Otherwise, an undesirable change in the water level would result and the reactor would be tripped. Should there be a rapid drop in the steam demand, provisions are made for the steam to bypass the turbine and to flow directly to the condenser. If an isolation valve in the main steam line of a BWR should close automatically, the system pressure would rise leading to the situation described in §12.62. In a PWR, a decrease in steam demand can result in an increase in temperature of the primary coolant water and an increase in the system pressure. The pressure would then be automatically reduced by operation of the pressurizer relief valve.

12.66. Such incidents, should they occur, are so-called *significant operating events* and must be reported to the Nuclear Regulatory Commission. Also, they are reported to and analyzed by the Institute for Nuclear Power Operations (§8.52) which maintains a data base useful for reactor safety improvement.

Events of Low Probability

12.67. As a result of quality assurance programs for reactor components and inspection during operation, the probability of mechanical failures is low. Nevertheless, the consequences of possible failures must be considered in the safety analysis. Among the more important events of low probability in the nuclear steam supply system, which could conceivably cause some fuel cladding damage, are the following: (1) a small-scale break in the reactor coolant circuit, steam-generator feedwater lines, or steam lines, and (2) complete loss of ac power leading to a loss of the normal flow of reactor coolant.

Small pipe breaks

12.68. *Pressurized-Water Reactors.* A small break or crack in one of the primary coolant loops in a PWR would be most serious if it occurred in one of the reactor vessel inlet lines. The reactor would be tripped, but some water would escape and be largely flashed into steam in the containment structure. In order to prevent damage to the fuel cladding, it would be necessary to keep the core covered with water while normal shutdown cooling proceeds. If the break were quite small, the water level could be maintained by operating the pumps in the intact loops at full capacity. Makeup water to compensate for that lost from the break would be provided by the CVCS. However, it should be recognized that should there be a problem in introducing makeup water, it would be necessary to lower the pressure to make use of the passive high-pressure ECCS subsystem.

12.69. For larger breaks in a PWR inlet line, the reactor vessel water level might drop temporarily to an extent that would permit some fuel cladding damage by overheating. As the system pressure is lowered by the escaping water, the high-pressure injection subsystem of the ECCS would operate and keep the core covered. The reactor shutdown cooling system would then be used to decrease the temperature and pressure. Again, the pressure status is important. The high-pressure ECCS subsystem has a limited capacity. If the leak rate is too small to reduce the pressure to the range of the low-pressure ECCS subsystem and the high-pressure capacity is depleted, a cooling problem could occur. Some of these considerations became evident during the course of the Three Mile Island small-leak accident (§12.179).

12.70. A break in a PWR feedwater line, which is within the containment, or in a steam line inside or outside the containment could result in the loss of all the water in that steam generator. The reactor would be tripped and the other steam generators would be able to provide sufficient cooling to permit safe shutdown. Since the radioactivity level in the steam from a PWR is very low, a break in the steam line outside the containment would not constitute a serious hazard.

12.71. *Boiling-Water Reactors.* If a relatively small break should occur in a feedwater or recirculation line, one of the auxiliary cooling systems, e.g., the reactor core isolation cooling system (§12.30), could maintain the water level in the reactor vessel. The reactor would be tripped and then be depressurized and cooled down in the normal manner. For larger breaks or those requiring the main steam-line isolation valves to be closed, water would be supplied from the suppression pool and from the condensate storage tanks. A break in a BWR main steam line would result in reactor trip and closure of the isolation valves to prevent escape of the radioactive steam from the containment. The core would be kept covered with water in the manner just described, and the system temperature and pressure would be decreased in the normal shutdown manner.

Loss-of-flow accident

12.72. Should there be a complete loss of both off-site and on-site power, all pumps would gradually coast to a stop and the result would be a *loss-of-flow* accident. However, within 30 s or so, emergency power should be available from the diesel generators to operate essential pumps. In the meantime, the reactor would have been tripped upon receipt of the loss-of-flow signal, and steam would be automatically dumped from the turbine. Since some onsite power would continue to be produced during the dumping process, the circulation pumps would normally be left connected to the main generator busbar for about 30 s. The circulation during coastdown together with some natural circulation of the coolant would normally be sufficient to prevent the critical heat flux condition being attained following reactor trip.

LICENSING DESIGN BASIS EVALUATION

12.73. The response of the reactor system to a number of postulated transients and accidents must be evaluated as one requirement for licensing. As we will see later, the methods used are intentionally conservative and differ from those to be discussed later (§12.132), that are used to obtain a "realistic" prediction of the response. Such a *licensing design basis evaluation* has evolved from what was commonly referred to once as *design basis accident analysis*. Since our present purpose is to identify the events and not the evaluation methodology, we will use the older term, *design basis accidents*. These are events of such low probability that they should not occur even once in an average reactor plant's lifetime, but which would have serious consequences if they were not controlled. These accidents are analyzed to assure the effectiveness of the engineered safety features and to evaluate the acceptability of the proposed plant site. The most severe design basis accident is considered to be a complete (double-ended) rupture of a large pipe, ranging in diameter from 0.61 to 1.07 m, in the primary coolant circuit of a PWR or a similar break in a recirculation pump intake line of a BWR. This accident is commonly referred to as the *loss-of-coolant accident*; it will be considered in some detail later. Other design basis accidents will first be examined briefly. It should be noted that design basis accidents do not result in core damage if the engineered safety features operate properly. So-called "severe accidents" in which core damage does occur will be discussed later (§12.95).

Control Element Ejection

12.74. The sudden ejection of a control element (or control-rod cluster), as a result of a mechanical failure, is a conceivable but improbable accident. In most situations, the increase in reactivity (and thermal power) would be small and could be handled by the reactor protection system. For example, when a reactor is operating at full power, the control elements are all partly to fully withdrawn. Sudden withdrawal of one element would therefore result in only a minor reactivity excursion, at worst. At the other extreme, when a reactor is in a shutdown state, the reactivity is sufficiently negative to compensate for the ejection of a control element.

12.75. One design basis accident is the ejection of a control element from an operating reactor leading to a power excursion of sufficient magnitude to cause some damage to the fuel cladding. Observations made with test reactors have shown that the amount of damage resulting from the ejection of a control element would be governed mainly by the energy generated as a result of the excursion. This in turn depends on the reactivity worth of the ejected element and the power distribution attained by the remaining control-element pattern. The test data are used in assessing the consequences of control-element ejection accidents. Because of the very low probability of an ejection accident, limited fuel cladding damage is considered an acceptable consequence. For a plant design to be approved, analysis must show that several criteria are met which are intended to ensure that there is little possibility of fuel dispersal in the coolant, gross core lattice distortion, or severe shock waves.

Spent-Fuel Handling Accident

12.76. During the regular refueling operation, about once a year, the head of the reactor vessel is open to the containment. Care must then be taken to avoid damage to the spent fuel rods that would result in the release of fission products, especially iodine vapor and noble gases present in the gap between the fuel pellets and the cladding and in the gas plenum above the pellets (§7.168). Such damage could conceivably arise from overheating caused by inadvertent criticality or by greatly impaired ability to remove decay heat. Core criticality is prevented by detailed administrative procedures for removal of control elements, and critical geometry of the withdrawn fuel-rod assemblies is avoided by proper spacing in storage racks. Adequate cooling for removal of decay heat (and radiation shielding) is provided by keeping the assemblies under water at all times. Furthermore, boric acid is added to the reactor water to ensure subcritical conditions.

12.77. Damage to a spent-fuel assembly could also occur if the assembly were dropped. The use of special gripping and handling devices makes this very improbable. Nevertheless, the Safety Analysis Reports are required to include a calculation of the radioactivity that might be released to the environment if a fuel-rod assembly suffered severe mechanical damage inside the containment vessel. Damage outside the containment might result from the assembly being dropped to the floor of the spent-fuel storage area. However, such an event is extremely unlikely since fuel is transferred from the reactor to the pool under water and not elevated above the pool.

Loss-of-Coolant Accident

12.78. Loss-of-coolant accidents fall into several categories depending on the size of the postulated break in the primary coolant circuit. Smaller breaks, which could lead to minor fuel cladding damage at worst, were considered earlier. For the design basis accident, however, a "guillotine" (or double-ended) break is postulated in one of the cold legs of a PWR or in one of the recirculation pump intake lines of a BWR (§12.73). As a result of such a break, the primary system pressure would drop and almost all the reactor water would be expelled into the containment. The drop in pressure resulting from such a loss-of-coolant accident (LOCA) would actuate the protection system and the reactor would be tripped. The fission chain reaction in the core would thus be terminated, as it would in any case because of the loss of coolant (moderator). Nevertheless, heat would continue to be released at a high rate from the sources mentioned in §12.28. The various ECCS subsystems must then provide sufficient cooling in time to minimize overheating and fuel cladding damage. The steam flow limiters and isolation valves, inside and outside the containment vessel, would close automatically to prevent the spread of possibly contaminated steam.

12.79. The thermal-hydraulic (and other) phenomena are very complex, and studies are being made to provide a better understanding for computer modeling (§12.126 et seq.). These phenomena are somewhat different for PWRs and BWRs and will be outlined separately.

Pressurized-water reactors

12.80. The events occurring within the first minute or two following a design basis LOCA in a PWR are divided roughly into three periods: (1) *blowdown*, in which the coolant would be expelled from the reactor vessel, (2) *refill*, when the emergency coolant water would begin to fill the vessel up to the bottom of the core, and (3) *reflood*, when the water level would rise sufficiently to cool the core.

12.81. *Blowdown*. Blowdown proceeds in two stages, namely, subcooled and saturated. During the initial *subcooled* blowdown stage (§9.93), the primary system pressure drops rapidly to the vapor (or saturation) pressure of the water at the existing temperature (Fig. 12.8). In this stage, decompression may be accompanied by the propagation of pressure waves; analysis must show that the mechanical loading caused by these waves would not damage the system in ways that could constitute a public hazard.

12.82. In the longer, *saturated* blowdown stage, steam voids are formed and the steam–water mixture flows out through the break; this stage would probably continue for roughly 15 to 20 s until the system pressure becomes approximately equal to the containment pressure. The two-phase flow, which is comparable to a steam-water mixture flowing through a pipe (§9.131), is called "choked" flow. It would continue for the period during which the velocity of the mixture is at the maximum sonic velocity. The fuel rods would be cooled to some extent by the steam–water flow, and the cladding temperature would drop for a short time. Subsequently, as the enthalpy of the fluid increases, the critical heat flux would fall below

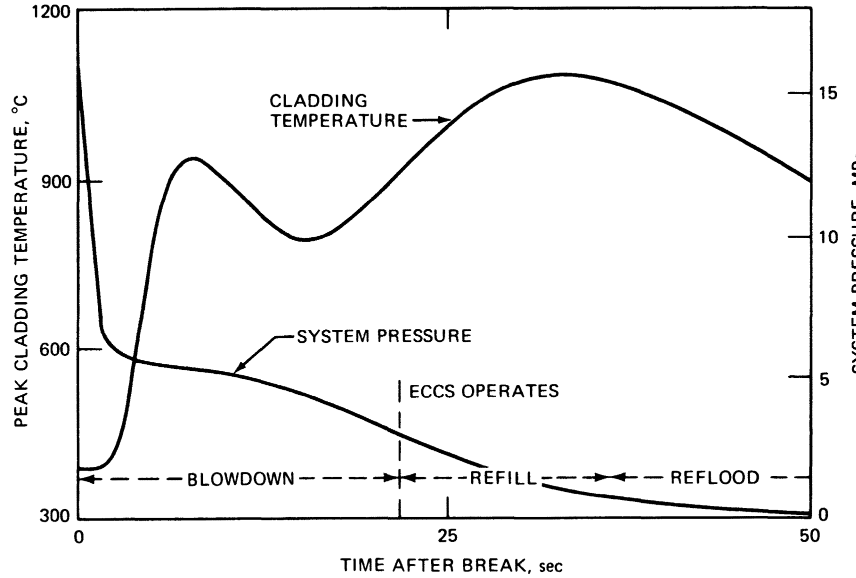

Fig. 12.8 Loss-of-coolant accident in a PWR. (Temperatures, pressures, and times are approximate.)

the maximum (hot channel) flux; there would then be a marked decrease in the heat-transfer coefficient and a corresponding increase in the cladding temperature, as seen in Fig. 12.8.

12.83. As the saturated blowdown proceeds, the fraction of liquid phase in the coolant would decrease until a froth of water and steam remained. The level of the froth would fall, leaving the upper part of the core dry. Heat removal from the core would then be primarily by radiation to the surrounding structure. The temperature of the cladding might then rise to the point (over 980°C) at which failure of some of the hotter rods could occur as a result of the loss of strength of the hot zircaloy. The heat liberated in the zirconium–water reaction would contribute to the temperature increase.

12.84. *Refill.* The decrease in the PWR primary system pressure during blowdown would actuate the ECCS. The accumulators would inject borated water into the reactor vessel through the intact cold legs (or directly into the downcomer); this would provide some cooling of the fuel, but in the initial stage most of the water would probably be expelled as a steam–water mixture through the pipe break.* When blowdown is complete, some

*In a cold-leg break, the desired flow of injected water down the downcomer and up through the core would be resisted by the blowdown flow in the opposite direction. If the resistance is large, *ECC water bypass* could occur, in which the injected fluid flows around the downcomer annulus and out through the pipe break.

of the borated water would begin to collect at the bottom of the reactor vessel and refill it. If off-site electric power is available, the low-pressure injection subsystem of the ECCS will have commenced to pump borated water into the reactor vessel. But if the offsite power fails and there is a delay in starting the emergency generators, the low-pressure system will not operate until late in the refill period. In any case, there is very little cooling of the fuel during the refill phase; such cooling as does occur is mainly by convection to the steam–water mixture.

12.85. *Reflood.* Reflooding of the core would commence when the water level reaches the bottom of the fuel rods. As the water rose, it would come into contact with the hot fuel cladding; the latter would consequently become covered with a film of vapor which the liquid cannot penetrate. A condition of pool film boiling would then exist (§9.89). The temperature near the core bottom, however, would soon drop to the point at which water can penetrate the steam film and nucleate boiling could occur. The temperature of the cladding could then decrease fairly rapidly. The accumulator tanks in a PWR would be empty about a minute after the pipe break, but water from other ECCS sources will permit the core to be recovered, i.e., to complete reflooding, within 2 min or so.

12.86. A problem associated with reflooding is that the pressure of escaping steam would tend to oppose the driving pressure for entry of water into the core during reflood. The phenomenon, called *steam binding*, would tend to decrease considerably the rate of rise of water in the core. In fact, flow oscillations could be set up as a result of the effects of the inertia of the injected water and the compressibility of the remaining vapor in the core. The estimated reflooding time given takes this into account.

Boiling-water reactors

12.87. If a design basis LOCA should occur in a BWR, the recirculation pump in the intact line would continue to function while coasting down, even if off-site power were to be lost. Some water would continue to flow through the reactor core and nucleate boiling could continue; there would be little change in the system pressure and in the fuel cladding temperature (Fig. 12.9). After 5 to 8 s (or so) the rate of water flow through the core would drop off and the temperature would rise.

12.88. During this time, water would be flowing out of the break, and the level in the reactor vessel would fall. When the level of water outside the shroud (see Fig. 1.5) reached that of the break, assumed to be at a recirculation pump intake line below the core bottom, blowdown would occur, at about 10 s after the break. There would then be an escape of steam from the reactor vessel and the pressure would decrease fairly rapidly. As a result, the water in the lower plenum, i.e., in the vessel below

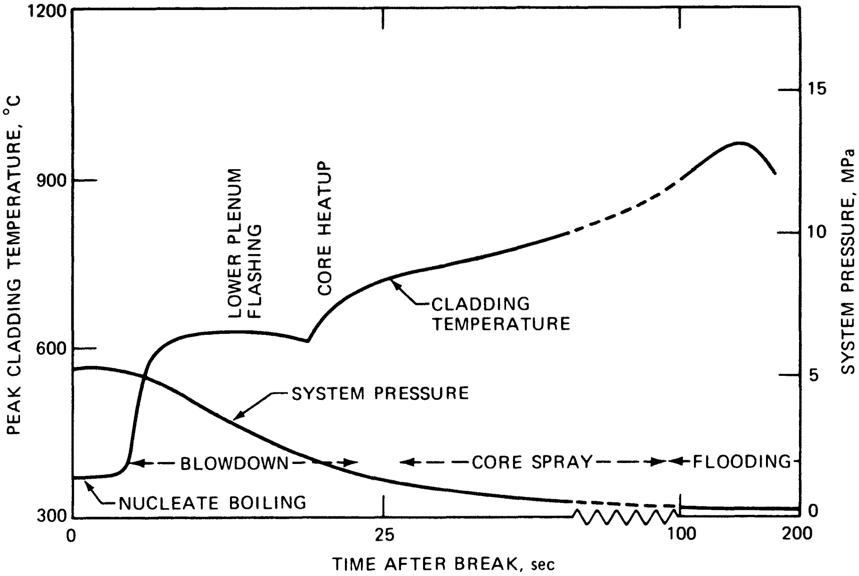

Fig. 12.9. Loss-of-coolant accident in a BWR. (Temperatures, pressures, and times are approximate.)

the core, would undergo violent boiling (flashing). Part of the mixture of steam and water would pass up through the core and reduce the fuel cladding temperature somewhat. This would be followed by a short core-heat-up period.

12.89. By the time blowdown and lower-plenum flashing were almost over, signals of the low water level in the reactor vessel and of the increase in drywell pressure would have initiated ECCS action. One of the core-spray subsystems would probably operate first, followed by the LPIS subsystem. Reflooding of the core, accompanied by effective cooling, would occur by accumulation of water in the lower plenum of the reactor vessel. When the water level reached the bottom of the core, the sudden heating would produce a frothy steam-water mixture. In flowing up through the core, this mixture would produce sufficient convective heat transfer to cool the fuel rods. Eventually, the water level in the core should reach that of the inlets to the jet pumps.

Emergency Core-Cooling Criteria

12.90. To provide assurance that the ECCS is designed so that its operation can prevent a significant release of radioactivity to the environment, the NRC requires in 10 CFR 50.46 that, in a postulated design basis LOCA in a light-water reactor (PWR or BWR), the following criteria should be met:

1. The calculated maximum fuel cladding temperature following the accident should not exceed 2200°F (1204°C).
2. The calculated total oxidation of the cladding, as a result of the interaction of the hot zircaloy with steam, shall nowhere exceed 0.17 times the total cladding thickness before oxidation. (The total oxidation is defined as the thickness of cladding that has been converted into oxide, assumed to be stoichiometric ZrO_2. A modified definition is used where the calculations indicate that swelling or rupture of the cladding is to be expected.)
3. The calculated total amount of hydrogen gas generated, by the chemical reaction of zirconium in the cladding with liquid water and steam, shall not exceed 1 percent of the hypothetical amount that would be generated if all the cladding material surrounding the fuel pellets, i.e., within the active core, were to react.
4. Calculated changes in the geometry, e.g., in fuel rod diameters and spacing, shall be such that the core remains amenable to cooling.
5. After successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value for the extended period of time required by the decay of the long-lived radioactivity remaining in the core.

The significance of these criteria will now be discussed.

12.91. The temperature of the fuel rod cladding would rise to a maximum in the reflood stage after an LOCA. The integrity of the cladding must be maintained so that it would not fragment under the thermal stress subsequently imposed when the very hot fuel rods are quenched. Embrittlement of zircaloy, which could lead to fragmentation, is a function of the temperature and degree of oxidation. Criteria (1) and (2) are intended to preclude the embrittlement (and melting) of the cladding.

12.92. The objective of criterion (3), which limits the amount of hydrogen gas produced by the zirconium–steam (or –water) reaction, is to keep the concentration of the gas in the containment vessel well below that at which a hydrogen–air mixture would ignite.

12.93. Local swelling (or ballooning) of the cladding, which might result from the expansion of contained fission-product gases, could affect the flow of coolant water through the core. According to criterion (4), calculations must show that despite changes in its internal geometry, the core should remain coolable during the reflood stage. The purpose of criterion (5) is obvious and does not require discussion.

12.94. The conservative nature of the foregoing criteria assures an adequate margin of performance of the ECCS should a design basis LOCA occur. This margin is provided by the criteria themselves as well as by conservative features of the evaluation models (§12.132 et seq.). The conservative features include the following:

1. The calculation of stored heat in the fuel is based on a preaccident reactor power level of 102 percent of the maximum operating power, with the highest allowed peaking factor (§9.167) and the lowest estimated thermal conductance between the fuel pellets and the cladding (§9.50).
2. Heat transfer during blowdown must be calculated using NRC-approved realistic models having an extensive experimental basis.
3. The fission-product decay heat is taken to be 20 percent greater than in the ANS standard (§2.217). The use of the Baker–Just relationship (§7.130) for calculating the heat generation rate from the zirconium–water reaction is also conservative.
4. The peak cladding temperature of 2200°F (1204°C) refers to the hottest region of the hottest fuel rod. This criterion ensures that there would be very little damage to the reactore core.

SEVERE ACCIDENTS [5]

12.95. As described in previous sections, design basis accidents are used to establish design requirements for plant safety systems. In the event of such an accident, the proper operation of such safety systems would prevent melting of the reactor core. However, as a result of the Three-Mile Island accident in 1979 (§12.179), in which partial core melting occurred, it became clear that studies were needed of accident scenarios that would lead to core degradation and the resulting possible transport of released fission products to the environment. The term *severe accident* has come to designate such an accident for analysis purposes. Such studies are helpful in establishing confidence levels and identifying possible failure modes. An accompanying research program has provided supporting data.

12.96. Severe accident scenarios commonly used evolved from those developed in the Reactor Safety Study (RSS), which is often referred to as WASH-1400 [6]. Since the engineered safety features are provided to prevent fuel degradation, to have a severe accident we must not only have an initial "fault" such as a loss-of-coolant break but an additional failure of one or more of these engineered safety features. We will deal with the three representative sequences for PWRs that are often studied as well as two important BWR sequences. We are including accident sequence symbols that originated with the Reactor Safety Study (§12.208) only as a convenience to readers who may be familiar with them.

PWR Sequences

Large-break LOCA with loss of all ac power (AB accident)

12.97. The accident sequence is somewhat different depending on whether the initiating break occurs in one of the pressure vessel outlets to the steam generator (hot-leg break) or in one of the vessel inlets (cold-leg break). In a hot-leg break, coolant, steam, hydrogen, and accident debris escape directly into the containment, while in the latter case, significant flow would be through the steam generator, with some opportunity for fission product deposition. On the other hand, in considering a LOCA design basis accident in which emergency core cooling is operative, the cold-leg break results in a higher clad temperature as a result of coolant flow reversal during the blowdown stage (§12.82).

12.98. Should there be loss of all ac power, the so-called *AB accident*, none of the engineered safety features (ESFs) would be functional except for the initial passive injection systems. The subsequent inability to remove stored and decay heat would result in a degraded core. Various chemical reactions among fission products and other core materials would take place. Non-noble gas fission products, their compounds, core debris, and water vapor are likely to condense partially in the cooler regions above the core and form aerosols upon escaping to the containment.

12.99. Depending on when power might be restored, the accident could proceed through a stage where molten fuel penetrates the reactor vessel, followed perhaps by interaction by the noncooled debris bed with the concrete base mat. Containment breaching would then be of concern. The nature of all of these steps is being studied and is mentioned here primarily for identification purposes and not to indicate risk levels. In fact, the AB sequence is considered relatively low in probability compared with the already extremely low probability of other severe accident scenarios. However, the sequence is significant because core melt starts relatively rapidly, about 30 minutes after the break.

Transient-induced accident (TMLB′)

12.100. This accident is caused by any event that requires reactor trip combined with a station blackout, i.e., the loss of all power, as well as the loss of capability of the secondary system to remove heat from the primary circuit. Under such circumstances, with no means for normal heat removal, the primary system pressure would rise until the relief value system would initiate blowdown. Subsequent events for a TMLB′ accident could follow those given above for the AB accident. However, in a more realistic variation, designated as a TMLB accident, electrical power is assumed to be recovered in 1 to 3 hours. Should this be the case, meltdown and the subsequent scenario might be avoided.

Containment bypass accidents (V)

12.101. This is an accident category in which the containment would be compromised by such possibilities as the failure of a series of valves connecting the high-pressure reactor coolant system with the low-pressure portions of the emergency core cooling system leading outside the containment. In this category, the event scenario and consequences depend greatly on the design details of the specific reactor system. It should be noted that in the Three Mile Island accident, there was some containment compromise through the level control and purification system (§12.179).

BWR Sequences

Transient with failure of the reactor protection system (TC)

12.102. Should there be a malfunction combined with failure of the reactor protection system, the reactor will continue to generate power. However, it is assumed that the main steam valves will isolate the reactor from the turbine generator and the recirculating pumps will trip. As a result of rising steam pressure, safety relief valves will open, releasing steam to the suppression pool, with the resulting core voiding reducing the reactor power. Subsequent events would depend on the ability of water makeup and coolant injection systems to keep the core covered. Insufficient suppression pool heat removal capacity could result in excessive pressurization of the containment and failure, while core uncovering would result in fuel failure and fission product release.

Transient with failure to remove residual core heat (TW)

12.103. In this type of sequence, the reactor would trip, but the suppression pool would fail to remove the continuing production of residual heat. Containment failure could occur as a result of excessive pressure and the ECC systems, which use the suppression pool as a water source, becoming ineffective because the water is now saturated and cannot be pumped. Core uncovering is then likely.

THE SOURCE TERM [7]

Introduction

12.104. The designations *source term* or *fission product source terms* refer to the amount and type of radioactive materials which would be available for escape to the environment from a reactor plant that has undergone a severe accident. Source term studies consider various accident sequences as described above and the chemical and physical processes that affect the transport of radionuclides from the point of origin, such as a damaged fuel element, to the environment. Probability considerations and the behavior of the radionuclides once they reach the environment are normally considered beyond the scope of source term studies.

12.105. Consideration of the fission product nuclides and their behavior during an accident release is a logical starting point in our study of accident consequences. However, space limitations prevent a detailed step-by-step discussion of the various processes that contribute to fission product transport during typical severe accident scenarios.

Barriers to the Escape of Radioactivity

12.106. Before considering fission product transport, mention should be made of the physical barriers in reactor systems to the escape of radioactivity. The first barrier is the fuel material itself, which in the case of uranium dioxide retains the solid fission products and inhibits release of the volatile radionuclides. The fuel rod cladding is the second barrier. During service, the cladding does develop small pinholes and cracks, which allows some radioactivity to escape into the cooling water. The third barrier, the primary coolant system boundary, contains the normally small amount of radioactivity in the circulating coolant, which is continuously monitored. The final physical barrier is the containment structure. However, as we discuss shortly, certain chemical and physical processes along the transport path also inhibit the fission product release.

Radionuclide Importance Factors

12.107. In considering the hazards of fission product release in an accident sequence, a number of factors determine the importance of individual radionuclides.

1. The total core inventory of the isotope. This depends on the balance between formation from fission, destruction, and decay as discussed in §2.203 et seq.
2. Chemical properties determine reactions that may occur during the accident sequence which could affect transport of the isotope.
3. Physical properties and characteristics such as volatility, particle size, and deposition behavior require attention.
4. Radiological and biological importance. Such matters as postaccident isotopic changes, biological uptake and half-life, and specific effects on organs are all relevant.

Fission Product Transport Overview

12.108. During the course of the Three Mile Island accident in 1979 (§12.178), a surprising small amount of iodine, about 18 Ci, escaped to the environment. Since this was so much less than the 6 million curies of the inert gas, xenon, that escaped, studies were instigated which showed that as a result of the reducing chemical state in the reactor vessel, the iodine was converted to cesium iodide, which is soluble in water and hence was not released. Therefore, it was recognized that chemical and physical phenomena during the course of an accident have a great deal to do with reactor safety, and major attention was devoted to source term studies.

12.109. In assessing the hazards associated with fission product release from the fuel, we must consider in a stepwise manner how each nuclide of radiological importance makes its way from the point of release, through barriers that may have failed, into the coolant, and then into the containment. Some release from the containment to the environment by some means would then constitute the hazard. An analysis of this sequence depends very much on the nature of the accident. Types of accidents are considered in §12.94 et seq. However, in all cases, the fission product transport depends first on chemical changes that affect their characteristics, then on their behavior when they are in the form of particles or aerosols. Therefore, we will examine in a general way the chemistry of the fission products. Then, an introduction to fission product particulate mechanics will be given.

Fission Product Chemistry

12.110. The fission product elements react chemically with each other and with other substances during the course of an accidental release from the fuel. These reactions affect the transport characteristics of the radionuclides that concern us. A starting point is to consider the reactor fission product inventory, which consists of nearly 40 different elements. Various tabulations and computer codes such as ORIGIN-2 [8] are available to provide these data. However, it is important to remember that chemical reaction rates depend on the amount of *element* present, not just the amount of a radioactive isotope. Therefore, the inventory must include *all* the isotopes. A typical PWR *element* inventory is given in Table 12.1. Based on their reactivity with oxygen, it is convenient to classify the elements into eight groups: the noble gases, halogens, alkaline earths, tellurium, noble metals, rare earths, elements with refractory oxides, and stable fission product elements.

12.111. The general chemical behavior of the elements in the various groups as they occur in the fuel, in the water coolant of a LWR, and in the containment is summarized as described below.

Noble gases

12.112. The noble gas fission products such as xenon are inert chemically. During irradiation, they are partially released from the ceramic oxide

TABLE 12.1. Typical Fission Product Inventory (kg) (Based on 780 MW(e) PWR)

Element	kg
Xenon (Xe)	260
Krypton (Kr)	13.4
Iodine (I)	12.4
Bromine (Br)	0.8
Cesium (Cs)	130.6
Rubidium (Rb)	14.7
Tellurium (Te)	25.4
Antimony (Sb)	0.7
Barium (Ba)	61.2
Strontium (Sr)	47.6
Ruthenium (Ru)	104.3
Rhodium (Rh)	20.9
Molybdenum (Mo)	154.9
Technetium (Tc)	37.1
Yttrium (Y)	22.9
Lanthanum (La)	62.3
Zirconium (Zr)	178.6
Cerium (Ce)	131
Praseodymium (Pr)	50.7
Neodymium (Nd)	171

fuel matrix and occupy grain boundary regions and whatever other free space is available within the fuel rod. Upon failure of the cladding and, in turn, the coolant system boundary, they would travel into the containment and be available for transport to the environment without acting chemically. Although the transition during this time period by radioactive decay to other elements that may be reactive is a possibility, the effect is small in light of other process uncertainties and may be neglected in approximation-level analyses.

Halogens

12.113. The halogens are very reactive and tend to combine with such metallic fission product species as cesium and zirconium. Iodine is the primary element of interest since its concentration in the fission product mixture is about 100 times greater than that of bromine. Also, the cesium/iodine concentration ratio in the fission products of about 10 to 1 favors the formation of cesium iodide.

12.114. During a loss-of-coolant accident, the reactor coolant system tends to have a reducing atmosphere. Hence, such strong oxidizing agents as the halogens are likely to form halides, which will be retained in the water present. For example, iodides transported into the containment with large amounts of water will tend to remain in the water as a nonvolatile component.

12.115. Since the release of iodine-131 to the atmosphere has traditionally served as a measure of radiological risk (§12.209), the chemical behavior of various forms of iodine has been studied extensively. Should there be a hydrogen burn in the containment, volatile methyl iodide might form, but the conversion would only be about 0.03 percent of the reactor iodine inventory [7]. Whether or not this constitutes a hazard depends on the integrity of the containment.

Alkali metals

12.116. The alkali metals, such as cesium and rubidium, are both reactive and volatile in their elemental form. In the fuel, gaseous cesium tends to exist in thermodynamic equilibrium with several cesium compounds. Upon failure of the fuel, a fraction of the cesium present will combine with most of the iodine, but the remainder is likely to react with steam present to form cesium hydroxide. Other reactions could occur with the various substances present but it is likely that the compounds formed will have low volatility.

Tellurium

12.117. Tellurium is the most significant fission product of the chalcogen class, which also includes selenium. This importance is a result of both the concentration of its several isotopes (about 1 percent of the fission products) and their radiotoxicity, primarily because they are iodine precursors. Under accident conditions, it is expected to be released from the fuel in elemental form and then react with various structural materials. Tellurium behavior appears to be dependent on the particular accident scenario and is the subject of ongoing studies.

Antimony

12.118. Antimony remains in elemental form upon being released from failed fuel elements and tends not to react with steam or hydrogen in the cooling system. Since its boiling point is 1380°C, it will vaporize under accident conditions and deposit on system surfaces. However, it can alloy with other metals, which would tend to reduce its vapor pressure.

Alkaline earths

12.119. Strontium and barium are released from failed fuel rods as oxides and could react in the cooling system with steam to form hydroxides which are moderately soluble in water. The presence of carbonates in emergency cooling water would result in precipitation of the alkaline earths from the hydroxides, thus preventing their release from the containment to the environment.

Other fission product groups

12.120. Noble metals, rare earths, and refractory oxides are generally chemically inactive. However, upon possible release into the containment during a severe accident, they could be transported as fine particles such as aerosols.

Fine Particle Dynamics

12.121. The mechanics of particle movement through a fluid has received a great deal of engineering attention for many years. For example, scrubbers and devices for removing mists from chemical plants are common applications of fine particle dynamics. A first approach to describing particle motion is to apply a force balance which includes an external force such as a gravitational force and a frictional or drag force. Thus, for a spherical particle having a diameter D and a density ρ falling through a fluid of density ρ_f with a velocity u, we can say that

$$\text{Weight of sphere} - \text{Buoyant force} = \text{Drag force}.$$

The drag force is expressed in terms of a drag coefficient C_D according to the relation

$$\text{Drag force} = C_D(\tfrac{1}{2}\rho_f u^2 A_P),$$

where u is the constant settling velocity and A_P is the projected area of the sphere, i.e., $\frac{1}{4}\pi D^2$. Since the particle volume is $\frac{1}{6}\pi D^3$, we have

$$\frac{1}{6}\pi D^3 g(\rho - \rho_f) = \frac{1}{2} C_D \rho_f \left(\frac{\pi D^2}{4}\right) u^2. \qquad (12.1)$$

In the particle diameter range of 3 to 100 μm, Stokes law applies and

$$C_D = \frac{24}{Re} = \frac{24\mu}{Du\rho_f}.$$

Then simplifying, we have

$$u = \frac{gD^2(\rho - \rho_f)}{18\mu}. \qquad (12.2)$$

For smaller particles, 0.1 to 3 μm in diameter, an empirical correlation known as the Cunningham factor is applied to account for slip and free molecule effects. An average factor value of 1.8 may be assumed for estimation purposes.

Example 12.2. Determine the settling velocity in air of a particle having a diameter of 10 μm and a density of 3000 kg/m³. The air at 288 K and 1 atm has a viscosity of 1.8×10^{-5} Pa · s. Compare with the velocity of a particle having a diameter of 1 μm.

$$u = \frac{gD^2(\rho - \rho_f)}{18\mu} = \frac{(9.81)(10 \times 10^{-6})^2(3000)}{18(1.8 \times 10^{-5})}$$
$$= 9.1 \times 10^{-3} \text{ m/s}$$

For the 1-μm particle,

$$u = \frac{(1.8)(9.81)(1 \times 10^{-6})^2(3000)}{18(1.8 \times 10^{-5})} = 1.6 \times 10^{-4} \text{ m/s.}$$

(A more accurate calculation yields 1.1×10^{-4} m/s.)

Aerosols [5]

12.122. An aerosol is a gaseous suspension of fine liquid or solid particles. Spherical diameters may vary from 1 mm to as small as 1 nm, but the range of interest is normally from 0.01 to 100 μm. Degraded core materials making their way into the containment as a result of a severe accident tend to form aerosols by several processes. For example, the volatile fission products, such as cesium, iodine, and tellurium, are likely to condense or chemically react to form lower-volatility compounds and then condense. Nonvolatile fission products and other materials also form solid aerosols or combine with the steam present to form liquid aerosols. Although solid aerosols can have various shapes, a spherical shape is first assumed in analysis and correlation factors applied.

12.123. The amount of fission products available for release from the containment depends greatly on aerosol formation and deposition processes that occur following a degraded core accident. However, the picture is complicated. The suspending gas is likely to be in turbulent motion while the particles change size and may deposit by several processes.

12.124. Deposition by sedimentation as a result of the action of gravity occurs in accordance with the Stokes law principle described in §12.121. However, both the particle density and diameter tend to change during the course of an accident affecting the settling velocity. Particles having a diameter of about 1 μm have a settling velocity of about 0.1 mm/s. A measure of fission product removal is the *decontamination factor*, defined as the ratio of activity before to that after a given process. In a PWR containment vessel having a volume of 70,000 m^3, it would take 17 days to achieve a decontamination factor of 100 for such particles [5].

12.125. Modeling of the deposition of aerosols has been carried out by the TRAP-MELT code [9]. In addition to sedimentation, other deposition processes include thermophoresis, turbulent deposition, and diffusion. Thermophoresis is the result of a force in a nonisothermal gas arising from unequal molecular impacts on the particle. Turbulent deposition is a wall effect resulting from the particles not being able to follow the fluid eddy flow adjacent to the wall. Diffusion become significant for particles having a diameter of less than 0.1 μm. It is the result of bombardment by the gas molecules, which results in a diffusive flux when the particle concentration varies in space. Aerosol behavior is the subject of continuing modeling development supported by experimental research.

Explosions

12.126. In an explosion, gases are formed so rapidly that the propagation front moves with supersonic velocity so that a shock wave is produced. The resulting kinetic energy can be destructive. In severe accidents, two types of explosions are possible. One is a vapor explosion, or so-called steam explosion. The other is a result of a chemical reaction, or combustion, normally of hydrogen produced from zirconium oxidation. If the speed of the combustion wave is subsonic, we have *deflagration* or burning rather than the supersonic *detonation*.

12.127. The postulated vapor explosion during a severe accident is a result of the interaction of molten fuel with liquid water coolant. Several steps in this interaction take place, the details of which are important is assessing the load on the containment structure. Since both experimental and theoretical research on mechanisms is ongoing, we can only present some of the principles involved.

12.128. When the molten fuel contacts the water, steam is formed very quickly at the interface that separates the two liquids during a short metastable period. This period may last from a few milliseconds to a few seconds, after which the film becomes unstable, resulting in very rapid fragmentation of the hot liquid fuel. The greatly increased interface surface area which quickly propagates throughout the molten fuel–water mixture then results in the formation of much more steam at a rate that can produce a detonation-like shock wave. Thus, adequate modeling requires a description of the fuel–coolant mixing, so-called "triggering" when the metastable condition is disturbed, and the subsequent explosion propagation step [35].

12.129. The role of hydrogen formed from the zirconium–steam reaction during a severe accident is reasonably well understood and is described by suitable modeling codes. Also, several countermeasures normally prevent explosions in the containment. For example, thermal or catalytic recombiner devices combine the hydrogen and oxygen present so that the hydrogen concentration remains below flammable limits. Another approach is to "inert" the containment by substituting nitrogen for air during normal operation. A number of glow or spark plug ignitors distributed throughout the containment deliberately burn the hydrogen in a controlled manner close to the point of formation, thus avoiding an explosion involving a large mass of reactant [36].

SAFETY MODELING METHODS

Introduction

12.130. The modeling of various aspects of reactor system behavior is the objective of a large number of safety-related computer codes. Also, by assembling specialized codes into related modules, it is common practice to provide a description of system behavior following a spectrum of disturbances and accidents of varying degrees of seriousness. Since modeling methods are continually being updated and many are proprietary, our primary purpose here is to provide orientation and describe some typical approaches.

12.131. Modeling methods may be classified by their intended application.

1. *Licensing requirements.* Licensing codes, which are primarily thermal-hydraulics in nature, confirm the protection provided by LWR emergency core cooling systems.
2. *Modeling as an integral part of design procedures.* Design codes provide "best estimate" or "realistic" predictions. They, too, normally model thermal-hydraulics behavior, although some deal with containment response.
3. *Severe accident analysis.* Analysis complements risk assessment and source term development activities.

Many codes, particularly those in the second category, have been developed by reactor vendors and are proprietary. Others have been developed with government or utility support. Some codes serve more than one application. Each of these general modeling categories are considered in the following sections.

Licensing "Evaluation" Models

12.132. An important requirement for LWR reactor licensing is to determine that the emergency core cooling system design meets the criteria specified in 10 CFR 50.46 and summarized in §12.90. The features of the so-called "evaluation" computer models for the required demonstration are specified in Appendix K of 10 CFR 50. Prior to 1988, evaluation models were required to embody very conservative assumptions and hence predicted results that were intentionally pessimistic, rather than representing the "best estimate" of the real conditions. In 1988, Appendix K was amended to permit more realistic modeling provided that the predictions could be compared with applicable experimental information and appropriate sensitivity studies performed. Correlations used in the modeling must be approved and shown to be reasonably conservative. This change had the effect of increasing allowable core peaking factors, which, in turn, provides greater flexibility for reload core design. However, the owner of a nuclear power plant may wish to evaluate safety margins for management purposes using realistic models which may not be as conservative as those approved for licensing. Such an activity is known as *safety margin basis evaluation*. This "nondesign basis" type of evaluation is also useful in studying containment response to severe accident scenarios (§12.95).

12.133. These models are based on complex system codes (§12.142) which calculate fuel and cladding temperatures (and other core properties, e.g., zirconium–water reaction) during the blowdown, refill, and reflood stages of a LOCA accompanied by operation of the ECCS. The computer programs must evaluate the various energy sources, provide for fuel dimensional changes, and describe in detail the various heat-transfer and fluid-flow phenomena at various stages of the hypothetical accident.

12.134. The energy sources must include the stored energy in the fuel prior to reactor shutdown as a result of the loss of coolant, the fission heat generated subsequently, the radioactive decay of the fission (and neutron capture) products in the fuel rods, and the heat generated by the zirconium–water (–steam) reaction. Allowance must be made for energy stored in the reactor vessel walls, internal hardware, and piping. In PWRs, heat removal in the steam generators must also be considered.

12.135. Among the hydraulic parameters are the rate of flow of water through a postulated pipe break in the primary coolant circuit, rate of flow through the core and other system components, and the pressure and water level in the reactor vessel during the blowdown, refill, and reflood stages. Flow out of the pipe break in a PWR must be calculated for two separate periods: (1) immediately following the break when the water is still subcooled and consists of a single phase and (2) when the discharge consists of a two-phase (liquid water and steam) flow.

12.136. The types of fuel-coolant heat transfer to be considered include those during and after blowdown. During the blowdown phase, the critical heat flux (CHF) is attained; different heat-transfer correlations must be used in the pre-CHF and post-CHF stages. Following blowdown, especially during reflood, the heat-transfer coefficients (or correlations) are based on experimental data modified to assure that they are conservative. The NRC regulations require that no heat transfer be assumed during refill.

System Modeling Methods

12.137. The only practical way to model transient behavior such as that described above in a system consisting of connected components is to use subprograms for the individual component volumes and then join them together at the component interfaces. Improved accuracy is obtained by subdividing the component and piping volumes shown in Fig. 12.10 to a level balanced by the resulting increased computational cost. Thus, we have the primary coolant circuit divided into a number of *control* or *nodal volumes* connected by *junctions* at which the properties change, and fluid mass, momentum, and energy are transferred as a function of time. Within each volume, so-called field equations describe in a numerical finite-difference manner the conservation of mass, momentum, and energy. A typical system code is likely to have hundreds of nodal volumes, with time steps taken to model a LOCA of the order of milliseconds. Should longer transients be modeled, the number of volumes must be reduced to keep the computing time from becoming excessive.

12.138. In Fig. 12.11 is shown a practical level of volume subdivision for the analysis of the blowdown phase in a large cold leg break LOCA for a two-loop PWR. The model for this blowdown analysis consists of 45 volumes, 56 junctions, and 28 heat slabs. Three volumes are used to model the core. Also, within the reactor vessel, the downcomer, upper and lower plenums, and core bypass are represented by other volumes. Discharge into the containment (volume 40) flows through two imaginary "valves" used to model various break sizes [36].

12.139. In modeling the various stages of a LOCA, the picture is complicated by the need for a two-phase description which must also account for the transfer of the conserved quantities between phases in a given volume. The reflood stage is a particular challenge. Since the physics of two-phase flow is very complex, it is necessary to apply simplifying assumptions for practical modeling. In the most drastic simplification, the so-called one-dimensional homogeneous, no-slip, equilibrium model, the steam–water mixture is treated as one fluid at the saturated temperature moving in one dimension. In attempting to develop greater modeling sophistication to account for nonequilibrium and nonhomogeneity, various

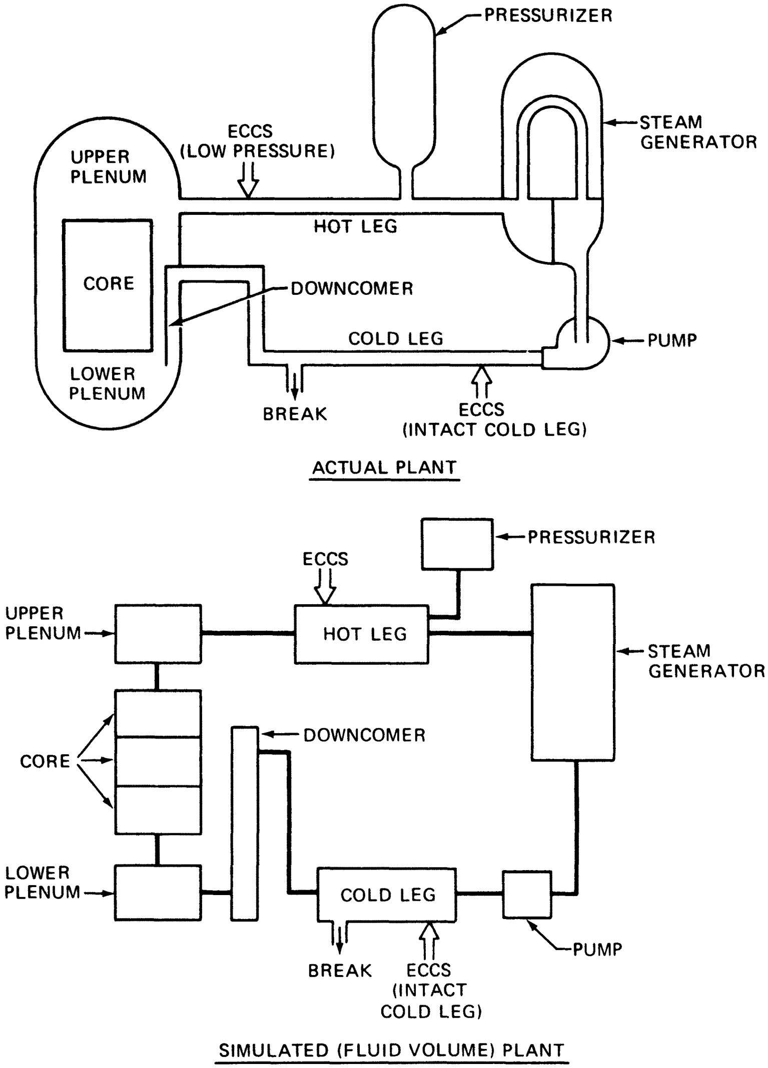

Fig. 12.10. Schematic representation of actual and fluid-volume models of a PWR coolant loop for LOCA calculations. (The intact loops are not shown but are included in the computations.)

averaging procedures have been developed to convert the local instantaneous governing equations written for each phase to suitable averaged governing equations for the mixture. This *lumped-parameter* approach is generally used for licensing-type calculations.

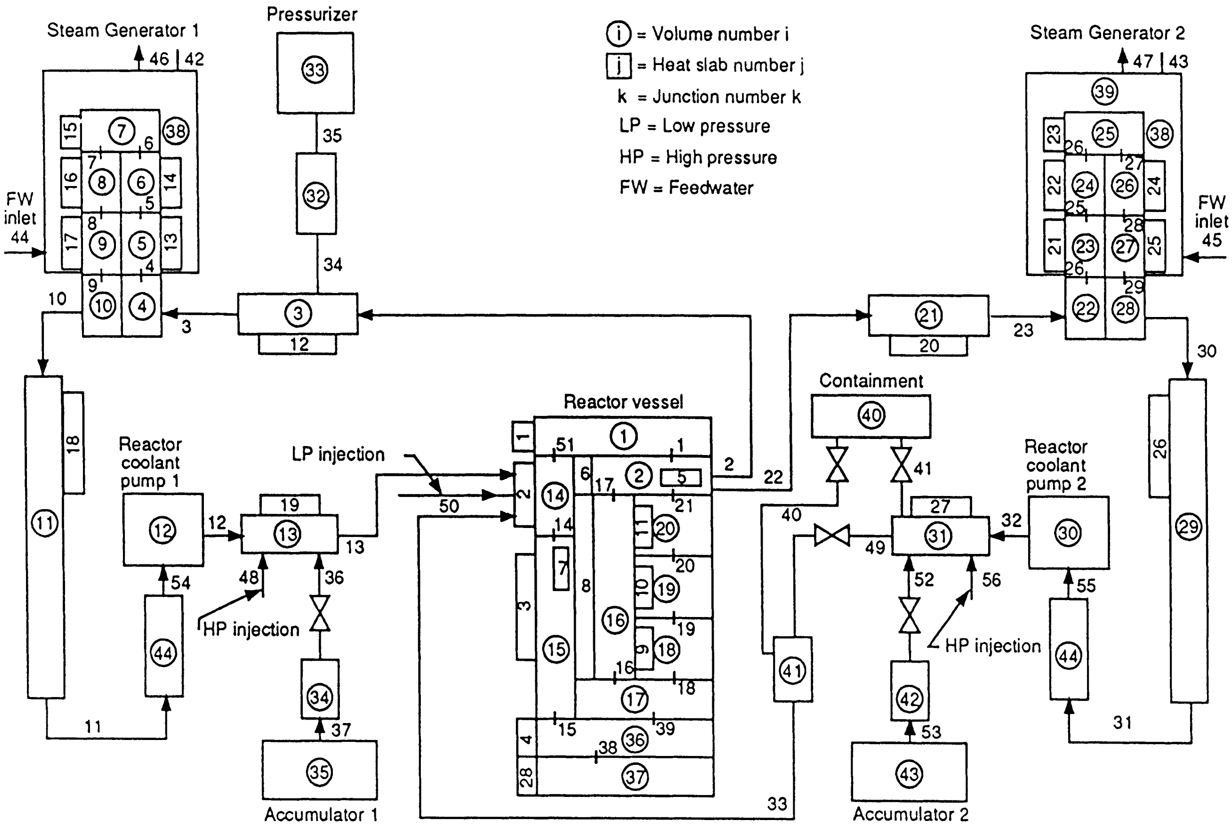

Fig. 12.11. Typical nodalization for two-loop PWR LOCA blowdown analysis [36].

12.140. Higher levels of sophistication are used in the best estimate codes. Here, one also starts with a *two-fluid* or two-phase model but retains the individual phase relations in the calculation with expressions for interfacial mass and energy exchange included. However, some simplification is still needed. This varies depending on the needs of the problem. In fact, the features of most codes in common use are being updated continually. The level of representation can vary among different subprograms describing behavior in individual coolant system components. Details may be found in other sources [10] and in the current literature.

12.141. A feature of the various system codes is the need to incorporate within the various subprograms available engineering correlations to predict such values as heat-transfer coefficients and vaporization rates as introduced in Chapter 9. Selection of the proper correlation for each region of application must be provided by the code system. Associated with the use of correlations is uncertainty analysis to determine the confidence level of the results. This may be done by estimating error bands for each correlation and determining the cumulative error through the computational process. Of course, modeling simplifications would contribute additional uncertainties. Therefore, experimental programs that simulate various types of LOCA behavior in individual components play an important role in system code development.

Representative Best Estimate System Modeling Codes

12.142. The RELAP class of codes developed at the Idaho National Engineering Laboratory (INEL) for the U.S. Nuclear Regulatory Commission is the nonproprietary workhorse for reactor safety thermal-hydraulic analysis for PWRs only [11]. TRAC codes developed by Los Alamos have a minimum level of modeling simplifications and therefore serve both for accident analysis and "benchmarking" or calibration of simpler modeling systems [12]. Both PWR and BWR versions are available. The Electric Power Research Institute has sponsored the RETRAN series of codes, which evolved from the RELAP effort. These are intended to provide electric utilities with a general-purpose tool for design and safety evaluation [13]. A feature is the continuing use of plant operating data for code calibration.

12.143. RELAP5/MOD2, is the result of a complete rewrite of the earlier RELAP4 to include nonhomogeneous and nonequilibrium features. It is applicable for modeling large and small break LOCAs, operational transients, transients in which the entire secondary system must be modeled, and system behavior simulation up to the point of core damage. The code includes a point reactor kinetics model (§5.7), a decay heat model, and the capability of modeling various functions of the reactor control system.

12.144. A challenge in the development of any system modeling code is to manage the numerical solution techniques so that there is a reasonable balance between accurate representation and computing effort, particularly when one considers the physical complexity of turbulent two-phase flow. In the case of RELAP5/MOD2, modeling with workstation-level computers has become possible. Various improvements are being incorporated into a new version, RELAP5/MOD3 [14].

12.145. TRAC-PF1 is intended to model PWR LOCAs and transients while TRAC-BF1 fulfills a similar function for BWRs. RETRAN-02 models all types of LWR transients. They are similar in the general nature of simulation approach to the RELAP codes but differ in modeling details and solution methods. TRAC is a useful design tool. It, like other system codes, includes a number of computing modules that model behavior in individual components. TRAC was extensively used in the analysis of the Three Mile Island accident.

Modeling of Fluid and Structure Interactions

12.146. Related to the system modeling codes there is the need to examine the mechanical effects on components resulting from the strong depressurization waves during a LOCA. Since dimensionality is important for such analysis, the approach used is to start with a two-dimensional, homogeneous, equilibrium code to model the two-phase blowdown and then use a special-purpose code to model the motion of such components as the core barrel of a PWR [15].

Severe Accident Modeling

12.147. Severe accident modeling has received increased attention by the U.S. Nuclear Regulatory Commission since the Three Mile Island and Chernobyl accidents. Such modeling has helped to provide guidance for reactor design and to supplement risk analysis studies. Generally, individual codes are used to describe the various steps in an accident sequence and coupled together. Since this is a rapidly changing field, and many of the codes in use differ significantly in their modeling approach, our purpose is merely to outline the functional requirements and cite some examples.

12.148. A major objective of the modeling is to describe the response of the containment building to a core melt accident in a light-water reactor. We are reminded that this building is the final barrier to fission product release to the environment. Therefore, it is necessary to describe, normally by coupled computer modeling, the stepwise progression of accident events that lead to the buildup of containment pressure. An associated objective is to provide a basis for risk assessment studies (§12.208). Detailed mechanistic code modules, developed in close connection with experimental programs, are used for benchmarking simplified codes used for source term quantification and risk assessment.

12.149. Typical code systems provide for input from a fission product inventory code such as ORIGIN to an in-vessel thermal-hydraulics model of the accident which may be initiated by such conditions as a station blackout sequence in which all power is lost. Following failure of the vessel, containment effects become important. A lumped parameter approach is followed both for the in-vessel and containment analyses. The respective volumes are divided into interconnected compartments or cells and mass, momentum, and energy conservations calculations are performed for each cell (§12.137).

12.150. Several different code module systems are available to model severe accident phenomena, with the pressure load on the containment generally as the first objective. The modeling approaches used vary in several ways, but a description of code details is beyond our scope. Therefore, we will merely identify several of the code packages used and indicate the "flavor" of the approaches followed. The MAAP code system was developed as part of an Industry Degraded Code Rulemaking Program (IDCOR) for predicting severe accident source terms [16]. The CONTAIN and MELCOR codes were developed for the NRC [17]. To determine containment pressurization, such in-vessel phenomena as blowdown and boil-off of the primary and secondary coolant must first be modeled. Next, the effects of zirconium oxidation, core heatup and meltdown, and debris relocation are described. Since the pressure load on the containment depends on both gases generated and energy produced, steam and hydrogen produced during debris–water interactions and molten core–concrete interactions are modeled. Energy production and removal as well as the impact of engineered safety features must be considered.

12.151. The modeling of fission product release to the environment from a severe accident is an essential feature of the evaluation of reactor risk (§12.231). In general, the same code systems described above for containment loading evaluation are used to describe fission product behavior. For example, chemistry considerations are integrated into the MAAP code with the concentrations of 34 species followed. It is emphasized that over the years, numerous other codes have been developed to describe fission product transport up to containment failure. However, for emergency response planning, meteorological considerations must also be modeled. Codes such as CRAC2 [32] develop off-site dose probabilities by sampling local meteorological data for a Gaussian dispersion model (§12.160).

12.152. As a result of the large number of interrelated processes that require description, severe accident modeling tends to be inherently complex. Furthermore, it is necessary to use many simplifying assumptions in the representation of individual processes which may differ among code packages. The methodology used may also differ. Therefore, the overall results yielded by one code package may differ somewhat from the results obtained from another. Thus, users should study the details of a given modeling procedure and appreciate the confidence levels developed.

SITING REQUIREMENTS

Introduction

12.153. The approval of a proposed reactor site is part of the licensing process and is likely to involve applicable state agencies. However, several requirements are concerned with the effects of postulated accidents on the surrounding population as well as certain accidents that relate to the site itself, such as the role of a possible earthquake. Therefore, it is useful to discuss such matters at this point. However, specific requirements are subject to change. Therefore, other sources, such as applicable NRC Regulatory Guides (1.3 and 1.4), should be consulted for the current regulations.

Radiological Criteria of Site Acceptability

12.154. Among the factors considered in evaluating a proposed site for a nuclear power plant are certain radiological criteria which serve as guidelines. These guidelines, as stated in 10 CFR Part 100, are calculated radiation doses to various population groups resulting from a hypothetical accident associated with a substantial release of fission products from the reactor core, followed by leakage from the containment vessel. The conditions postulated for these site acceptability calculations are such that they could be realized only in the highly improbable circumstances of a major LOCA accompanied by failure of essentially all the components of the ECCS.

12.155. In considering possible radiation exposures to population, it has been traditional *for siting purposes* to designate certain areas or zones surrounding the reactor plant following the philosophy that only a low population density would be acceptable close to the reactor where the radiation exposure from an accident might be significant. Immediately surrounding the plant is an *exclusion area*, a region in which the plant management has control of all activities.

12.156. The *low-population zone* is the region just outside the exclusion area; the total number and density of residents (if any) in this zone should be such that appropriate protective action, e.g., taking shelter or evacuation, would be possible in the event of a major accident. Radiation exposure limits have been prescribed for the outer boundaries of each zone. For example, at the outer boundary of the exclusion area, a limiting whole-body dose to a person of 25 rem from all sources for a 2-hour period has been designated. A whole-body dose limit of 25 rem at the outer edge of the low population zone was based on the person remaining in place during the entire period of passage of the radioactive cloud resulting from the accident. The philosophy of the low population zone is incorporated in the requirement for an Emergency Planning Zone (§12.171).

12.157. Another siting consideration is the distance of the plant site from population centers. Recognizing that in the event of a serious accident, the societal risk from delayed cancers may be significant up to a distance of 80 km (50 miles) from the plant site, a remote location is favored by the NRC. Candidate locations are evaluated on a case-by-case basis with the Emergency Response Planning requirement (§12.171) applicable.

12.158. To test the radiological acceptability of a given site, a procedure described in NRC Regulatory Guides 1.3, 1.4, and 1.23 is followed. An accident in which substantial core meltdown occurs is assumed, such as a LOCA with complete failure of all emergency core cooling systems. Following the accident, highly conservative (worst-case) assumptions are prescribed for the fractions of the core radionuclide equilibrium inventory that would be *immediately* available for leakage from the containment using maximum leak rate design specifications. Allowance may be made for radioactive decay during holdup in the containment vessel and material removed by containment vessel sprays and filters. However, source term studies (§12.104) show that these site evaluation radionuclide release assumptions are unrealistically conservative. Therefore, it is important not to confuse these site evaluation procedures, which are prescribed to assure a consistent treatment, with risk assessment studies to be described later (§12.231).

12.159. In our procedure, the containment leak *rate* acts as a source of a plume that is dispersed by atmospheric conditions, leading to a radioactive concentration downwind which is ingested by a person, say at the exclusion area boundary. With the aid of further assumptions in our model, a thyroid dose commitment at this point can be derived. For this purpose a simple Gaussian plume dispersion model is assumed [18]. More sophisticated (and realistic) models are used in risk assessment studies.

Radiation Dose Calculations

12.160. In calculating the expected radiation doses received from any airborne radioactive material that might escape from the reactor containment structure, it is assumed that the material forms a plume similar to that from a smokestack but closer to the ground. The plume is carried forward by the wind while diffusion causes it to spread in two perpendicular directions, i.e., laterally (crosswind) and vertically. Observations made on small plumes indicate that diffusion would result in a Gaussian distribution (§9.165) of the radioactivity concentration about a centerline. According to the Gauss plume model, assuming that the containment vessel constitutes a *continuous point source*, the distribution of the radioactivity on the ground, from a specific radionuclide, would be represented by

$$\chi(x, y) = \frac{Q}{\pi u \sigma_y \sigma_z} \exp\left[-\frac{1}{2}\left(\frac{h^2}{\sigma_z^2} + \frac{y^2}{\sigma_y^2}\right)\right], \tag{12.3}$$

where $\chi(x, y)$ $\mathrm{Bq/m^3}$ is the ground-level concentration of radioactivity at a point x, y; Q Bq/s is the source strength of the given nuclide, u m/s is the wind speed, assumed to be uniform in the x direction, h m is the effective release height of the radioactivity, and y m is the lateral distance of the receptor from the plume centerline; σ_y and σ_z are the standard deviations of the distribution in the plume in the y (lateral) and z (vertical) directions, respectively.

12.161. The values of σ_y and σ_z, which are functions of the distance x of the receptor from the source, have been determined from experimental studies; they are presented in graphical form in Figs. 12.12 and 12.13, for various atmospheric conditions, designated Pasquill categories A through F. Pasquill A refers to an extremely unstable atmosphere whereas Pasquill F is a moderately stable condition. Thus, if Q, u, and h are known, $\chi(x, y)$ can be determined for various atmospheric conditions from equation (12.3).

12.162. The source strength Q for a particular radionuclide is determined in the following manner. The saturation (or equilibrium) activity in a reactor which has been operating at a specified thermal power can be calculated along the lines indicated in §2.122. Values for the radioiodines obtained in this manner are given in Table 12.2. From the equilibrium activity of a particular radionuclide, the assumed release fraction, the dimensions of the containment structure, and the maximum design leakage rate from the containment vessel, the source strength Q in Bq/s can be readily evaluated.

12.163. In making radiation dose calculations, y in equation (12.3) is

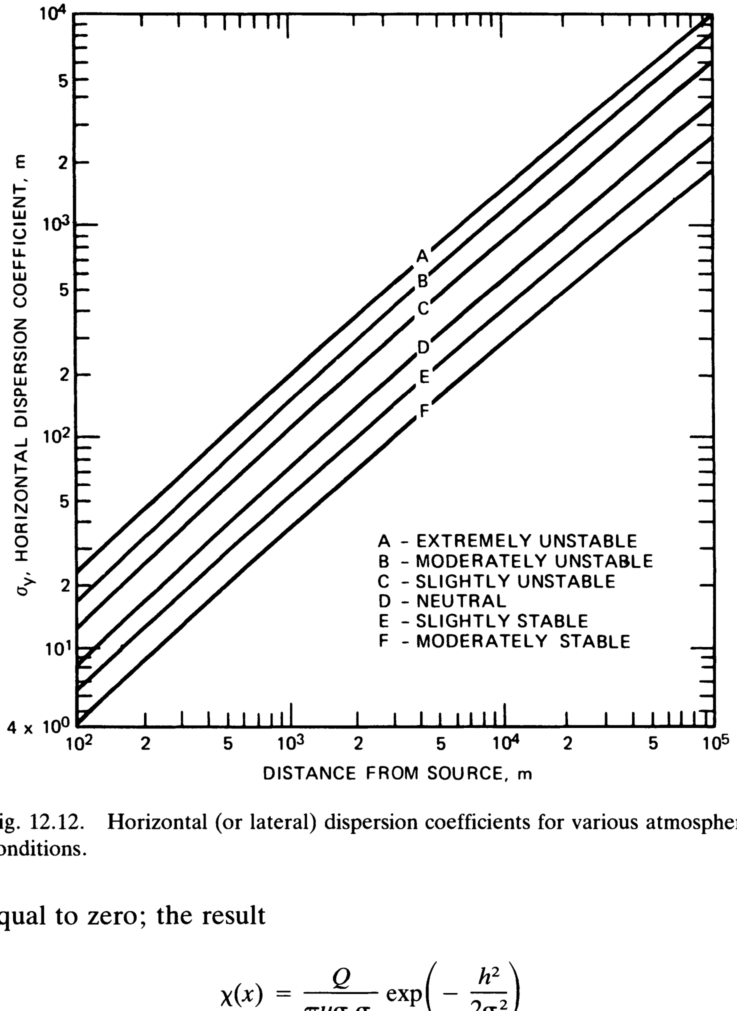

Fig. 12.12. Horizontal (or lateral) dispersion coefficients for various atmospheric conditions.

set equal to zero; the result

$$\chi(x) = \frac{Q}{\pi u \sigma_y \sigma_z} \exp\left(-\frac{h^2}{2\sigma_z^2}\right) \tag{12.4}$$

then applies to the radioactivity concentration on the ground along the plume centerline where the dose would be a maximum for a given value of x. This expression is assumed to hold for the first 8 hours after a presumed release of radioactive material, e.g., after an LOCA. For times greater than 8 hours, the wind is supposed to vary somewhat in direction so that the plume is spread uniformly over a 22.5° sector; the resulting equation is then

$$\chi(x) = \frac{2.302\, Q}{x u \sigma_z} \exp\left(-\frac{h^2}{2\sigma_z^2}\right). \tag{12.5}$$

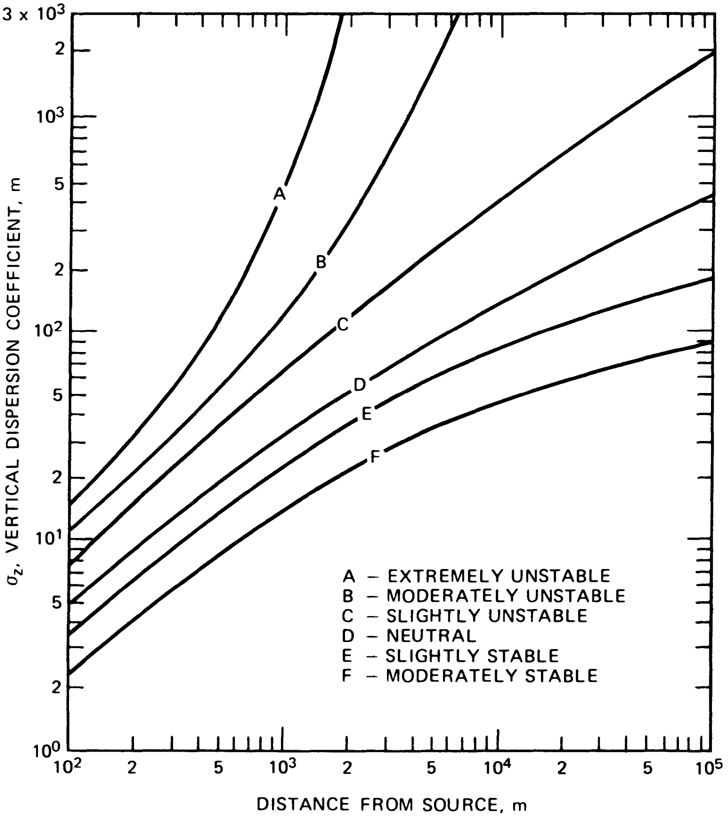

Fig. 12.13. Vertical dispersion coefficients for various atmospheric conditions.

12.164. For radioactivity escaping from a containment structure, the release is fairly close to ground level; hence h in equations (12.4) and (12.5) is taken as zero and the exponential terms are then equal to unity. The atmospheric conditions for the first 24 hours are assumed to correspond

TABLE 12.2. Equilibrium Activities of Fission-Product Radioiodines

Nuclide	Radioactive Half-Life	Effective Fission Yield	Equilibrium Activity [10^{13} Bq/MW (thermal)]
Iodine-131	8.05 d	0.028	89
Iodine-132	2.3 h	0.041	130
Iodine-133	21 h	0.068	211
Iodine-134	52 min	0.072	226
Iodine-135	6.7 h	0.064	200

to Pasquill F with a wind speed of 2 m/s (about 4.5 mph); under these conservative conditions the calculated doses will be much larger than would be generally expected. More realistic atmospheric conditions are postulated for later times.

12.165. In the case of a release near ground level, allowance must be made of the turbulent wake resulting from the presence of the reactor building. The wake will cause additional dispersion of the radioactivity, thereby reducing the concentration on the ground. For the first 8 hours after a release, a dispersion factor, not exceeding 3, is introduced into the denominator of equation (12.4) with $h = 0$; the factor decreases with the (minimum) cross-sectional area of the building (at a given distance from the source) and with the distance (for a given area). After 8 hours, the effect of the building wake is ignored, i.e., the factor is unity.

12.166. Examination of equations (12.4) and (12.5) shows that, for a given (or zero) release height h, the quantity $\chi(x)u/Q$ is a function only of the distance in the x direction from the release point and of the Pasquill criterion. Curves of $\chi(x)u/Q$ versus x for various release heights and atmospheric conditions have been published. These curves greatly facilitate the evaluation of $\chi(x)$ from a known source.

Whole-body dose

12.167. For a radioactive cloud (or plume) with dimensions that are small relative to the effective range of the gamma rays emitted from the cloud, the dose at a given point will include radiations received from various parts of the cloud. Calculations of the dose rate (or dose) are then very complex. If, however, the cloud is large and uniform, an equilibrium condition exists, and the rate of energy absorption per unit volume of air will be the same as the rate of energy release. Suppose that the concentration of a given radionuclide in a large uniform cloud is χ Bq/m^3, i.e., χ dis/s $\cdot$ m^3, and the average gamma-ray energy per disintegration is $\overline{E}_\gamma$ MeV, i.e., $1.60 \times 10^{-13}\, \overline{E}_\gamma$ J/dis, then

$$\text{Rate of energy release (or absorption) in air} = 1.6 \times 10^{-13} \frac{\chi \overline{E}_\gamma}{\rho} \text{ J/kg} \cdot \text{s},$$

where ρ is the density of air in kg/m^3. In soft-body tissue, the energy absorption would be 1.1 times that in air;* hence, upon taking ρ as about 1.3 kg/m^3 and recalling that 1 rad represents the absorption of 10^{-2} J/kg,

*According to the data in Table 6.1, the mass energy-absorption coefficient of gamma rays in soft body-tissue is roughly 1.1 times that in air over a wide range of energies.

it follows that

$$\text{Absorbed dose rate in body tissue} \approx 1.2 \times 10^{-11}\ \chi\overline{E}_\gamma\ \text{rad/s}. \qquad (12.6)$$

12.168. The value of χ obtained from equation (12.4) or (12.5) is for a receptor at ground level. The gamma-ray dose received by an individual on the ground at the center of an "infinite" cloud would be roughly half that given by equation (12.6) owing to the presence of the ground which limits the source to a 2π solid angle. The gamma-ray dose rate in body tissue (or the whole body) would then be

$$\text{Whole-body dose rate} \approx 0.6 \times 10^{-11}\ \chi\overline{E}_\gamma\ \text{rem/s},$$

where rems have been used in place of rads since they are essentially equivalent for gamma rays (§6.37). If the radionuclide concentration is assumed to remain constant over the exposure period of t s, the gamma-ray dose received in that time is

$$D_\gamma \approx 0.6 \times 10^{-11}\ \chi\overline{E}_\gamma\, t\ \text{rem}.$$

In view of the various assumptions and approximations made, this expression gives doses that are substantially larger than would realistically be expected.

Thyroid dose

12.169. The dose to the thyroid would arise mainly from breathing air containing radioiodines (see Table 12.2). If χ_i Bq/m^3 is the average concentration of radioiodine in the air, B m^3/s is the breathing rate, and t s is the time during which the iodine-laden air is breathed, the initial amount $C_{0(i)}$ Bq of radioiodine taken into the lung is given by

$$C_{0(i)} = \chi_i B t\ \text{Bq}.$$

For a conservative calculation, no allowance would be made for depletion of radioactivity from the plume as a result of deposition on the ground or for radioactive decay in transit from the reactor site to the point at which $C_{0(i)}$ is to be determined. In a realistic calculation, however, such factors would be taken into account.

12.170. Hence, in a conservative calculation, χ_i is obtained from equation (12.3), etc., with Q_i Bq/s equal to the rate of escape of radioiodine from the containment vessel (cf. §12.159). For the first 8 hours after a postulated accident, B is assumed to be 3.47×10^{-4} m^3/s, and from 8 to 24 hours it is 1.75×10^{-4} m^3/s. Subsequently, until the plume has passed, the normal breathing rate of 2.32×10^{-4} m^3/s is assumed. From these data, the value of $C_{0(i)}$ for a particular radioiodine can be determined. For iodine-131, this is related to the thyroid dose commitment by equation (6.8) in the form

$$D_\infty \text{ (in rem)} = C_{0(i)} \text{ (in Bq)} \times 4.1 \times 10^{-5},$$

where rads are replaced by rems because they are essentially equivalent in this instance. Similar evaluations may be made for the other radioiodines or, more simply, the dose from iodine-131 may be multiplied by 1.9 to obtain the total iodine dose. The factor 1.9 takes into account the equilibrium amounts (Table 12.2) and the dose equivalents per becquerel of the various isotopes.

Emergency Response Planning

12.171. In the United States, applicants for reactor licenses must provide plans for coping with emergencies. Requirements are given in 10 CFR 50.47 and in its Appendix E. An Emergency Planning Zone (EPZ) must be specified related to a plume exposure pathway following a severe accident. The radius of such a zone would depend on the typographical and access route features of a specific site but is likely to be on the order of 16 km (10 miles). However, an EPZ for a food ingestion pathway must also be considered, which may have a radius of about 80 km (50 miles).

12.172. Plans must be established for both on-site and off-site organizations for managing emergency situations, including protective measures, communication, transportation, and many other considerations. Lessons learned from the Chernobyl accident response (§12.187) have been helpful in identifying needs. The general trend of NRC regulations has been to favor new plant siting significantly away from population centers. However, in those European countries that are densely populated, it is not possible to have the luxury of remote siting. Therefore, reliance must be placed on a defense-in-depth design philosophy, which includes a very conservatively designed containment structure as backup.

Seismic Design Criteria

12.173. Commercial nuclear power plants are designed to withstand the ground motion caused by the most severe earthquake that is likely to be experienced. From historical records of seismic events in the plant vicinity, the earthquake that would be expected to produce the largest ground motion at the reactor site is predicted. This is called the "safe shutdown earthquake." Analysis must show that the reactor can be tripped and the engineered safety features will function properly if such an earthquake should occur. Furthermore, the plant must be capable of remaining in full operating condition should a specified "operating basis earthquake" be experienced. Licensing requirements are specified in 10 CFR 100, Appendix A.

12.174. In developing the design necessary to meet these requirements, dynamic analysis based on expected ground acceleration spectra is applied to those components and structures (called Class I) vital to safe shutdown. There must be assurance that the resulting stresses will comply with applicable codes. Furthermore, special attention must be paid to the prevention of common-mode failure (§12.18) in the event of an earthquake. If the reactor plant is designed to withstand a maximum horizontal acceleration of 0.3 of gravity as the result of a safe shutdown earthquake, it is considered to be satisfactory for most sites in the United States.

12.175. The design philosophy used in the United States to accommodate possible earthquake motion is to render the structure and its components as rigid as required to prevent damage. In addition to planning for ground motion in the basic structural design, various devices such as struts and springs are used to manage stresses. In the case of piping which must expand and contract with temperature changes, mechanical shock arrestors, or *snubbers*, are used to attach the pipes to rigid structures. In some other countries, a design philosophy of decoupling the entire plant from the surrounding soil by the use of energy absorbing materials is favored [19].

12.176. The seismic design of nuclear power plants has become a specialized engineering subdiscipline. Computer models are available for the dynamic analysis of proposed structures and their components [20]. Such analysis is supported by various experimental programs that employ components and scale models on shake tables. An important design step is to characterize the damage produced by an expected earthquake to a candidate design and make changes as necessary. However, the prediction of the ground acceleration to be expected at a specific site is a technical challenge. For example, during the 1994 Los Angeles earthquake, unexpected high accelerations were measured, which varied considerably over short distances.

Other Natural Events

12.177. A nuclear power reactor installation must also be able to withstand the substantial wind and pressure loading of a significant tornado. Systems vital to the safety of the plant must be protected from missiles generated by the tornado. Plants located near rivers must be able to survive a postulated flood, called the "probable maximum flood." The possibility must also be considered of flooding that could result from a hurricane or failure of an upstream dam.

ACCIDENT EXPERIENCE AND ANALYSIS

Introduction

12.178. Over the course of nuclear reactor development, some accidents have occurred, which is not entirely unexpected for a new technology. In all cases, the effects were managed and the general public was not affected [21]. Commercial reactors earned an excellent record until the Three Mile Island accident in 1979, in which there was a negligible release of radiation to the public but major economic losses occurred. The accident also led to important changes in plant design, management, and regulation. In 1986, the Chernobyl accident, as a result of which there were 32 direct fatalities, had worldwide consequences on public acceptance of nuclear power and has had an effect on the design of future reactors. We will examine the effects of each of these two accidents on nuclear reactor engineering practice.

The Three Mile Island Accident [22]

12.179. The accident, which occurred in March 1979 near Harrisburg, Pennsylvania, to a PWR of Babcock and Wilcox design, was the result of an unusual combination of operator errors and equipment deficiencies. A lengthy chain of events began with an inability to pump feedwater into the secondary system, which resulted in a turbine trip shortly thereafter. The pressurizer relief valve (§13.16) then opened in response to the rise in primary system pressure but was stuck in the open position, which was unrecognized by the operators. Thus, we had what was comparable to a small-break loss-of-coolant accident.

12.180. As a result of operator confusion, the coolant inventory was further reduced by their throttling the high-pressure injection flow and draining water through the letdown line in the system used to regulate the coolant boron concentration. Boiling in the core as a result of the reduced pressure caused the operators to shut down the primary circulating pumps, leading to further steam buildup, core uncovering, and severe core damage. Earlier, overpressure on a tank receiving discharge from the pressurizer led to a minor radioactive release to an auxiliary building and then to the environment through a vent stack.

12.181. The loss of coolant was finally halted after several hours and the core was subsequently reflooded. However, it was still difficult to cool the core with the primary pumps because of the large quantities of steam and some hydrogen in the system as well as the degraded core geometry. A number of "feed and bleed" maneuvers in which water would first be injected and then the pressure reduced proved helpful. The operators were then able to activate a primary pump and achieve a relatively stable condition after about 16 hours.

12.182. Feed-and-bleed cooling as an operator option has since been modeled extensively with the aid of such system codes as MAAP [16]. These analyses provide a picture of the response of a given system should feedwater be unavailable and depressurization be attempted by alternatively injecting emergency coolant and bleeding off inventory through the pressurizer. Such a picture provides guidance for the emergency measures required, particularly the time "window" that may be available for countermeasures.

Impact of the Three Mile Island Accident [23]

12.183. Following the accident, major reviews were carried out which proposed many reforms covering both industry and the Nuclear Regulatory Commission (NRC), most of which have since been implemented. In the industrial area, improvements fall into four categories:

1. Plant performance review and personnel training
2. Operational regulations
3. Plant equipment improvements
4. Research

12.184. In the first category, the Institute of Nuclear Power Operations (INPO) was established to perform in-depth analysis of operating experience and serve as a clearinghouse for information. Training and accreditation programs were established. Second-category changes include provision for additional technical staff on each shift and extensive improvements in operational procedures.

12.185. The third category is concerned with hardware. Major changes were made in control rooms and system instrumentation. New venting in the primary system was provided, for example. Hundreds of changes were required for each plant at a cost in the range of hundreds of millions of dollars per plant. A measure of improvement has been a reduction in the number of *significant events*, defined by NRC as one that has a potential safety implication. Research in such areas as source terms and degraded core analysis was expanded.

12.186. All of the measures taken certainly improved the safety of operating reactors in the United States and certain other countries that adopted the same measures. The accident demonstrated the value of the containment since the public was not affected by an almost worst-case accident. In addition, the "cleansing" of the fission product mix through the conversion of volatile iodine proved the need for extensive source term studies. Yet the fact that the accident was even possible seriously eroded the public image of nuclear power as an almost foolproof technology.

Chernobyl [24]

12.187. The destruction of Unit 4 of the Chernobyl Atomic Power Station in Ukraine on April 26, 1986 was the worst nightmare of nuclear engineers throughout the world. Attempts to convince a skeptical public that nuclear power is indeed a safe energy option would now be more difficult than ever, even though the reactor design is unique to the former Soviet Union and inherently unstable. A first step in discussing the accident and its impact is to become familiar with the system.

12.188. The reactor, one of the RBMK-1000 type, was boiling light-water-cooled and graphite-moderated. Fuel element subassemblies, within pressure tubes through which the coolant flowed, contained 18 rods arranged in concentric rings. At full power, the coolant channel exit steam quality was 14 percent. The core, 7 m high and 11.8 m in diameter, contained 1661 such channels. Refueling was accomplished at full power by a machine in a bay above the core, as shown in the Fig. 12.14 plant arrangement. A 2 percent enrichment was used, with 211 control rods of various types needed for reactivity compensation. The rated thermal power was 3200 MW.

12.189. As a result of the massive size of the reactor and fuel handling equipment, it was considered impractical to have a complete containment such as used for all U.S. reactors. Instead, an *accident localization system*, as indicated in Fig. 12.14, was provided. This consisted of various sealed compartments that enclose the circulating pumps, large piping, and other components, the failure of which could lead to a LOCA. This compartment system vents to a suppression pool. The cooling channel refueling connections were unprotected. Hence, the accident localization system was designed for a different type of accident, i.e., a large pipe LOCA, rather than the massive fuel channel failure that occurred.

12.190. Since graphite is the moderator in the RBMK-1000 reactor, voiding of the coolant, which is a neutron absorber, increases the reactivity. This positive reactivity coefficient was significantly greater at low power. Therefore, 700 MW(t) was specified as the minimum permissible continuous operating power level. An interesting positive reactivity effect is introduced when control rods are inserted as a result of the displacement of water at the bottom of the control rod channels by graphite control rod

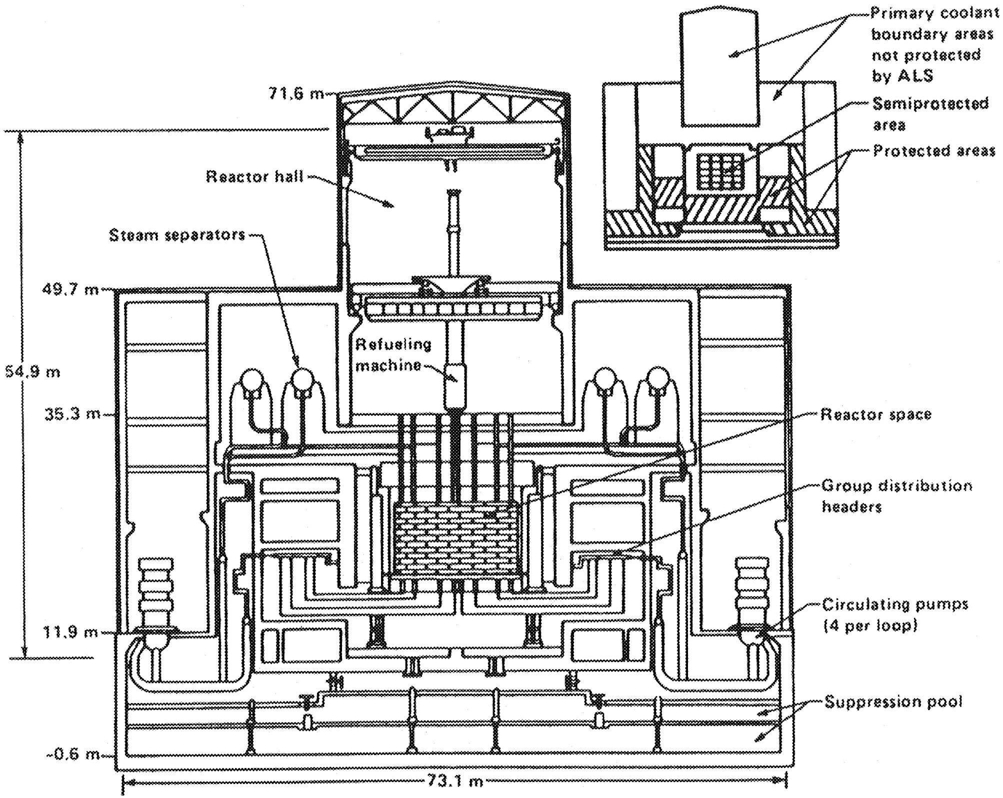

Fig. 12.14. Reactor building of Chernobyl Atomic Energy Station Unit 4. The inset shows the primary coolant boundaries enclosed by the accident localization system [24].

followers. During post-accident analysis, it was also concluded that the emergency rod insertion rate was slow by western standards.

12.191. The accident occurred during a test of turbine-generator coasting-down power, which was used to drive an emergency feedwater pump for about 1 minute in the event of the loss of off-site power, with some of the desired electrical conditions simulated. Although the intended reactor power for the test was 700 MW(th), errors by the operators resulted in a power loss to 30 MW(th), where xenon growth, particularly at the bottom of the core, from the previous higher power operation made it difficult to increase the power without withdrawing almost all of the control rods. Even with such action, only a power level of 200 MW(th) could be achieved, a level in violation of operating procedures because of the inherent instability of the reactor.

12.192. The test plan was initiated at this low power level by starting additional recirculating pumps as called for by an electrical simulation, which had the effect of reducing core voids and causing additional control rod withdrawal. As a result of operating difficulties, various protection devices were blocked out by the operators. When the recirculating pumps were allowed to coast down, as planned, coolant flow decreased and voids re-formed very rapidly in the pressure tubes, which increased reactivity because of the positive void coefficient, particularly at the bottom of the core. An emergency scram (trip) was initiated manually almost immediately, but the almost fully withdrawn rods could not be inserted fast enough to prevent a prompt critical power excursion. In fact, the rod insertion introduced some additional reactivity at the core bottom as a result of water displacement by the graphite followers.

12.193. The rapid vaporization of the coolant in the pressure tubes generated a shock wave that ruptured most of the tubes. Apparently, there were two excursions, seconds apart, the second a result of almost complete coolant voiding. The fuel became molten and generated an immense quantity of steam, which blew the 1000-ton steel and concrete biological shield off the top of the reactor. Hydrogen, formed by the reaction of fragmented cladding and water, exploded, severely damaging the building. Pressure tube ruptures provided an inlet and an outlet for air to feed combustion of the graphite, which was probably ignited by the exothermic zirconium–niobium oxidation reaction. The fire continued for several days and certainly complicated the management of the accident. The accident was categorized as level 7 on the International Nuclear Event Scale (INES).*

*The scale developed by the International Atomic Energy Agency provides for seven levels, with each step representing a factor of 10 in severity. Thus a level 3 incident would be roughly 10,000 times less serious than the level 7 Chernobyl accident.

12.194. In late 1986, after radiation levels had decreased somewhat, the damaged reactor building was enclosed by a concrete and steel shell, or "sarcophagus." Several years later, explorations inside the sarcophagus revealed that about 96 percent of the original fuel is contained in solidified "lava flows" in chambers below the reactor vault and in the form of dust and particles distributed inside the building. Since it was necessary to construct the sarcophagus as an emergency measure upon damaged foundations, it is expected that replacement will eventually be necessary.

12.195. The total release of particulate fuel from the core is estimated at 3.5 percent of the original inventory. This corresponds to a radioactivity release into the environment on the order of 50 million curies. Fallout over parts of the former Soviet Union and other countries was widespread. Considering the most biologically sensitive fission products, 100 percent of the rare gases was released, about 20 percent of the iodine, and roughly 13 percent of the cesium. The estimated 2 million curie release of the 30-year half-life cesium-137 is the most significant long-term contamination contribution. The initial 32 fatalities all occurred on-site. The long-term consequences of the exposure received by about 200 plant workers treated for radiation sickness at the time of the accident and off-site exposure to fallout by some of the nearby general population remains uncertain.

Impact of the Chernobyl Accident

12.196. Numerous changes have been made to the remaining RBMKs in the former Soviet Union to improve their safety. These include reducing the void reactivity coefficient by adding fixed absorbers to the core, increasing the fuel enrichment from 2.0 percent to 2.4 percent, adding 24 fast-acting shutdown rods, and modifying the control rod design. The new core physics characteristics are claimed to make the potential for prompt criticality extraordinarily low for all design-basis accidents [39]. The need for improved management controls, operator training, and discipline was also recognized. Since some of the radiation released in the accident affected the food chain in other countries, it became clear that there was a need for better international cooperation in exchanging information, developing standards, and other safety-related matters. The Chernobyl accident provided few technical lessons applicable to U.S. LWRs since their design is so different. Also, many institutional improvements had already been made as a result of the Three Mile Island experience. Emergency response to the Chernobyl accident provides useful data applicable to all reactor operations. In very many ways, the Chernobyl accident provided lessons and experiences that improve nuclear power safety. Unfortunately, it also served to reinforce the fears of nuclear reactors held by many members of the public.

SEVERE ACCIDENT MANAGEMENT [25]

Introduction

12.197. During recent years, the nuclear industry, together with NRC, has been implementing plans to manage severe accidents. By *accident management* is meant the innovative use of various resources, systems, and actions to prevent or mitigate a severe accident. Although such accidents are "beyond design basis" and would involve some degree of core melting, an understanding of the various possible accident mechanisms can permit the development of strategies to control the release of radionuclides to the environment. For example, the Three Mile Island accident could have been prevented by proper operator action, but once it progressed, it could have been better managed if needed strategies had been developed previously.

12.198. A severe accident management program consists of several different elements.

1. Information and analysis
2. Supporting instrumentation
3. Accident management strategy development
4. Equipment modification
5. Personnel training

These are interrelated and may be expressed somewhat differently. Our primary purpose here is to identify the features needed for a severe accident program.

Information and Analysis

12.199. Continuing research and studies contribute to a large body of severe accident information for various reactor types. The work has been carried out under the auspices of the Nuclear Regulatory Commission, the Nuclear Management and Resources Council (NUMARC), which serves as liaison between industry and the NRC, and the Electric Power Research Institute. NUMARC has a group that coordinates severe accident activities.

12.200. Basic research has been directed toward understanding the physical mechanisms occurring during severe accident scenarios, such as those described earlier (§12.95 et seq.) and to model the various steps analytically. Many projects are thermal-hydraulics oriented. For example, the thermal behavior of a pool of molten core debris at the bottom of a reactor pressure vessel can affect the practicality of preserving the vessel integrity by external cooling.

12.201. The assessment of severe accident risks carried out under the NUREG-1150 Program (§12.234) can identify areas of vulnerability specific to individual plants. Such work can then serve as the focus of further studies which can then lead to the subsequent development of appropriate remedial actions.

Supporting Instrumentation

12.202. During a severe accident, plant instrumentation must provide the operating staff with an indication of the plant condition. The program must therefore address the need for additional instruments, the survivability of vital instruments during an accident, and the interpretation of readings.

Accident Management Strategy Development

12.203. Should an accident occur, what is to be done? Strategies need to be considered carefully since some adverse effects may accompany the apparent advantages of a given strategy. For example, should the core slump to the bottom of a PWR pressure vessel, a logical strategy is to flood the surrounding reactor cavity, which will submerge the lower vessel head. Hopefully, this would prevent vessel failure and help cool the contained debris. However, should the vessel breach, a steam explosion is now a possibility as a result of the reaction between the molten fuel and the water that has been introduced.

12.204. Assessment of severe accident management strategies by various means is therefore a necessary feature of planning. Procedures based on decision trees (§12.212), an operations research method, can be used to evaluate systematically all of the options and to develop choices. Sensitivity analysis may also be included [38]. Then guidelines in a form useful for operators can be prepared.

Equipment Modification and Personnel Training

12.205. The role of these elements is clear. As a result of extensive studies and strategy development, some equipment modifications may appear desirable. For example, one study showed that the addition of one or more high-pressure ECCS pumps reduces the core melt probability. Also, once management strategies have been developed, operating personnel need to be trained in their use.

RELIABILITY AND RISK ASSESSMENT [26, 27]

Introduction

12.206. We may define *reliability* as the *probability* that a system or component will perform a specified function (or not fail) for a prescribed time. Now, if in addition, we consider the consequences of failure, i.e., a financial loss or injury to people, we have the concept of risk. Mathematical reliability models as design tools evolved from statistical sampling procedures used for quality control and biological research. The underlying probability theory, of course, is centuries old.

12.207. The aircraft industry provided an early incentive for the development of reliability engineering since in aircraft, one cannot accept overly conservative design if additional weight would be required. In the German World War II missile program, the concept of integrated system reliability analysis was pioneered. In the United States during subsequent years, the requirements of both the space and military missile programs led to the development of sophisticated systems studies which included the fault tree concept.

12.208. The application of probabilistic risk assessment (PRA) to nuclear reactor safety began in the 1960s in the United Kingdom with the study of advanced gas-cooled reactors [28]. As we shall see shortly, PRA is an analytical technique for modeling the possible failure of subsystems and components where there are complex interactions. Following the British work, the U.S. Atomic Energy Commission sponsored a number of studies of reliability techniques applied to U.S. reactors. Next, a major advance in PRA development was made in a 3-year effort known as the Reactor Safety Study, the Rasmussen Report, or WASH-1400, published in 1975 [6]. The performance of a PRA as a licensing requirement was instituted in 1982. Also, ongoing PRA studies applicable to existing plants have resulted in various modifications to improve reliability. We will discuss various PRA efforts after we have presented additional introductory material.

Deterministic and Probabilistic Analysis

12.209. Before we introduce some PRA principles, it is desirable to clarify the difference between deterministic and probabilistic safety analysis. The term *deterministic* has a philosophical basis which refers to the mechanical correspondence between causes and effects. For example, we could consider a small-break LOCA in a PWR as a "cause," and by suitable analytical modeling determine the maximum fuel clad temperature as a function of the break area. The clad temperature would be the "effect" and when related to prescribed limits provides us with a "safety margin." An evaluation of numerous safety margins is required in licensing applications (§12.132). In contrast, probabilistic safety analysis utilizes statistical methods to evaluate failure *probabilities* resulting from various initiating *events*. Here we are concerned with binary states; i.e., an initiating event might be the transition of a given component from an operating state to a nonoperating state. Then this state could affect the condition of related components, as we shall see.

Elementary Binary State Concepts

12.210. As an introduction to failure concepts, let us consider a component that is either functioning normally or failed. We define the *reliability*, $R(t)$, as the probability of survival to age t. Then,

$$R(t) = \frac{\text{number surviving at } t}{\text{total sample (population)}}.$$

We can define *unreliability*, $F(t)$, as the probability of failure up to age t (t not included):

$$F(t) = \frac{\text{number of failures before } t}{\text{population}}$$

Now, $R(t) = 1 - F(t)$. If we consider the proportion of the population, or sample, that will fail between t_1 and t_2, we can introduce the *failure probability density*, $f(t)$, where

$$F(t_2) - F(t_1) = \int_{t_1}^{t_2} f(t)\, dt, \tag{12.7}$$

where $f(t)\, dt$ is the probability of failure in dt about t:

$$f(t) = \frac{dF(t)}{dt}.$$

We are also interested in the *rate* of failure, $r(t)$, which is sometimes called the hazard rate. This is the probability of failure *per unit time* at age t; i.e., the device must have survived to time t:

$$r(t) = \frac{f(t)}{1 - F(t)} = \frac{f(t)}{R(t)}. \tag{12.8}$$

Here $R(t)$ is the number of survivals at t divided by the initial population.

12.211. The behavior of $r(t)$ for many devices is described by the classic "bathtub curve" shown in Fig. 12.15. Characteristically, there are significant early failures during a burn-in period arising from poor manufacturing quality control. Subsequently, there is a flat period of random failures followed by a rising rate in the wear-out range. This concept provides only

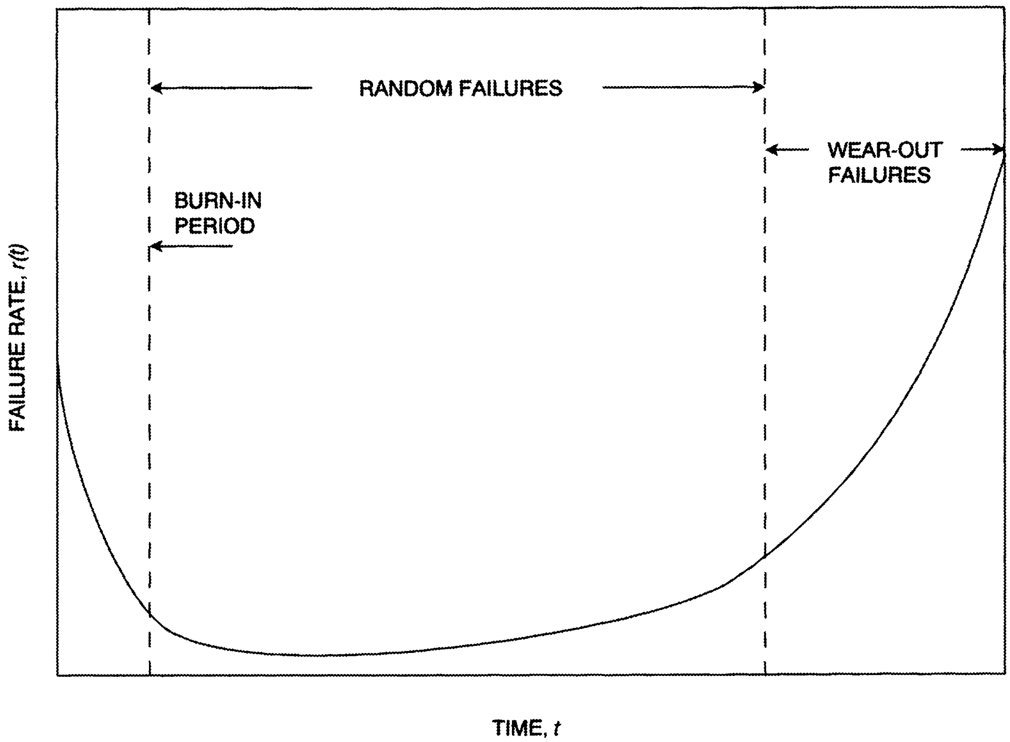

Fig. 12.15. Bathtub curve.

a "taste" of a discipline known as reliability engineering, in which component and system performance are analyzed.

Fault Tree Analysis

12.212. Fault tree analysis, which is essentially a graphical communication tool based on Boolian algebra, is a key ingredient of reliability analysis and risk assessment. It has the value of identifying weak links in complex system interactions as well as providing insight into system behavior. Since risk is determined by relating the system failure probability to consequences, fault tree analysis is the first stage of risk assessment.

12.213. A *system fault tree* is a logic diagram which depicts the component failure modes (or, in general, faults*) that combine to produce a failure of the system. First, an undesired event or failed state of a system is postulated; this is called the *top event*. The latter is then traced back, step by step, to identify the combinations of sequences of other events or failures that could lead to the top event. After proceeding through a number (often in the tens) of secondary stages, a set of primary failures is reached which can not (or need not) be traced further. In many cases, the failure of a complex system depends on the failures in several subsystems. Fault trees for the latter can then be analyzed separately, and the results provide part of the "primary" input to the system fault tree.

12.214. The first few stages of a fault tree are shown in Fig. 12.16, for which the top event is an insufficient flow of water from the spray intended to cool the containment atmosphere of a PWR. There are two redundant systems, A and B, either of which alone is capable of providing the necessary cooling. Hence, both systems must fail in order for the top event to occur; this is indicated by the AND logic gate, with a rounded top, relating the second level events to the undesired (top) event. The most immediate cause of the failure of the spray system would be insufficient water to the header to which the spray nozzles are attached; hence, this is regarded as the second level of the fault tree.

12.215. The third level identifies four different faults, each of which is sufficient to cause the second level event. The third level is therefore related to the second level by an OR gate, with a pointed top. Events within circles or diamonds need not be developed further. A circle indicates the failure of a component for which the probability of occurrence is available or can be determined. An event in a diamond is a fault, in its general sense, which is not developed either because of its minor significance or because of lack of information. An event within a rectangle in a fault tree is one that must

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*The term *fault* as used here refers to a mechanical failure or malfunction but also includes any type of abnormal situation, such as deviation from operational procedures.

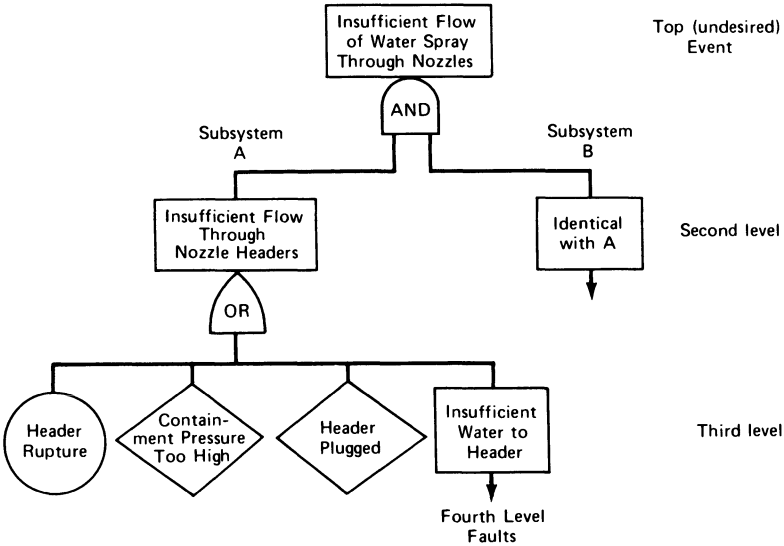

Fig. 12.16. Simplified fault tree.

be traced down to a lower level (or levels). Levels beyond the third are not shown in Fig. 12.16, although they have been traced in some instances through more than 20 stages.

12.216. Fault tree analysis has both qualitative and quantitative aspects. From the qualitative standpoint, the analysis can often identify a critical subsystem or component where a failure could have a marked influence on the failure of the entire system. Similarly, in a complex network, one event path may be found to have a controlling effect on the total failure. In these circumstances, it would be desirable to introduce redundant elements.

Quantitative Fault Tree Analysis

12.217. Fault tree analysis may be used to predict the probability of system failure based on data or estimates of failure at the subsystem level. For this purpose, fault tree models, like the one in Fig. 12.16, are constructed and a probability is associated with each event; the results are then combined, along the following lines. Suppose that a top event A is connected through an AND gate to three events at the second level for which the probabilities are B_1, B_2, and B_3, respectively. The probability of A is then $B_1 \times B_2 \times B_3$. On the other hand, if the three second-level events are connected with A by an OR gate, the probability of A is $B_1 + B_2 + B_3$. The second-level events are themselves connected with third-level events and the probabilities of these events in turn determine B_1, B_2, etc. By proceeding in this way through all the levels, down to the lowest, of a fault tree, the probability of the top event is determined.

12.218. It is evident that when a tree consists of many levels, as is commonly the case, the calculation of the probability can become very complex. As a general rule, therefore, fault tree models are first simplified as far as possible by using Boolean algebra techniques. Such simplification is aided by eliminating events of low probability and combining related events into a single event. Although the approach is straightforward, computer representation is generally necessary, even after simplification. Probabilistic analysis also makes use of event trees which are discussed in the next section.

12.219. There are a number of uncertainties in fault tree analysis that place error limits on the results. For example, because of the lack of experience of an event, e.g., a double-ended pipe break in the primary coolant system of a reactor, the probability of failure cannot be established and must be estimated. Allowance should also be made for human error and the unavailability of systems or components because of test or maintenance requirements. Other uncertainties may arise from possible omission of important faults and from unforeseen common-mode failures. Furthermore, in the fault tree model a component or subsystem is regarded as either failed or not failed; there is thus no allowance for a partial failure in which a subsystem may be operative but not at its full efficiency.

Event Trees

12.220. A technique for the quantitative assessment of the risks (or probabilities) of specified accidents is based on the use of *event trees*, in conjunction with the fault tree method just described. An event tree is a graphical means of identifying the various possible consequences of a particular event (or failure) called the *initiating event*. These consequences depend on the different options that are applicable following the initiating event. Event trees are similar in principle to the "decision trees" widely used in making business decisions. Since the event tree starts with an initiating event, it represents a *deductive* logic process, whereas the fault tree is inherently *inductive*.

12.221. An outline of an event tree is depicted in Fig. 12.17, where the initiating event is a large pipe break leading to an LOCA in a nuclear power plant; let P_1 be the probability of the occurrence of this event, as determined from a fault tree analysis. The next step is to consider whether or not electric power will be available to operate the active engineered safety features. The probability that electric power will fail, as estimated from the appropriate fault tree, is taken as P_2. If power (including auxiliary

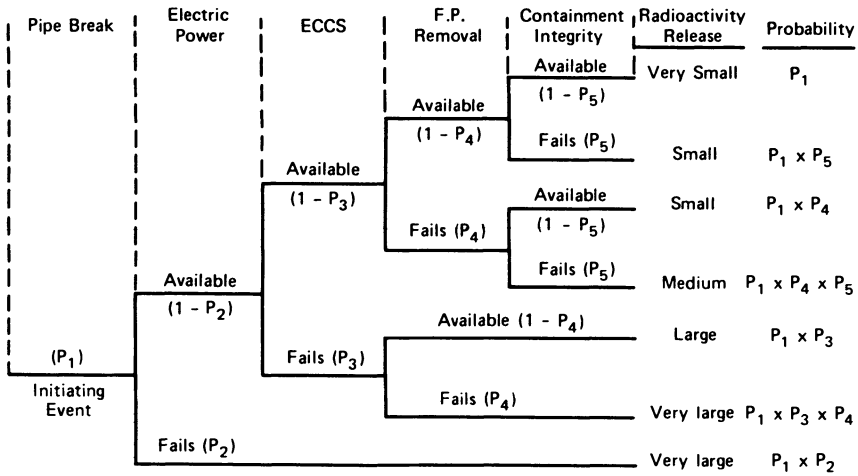

Fig. 12.17. Simplified event tree.

power) is not available, the active safety features will not operate and the core will be disrupted, leading to a very large release of radioactivity. The probability of this sequence of events, as indicated on the bottom line of Fig. 12.17 is thus $P_1 \times P_2$.

12.222. If electric power is available, the next event to consider is whether or not the ECCS will operate; suppose the probability of failure is found to be P_3. Subsequent options are concerned with the fission-product (F.P.) removal system and the containment integrity; the probabilities of failure are P_4 and P_5, respectively. At each stage, the probability that the system *will* be available is $1 - P$, where $P = P_2, P_3$, etc.

12.223. The overall probability of a chain of events, as given at the right of Fig. 12.17, is the product of the probabilities of the individual events in the chain. Hence, the second line from the bottom should include the factor $1 - P_2$, and the third from the bottom should include $(1 - P_2)(1 - P_4)$. However, P_2, P_3, etc., are small and so the values of $1 - P$ are taken to be unity in each case.

12.224. It is apparent that if the failure probabilities, i.e., P_1, P_2, P_3, etc., were known from the appropriate fault tree analyses for each of the systems in the event tree, the overall probabilities of the different failure consequences could be calculated. Thus, a combination of an event tree with a number of fault trees provides a means for evaluating the risks associated with various conceivable consequences of accidents.

12.225. The foregoing description of an event tree is presented in a simplified form to illustrate the general principle of risk assessment. In practice, several of the systems shown in Fig. 12.17 would be divided into subsystems, each of which would have its own event tree supplemented by the requisite fault trees. As is the case with fault tree analysis, uncertainties are involved in the use of event trees. For example, no allowance is made for the possibility of a partial failure or for a delay before a safety system becomes operable. Nevertheless, risk assessments based on fault trees and event trees are useful in estimating the probabilities of various accident sequences.

Computer Modeling

12.226. Many code packages are available for the development and analysis of system logic models. Many are specialized depending on the application desired. Although a discussion of such applications is beyond our scope, it is useful to mention several characteristics of fault trees that affect modeling.

12.227. The concept of *cut set* is important in fault tree analysis. In a collection of basic events, called a cut set, if all of these basic events occur, the top event is guaranteed to occur. In words, a cut set is defined as a set of system events that, if they all occur, will cause system failure. For example, in the simplified fault tree shown in Fig. 12.16, if we have the rupture of *both* headers *A* and *B*, the top event will occur and we have a cut set. A *path set* is complementary to a cut set in that it consists of a group of events (or failures) which must *not* occur in order to ensure that the top event will *not* happen. The path set usually includes events in addition to those in the cut set.

12.228. Since a large system may have thousands of cut sets, it is necessary to simplify the analysis by identifying and eliminating subsets which tend to duplicate simpler logic paths and therefore are not essential. Therefore, one class of computer program is used to generate *minimal cut sets*, usually by Boolean manipulation techniques. A classic code for this purpose, known as qualitative evaluation, is MOCUS [29]. Although various codes are available to assist in the actual construction of fault trees by using automated procedures, some level of manual involvement is usually required.

12.229. The next requirement is to carry out a quantitative evaluation by introducing probabilities to the logic gates. For example, an extensive data base exists for the reliability of mechanical and electrical components which can be used in a suitable computer model to calculate the failure probability of the top event. As pointed out in §12.217, the probabilities of each event in a cut set are multiplied together which could result in some very small values to be added at OR logic gates. Thus, some simplification is possible. A typical code for quantitative evaluation is WAM-BAM developed by the Electric Power Research Institute [30].

12.230. Various other codes are available for uncertainty analysis and consideration of common cause failures. Since often no data base is available to establish probabilities for many logic gates, expert system procedures have been used in recent studies as described in the following section. At any rate, the establishment of confidence levels for the results continues to receive major modeling development attention.

Risk Assessment Studies

12.231. To assess risks, we need to relate failure probabilities with consequences. For this purpose, the event tree (§12.220) links an initiating event, with a probability determined by fault tree analysis, through a chain of events with evaluated probabilities, leading to radioactive release to the environment and transport to populated areas. Finally, the dosage received and known health effects provides a measure of the consequences of the initiating event. An example of such a result is shown in Fig. 12.18, which is a risk curve from the 1975 Reactor Safety Study. Here, the ordinate is the frequency (or probability) that fatalities of magnitude X or *greater* will be produced.

12.232. Modeling of the various steps leading to risk assessment results in a considerable challenge. We need to consider the source term and severe accident progression steps, atmospheric transport, and finally, health effects. Therefore, results such as those shown in Fig. 12.18 have substantial uncertainty bands, the magnitude of which have been debated. Typically, such bands cover about two orders of magnitude. However, since the failure probabilities are so small, the reactor risks are still orders of magnitude less than natural or other industrial risks.

12.233. As mentioned, the pioneering reactor risk assessment study was the 1975 Reactor Safety Study. Subsequent critiques identified various weaknesses in the methodology and questioned the confidence levels estimated and some of the results. However, follow-up studies tended to confirm the general order of magnitude levels of the RSS results. Considering that the RSS prediction of a meltdown probability of once in 20,000 reactor-years of operation, it is logical to conclude that the systems are acceptably safe, even if there is an order-of-magnitude uncertainty in the probability prediction.

12.234. A comprehensive risk assessment study with emphasis on severe accidents was completed in 1990 [31]. This effort, commonly referred to as the NUREG-1150 study was the most ambitious since the 1975 Reactor Safety Study (RSS). It utilized many methodology improvements developed since the earlier study, particularly in the areas of core damage evaluation, source terms, containment event tree development, and consequence modeling. New codes such as CRAC2 for consequence modeling,

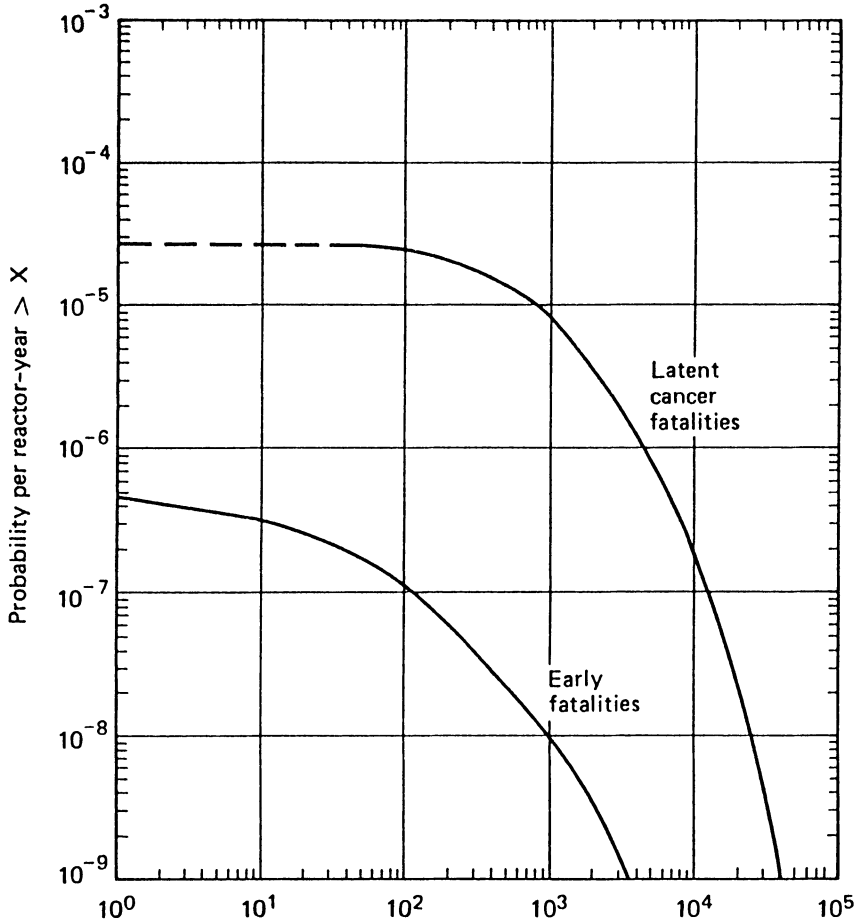

Fig. 12.18. Result of typical consequence modeling of accidental radioactivity release showing probability distribution for early fatalities and latent cancer fatalities (abscissa, X) [6].

including atmospheric dispersion, were utilized [32]. An important new feature was the use of an expert system approach as part of the uncertainty analysis.

12.235. The results, developed for five representative LWRs, predict risk levels somewhat lower than those estimated in the RSS. However, the overall uncertainty range envelopes the results of both studies. A great deal of detail was provided regarding the behavior of each of the plants studied and possible safety-related backfits to reduce some of the uncertainties identified.

12.236. Although general confirmation of earlier study meltdown probability predictions was important, the NUREG-1150 study's primary significance was to provide a much improved picture of plant component performance under severe accident conditions that provides guidance for design and regulation. Possible equipment backfits for each plant were identified and their cost-effectiveness determined. Information on the harsh environment during an accident provided a new basis for equipment qualification as part of the regulatory process. In fact, the study provides a basis for the reexamination of many safety-related equipment licensing requirements.

LICENSING AND REGULATION OF NUCLEAR PLANTS [33]

Introduction

12.237. We have already mentioned several aspects of the licensing and regulation of nuclear plants, which in the United States are the legal responsibility of the Nuclear Regulatory Commission (NRC). Consistent with the need to assure a high level of safety, the procedures required are lengthy, with much documentation required. In the past, two stages of approval have been required, the first for a construction permit, the second after the plant was built, for an operating license. Since public input opportunities at each stage permitted various groups who were opposed to nuclear power for philosophical and other reasons to obstruct the licensing process, long delays were common. These delays, particularly after the plant was built, added greatly to the financial burden to both the owning utility and the eventual ratepayers. An additional complication has been the need to satisfy requirements at the state level. For example, as a result of objections by New York State authorities regarding emergency planning, the Shoreham Plant, a new 800-MW(el) BWR, was dismantled, starting in 1992, without ever being operated.

12.238. Such experiences have discouraged investment in new nuclear power plants. In fact, the establishment of a stable and predictable licensing environment is considered essential by industry before a new plant is likely to be ordered. There is general agreement that reforms are needed, but they are still evolving during the early 1990s. However, we will describe the essential features of likely licensing reforms that will improve the investment environment and yet satisfy the need to "assure public health and safety."

12.239. In 1989, the NRC issued a landmark rule, 10 CFR 52, which provides for the certification of standardized plant designs, the issuance of early site permits, and the issuance of a combined construction and operating license. To avoid delays *after construction was completed*, the procedures were clarified in 1992 to provide for an informal post-construction hearing but limit it to issues of nonconformance to the previously approved combined license. We will describe these steps in the following sections.

Design Certification

12.240. We are concerned here with an "evolutionary" or "advanced" plant as described in Chapters 13 and 15, as no new plants of current design are likely to be built. Since standardization is a major feature of all new plants, it is practical to certify a standard design based on a comprehensive safety analysis report. The amount of information required is formidable and is normally contained in many volumes. It is comparable to what has been required in a Final Safety Analysis Report in an application for an operating license. A complete description of the plant must be included. The safety analysis describes the response of the system to a wide range of accident situations. Details of operation plans, test programs, and technical specifications are to be provided. The technical specifications list limits to be placed on all process variables. In particular, the core peaking factors as determined by emergency core cooling criteria are specified.

12.241. Review by the NRC staff of the Application for Design Certification is a lengthy process which normally includes requests by the staff to the applicant for additional information. It may be necessary to resolve issues by amending the application. In addition, the application is evaluated by the Advisory Committee on Reactor Safety (ACRS), a statutory body of up to 15 specialists in reactor physics, engineering, materials sciences, and other areas related to reactor safety. Finally, after approval is recommended, public hearings are to be held. In the past, an Atomic Safety and Licensing Board would be specially convened for this purpose.

Early Site Permit [37]

12.242. Site-related matters are evaluated separately. Documentation required includes such topics as a seismic description of the proposed site. An Environmental Report covers all aspects of the effects of the plant on the surroundings. In addition to radiological and thermal effects of plant operations, the effects on the environment of a spectrum of postulated accidents must be described. Also to be provided are economic and social effects as well as a cost-benefit analysis for the plant.

12.243. Emergency plans must be described. It should be recognized that requirements of the Environmental Protection Agency (EPA) and the Federal Emergency Management Agency (FEMA) also apply to the proposed site. Work is ongoing to minimize the impact on the nuclear industry due to the overlap of regulatory responsibilities among NRC, EPA, FEMA, and state safety and environmental agencies.

12.244. At any rate, after the site application has been approved provisionally, a public hearing is held to allow for additional input. After any issues that may be raised are resolved, an early site permit is issued.

Combined Construction and Operating License

12.245. After an order has been placed for a certified plant to be located at an approved site, an application for a combined construction and operating license (COL) may be submitted to NRC. There is then an opportunity to evaluate all remaining safety issues prior to the start of construction, including those that may arise in another public hearing conducted at this stage. After a combined license is issued, construction of the plant may proceed.

12.246. An important concern of owner utilities is the role of any post-construction, pre-operation public hearing. The objective of both NRC and the utilities is to avoid any "unreasonable" delays in operational startup after the very substantial investment in plant construction has been made. Therefore, present plans are to provide for only an informal hearing, with consideration limited to whether the facility has been constructed, and will be operated, in *conformity* with the license. Furthermore, unless the NRC determines that there will not be adequate protection of the public health and safety, operation may proceed, with the rationale that minor issues of nonconformity can be resolved later.

12.247. During operation, the NRC continues to play a major regulatory role as described in various sections of this book (see, e.g., §14.47). Also, so-called Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC) are specified in 10 CFR 52 "to provide assurance that the plant will operate in accordance with the design certification."

State-Level Regulation [34]

12.248. Investor-owned electric utilities are regulated by state public utility commissions, which have the responsibility for approving rate structures that will provide a fair rate of return on investment (§10.107). In most states, they also are required by statute to judge whether or not new generating facilities are indeed necessary. If approval is obtained, the issuance of a Certificate of Public Convenience and Necessity (CPCN) assures that the needed investment will be added to the rate base. Generally, an electric utility will obtain a CPCN before initiating NRC application procedures.

12.249. State agencies also have various requirements that affect site selection, particularly within coastal areas, where land and water use might be affected. Also, such matters as cooling water supply and effluent discharges usually require state approval. In some states there are restrictions on nuclear plant construction on radiological health and safety grounds. Overlapping state and federal jurisdiction can result in regulatory complications.

NUCLEAR REACTOR SAFEGUARDS

Introduction

12.250. Deliberate destruction of a reactor by sabotage could have consequences similar to those of a severe accident; hence, prevention of sabotage is an aspect of nuclear safety. The owner-operator of a nuclear plant (or other nuclear facility) is required to develop a physical security plan for the plant. According to NRC regulations, details of the plan must be included in the application for an operating license but they are withheld from public disclosure. Specific guidance for implementing plant security criteria is provided by the NRC and by the American National Standards Institute. Although protection against possible sabotage is primarily an industrial security problem, it is also an aspect of the nuclear safeguards activity concerned with the prevention of unauthorized use of "special nuclear material" (§10.109).

Protection Against Sabotage

12.251. In order to provide protection against possible sabotage, the reactor installation must be enclosed within at least two separate barriers with a "protected area" between them. An *isolation zone*, clear of all objects, must surround the protected area. Both the protected area and the isolation zone must be illuminated at night and continuously monitored to detect the presence of unauthorized persons or vehicles. Access to the protected area must be strictly controlled, and packages delivered to the area must be checked. Barriers are provided to control access by vehicles to vital plant areas. Alarms must be located at important points to indicate any unauthorized entry.

12.252. The capability of continuous communication must be maintained between guards at the nuclear plant and a central alarm station. In addition, the alarm station must maintain communications contact with local law enforcement authorities. The owner of the plant is required to maintain liaison with these authorities so that they can be called upon to provide assistance should it be necessary.

12.253. Safeguards for the protection of nuclear installations are under continuous study and review by the NRC. As better techniques are developed, plant licensees are required to adopt them. If the requirements are properly implemented, the hazard to the general public from sabotage of a nuclear reactor (or a spent-fuel reprocessing) plant should be extremely small.

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PROBLEMS

1. Serious core heat up began about 2 hours after the TMI accident reactor trip when the coolant pumps were shut off. At this time, decay heat generation was about 1.0 percent of the operating maximum linear heat rate of 43 kW/m for the fuel rods, which may be assumed to have the same dimensions as given in Example 9.5 (§9.76). (*a*) If zirconium melts at 1850°C and the coolant temperature at this point was estimated at 330°C, estimate the coolant heat transfer coefficient when clad melting occurred. (*b*) Estimate the central temperature of the fuel pellet.
2. A BWR has a coolant inventory of 230 Mg, of which 9.9 Mg is normally in the form of saturated steam. If 96 percent of the coolant inventory is condensed in the pressure suppression pool during a LOCA, estimate the temperature rise of the pool water, which has a volume of 3400 m^3. At the reactor pressure of 7.2 MPa, h_f = 1277 kJ/kg and h_g = 2769 kJ/kg. At the containment design pressure of 0.17 MPa, h_f = 483 kJ/kg, h_g = 2699 kJ/kg, and c_p = 4.3 kJ/kg · K.
3. The PSAR for the Diablo Canyon 1 PWR lists the primary coolant inventory volume as 351 m^3. With the aid of reasonable assumptions, estimate the internal diameter of a suitable cylindrical dry containment building.
4. A PWR containment design provides for an internal air cleanup system to reduce the airborne iodine concentration before a purge by recirculating the air through filters and absorbers. For such a system, the rate of change in iodine activity A per unit volume can be expressed by the relation

$$\frac{dA}{dt} = -A\frac{FE}{V},$$

where F is the internal recirculation flow rate, V the containment volume, and E the iodine removal efficiency. If the initial iodine activity in a PWR containment having a volume of 2.83×10^4 m^3 is 1.2×10^{-8} Ci/m^3, what will be the activity after 16 hours of operation of the cleanup system? The internal recirculation flow rate is 4.8 m^3/s and the removal efficiency is 0.63 in this case.

5. In PWRs, a large condensate storage tank normally provides a supply of feedwater to the steam generators to permit reactor cooldown for about 8 hours. Consider a typical 1000-MW(el) PWR as described in Table 13.1. By making approximate calculations, estimate the capacity of a suitable condensate storage tank.
6. In a PWR, the coolant is heated from 550°F to 600°F. To control a transient that results in the pressure rising from 2000 to 2100 psia, 200 lb of spray water is injected into the pressurizer. At the higher pressure, the total H_2O mass in the pressurizer has increased by 3080 lb. Initially, the pressurizer contained 22,150 lb of H_2O, of which 88 percent by weight was liquid. Finally, 9 percent by weight of H_2O was vapor. Determine the heat that was added. Some properties follow:

	Enthalpy (btu/lb)		Internal Energy (btu/lb)	
Condition (Sat.)	**Liq.**	**Vap.**	**Liq.**	**Vap.**
550°F	549	1190	545	1108
610°F	632	1158	624	1084
2000 psia	672	1135	662	1065
2100 psia	683	1127	673	1060

7. A 3000-MWt reactor has been operating at full power for 6 months when an accident occurs which releases 10 percent of the inventory to the containment. The containment leak rate is 0.1 percent of the volume per day. Calculate the direct, 2-hour inhalation dosage at the site boundary 700 m downwind from the containment using a conservative assumption of atmospheric conditions. The effective release height is 10 m. Neglect natural iodine removal mechanisms provided within the containment.
8. A centrifugal pump is used to inject water from a reservoir into a tank for safety purposes under certain circumstances. A motor-driven valve is between the pump exit and the tank. The protection system sends a signal actuating the pump and valve, but no water is injected. Using this failed state as the top event, construct a fault tree diagram to identify possible contributing causes for this failure.
9. An emergency cooling system consists of two centrifugal pumps in parallel, each drawing water from a single reservoir. Each pump has two parallel motor-driven valves on lines leading to the primary system. Thus, there are two pumps and four valves. Draw a fault tree for the failure of the system to deliver coolant to the primary system in the event of an accident.

10. (*a*) A mist consists of 10-μm-diameter water droplets in air at 20°C and 1 atm. Determine the settling velocity. (*b*) A mist contains a concentration of 0.3 kg of water per m^3 of air, the water being present at 2-μm-diameter droplets. Assuming an air temperature of 20°C, estimate the time required for 90 percent of the mist particles to coalesce.
11. One of the safety improvements that have been made to Chernobyl-type (RBMK) reactors is the addition of fixed strong-neutron absorbers to the solid graphite-moderated core. Considering spectral effects and the neutron economy, explain qualitatively when this change leads to a reduction in the void reactivity effect. What is the effect on the control system?

CHAPTER 13

Power Reactor Systems

INTRODUCTION

13.1. The practice of nuclear engineering is focused on the design and operation principles of commercial nuclear power plants. In previous chapters we have discussed many such principles that apply particularly to light-water reactors (LWRs). A brief overview of pressurized-water (PWR) and boiling-water reactors (BWR) was given in Chapter 1 to provide a general background and the material in subsequent chapters tended to further develop an understanding of such systems.

13.2. The purpose of this chapter is to provide additional details of some typical commercial LWRs as well as the Canadian heavy-water moderated reactor (CANDU). Although no new reactors have been ordered in the United States for some years, vendors do offer various improvements for operating reactors from time to time, particularly in fuel assembly design. The specifications listed here are primarily for orientation purposes, and are not necessarily representative of the latest trend.

13.3. Space limits the discussion of details of reactor technology, which are available in other sources [1]. Descriptive information is included with safety analysis reports submitted to the NRC and available as public documents. Reactor vendors also generally provide technical brochures in response to commercial inquiries.

13.4. So-called evolutionary designs have been developed by reactor vendors for new commercial LWRs in the 1000-MW(el) size class. These incorporate many improvements, particularly in the safety and cost reduction area, which will be discussed in this chapter. Advanced reactors, many with passive safety features, will be covered in Chapter 15.

PRESENT PRESSURIZED-WATER REACTORS

Introduction

13.5. A description of the PWR system starts with a summary of the significant design specifications as listed in Table 13.1. As an aid in comparing the systems described in this chapter, a consistent format has been followed which indicates general specifications, as well as values for the core, fuel, thermal-hydraulic, and control features. Specifications for different nuclear steam supply system (NSSS) models offered by the same manufacturer vary somewhat. Those listed in Table 13.1 are for a four-coolant-loop Westinghouse model, as described in a reference safety analysis report submitted to the NRC in 1974 [2]. This was about the end of the period when new reactors were ordered. Many currently operating Westinghouse four-loop PWRs were ordered somewhat earlier and have a core 12 ft in length to provide a thermal output of about 3400 MW. However, the 3800-MW version, which operates at the maximum power that the NRC will authorize, takes advantage of economy-of-scale savings. Such savings arise from the principle that fixed and operating expenses scale up as only a weak function of size. Therefore, if the plant is base loaded, the generating cost, on a unit energy basis, would be less for the larger plant.

Reactor Vessel and Core

13.6. The reactor vessel and internal components for a typical PWR are shown in Fig. 13.1. Control rod drive mechanisms are integral with the removable upper reactor head. Magnetic couplings for these drives are used across the pressure boundary. There are steel pads integral with the coolant nozzles for vessel support. These pads rest on steel base plates atop a structure attached to the concrete foundation. The low-alloy carbon steel vessel is clad on the inside with a minimum thickness of 3 mm of austenitic stainless steel. Neutron shield panels are attached to the core barrel opposite the core corners, where the flux tends to be higher.

TABLE 13.1. Typical Design Specifications for Pressurized-Water Reactor System

General	
Power	
Thermal	3800 MW
Electrical	1300 MW
Specific power	33 kW(th)/kg U
Power density	102 MW(th)/m^3
Core	
Length	4.17 m (13.7 ft)
Diameter	3.37 m (11.1 ft)
Fuel	
Rod, OD	9.5 mm (0.374 in.)
Clad thickness	0.57 mm (0.0225 in.)
Pellet diameter	8.19 mm (0.3225 in.)
Rod lattice pitch	12.6 mm (0.496 in.)
Assembly width	214 mm (8.43 in.)
Rods per assembly	264 (17×17 array)
Assemblies	193
Fuel loading, UO_2	115×10^3 kg (2.54×10^5 lb)
Ave. feed enrichment	~3.3%
Ave. core enrichment	~2.8%
Burnup 2.85 TJ/kg	(33,000 MW · d/t)

Thermal-Hydraulic	
Coolant	
Pressure	15.5 MPa(a) (2250 psia)
Inlet temp.	293°C (560°F)
Outlet temp.	329°C (624°F)
Flow rate	18.3 Mg/s (1.45×10^8 lb/hr)
Mass velocity	3.67 Mg/s · m^2 (2.7×10^6 lb/hr-ft^2)
Rod surface heat flux	
Ave. 0.584 MW/m^2	(1.85×10^5 Btu/hr-ft^2)
Max. 1.46 MW/m^2	(4.63×10^5 Btu/hr-ft^2)
Linear heat rate, ave.	17.5 kW/m (5.33 kW/ft)
Steam pressure	7.58 MPa (a) (1100 psia)
Control	
Rod cluster elements	24 per assembly
Control assemblies	61 full length, 8 part length

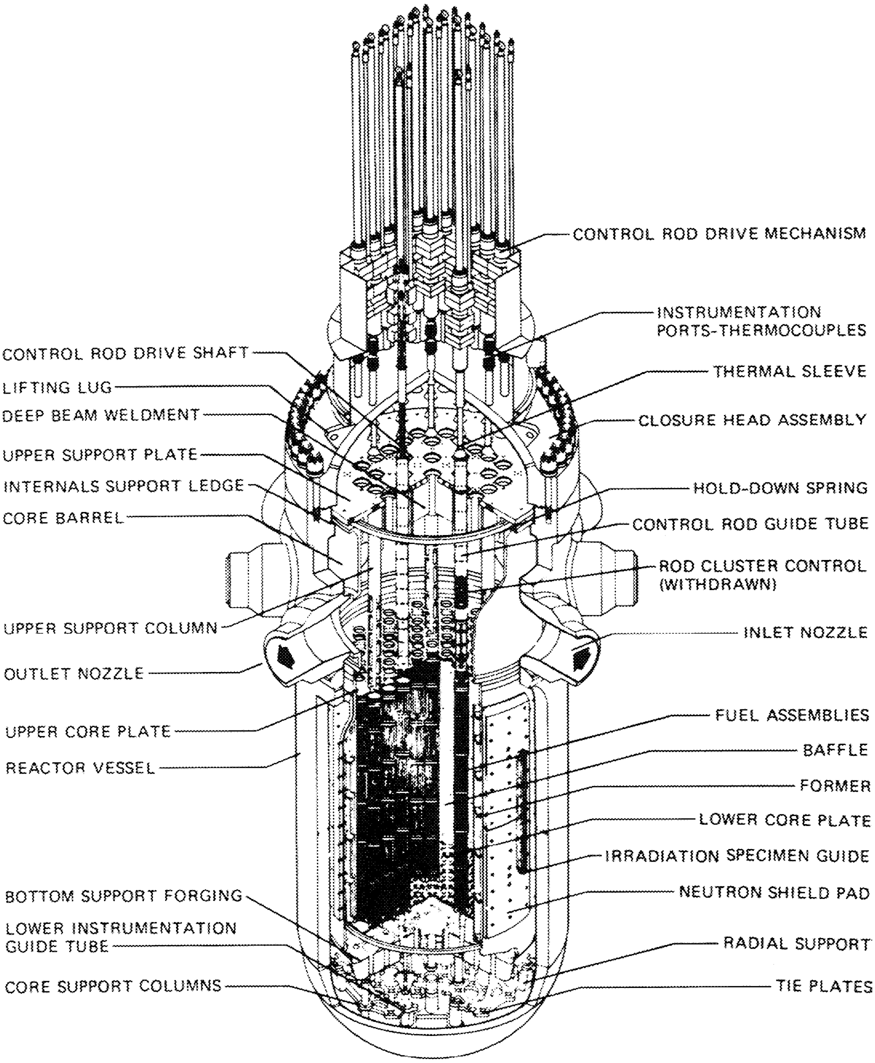

Fig. 13.1. Internal structure of a PWR (Westinghouse Electric Corp.).

13.7. The coolant water leaving the steam generators flows down the annular region (downcomer) between the vessel wall and the lower core barrel and then upward through the core. Flow holes in the lower core plates are sized to permit a higher coolant flow rate through the center of the core where the power generation is greater than at the periphery. After passing through the core, the coolant enters a common upper plenum and exits through the outlet nozzles to the steam generators.

13.8. Thermocouples entering through the vessel head indicate coolant outlet temperatures from fuel assemblies at selected locations. Movable neutron flux detectors, in guide tubes entering through the bottom of the vessel, can be inserted at various points to determine the power distribution in the core.

13.9. The core contains about 200 assemblies of fuel and control rods; in most of the Westinghouse designs, each assembly consists of a 17 × 17 array. Of the 289 spaces available in an assembly, 264 are occupied by fuel rods; the remaining spaces contain guide tubes (thimbles) for control rods with a central tube available for instrumentation. About one third of the assemblies in the core include control rods; in the other assemblies the guide tubes are partially blocked. The fuel rods in an assembly are supported and separated by grid assemblies at intervals along the length (Fig. 13.2). The top and bottom "nozzles" control the flow of coolant water through the fuel assembly. Because the assemblies are open at the sides, lateral flow of coolant is possible from one assembly to another. This arrangement is in contrast to that in a BWR where the fuel assemblies are enclosed by vertical "channel separators" (§13.32).

13.10. The ratio of hydrogen atoms (in the water) to uranium (in the fuel) is an important parameter in PWR core design [3]. This ratio determines the neutron spectrum which, in turn, affects the extent of resonance capture, the Doppler coefficient, and the fraction of fast-neutron fissions. An increase in the H/U ratio results in decreased resonance capture and hence an increase in reactivity. This means that a lower uranium enrichment is required in the fuel. On the other hand, should the H/U ratio be decreased, the harder spectrum would result in an increase in the conversion of uranium-238 to plutonium-239. However, with no plutonium being salvaged through reprocessing, the present trend is to increase the water/fuel ratio. Present designs have atomic H/U ratios in the range 4.0 to 4.3. This corresponds to a H_2O/UO_2 volumetric ratio of about 2. The assemblies listed in Table 13.1 have a water/fuel volumetric ratio of 1.95. One newer design for this 17 × 17 lattice features rods having an outside diameter of 9.144 mm, which would provide a 6 percent increase in the water/fuel volumetric ratio.

13.11. Thermal and hydraulic considerations influence core design. The fuel-rod diameter determines the lattice spacing and thus affects the resistance to coolant flow. The selected rod diameter depends on such parameters as fabrication and cladding costs, desired core power density, and surface heat flux limitations. As the diameter of the fuel rod is decreased, the specific power and power density are increased, assuming a constant volume ratio of water to fuel. The linear heat rate remains constant, but the number of rods per unit core volume is increased as the diameter is decreased. A decrease in fuel-rod diameter, however, also results in an increase in the surface heat flux and thus a closer approach to the DNB

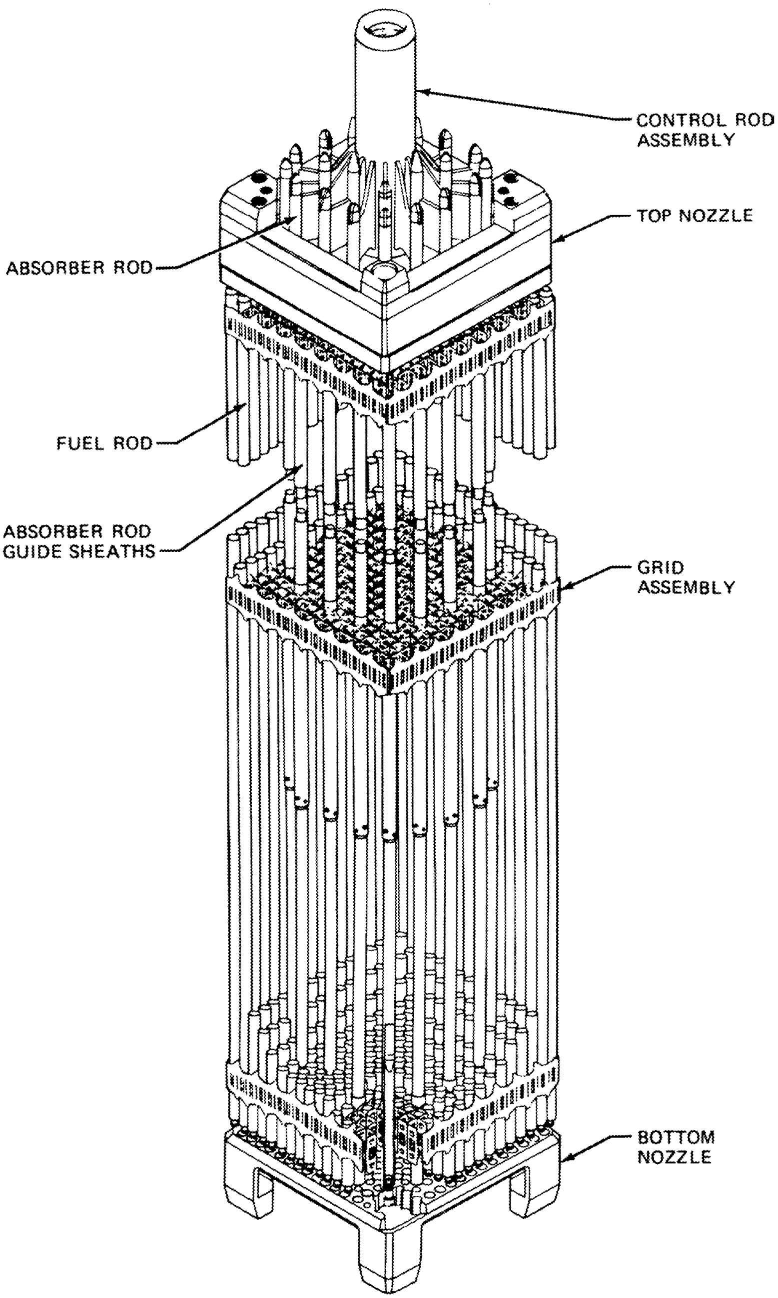

Fig. 13.2. Fuel and control rod assembly of a PWR (Westinghouse Electric Corp.).

condition (Chapter 9). This factor and the increase in fabrication cost of the larger number of fuel rods set a lower limit to the fuel-rod diameter.

Control and Safety Systems

13.12. Operational reactivity control in a PWR is provided primarily by boric acid dissolved in the coolant water supplemented somewhat by control rods. In a procedure known as chemical shim (§5.187), the boric acid concentration is varied to control reactivity changes during the operating cycle, such as those resulting from fuel depletion and fission product buildup. However, selected fuel assemblies contain burnable absorber rods to limit power peaking (§10.31). Therefore, the soluble boron concentration adjustment is made in accordance with the effects on reactivity contributed by these burnable absorber rods as shown in Fig. 13.3. Although

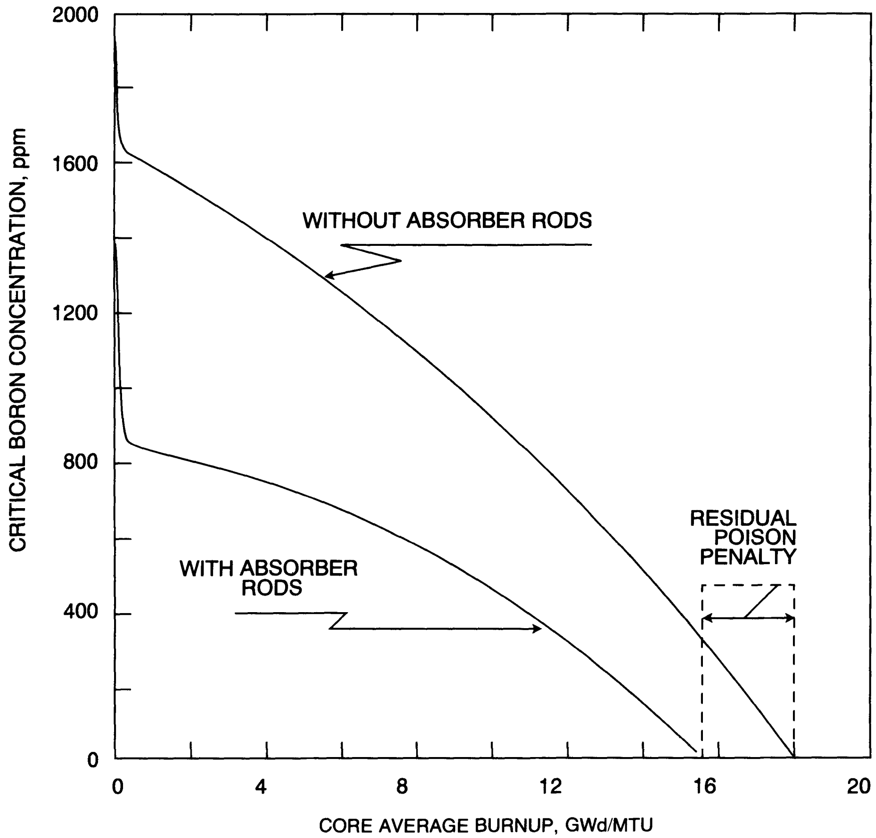

Fig. 13.3. Typical effect of burnable absorber rods (boron) on PWR first core soluble boron concentration.

this figure is for a first core in which solid burnable poison is needed to substitute for the fission products accumulated in subsequent cycles, the general effects on soluble boron concentration are typical of boron burnable absorber rod use [2]. Also shown in Fig. 13.3 is the residual poison reactivity penalty associated with this practice. Gadolinia rods behave somewhat differently. The control rods are used primarily for startup, safety shutdown, and to follow load changes. Some rods are partial length or partial strength to aid power distribution "shaping."

13.13. The control rods are stainless steel tubes encapsulating an absorber material such as hafnium, boron carbide, or a silver-indium-cadmium alloy. These rods are arranged in clusters which move within guide tubes (thimbles) which replace some fuel rods in the rod lattice of the fuel assemblies. Perforations over a portion of the thimble length allow the escape of water as the rods are inserted. However, the lower end of the tubes is closed to decelerate the rods at the end of their drop. The clusters are operated in groups to accomplish their various functions. In assembly positions not reserved for control rod insertion, the empty thimbles may be used for burnable absorber rods. However, other fuel designs provide for more flexibility (§10.32).

Coolant Circulation and Steam Generating Systems

13.14. Most Westinghouse PWR systems have either three or four coolant loops, each with its own circulation pump and steam generator (Fig. 13.4). These pumps are normally of the single-stage, vertical centrifugal type, each driven by a 6-MW (8000-hp) motor. They are rated at about 6.2 m^3/s (98,000 gpm), with a pressure developed of 7.4×10^5 Pa (108 psi). Not shown in the simplified coolant loop figure are connections to numerous auxiliary systems, such as those for safety injection, residual heat removal, and chemical and volume control.

13.15. A typical 1000-MW(el) plant steam generator design, with an overall height of about 20.7 m (68 ft) and an upper shell diameter of 4.4 m (14.5 ft), is shown in Fig. 13.5. Heated primary system coolant from the reactor vessel passes through the inverted U-shaped tubes, and saturated steam at 7.6×10^6 Pa(a), i.e., 1100 psia, is formed in the outer "shell" side. The upper part of the steam generator consists of a steam-drum section where centrifugal moisture separators and a steam dryer remove entrained water from the steam. The steam temperature is 291°C (556°F).

13.16. An important component of the steam-generation system is the system pressurizer which is connected to one of the primary coolant loops (see Fig. 13.5). A typical pressurizer is a cylindrical tank, about 16.2 m (53 ft) high and 2.44 m (8 ft) in diameter. During normal steady-state

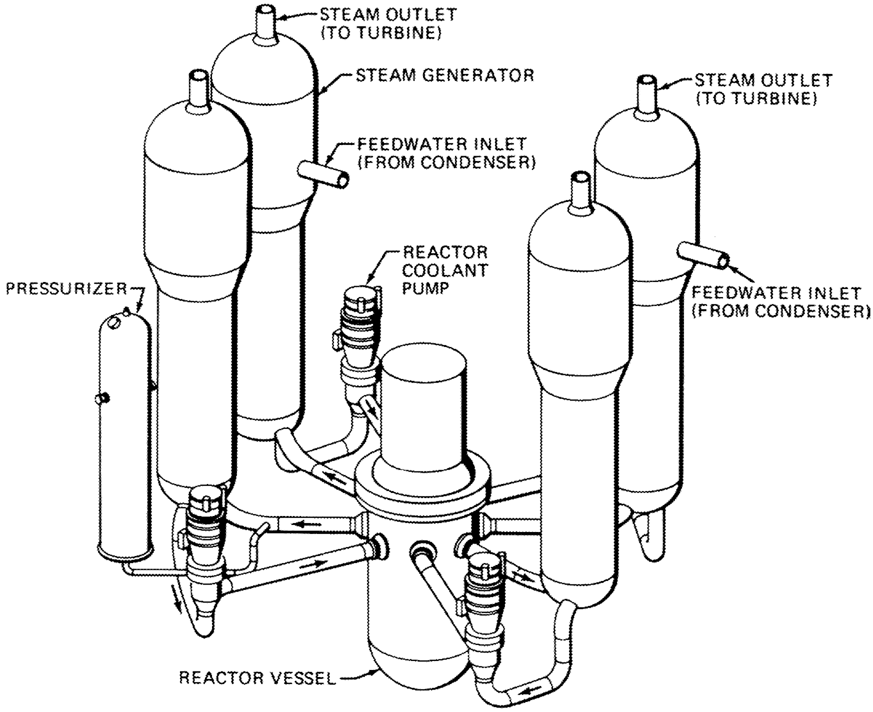

Fig. 13.4. Nuclear steam-supply system of a PWR with four steam generators (Westinghouse Electric Corp.).

operation it contains roughly 60 percent of liquid water and 40 percent steam (by volume). Electric immersion heaters in the lower part of the vessel can provide heating, whereas cold water sprayed into the steam space can provide cooling, as required. Both heaters and cooling sprays are operated by pressure signals. If, for any reason, the system pressure should drop, a low-pressure signal would actuate the heaters. The steam generated would then increase the system pressure. On the other hand, if the pressure should rise, a high-pressure signal would operate the cooling water spray. Some of the steam in the pressurizer would be condensed and the system pressure would decrease. if the pressure should increase beyond the control capability of the sprays, one or more motor-operated relief valves would open automatically and steam would be vented to a quench tank partially filled with water. Should the pressure in the quench tank exceed the design value, a rupture disc would allow the tank to vent to the containment sump. An important contribution to the Three Mile Island accident (§12.179) was the failure of the pressurizer relief valve to close

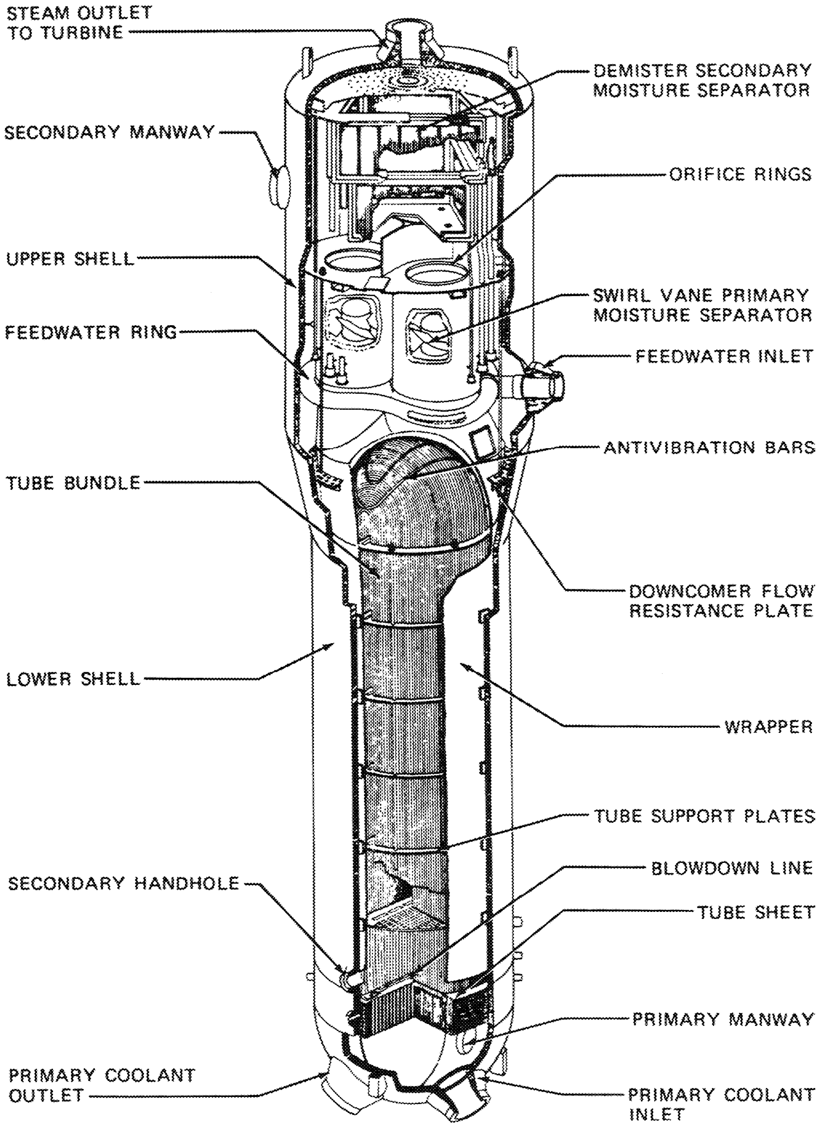

Fig. 13.5. Cutaway representation of a PWR steam generator (Westinghouse Electric Corp.).

when the pressure was restored to normal. Therefore, assurances that valves would operate reliably have received a great deal of attention since that time.

13.17. Steam from the generators passes to the turbine system which consists of high-pressure and low-pressure stages on the same shaft. Partially expanded steam leaving the high-pressure turbine goes through moisture-separator and reheater units before further expansion in the low-pressure turbines. Steam drawn from the high-pressure section and from the steam generator serve as the heat sources for reheating. The exhaust steam from the low-pressure stage passes to the condensers, normally located under the turbines. The condensate provides the feedwater for the steam generators. Before entering the generators, however, the condensate temperature is increased in a series of feedwater heaters which use steam drawn from the turbines to heat the water.

EVOLUTIONARY PRESSURIZED-WATER REACTORS

Introduction

13.18. An evolutionary reactor design is one that draws heavily upon a successful previous design but incorporates many improvements that are desirable as a result of technological development, experience, and possibly new requirements. The basic features are retained so that there would be a high level of confidence that a new plant would operate reliably without the need to first build a demonstration plant.

13.19. An example of this approach for a PWR that is to be available whenever orders for new reactors are forthcoming is the Combustion Engineering System 80 Plus™ reactor model [4]. Several general needs were addressed in developing this design for a so-called "next-generation plant" which provide some background for new reactor trends. A European PWR design meets similar needs.

13.20. The opinion of utilities regarding desirable characteristics of future plant designs was determined by EPRI. Of major importance was the need to satisfy public acceptance with regard to safety, economics, and reliability. Design simplification, standardization, increased safety margins, and various technical improvements were felt to help satisfy public concerns. However, other issues also affect the attractiveness of utility investment in new plants. Predictable licensing and a stable regulation climate were considered necessary to avoid nontechnical and expensive delays that have plagued the industry in the past. Also recognized was the need to have fair treatment by state regulatory agencies on financial and planning matters.

13.21. Design simplification is accomplished by eliminating unnecessary duplication of functions performed by separate systems or in some cases combining component functions. Both simplification and improved planning should lead to shorter construction schedules, which are essential for improved economics. It is also desirable to include in a new plant features that will mitigate the effects of a severe accident. For example, changing the reactor vessel geometry to deal with molten core material can be accomplished in a new design at relatively little cost.

Design Features

13.22. The Combustion Engineering design evolved from that of the three-unit Palo Verde Generation Station in Arizona, which began commercial operation in 1986. A summary of design specifications for the proposed system is given in Table 13.2. The format follows that used in Table 13.1.

13.23. The fuel assemblies are in a 16 × 16 array with rod dimensions similar to those used in the Westinghouse core. However, as seen in Fig. 13.6, there are five large thimbles in each assembly. Four of these are for control rods and one for instrumentation. There are clusters of both four and 12 elements in the core. The 12-element cluster spans five assemblies. Each rod cluster is connected by a "spider" to the drive mechanism. Groups of these clusters are moved as units for power regulation. Burnable absorber rods are substituted for fuel rods in the lattice, as needed, rather than inserted in the control rod thimbles.

13.24. The reactor cooling system arrangement is shown in Fig. 13.7. Four circulating pumps serve two large steam generators. The steam generator secondary-side water inventory has been increased by 25 percent to improve the response to upset conditions and to extend the time available for countermeasures should the feedwater supply be interrupted. Similarly, the pressurizer volume has been increased by 33 percent to reduce the pressure changes during transients such as reactor trip and load rejection.

13.25. The reactor operating margins relative to previous designs were increased by the foregoing improvements as well as by a reduction in the hot-leg temperature and improved monitoring methods. Also, the new design provides for carrying out operating power level maneuvers using control rods only. Thus, the need for short-term boron-level adjustments is reduced.

13.26. Numerous other evolutionary design features contribute to simplification and improved safety margins. A probabilistic risk analysis indicated a reduction in the risk of severe accidents by two orders of magnitude compared with that for present systems.

TABLE 13.2. Design Specification for Evolutionary PWR

General	
Power	
Thermal	3800 MW
Electrical (net)	1280 MW
Specific power	37 kW(th)/kg U
Power density	95.4 MW(th)/m^3
Core	
Length	3.81 m (12.5 ft)
Diameter	3.66 m (12 ft)
Fuel	
Rod, OD	9.7 mm (0.382 in.)
Clad thickness	0.64 mm (0.025 in.)
Pellet diameter	8.3 mm (0.325 in.)
Rod lattice pitch	12.8 mm (0.506 in.)
Rods per assembly	236 (16 × 16 array)
Assembly overall width	202 mm (7.97 in.)
Assemblies	241
Fuel loading, UO_2	116 × 10^3 kg (2.57 × 10^5 lb)
Enrichment levels	3.3, 2.8, 1.9 percent by weight

Thermal-Hydraulics	
Coolant	
Pressure	15.5 MPa(a) (2250 psia)
Inlet temp.	292°C (558°F)
Outlet/temp.	324°C (615°F)
Flow rate	21.4 Mg/s (1.61 × 10^8 lb/hr)
Mass velocity	3.6 Mg/s · m^2 (2.64 × 10^6 lb/hr)
Rod surface heat flux	
Ave.	0.599 MW/m^2 (1.90 × 10^5 Btu/hr-ft^2)
Max.	1.41 MW/m^2 (4.46 × 10^5 Btu/hr-ft^2)
Steam pressure	6.90 MPa(a) (1000 psia)
Control	
Control assemblies	68 full length, 25 part strength

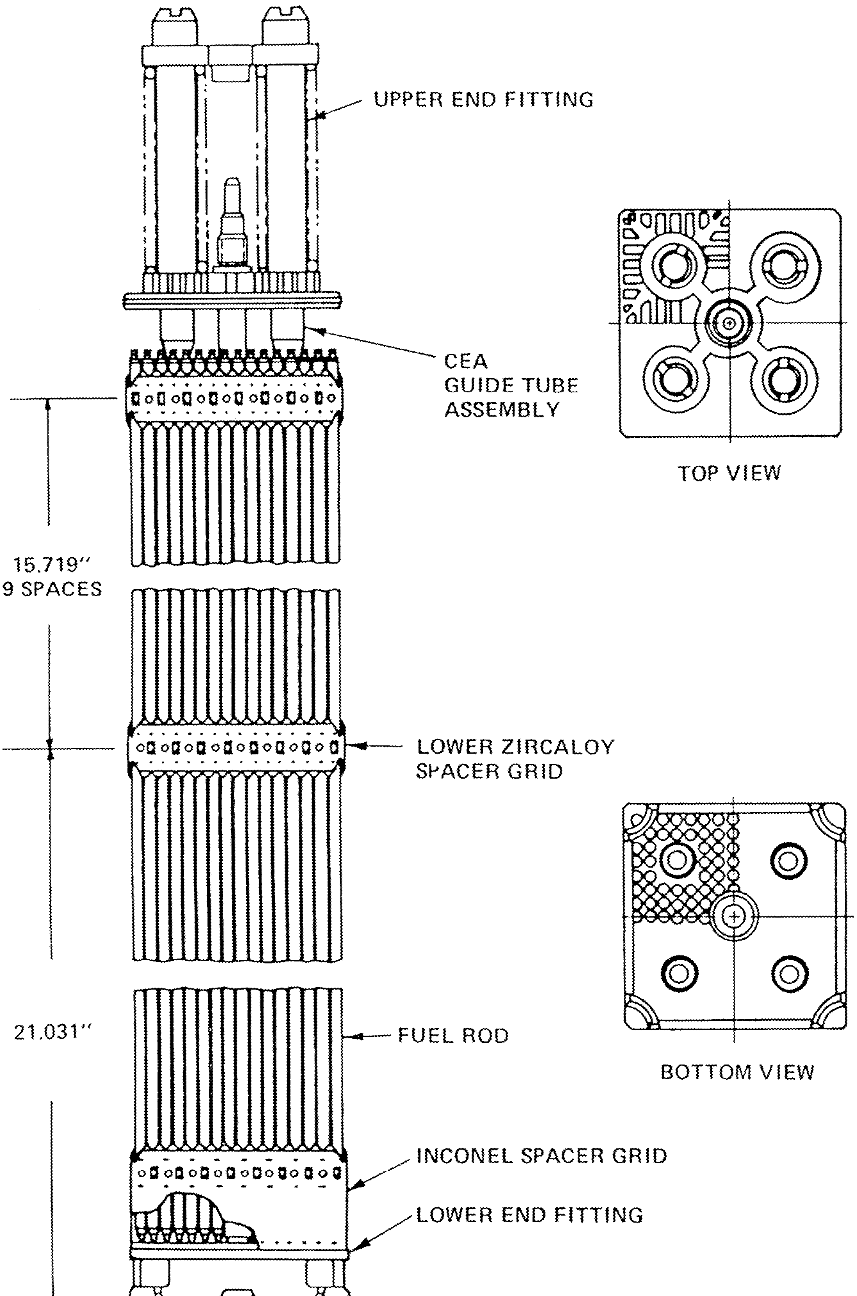

Fig. 13.6. System 80+™ evolutionary PWR fuel assembly (© 1989 Combustion Engineering, Inc.).

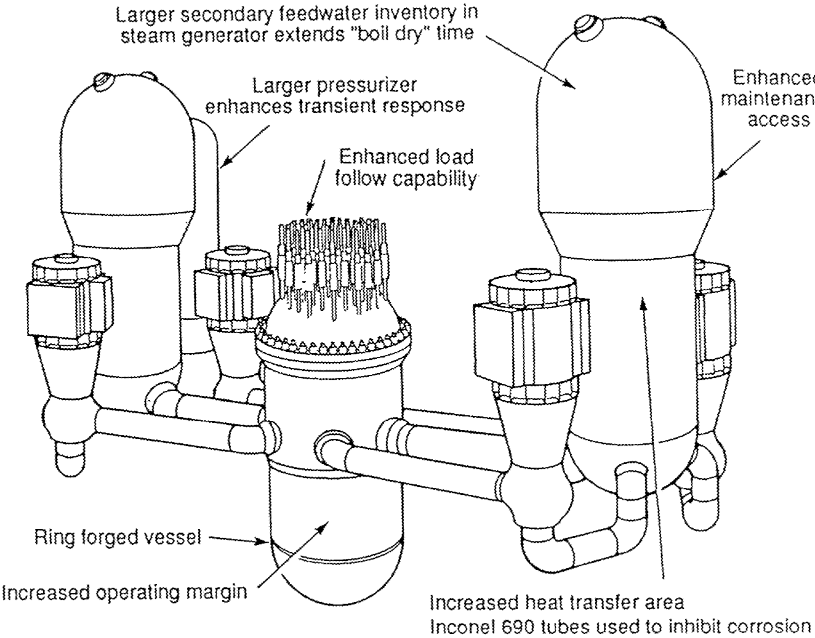

Fig. 13.7. System 80+™ reactor coolant system showing improvements (© 1989 Combustion Engineering, Inc.).

13.27. A 61-m (200-ft)-diameter spherical containment for this system provides both safety and cost advantages. As shown in Fig. 13.8, a concrete shield building encloses an inner steel sphere which has a relatively large internal free volume of 9.5×10^4 m^3 (3.4×10^6 ft^3). The steel shell acts as a heat sink and the large volume provides energy-absorbing capability in the event of an accident. An in-containment refueling water storage tank (IRWST) combines the functions of a refueling water tank and a post-LOCA containment sump. Economic advantages also result from efficient space utilization and ease of construction.*

13.28. To control costs, this and other new reactor proposals feature extensive standardization, not only for the Nuclear Steam Supply System, but the entire plant. This is in accordance with NRC's standardization rules given in 10 CFR 52, which also provides for design certification so that the plant can be pre-licensed prior to purchase. A 48-month construction period is estimated, which should help meet a construction cost goal for new plants established by EPRI of \$1500 per kW(el).

*HVAC = heating, ventilation, and air conditioning.

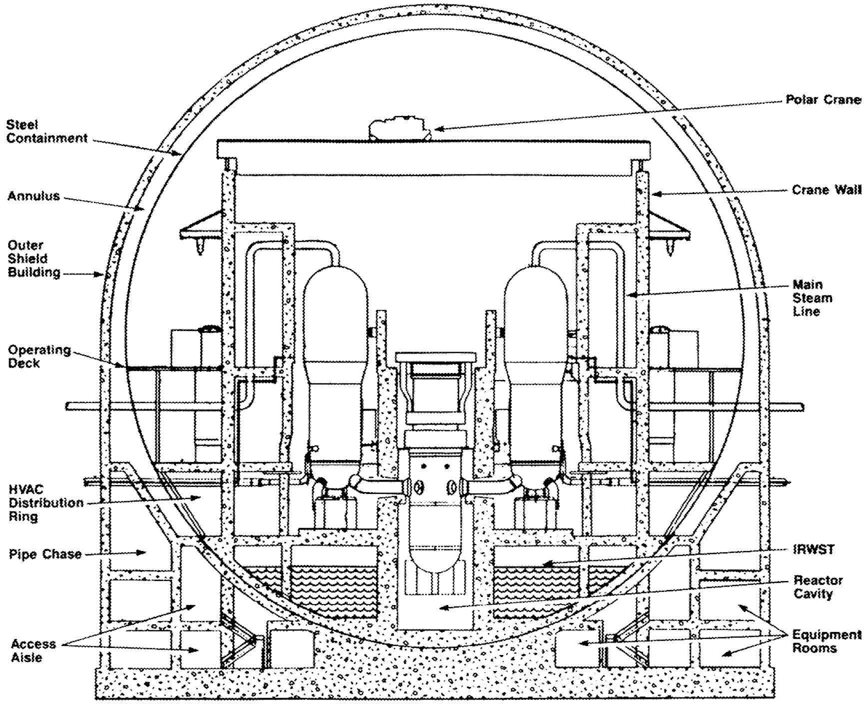

Fig. 13.8. Elevation view of System 80+™ spherical containment (© 1989 Combustion Engineering, Inc.).

PRESENT BOILING-WATER REACTORS

Introduction

13.29. The boiling-water reactor (BWR), as the name implies, generates steam directly in the core. Since this steam then goes directly to the turbine-generator, the system is inherently simpler than a PWR. However, a complication is the need to maintain the fraction of steam in the core less than about 14 percent by weight for stability purposes. Thus, the exiting water, which has not been vaporized, must be separated from the steam at the top of the core and recycled.

13.30. Since the core steam conditions are comparable to those on the secondary side of a PWR steam generator, the BWR core pressure is about one-half that of a PWR. Therefore, a larger core and lower power density are practical as well as larger-diameter fuel rods. In Table 13.3 are listed typical design specifications for a large BWR of the type marketed in the 1970s. The thermal and electrical powers are essentially the same as for

TABLE 13.3. Typical Design Specifications for Boiling-Water Reactor System

General	
Power	
Thermal	3830 MW
Electrical	1330 MW
Specific power	25.9 kW(th)/kg U
Power density	56 MW(th)/m^3
Core	
Length	3.76 m (12.3 ft)
Diameter	~4.8 m (15.8 ft)
Fuel	
Rod, OD	12.52 mm (0.493 in.)
Clad thickness	0.864 mm (0.034 in.)
Pellet diameter	10.57 mm (0.416 in.)
Rod lattice pitch	16.3 mm (0.64 in.)
Rods per assembly	62 (8 × 8 array)
Assembly overall height	4.47 mm (14.7 ft.)
Assembly width	134 mm (5.28 in.)
Coolant flow area/assembly	0.01 m^2 (15.5 $in.^2$)
Fuel loading, UO_2	168 × 10^3 kg (3.69 × 10^5 lb)
Ave. feed enrichment	~2.6%
Ave. core enrichment	~1.9%
Burnup 2.38 TJ/kg	(27,500 MW · d/t)

Thermal-Hydraulic	
System pressure	7.17 MPa(a) (1040 psia)
Steam temp.	289°C (551°F)
Feedwater temp.	216°C (420°F)
Steam flow rate	2.08 Mg/s (1.65 × 10^7 lb/hr)
Exit quality	0.147
Coolant flow rate	14.3 Mg/s (1.13 × 10^8 lb/hr)
Jet pumps	24
Rod surface heat flux	
Ave.	0.51 MW/m^2 (1.6 × 10^5 Btu/hr-ft^2)
Max.	1.12 MW/m^2 (3.54 × 10^5 Btu/hr-ft^2)
Linear heat rate, ave.	20.7 kW/m (6.04 kW/ft)
Control	
Movable cruciform elements	193
Overall length	4.42 m (14.5 ft)
Poison section	3.66 m (12.0 ft)

the PWR described in Table 13.1. Hence, the parameters quoted for the two types of light-water reactors may be compared.

13.31. A variety of BWR fuel designs are offered by various vendors. Worthy of note is a 9 × 9 lattice design containing 272 fueled rods and a central water channel (§10.57). An average linear heat rating of 15.9 kW/m is specified for rods having an outside diameter of 11 mm.

Core and Vessel

13.32. The internal structure of a typical BWR core and reactor vessel is depicted in Fig. 13.9. The core consists of almost 800 fuel assemblies in square 8 × 8 arrays; 62 fuel rods are contained in each assembly with two hollow central ("water") rods designed so that water flows through them to provide additional moderation (§10.52). Each fuel assembly is contained in a four-sided channel separator or "can" to prevent crossflow of coolant between assemblies. A control rod (or element) with a cruciform cross section is located within a set of four assemblies (see Fig. 5.16).

13.33. The fuel assemblies rest in support pieces mounted on top of the control-rod guide tubes. The guide tubes, in turn, rest on the control-rod penetration nozzles in the bottom of the reactor vessel. The core plate (Fig. 13.9) merely provides lateral support. Orifices in the fuel-assembly support permit adjustment of the coolant flow distribution among the assemblies in the core. These orifices can be changed, if necessary, but only by disassembly of the core structure.

13.34. An important BWR consideration is the power stability based on the coupled hydrodynamic-neutronic feedback response resulting from the formation of steam in the core (§5.109). Power oscillations, which would cause fluctuations of the steam voids in the moderator, and hence in the reactivity, should be damped by a proper combination of feedback parameters (§5.132 et seq.); a detailed discussion is beyond the scope of this treatment. Stability tends to be improved by a decrease in power density, increase in coolant flow rate, and increase in rod diameter. Stability considerations govern the thermal power–coolant flow combination during startup and operation as shown in a "power map" (Fig. 14.1) which defines the acceptable operating region. Generally, a BWR of present design can be operated on natural circulation up to about 25 percent of rated power.

13.35. As with the PWR (§13.10) the H/U ratio in a BWR affects the fuel utilization. However, as a result of the low hydrogen density in the vapor fraction of the coolant (~0.4 by volume), the H/U ratio tends to be slightly lower than that for a PWR although the H_2O/UO_2 volume ratio (~2.4) is higher.* The void (vapor) fraction, and the H/U ratio which

*Water in the channels between fuel assemblies is included in the calculation of the ratio.

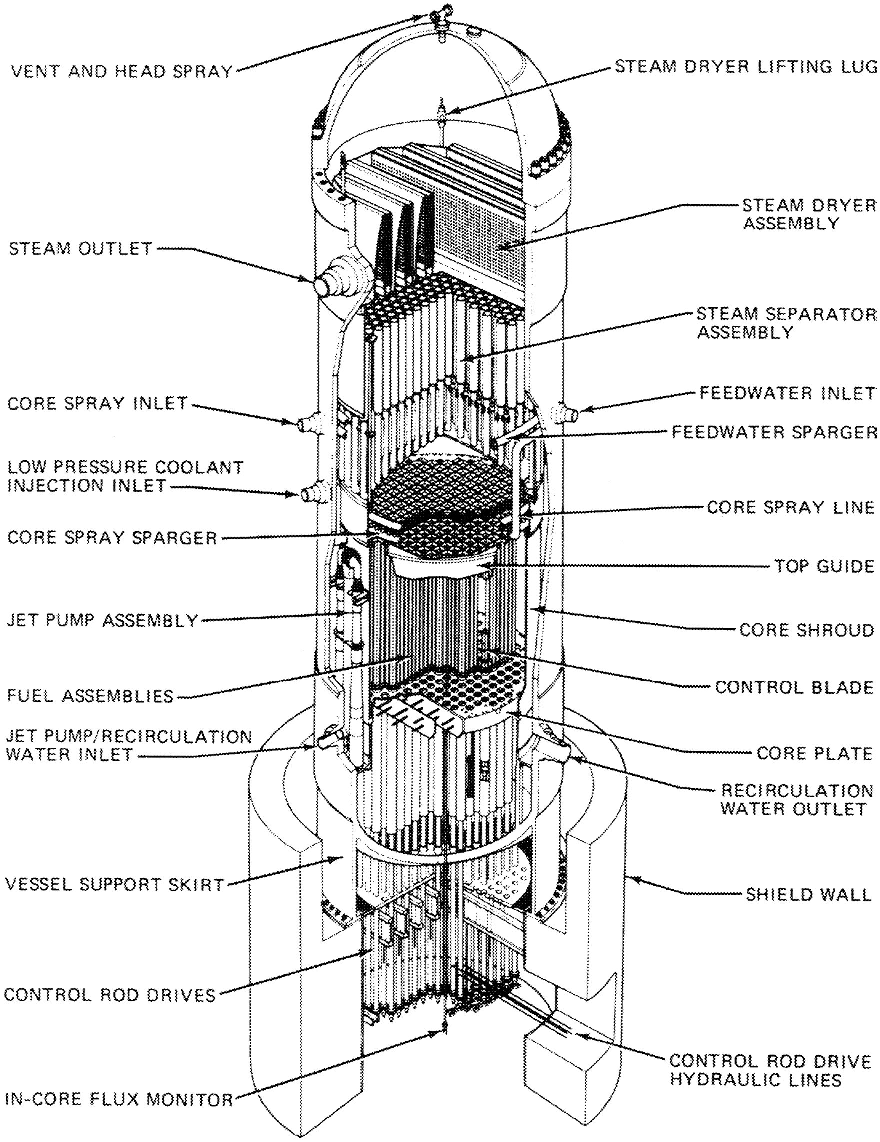

Fig. 13.9. Internal structure of a BWR (General Electric Co.).

depends on it, varies both axially and radially in the core, as well as with burnup.

13.36. A stainless steel cylindrical shroud within the reactor vessel surrounds the core and separates the upward flow of coolant through the core from the downward flow in the annulus between the shroud and the vessel. The jet pumps (see §13.39) within the annulus are supported by the shroud.

13.37. Liquid water entrained in the steam–water mixture leaving the fuel assemblies is separated in the upper part of the reactor vessel. The mixture first enters vertical standpipes in the shroud head and then passes through axial flow centrifugal separators. The swirling motion drives the water droplets to the outer wall from which they flow back to the core via the downcomer annulus outside the shroud (Fig. 13.9). The steam, still containing some moisture, passes on through a dryer assembly of vanes and troughs; here most of the remaining moisture is separated and returned to the downcomer.

13.38. The two-stage system represents a design challenge since the exit steam must contain no more than 0.1 percent water by weight. Steam leaving the first stage contains less than 10 percent water by weight. Since the system must process the entire two-phase mixture from the core, the drying capacity of the system has been one of the constraints on core power.

Coolant Recirculation System

13.39. An important feature of BWR operation is the coolant recirculation system, shown schematically in Fig. 13.10. This system provides the forced convection flow through the core necessary to achieve the required power density in a BWR. About 14 weight percent of the water passing through the core is vaporized and the remainder must be recirculated. A portion of the coolant is withdrawn from the lower end of the shroud region and forced by a centrifugal recirculation pump into the jet pumps. Most BWRs have two recirculation pumps, each of which provides by way of a manifold the "driving flow" for 10 to 12 jet pumps. The flow of coolant from the pumps, up through the core and back by way of the annular region, is indicated in Fig. 13.10.

13.40. In the German KWU-BWR design, the recirculation pumps are within the reactor vessel. The electric drive motors, which are outside the vessel, are connected to the axial flow pumps by mechanical seals flanged to the bottom of the pressure vessel. By eliminating large external coolant circulation loops, the loss-of-coolant accident becomes less probable, and the coolant depressurization rate is reduced. This internal circulation pump arrangement has been adopted by the General Electric Co. for its evolutionary Advanced BWR design (§13.49).

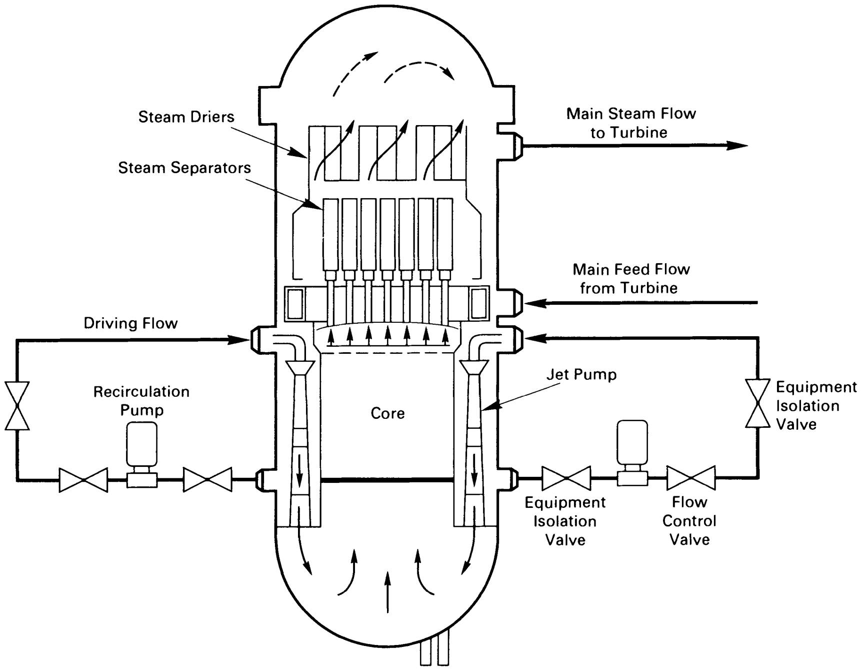

Fig. 13.10. BWR/6 steam and recirculation water flow paths (General Electric Co.).

Control System

13.41. The control rods in a BWR, driven by hydrostatic pressure, enter from the bottom of the core; this is both necessary and desirable. In the first place, the reactor vessel above the core is occupied by the steam-water separators and dryers; furthermore, movement of the controls in and out of the lower part of the core permits compensation for the reactivity decrease in the upper part arising from the steam voids. The absorber rods are used for reactor startup and shutdown and also to flatten the power distribution. Power adjustment during operation is commonly made by changing the coolant recirculation rate, as will be explained shortly. Boric acid solution is not used in BWRs because solid would be deposited on the fuel rods in the boiling region, thereby interfering with heat transfer. However, in newer designs for the advanced BWR, an electrical-hydraulic control rod drive system provides motion in fine steps suitable for power adjustment purposes.

13.42. Early BWR practice was to "scatter load" the core so that fuel assemblies having different burnups would be adjacent to partially inserted control rods. Since the influence of the neighboring absorber would cause uneven burnup, particularly with fuel of higher enrichment, it was common practice to exchange or "swap" groups of control rods used for deep or shallow insertion during the operating cycle. In this way, the effects on adjacent assemblies would be reduced. However, in the newer control cell core (CCC) loading (§10.55), control rod groups intended for partial insertion service have *only* low enrichment assemblies in the adjacent four positions. Therefore, burnup effects are minimal and these rods may be used for reactivity and power distribution control throughout the operating cycle, with no need for rod pattern exchanges.

13.43. The excess reactivity of fresh fuel is partly compensated by the inclusion of gadolinium oxide (Gd_2O_3) as a burnable poison mixed with uranium dioxide in some of the fuel elements (§5.197). These rods are located where they will improve the uniformity of the power distribution.

13.44. The use of the coolant recirculation rate in the adjustment and automatic control of BWR power output is based on the following considerations. Let w_f be the feedwater mass flow rate, w_r the recirculation flow rate, and $w_c(= w_r + w_f)$ the core coolant mass flow rate. The reactor power P can then be represented by

$$\begin{aligned} P &= w_f(h_r - h_f) + w_c X h_{\text{vap}} \\ w_c &= w_r + w_f, \end{aligned} \tag{13.1}$$

where h_r, h_f, and h_{vap} are the enthalpies of the saturated recirculating water, the feed water, and of water vaporization, respectively; X is the quality (weight fraction of vapor) of coolant leaving the core. If the recirculation flow rate w_r is increased without changing w_f, the quality will be momentarily lowered, since there will not have been time for P to change. The reduction in steam voids will tend to increase the reactivity and hence the power; as a result, X will increase toward the original value and the reactivity will decrease. The reactor will thus become stabilized at a higher power level [5].

13.45. The foregoing behavior, which leads to a roughly linear dependence of the BWR power on the recirculation flow rate, is effective over a range of about 25 percent of the reactor design power without movement of the control rods. The coolant recirculation rate is determined by the injection rate into the jet pumps, and this is controlled by flow control valves at the recirculation pumps.

13.46. The critical heat flux (§9.99), at which DNB occurs, depends on the core coolant flow rate and the exit quality. Either a decrease in the flow rate or an increase in the quality (or both) tends to decrease the critical heat flux. Changes in the recirculation rate, w_r, and consequently in the core flow rate w_c (at constant w_f), must therefore be such that the critical heat flux ratio (§9.179) is maintained above the design minimum of 1.90 at 120 percent of rated power. However, this classical figure of merit does not give a true picture of the thermal margin since the axial heat flux profile changes with power. Therefore, the *critical power ratio* (CPR), defined as the critical power divided by the operating bundle power at the condition of interest, is also used (§9.181).

Feedwater Temperature and Fuel Cycle Length

13.47. The length of a fuel cycle in a BWR can be extended beyond that normally available from reactivity limitations by reducing the reactor feedwater temperature and hence its enthalpy. As can be seen from equation (13.1), a decrease in h_f will be accompanied by a decrease in the quality X, assuming other conditions are not altered. The thermal power of the reactor would thus remain unchanged, but the reactivity would be increased. This means that the useful life of a BWR fuel loading would be extended by reducing the feedwater temperature toward the end of the normal operating cycle, a practice known as "coast-down."

13.48. The decrease in feedwater temperature is, however, accompanied by a decrease in steam flow and in the electric power generated, for a given thermal power. For example, a reduction in feedwater temperature from the normal 216°C (420°F) to 121°C (250°F) toward the end of a cycle permits an extension of 6 weeks to the regular fuel cycle length of about 18 months. However, during this period, the steam flow decreases to 82 percent of the normal value and the electrical power output to 91 percent. There is, nevertheless, a net cost benefit as a result of a reduction in the enrichment required for the reload fuel to achieve the same cycle length.

EVOLUTIONARY BOILING-WATER REACTORS

Introduction

13.49. The advanced boiling-water reactor (ABWR) is a large, 1300-MW(el) system developed by an international team of BWR manufacturers benefiting from the extensive safety and performance experience of the many operating systems throughout the world. Two units are under construction in Japan, with commercial operation scheduled for the late 1990s. Some technical data are summarized in Table 13.4. The core power density has been reduced by about 10 percent compared with present BWR values given in Table 13.3. Various fuel assembly design options are available, but for orientation purposes, the fuel dimensions for the 8 × 8 lattice given in Table 13.3 are adequate.

TABLE 13.4. ABWR Design Specification Summary

General		Thermal-Hydraulic	
Power		System pressure	7.17 MPa(a) (1040 psia)
Thermal	3926 MW	Steam temp.	288°C (550°F)
Electrical	1356 MW	Feedwater temp.	216°C (420°F)
Power density	50.5 MW/m^3	Recalculation flow	13.2 Mg/s
Core		**Control**	
Length	3.81 m	Control rod number	205
Diameter, equivalent	5.14 m	Neutron absorber	B_4C
Fuel		Control rod form drive	Cruciform electro-hydraulic fine-motion
Assemblies in core	872		
Lattice type	8 × 8		
Rod, OD	12.3 mm (0.484 in.)		

System Features

13.50. Core and vessel features are shown in the Fig. 13.11 assembly. Internal pumps at the bottom of the core shroud force the recirculating water–feedwater mix up through the core. A two-stage steam separator and dryer arrangement is provided as in present BWRs. Since there are no large coolant nozzles below the top of the core, a loss-of-coolant accident is unlikely. Control rod drives have been redesigned to permit fine motion. Numerous other design innovations have been made that improve safety and reduce construction time.

Standardization and Certification

13.51. Standardization regulations provided in 10 CFR 52 have been followed for the entire plant package, including the nuclear steam supply system, or "nuclear island," the "turbine island," and the radwaste facility. The resulting certification by NRC is a type of prelicensing which should also reduce construction time significantly (§12.240).

HEAVY-WATER MODERATED REACTORS

Introduction

13.52. The use of heavy water as a moderator instead of light water permits natural uranium to be used as the fuel. This freedom from the need for enriched uranium has made the concept attractive in several countries, especially in Canada where heavy-water moderated reactors have been developed and operated successfully in commercial power plants. Although other coolants have been suggested and used to some extent, the present discussion will be devoted to reactors of the CANDU (Canadian-Deuterium-Uranium) type in which heavy water is the coolant as well as the moderator.

13.53. A special feature of the CANDU (pressure-tube) reactors is that the heavy-water coolant and the heavy-water moderator are separated from each other forming two completely independent circuits. This is possible because heavy water is a poor neutron absorber; consequently, the moderator-to-fuel ratio can be higher than is acceptable in light-water reactors.* Hence, the fuel rods in a core moderated by heavy water are less tightly packed. Space is thus available for confining the heavy water coolant under pressure (to prevent boiling) in tubes containing the fuel; the moderator, which does not need to be pressurized, surrounds the pressure tubes.

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*A similar situation arises in graphite-moderated reactors.

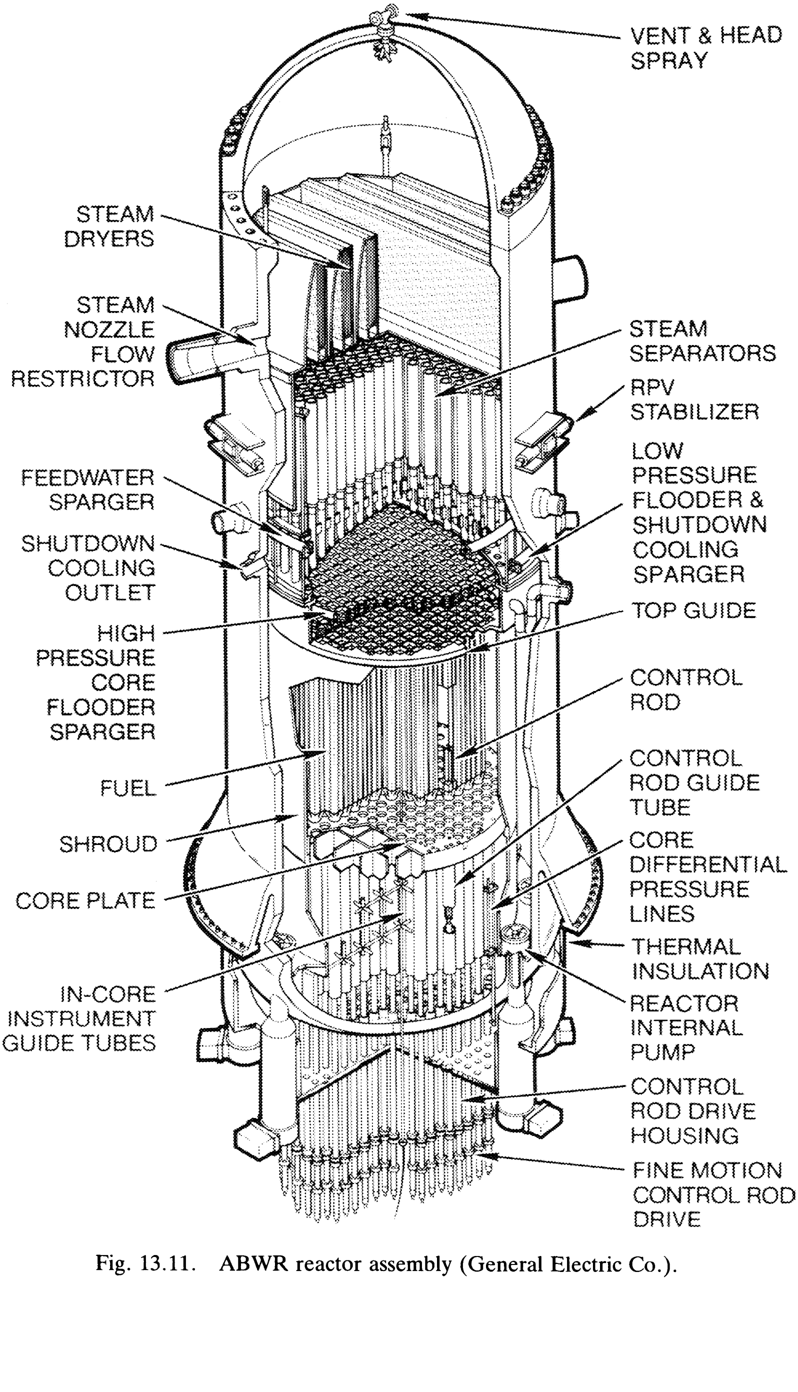


Fig. 13.11. ABWR reactor assembly (General Electric Co.).

Design Specifications and Core Features [6]

13.54. Typical design specifications of an advanced CANDU type reactor are given in Table 13.5, and the reactor core arrangement is shown in Fig. 13.12. The zircaloy pressure tubes containing the fuel rods are horizontal, thereby facilitating fuel handling (charge and discharge) while the reactor is operating at power. Each pressure tube contains 12 fuel bundles; in the older CANDU reactors there are 28 rods per bundle, but the number has been increased to 37 in the newer designs. The fuel rods, which are separated laterally by spacers, consist of natural uranium dioxide pellets clad with zircaloy. A number of vertical "booster" fuel rods containing slightly enriched uranium dioxide are not normally in the core; they are used only to override xenon poisoning if the reactor is restarted soon after an unexpected shutdown.

13.55. On-line refueling, i.e., without reactor shutdown, is conducted by means of two machines, one at each end of the core, operating in tandem. One machine inserts a fuel bundle in a pressure tube while the other machine removes the spent bundle displaced at the other end. Axial power flattening is achieved by limiting the number of bundles that are replaced at any time and by using bidirectional loading in adjacent channels. Refueling strategy is determined on the basis of calculated channel fuel burnup and reactivity demands.

Heat Removal

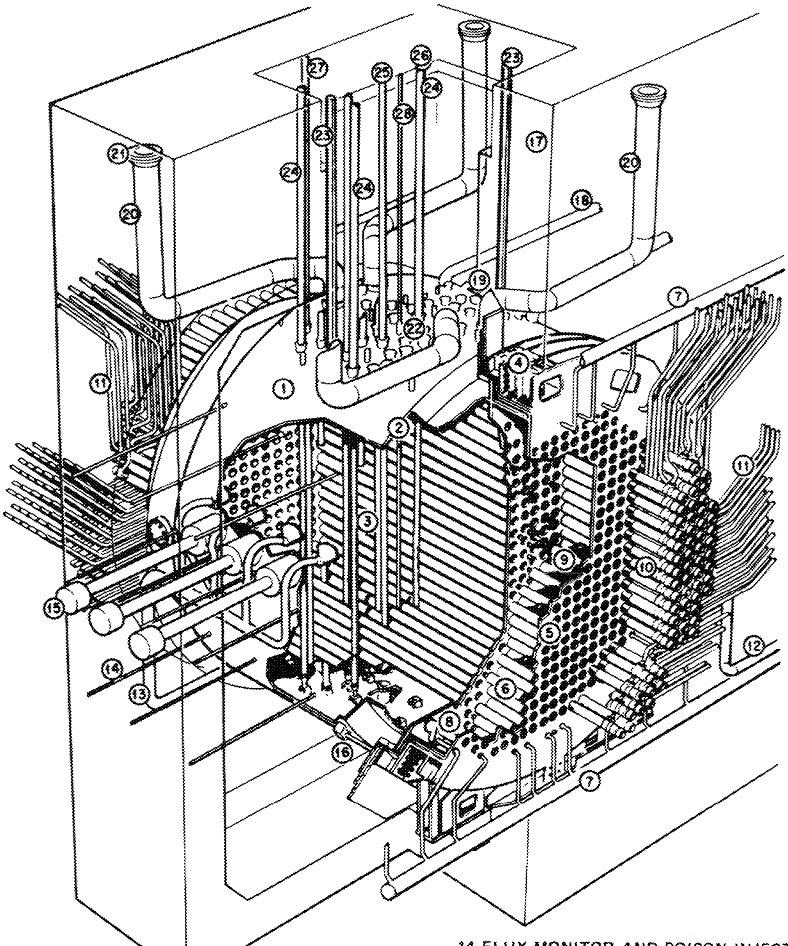
13.56. The pressure tubes are supported by tube sheets at the two ends of the cylindrical reactor vessel, called a calandria, with its axis horizontal. The calandria contains the heavy water at essentially atmospheric pressure. Some heat (about 5 percent of the reactor total) is generated in the moderator by the slowing down of fast neutrons. The moderator is consequently circulated through a heat exchanger and also through a purifier.

13.57. A fairly complex piping system serves to connect the inlet and outlet ends of each pressure tube to coolant manifolds (see Fig. 13.12). Pressurized heavy-water coolant flows between and around the fuel rods in each bundle; the flow is in opposite directions in adjacent channels. The heated coolant leaving the reactor is collected in a header (or manifold) from which it passes to a steam generator of the conventional type. The coolant is then pumped back to the reactor. A simplified representation of the CANDU cooling system is shown in Fig. 13.13. Two coolant exit headers are provided, one at each end of the calandria, and each header is attached to a steam-generator-pump loop. A single pressurizer, of the same type as is used in PWRs (§13.16), maintains the coolant system pressure.

TABLE 13.5. Typical Design Specifications for Heavy-Water Pressure-Tube Reactor

General	
Power	
Thermal	2064 MW
Electrical (net)	600 MW
Specific power	24 kW(th)/kg U
Power density	7.46 MW(th)/m^3
Core	
Length	5.94 m (19.5 ft)
Diameter	7.7 m (25.2 ft)
Fuel	
Rod, OD	13.1 mm (0.515 in.)
Clad thickness	0.38 mm (0.015 in.)
Spacing between rods	1.02 mm (0.0047 in.)
Rods per bundle	37
Bundle length	0.495 m (19.5 in.)
Bundle diameter	0.102 m (4.0 in.)
UO_2 per bundle	21.3 kg (46.5 lb)
Fuel loading, UO_2	97×10^3 kg (214×10^3 lb)
Ave. burnup 650 GJ/kg	(7,500 MW · d/t)

Thermal-Hydraulic	
Coolant pressure	
Inlet	11.3 MPa(a) (1640 psia)
Outlet	10.0 MPa(a) (1450 psia)
Temp.	
Inlet	266°C (512°F)
Outlet	310°C (590°F)
Flow rate	7.7 Mg/s (6.1×10^7 lb/hr)
Rod surface heat flux	
Ave.	0.73 MW/m^2 (2.3×10^5 Btu/hr-ft^2)
Max.	1.3 MW/m^2 (4.1×10^5 Btu/hr-ft^2)
Linear heat rate, ave.	30.5 kW/m (10 kW/ft)
Heat generated in moderator	116 MW
Moderator outlet temp.	71°C (160°F)
Control	
Primary shutdown	28 cadmium rods
Secondary shutdown	$Gd(NO_3)_3$ injection
Adjuster rods	21 stainless steel absorbers
Control absorbers	4 cadmium rods
Zone control	14 H_2O chambers
Heavy Water Inventory	
Moderator	264,000 kg
Heat transport	179,000
Fuel handling	5,000
D_2O management	15,000
Total	463,000 kg



1 CALANDRIA
2 CALANDRIA SHELL
3 CALANDRIA TUBES
4 EMBEDMENT RING
5 FUELING TUBESHEET
6 END SHIELD LATTICE TUBES
7 END SHIELD COOLING PIPES
8 INLET-OUTLET STRAINER
9 STEEL BALL SHIELDING
10 END FITTINGS
11 FEEDER PIPES
12 MODERATOR OUTLET
13 MODERATOR INLET
14 FLUX MONITOR AND POISON INJECTION
15 ION CHAMBER
16 EARTHQUAKE RESTRAINT
17 VAULT WALL
18 VAULT COOLING PIPES
19 MODERATOR OVERFLOW
20 PRESSURE RELIEF PIPES
21 PRESSURE RELIEF DISC
22 REACTIVITY CONTROL ROD NOZZLES
23 VIEWING PORT
24 SHUTOFF ROD
25 ADJUSTER ROD
26 CONTROL ABSORBER ROD
27 ZONE CONTROL ROD
28 VERTICAL FLUX MONITOR

Fig. 13.12. Internal structure of a CANDU reactor (Atomic Energy of Canada, Ltd.).

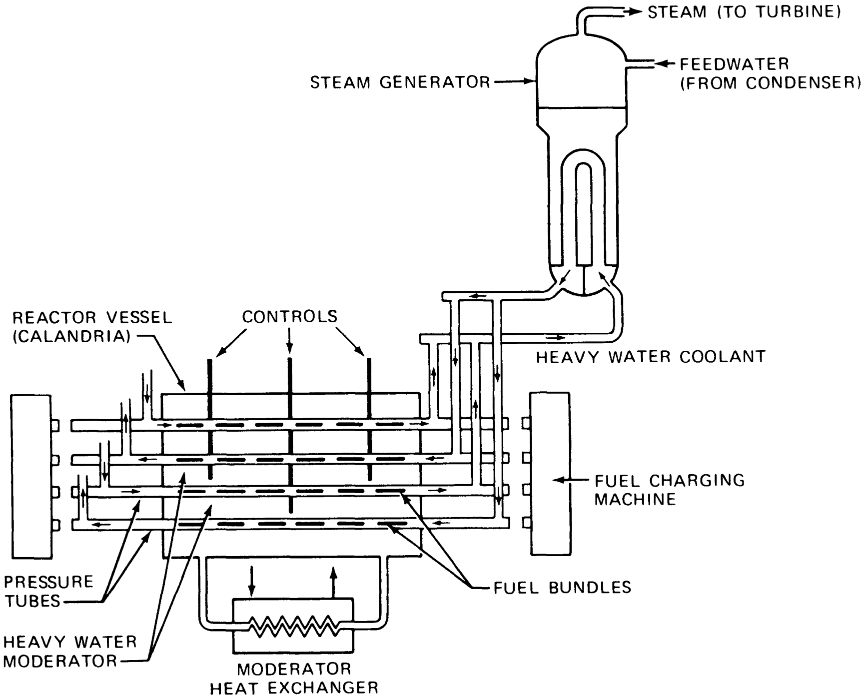


Fig. 13.13. Schematic representation of CANDU coolant and steam-generator systems.

Control System

13.58. Since there is only slight neutron absorption in the coolant and the moderator is kept cool, there is little decrease in reactivity from shutdown to power in contrast to that for LWRs. Continuous refueling also avoids reactivity burnup changes. The core reactivity inventory requiring compensation by the control system is therefore small. The coolant temperature and void coefficients are slightly positive; nevertheless, the power coefficient is negative, although small, as a result of the overriding effect of the negative Doppler fuel coefficient. Thus, the system is inherently stable.

13.59. Several types of control elements are provided in the CANDU reactor. Primary shutdown capability is provided by 28 vertical absorber (shutdown) rods with secondary shutdown, if necessary, by high-pressure injection of gadolinium nitrate solution into the moderator. Since the core is large and the negative power coefficient is small, xenon oscillations, which are slow, require control (§5.76 et seq.). The neutron flux level is therefore measured continuously at many points (~100) in the core and the flux level adjusted as necessary. Flux flattening is provided by 21 absorber rods called "adjusters." In addition, there are 14 "zone control" absorbers for bulk reactivity control and local suppression of flux oscillations. These absorbers consist of vertical chambers which can be filled with ordinary water to any desired level. Four vertical (solid) absorber rods supplement the zone control absorbers.

Safety Features [7]

13.60. Containment of the fuel bundles within individual pressure tubes localizes the consequences of possible fuel cladding failures and contributes to overall safety. Further, the combination of natural uranium fuel and on-line refueling results in a fuel burnup only about one fourth that of an LWR. Hence, the fission-product inventory is much lower.

13.61. In addition to the normal shutdown system, a backup system is provided for emergency shutdown. In earlier CANDU designs, the moderator could be dumped in an emergency. In the larger cores of recent design, injection of gadolinium nitrate solution into the moderator is preferred.

13.61. An emergency core-cooling system is available in the event of a loss-of-coolant accident. However, the separate moderator system provides a substantial heat sink should the coolant circuit be breached. Since there are many coolant inlet and outlet lines of relatively small diameter, compared with those in a LWR, the probability of a rapid depressurization accident is greatly reduced. A containment structure, consisting of a prestressed-concrete enclosure containing the nuclear steam supply system, combined with a water dousing arrangement, provides a pressure suppression feature should there be a break in a coolant manifold.

The Evolutionary CANDU 3

13.63. An evolutionary CANDU-type reactor system is being developed to be competitive in the international market with other medium-sized advanced concepts. This system, designated CANDU 3, is a 450-MW(el) plant that features a standardized design and numerous design features intended to achieve a short construction schedule. Prefabricated modules are extensively employed. The basic design is similar to that of the CANDU described previously, and thus takes advantage of proven technology. A new control room includes extensive human factors features similar to those incorporated into evolutionary LWRs in the United States. Design certification by the US NRC is being pursued [8].

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CHAPTER 14

Plant Operations

INTRODUCTION

14.1. As the nuclear power industry has matured, engineering emphasis has shifted from new construction-related design to operational considerations. Although many of the topics covered in previous chapters are relevant to plant operations, we are devoting a separate chapter to subjects that are particularly operations related.

14.2. Operational strategy to meet system load, refueling, and maintenance requirements is an important area of utility engineering management. Control room operations and associated staff activities are another vital area. Although the subject of reactor control was introduced in Chapter 5, we can now provide additional depth, building upon the material of reactor safety and regulation presented in Chapter 12. Practically all aspects of operation is subject to strict licensing requirements, another topic worthy of at least introductory attention.

14.3. As plants approach the end of their original design lifetime, there are some special operational considerations primarily concerned with materials behavior. Also, with no new reactors being built, there is a substantial economic incentive to extend the operating lifetime of existing reactors. Both of these related subjects will be discussed in this chapter. Finally, the engineering aspects of decommissioning presently operating plants will be treated.

PLANT OPERATIONAL STRATEGY

Generation Dispatching

14.4. A commercial nuclear power plant station usually consists of one or more generating units connected to the utility's electrical distribution grid, which is also supplied by other units at various locations. Interconnections with grids of neighboring utilities are also provided to meet emergency requirements as well as to provide a means to purchase or sell energy. In dispatching load to generating units, the utility's objective is to provide electricity to its customers by operating individual units in such a manner that generating costs will be minimal.

14.5. As pointed out in §10.99 and §10.106, the capital costs of the entire utility system must be paid on a unit time basis, whether or not an individual generating unit is operating. However, fuel is expended only when a unit is operating and the rate of expenditure for a given power level depends on the efficiency of the generating unit. Therefore, neglecting transmission considerations, it is cost-effective to dispatch the grid electrical load so that the most efficient plants operate as much as possible. The load changes during the course of a day, being lightest during the late night and early morning hours. Also, there are seasonal changes with peak loads occurring during hot weather when many air-conditioning units are operating. Thus, the most fuel-efficient units will be operated as much as possible to meet the constant *base load*, while less efficient units will be assigned to operate as needed to meet the variable *peak load*. As a result of balancing capital and fuel costs, it is common practice to use low-capital-cost but fuel-inefficient gas turbine generators to meet the peak loads of shortest duration, while high-capital-cost but fuel-efficient nuclear units are base-loaded.

Operating Cycle Length and Outage Management

14.6. The operating cycle length is the planned operating time before shutdown for refueling is required. It is an important fuel reload design parameter and sometimes is expressed in units of fuel burnup. For many years, operating cycles of 12 months were common. However, in recent years, longer cycles became economical as the cost of uranium and enrichment remained relatively low and the advantages of a higher capacity factor became more significant. Also, longer cycles offer more flexibility for generation management.

14.7. Generation management is important for a utility having several nuclear units so that overlapping refueling and maintenance shutdown periods will be avoided. Such outage periods are normally 1 to 2 months long. Also, less frequent refueling has the advantage of reduced radiation exposure to operating personnel. Therefore, many utilities have adopted operating cycles of 18 months and in some cases as long as 24 months [1]. The increase in permissible core peaking factors as a result of changes in 10 CFR 50 Appendix K tends to simplify reload core design for extended cycles.

14.8. Outage management requires long-range strategic planning to maximize plant availability by making the most efficient use of the time available for plant maintenance. For example, project priorities, many to meet NRC requirements, are generally established years in advance, with individual projects scheduled over a series of outages, as appropriate [2].

PLANT CONTROL

Normal Operational Maneuvers

14.9. In Chapter 5 we introduced reactor kinetics and some features of the control system. However, in a nuclear power plant, we must deal with the control of the entire plant, not just the nuclear reactor component. All aspects of plant control emphasize safety, with all operations conducted according to written procedures approved by the NRC. Normal operations are those routinely performed, such as starting up and shutting down the plant, as well as changing power levels. In contrast, an abnormal condition is one in which an unplanned transient requires corrective action to prevent serious problems. Some of these were considered in Chapter 12. We will consider here only normal operations as a way of introducing some of the features of plant control. However, we will not deal with detailed plant procedures necessary to carry out the operations. Also, a description of corrective actions for a wide variety of abnormal conditions is beyond our scope.

PWR plant start-up

14.10. Routine startups for LWRs may either be from a cold condition or from a so-called "hot standby" condition. When the reactor is started for the first time or following a shutdown that required the system to be cooled down and depressurized, we have a cold startup. *Hot-standby startup* refers to a return to power operation when the cooling system is still near operating temperature and pressure such as the condition following a reactor trip resulting from a minor transient, instrument malfunction, or turbine trip.

14.11. Prior to initial "commercial" startup, extensive testing is prescribed by the NRC to meet licensing requirements. Such tests include not only functional tests of all components and instruments, but also physics tests at zero power and testing at various power levels.

14.12. Precriticality checks are carried out before any cold startup to assure that the plant is ready to function. In a PWR, the coolant system boron concentration is adjusted and various auxiliary systems prepared. Frictional heat from the circulating pumps is then used to raise the primary coolant temperature to about 200°C at a pressure of about 2.8 MPa over a period of about 6 hours. After the pressurizer is adjusted, systematic withdrawal of the control rods to approach criticality is initiated.

14.13. In connection with control rod withdrawal, the importance of neutron detection instrumentation must be emphasized. As pointed out in §5.216, appropriate different neutron detectors are used in the source range, intermediate range, and power range. These are monitored by control room instrumentation, which has been significantly modernized during recent years (§14.25).

14.14. The control rods are withdrawn intermittently rather than continuously, with the neutron level allowed to stabilize before the rods are withdrawn further. A neutron-level startup rate well below criticality of about 0.5 decade per minute, corresponding to about a 50-s period, is common. However, when criticality is approached, the neutron count rate no longer reaches equilibrium between control rod withdrawals as a result of the contribution of the core neutron source.

14.15. Reactor physics data are taken at criticality and the power is raised to about the 1 percent level, where various adjustments are made to the secondary system. Reactor power is increased by further manual control rod withdrawal and the pressure and temperature raised to operating conditions while the power level is still relatively low. The reactor is now at essentially hot-standby conditions at zero electrical load. The turbine-generator may be brought up to speed and connected to the grid when the power level is about 15 percent. At this point, additional power increase is normally switched from manual to automatic control. The loading rate to full power is normally limited to 5 percent per minute, determined by turbine considerations. To compensate for the reactivity power defect, some dilution of the coolant boron concentration is needed before reaching 100 percent power. A cold startup requires a minimum of about 13 hours.

14.16. Startup from the hot-standby condition follows a similar procedure except rod withdrawal is initiated when the cooling system is close to operating conditions. However, the relative negative reactivity contributions of the dissolved boron and control rods must first be balanced properly so that criticality will occur with the rods above a certain limit. In the event of a trip, there will then be adequate rod negative reactivity available. Hot-standby startup time can be as short as 1 hour.

BWR startup

14.17. The general startup approach for a BWR is similar to that for a PWR except for the sequence of operations needed to bring the plant on-line. Also, for a BWR, nuclear heat is used in a cold startup to bring the coolant system to temperature and pressure. Therefore, the reactor is brought to critical by control rod removal when cold, but after thorough prestartup checks are made and recirculation at a low rate is initiated.

14.18. An extensive startup procedure calls for bringing the various systems on-line at appropriate stages. The heat-up rate is usually limited to 50°C per hour to avoid thermal stresses in the reactor walls. In the approach to full power following heat-up, there may be a rate restriction to avoid a too rapid increase in the fuel temperature, which could lead to pellet–clad interaction (§7.172) problems.

14.19. During the approach to power, thermal-hydraulic stability must be considered. Many stability studies have been made over the years to predict acceptable conditions which have led to plots using dimensionless parameters. Such variables as operating pressure, heat input, flow rate, and coolant conditions are relevant. More recently, various computer codes such as TRACG [3] have been used for stability analysis. A different type of plot, known as a *power map*, is shown in Fig. 14.1. This shows typical BWR operating conditions with the natural circulation and minimum recirculating pump speed ranges indicated. A particularly important feature of such a power map is to show the role of the reactor protection system at varying core power levels and recirculating flow rates. Withdrawal of the control elements is "blocked" at the combination of power and flow indicated by the so-called *block line*. Conditions leading to reactor trip are also shown. Since the power is measured by several (normally four) average power range monitors, each receiving signals from many detectors throughout the core, the term *APRM rod-block* is commonly used.

Load changes and load following

14.20. During normal operation, most nuclear plants run continuously at full rated power (base loaded). Should there be unexpected sudden transients or disturbances affecting the load, various procedures are provided to accommodate them. Our purpose here is not to consider such

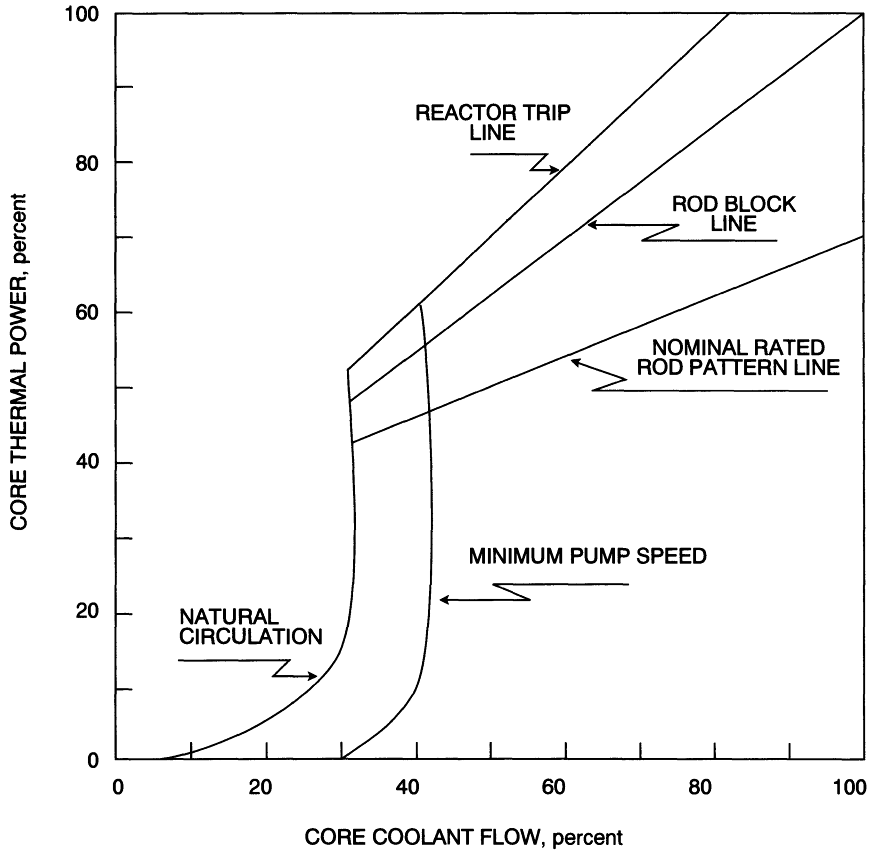

Fig. 14.1. Representative BWR power map showing operating conditions for control element blockage.

cases but to examine the inherent ability of a system to "follow" changes in load should the particular generating unit not be in base-loaded service.

14.21. An example of the natural load-following capability of a PWR is initiated by an opening of the steam throttle leading to the turbine in response to a sensor detecting a slight slowing of the generator as the load is increased. As the pressure is reduced on the secondary side of the steam generator, there is some flashing of water leading to additional heat removal from the primary system. As the cooler water returns to the core, the increase in reactivity resulting from the negative temperature coefficient causes an increase in reactor power. For a load reduction, the process is reversed. In practice, some "fine tuning" by the control system is usually carried out involving control rod movement or boron concentration adjustment. However, the picture is not quite so simple. The partial insertion of rods during part-load operation can lead to an undesirable upper core axial power shift on return to full power since the upper region xenon concentration had previously been reduced. The excess upper half power is known as *axial offset*. However, axial problems can be avoided and load following accomplished without boron concentration adjustment by balancing the operation of two banks of control rods, with each bank containing some weak absorbers or "gray" rods. This is known as MSHIM (mechanical shim) strategy.

14.22. By contrast, in a BWR, an opening of the steam throttle reduces the core pressure, causing more boiling, which, in turn, *reduces* the reactivity as a result of the void coefficient. Therefore, BWR load following, generally in the range of 75 to 100 percent power, is accomplished by the control system, which can reverse this tendency by increasing the recirculating flow rate and moving the rods an appropriate amount. Although the recirculating pumps operate at constant speed, valving is provided for flow rate adjustment.

Shutdown

14.23. PWR shutdown to either a hot-standby or cold condition is initiated by unloading the turbine-generator and corresponding plant systems automatically. For a hot-standby subcritical condition, the control rods are inserted, cooling loop flow is maintained with two of the four pumps, and the residual heat removal system is placed in service. To place the plant in cold shutdown condition, numerous adjustments to plant systems are needed, including blocking the action of the safety injection systems and increasing the dissolved boron concentration.

14.24. A BWR *operational shutdown* is one of short duration in which the reactor remains critical at about 0.01 percent of rated power and the turbine-generator is removed from the grid. Pressure is maintained and the steam produced is bypassed to the condenser, as needed. In an *isolated shutdown*, the reactor is not critical but the system is maintained at pressure with the residual heat removal system activated. For cold shutdown, the pressure is slowly reduced, with the cooling rate limited to about 50°C per hour and the residual heat removal system utilized.

The Control Room

14.25. The Three Mile Island accident focused attention on the shortcomings of most nuclear power plant control rooms existing at that time since operator response during the accident was complicated by confusing displays, many alarms, and an "overload" of information. Diagnosis of the problems during the stress of the accident was difficult. As a result, practically all control rooms were upgraded to meet new NRC requirements. Human factors engineering, which is concerned with the interface between people and machines, was recognized as an essential contribution to control room design.

14.26. Control room designs for new nuclear power plants have been developed with important features that we will examine. Much attention has been given to providing the operators with an easily understood structured information display arranged in order of importance, taking advantage of extensive computer modeling of plant systems. Expert system and neural network features are used in some systems to provide guidance for remedial actions [3].

14.27. The primary control room objective is to help the operator make the decisions that are necessary, particularly under unexpected circumstances. To accomplish this, designers have made the most important plant safety and power production information readily apparent on fixed indicators, while reducing the number of distracting displays. Less vital information is made available by a data processing system on color graphic and other video displays. A third level of information is provided on cathode ray tube (CRT) monitors by calling up suitable "pages." Similarly, alarms are categorized by priorities so that operators are alerted to problems but not overwhelmed with many signals. A "safety panel" informs the operator of the status of the plant during an accident with data processed so that guidance to corrective action is provided. Feedback indicates the effect of such actions.

14.28. Another new feature is a large screen, visible throughout the control room, which provides a graphical color display of the status of the plant and critical functions. This display overview links the vital fixed indicator information with that provided by CRT "pages." Past major dependency on such CRT access to a great deal of information diminished the operator's "feel" of the plant condition. Thus, the large display aids coordination and diagnostics. The development of an optimum balance between the various informational and operating tools has required much experimentation using computer-driven plant simulators.

14.29. A typical advanced design control room layout is shown in Fig. 14.2. The operation of major plant functions, such as the core, primary coolant system, and turbine generator, is centered at the master control console (MCC). Other panels serve the safety and auxiliary systems. The MCC is engineered using anthropometric standards for a seated operator

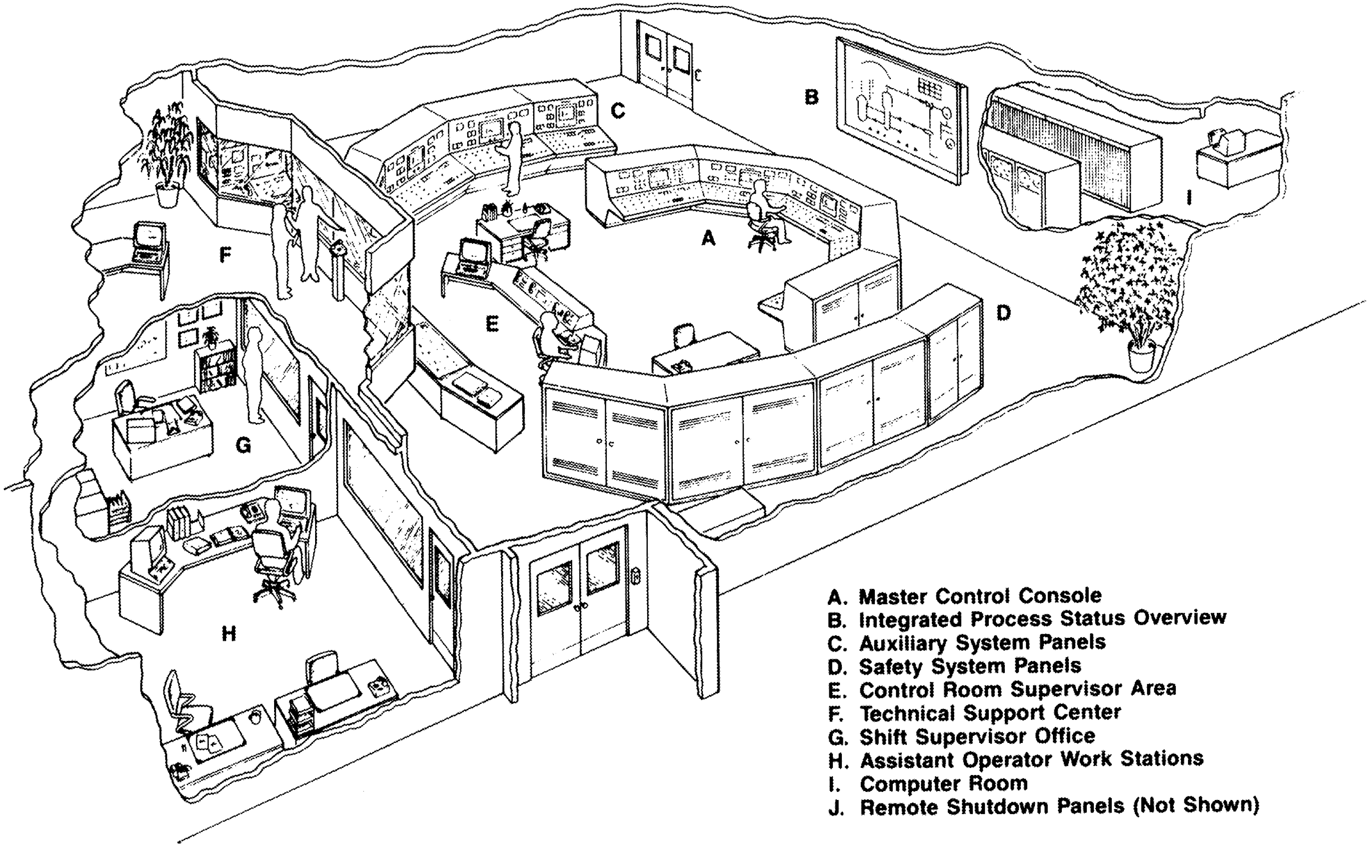

Fig. 14.2. System 80+™ controlroom (© 1989 Combustion Engineering, Inc.).

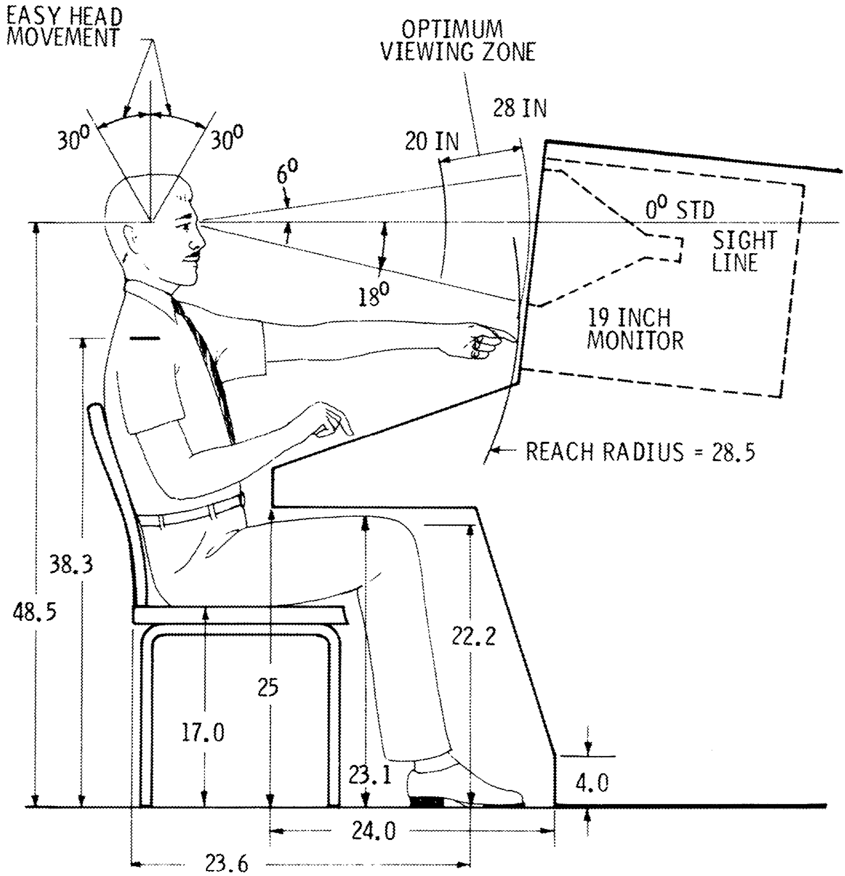

Fig. 14.3. 50th percentile adult male seated at control console (© 1989 Combustion Engineering, Inc.).

as shown in Fig. 14.3. Anthropometry deals with the study of human body measurements.

14.30. All new control rooms use digital multiplexing for signal transmission. In multiplexing, many different signals are sent over the same wire using different frequencies. Individual sensors are wired to remote terminals where they are sequentially sampled for transmission to the control room area, where the signals are separated and distributed to information-processing equipment. In this way, it is possible to save hundreds of miles of hardwiring.

EXPERT SYSTEMS AND NEURAL NETWORKS IN PLANT OPERATIONS [4]

Introduction

14.31. Expert systems and neural networks were introduced in Chapter 8 as information tools. Expert, or so-called "knowledge-based" systems play an important role in nuclear operator support systems and provide diagnostic aids in plant maintenance activities. Their role is to take advantage of various knowledge sources and to provide guidance in solving problems. Neural networks, on the other hand, utilize highly interconnected elements that process information in a manner similar to that used in the human brain.

14.32. A feature of neural networks is their ability to recognize patterns, even when the incoming data are incomplete or distorted, which is not possible with normal computer programs. Therefore, the information-gathering power of expert systems, with the disadvantage of some input uncertainty or "fuzziness," complements the processing power of neural networks, which can quickly yield results despite incomplete input. Although application of expert systems to nuclear power plants has become fairly common, neural network use has been mostly developmental in nature. Our primary purpose is to draw attention to the features of such applications.

Expert Systems for Operator Support

14.33. The reactor operator must be able to react quickly to indications of abnormal conditions. Expert systems can help provide the information needed for effective action. For example, an expert system for PWR operator support would consist of several subsystems, each having a different function. A *diagnosis* subsystem identifies an abnormality. Without assistance, an operator confronted with various offnormal meter readings would have to *infer* the basic cause. Similarly, multiple alarms triggered by limit exceeding conditions may not be helpful in pinpointing the malfunction.

14.34. In one type of diagnosis subsystem or module, logic trees are prepared from a knowledge or "rule base" relating *symptoms*, such as those indicated by meters or alarms, to basic *faults*. A pattern search then seeks one or more matches. As a simplified example, consider leakage in the reactor coolant system (RCS) as the fault. If we had a high charging-rate flow alarm and an indication that the pressurizer level was decreasing, the search would conclude that RCS leakage was likely, which then could be verified by additional signals.

14.35. Once a fault has been identified by the diagnosis module, the computation moves to an *operational guidance* module which calls upon an expert knowledge base to provide corrective measures. In our simplified example, a load reduction and perhaps reactor trip would be indicated. However, for a more sophisticated case, a less obvious action might be recommended.

Expert Systems Development

14.36. The electric utility industry has recognized that there is a need for developing, testing, and applying expert systems. Quality assurance is a consideration since whenever subjective opinions are included in a knowledge base, the matter of confidence levels needs to be addressed. To meet a variety of needs, the Electric Power Research Institute established in 1989 the Knowledge-Based Technology Applications Center (KBTAC), located on the campus of Syracuse University.

Neural Network Development

14.37. Neural networks as an information-processing technology have attracted a great deal of recent developmental attention and appear to have significant applications to nuclear power plants, particularly in those areas in which expert systems are used, such as control diagnostics. Improvements in the monitoring of various systems as a result of neural network pattern recognition capability also appear likely.

PLANT MAINTENANCE

Introduction

14.38. Equipment maintenance is a very important plant operations activity, but is generally given little attention in engineering texts. In fact, accommodation of maintenance requirements is a vital plant design requirement. Components, piping, wiring, and auxiliary equipment must be arranged in such a manner that routine maintenance and component replacement can be carried out efficiently. The presence of a radiation field can be an important design consideration.

14.39. Maintenance is usually planned during scheduled refueling shutdowns. Unscheduled shutdowns are expensive, with the economic penalty generally based on the cost of replacement energy. Therefore, measures to assure trouble-free equipment performance are justified and should be accomplished during planned shutdowns. During such shutdowns, careful work scheduling is necessary to avoid excessive outage time. Radiation exposure limits to personnel and the need for protective clothing require attention.

14.40. During the late 1980s, the NRC increased its involvement in nuclear plant maintenance. As a result, EPRI formed the Nuclear Maintenance Applications Center (NUMAC) to improve maintenance effectiveness by coordinating information and developing technical repair guides for specific equipment. Also, the Institute of Nuclear Power Operations (INPO) has developed standards for maintenance. It is recognized that there is an optimum maintenance effort as a result of the cost of failures balanced by the expenses of preventive maintenance. A goal is to devote about 70 percent effort to preventive maintenance and 30 percent effort to corrective maintenance.

Plant Aging

14.41. When present nuclear power plants were put into service, a 40-year operating lifetime was projected, essentially on an arbitrary basis. However, a maximum of 40 years was established by Congress in 1954 as the duration of an operating license. With no new plants being built and some plants approaching 30 years of service, operating license renewal for an additional 20 years has been recommended by the U.S. Department of Energy [5]. To allow ample time for regulatory decisions, NRC has developed rules for considering license renewal. Essentially, the principle applies that the present adequate level of safety must be maintained during any extended operating period [6].

14.42. Extended life studies have concluded that it is feasible to maintain safety levels, regardless of age, through repair, refurbishment, and replacement. Degradation in component performance can be detected by extensive routine surveillance and testing. Various aging mechanisms were identified which fall into categories of fatigue, erosion, corrosion, service wear, radiation embrittlement, thermal embrittlement, chemical effects, material creep, and environmental degradation [7]. Among these, PWR pressure vessel radiation embrittlement is probably the most important and will be discussed further in the next section. Corrosion problems have required the replacement of some PWR steam generators. Although such replacement is expensive, with about a 6-month outage required, it is a proven procedure.

14.43. In 1985, the NRC added rule 10 CFR 50.61 entitled "Fracture Toughness Requirements for Protection Against Thermal Shock Events." The rule establishes pressure vessel screening criteria and calculation methods for a reference temperature for nil-ductility transition (§7.16). Of concern is the possibility of brittle fracture during a pressurized thermal shock transient as a result of the sudden injection of cold water during operation of the Emergency Core Cooling Systems. For plates and axial welds, the maximum NDT temperature is 132°C (270°F), while 149°C (300°F) is the limit for circumferential welds at the core beltline. Licensees are required to submit projected values up the time of license expiration. Should the criteria be exceeded, neutron flux reduction programs are required.

14.44. The pressure vessel wall fast-neutron flux exposure can be reduced by lower power reactor operation, but a preferred approach is to use special reload core designs to provide very low neutron leakage. A number of design options are available but generally feature loading the most depleted fuel or special blanket assemblies around the periphery of the core (§10.26).

14.45. In-place thermal annealing of a reactor vessel to remove some or all of the effects of neutron embrittlement is a "last ditch" option being investigated [8]. A technique that has been used successfully in Russia on defueled PWRs involves the use of special heaters in the region of critical vessel welds to raise the metal temperature to 454°C for about 6 days. Wet annealing is another possibility but would lead to lesser benefits. In this technique, the vessel and primary cooling circuit is held for about 6 days at a temperature of 340°C using main reactor cooling pump friction heat. Regulatory considerations associated with annealing methods would require resolution.

Life-Cycle Management

14.46. As the economic desirability of extending the operating lifetime of existing plants became recognized, a longer view for nuclear plant maintenance has been adopted. Rather than merely meeting regulatory requirements on an operating cycle-by-operating cycle basis, utilities have implemented aggressive long-term maintenance programs to avoid age-related degradation of safety-related systems, structures, and components. Such life-cycle management programs have as their objective the continuation of an "as new" plant condition so that a 20-year operating license extension can be applied for some years before the current 40-year license expires.

REGULATORY ASPECTS OF OPERATIONS

14.47. We have seen that licensing and other regulatory considerations affect practically every aspect of reactor operations. In addition to the activities described, several additional points are worthy of mention here. Adherence to regulations is monitored by on-site NRC personnel. Reactor operators are licensed and must meet strict fitness for duty requirements. The industry has provided extensive training programs to qualify operating personnel which are subject to accreditation by INPO.

14.48. During maintenance operations, radiation exposures must be kept "as low as reasonably achievable" (ALARA) (§6.63). Radiation protection and exposure times must be such that doses are as far below the numerical limits as is practical, as confirmed by appropriate monitoring and reports to the NRC.

14.49. In-service inspection of components and auxiliary systems is required to assure the continued integrity of a reactor's primary system and safety-related equipment. In connection with this and other license-related activities, reports to NRC are required.

REACTOR DECOMMISSIONING

Introduction

14.50. The cost of decommissioning and the problems associated with the retirement of nuclear power reactors have become issues that have affected public acceptance. Therefore, it is useful to examine some of the considerations involved. One of the requirements for an operating license is to provide reasonable assurance that funds will be available to decommission the facility. Detailed requirements for a minimum guarantee of the order of $150 million, with provisions for escalation, are given in 10 CFR 50.75. Therefore, with funding assured, the important considerations are the technical options available and the availability of an experience base to confirm the methods to be used.

Technical Options [9]

14.51. For all options, the reactor fuel is first removed. Loose contamination would be cleaned from accessible plant areas and radioactive wastes would be processed and shipped off-site. The subsequent strategy depends on the urgency to clear the site and comparative economics. A 20-year delay would reduce the residual radioactivity levels from major components by about a factor of 10. However, cobalt-60 activity would still require remote handling of equipment. A decay period of about 100 years would be needed to permit manual handling. Although a 20-year delay might simplify removal operations somewhat, the unavailability of the site could carry an economic penalty. Therefore, we have the option of prompt dismantling and removal or so-called "mothballing" for a period of years prior to dismantling. A third option is to enclose the vessel and other radioactive components with a concrete barrier and allow the plant to remain for an indefinite period.

14.52. All options would require a possessions-only license from the NRC which would call for security provisions, environmental surveys, and reports until all radioactive materials are removed from the site. Therefore, the expense of site maintenance and public relations considerations are likely to favor the early dismantling option.

Decommissioning Experience

14.53. Many small developmental reactors in a number of countries have been decommissioned without problems. A small (58-MWt) BWR with a 118-ft-high, 85-ft-diameter containment at Elk River, Wisconsin was shut down in 1968 and completely dismantled from 1972 to 1974, with the site converted to a parking lot. The cost was $5.7 million.

14.54. The 6-year decommissioning of the Shippingport, Pennsylvania Station, starting in 1984, one year after shutdown, served as a demonstration of the prompt dismantling option [10]. Although the reactor operated at only a modest rating of about 500 MW(t), the pressure vessel had a diameter of 10 ft, which is comparable to the 15-ft-diameter of present large PWR vessels. Other comparisons indicate that the decommissioning operation was indeed relevant.

14.55. Since the reactor pressure vessel is the largest and the most highly irradiated component in a reactor plant, its removal and disposal is the most challenging decommissioning operation. In the case of Shippingport, it was possible to prepare a monolithic package including the vessel, internals, and concrete shield, remove it from the containment, and ship it by barge to a burial site at Hanford, near Richland, Washington. For another reactor, rail transportation might be necessary. This is likely to require segmentation, which would lengthen the project. Although steam generator removal is also a challenge, there has been ample experience with removal and replacement in existing plants.

14.56. The approximate $100 million cost of the Shippingport project provides a useful reference point for estimating decommissioning costs for presently operating reactors and those to be ordered. However, cost estimates often show large variations, with site-specific considerations and regulatory uncertainties playing a factor [11]. A logical approach is to include an expense for future decommissioning as part of the annual energy generating costs. For example, in Table 10.2, we have *arbitrarily* included an annual allowance of $5 million for future decommissioning in the listing of energy generation costs. Considering the time value of money at an annual compound interest rate of 4 percent, this will yield $149 million in 20 years, $280 million in 30 years, and $475 million in 40 years for decommissioning. Of course, escalation (inflation) could add to the future cost of decommissioning. On the other hand, should this become significant, a higher compound interest rate may be appropriate for the accumulating funds. Since such cost projections are in the range of most estimates, it is reasonable to assume that nuclear power plants will not become an economic or environmental burden after they have completed their useful life.

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CHAPTER 15

Advanced Plants and the Future

INTRODUCTION

What Is Needed

15.1. "Next generation" reactor concepts have been developed to provide some enhanced safety and economic features for new plants that may be ordered during the 1990s and subsequent years. However, before considering such designs, we should examine the need for additional generating capacity and the likely competitiveness of nuclear plants in meeting this need.

15.2. New generating plants are needed to replace plants that are retired from service and to meet growth in demand. Need projections have been the subject of numerous studies [1]. In the United States an annual growth rate of electricity sales of about 3 percent during the late 1980s and early 1990s indicates a likely need for additional generating capacity after the year 2000.

15.3. A factor in energy planning is the effectiveness of *demand-side management*. In such management, utilities attempt to influence customer use patterns in various ways so that the efficiency of utilization is improved.

The term *efficiency* is preferred to *conservation* to avoid the impression that energy is being saved by sacrificing the quality of life. For example, lower rates during hours of low demand provide an incentive to shift electricity use away from peak hours, thus making better use of generating capacity. Aggressive programs to help individual customers improve their utilization efficiency are being carried out by many utilities. However, despite such efforts, it is reasonable to assume that the demand will outgrow supply in the near future and that there will be a need for new generating capacity. Such a need must be anticipated some years in advance to allow time for construction. Whether a significant number of nuclear power plants are constructed to meet the energy demand will depend on the resolution of several challenges.

15.4. New nuclear power plants will be built only if *both* the public and electric utility management are convinced that such new construction is in their own best interest. Public attitudes vary from strongly negative by a militant small fraction to various levels of uneasiness by other groups as a result of the Three Mile Island and Chernobyl accidents and delays in managing high-level wastes. However, motivated by concerns regarding atmospheric pollution and the greenhouse effect caused by fossil-fueled power plants, there appears to be an increasing willingness by those who hold reservations to accept the construction of new plants provided that they would be extraordinarily safe.

15.5. The safety of present LWR reactors is considered completely adequate, particularly following numerous systems improvements made as a result of the lessons learned from the Three Mile Island accident. However, the poor economic experience of some utilities has raised some questions regarding the wisdom of new investment for nuclear power plants. Although a high rate of inflation during the 1970s, as well as orders unjustified by energy demand, contributed to plant cancellations, poor management, a lack of standardization, and regulatory inconsistencies were factors. Also, for many U.S. operating reactors, plant capacity factors were less than those for similar plants in other countries. Therefore, from the utility viewpoint, there is a need for financial risk reduction.

15.6. New reactor designs with "passive" safety features do not require prompt operator intervention in the event of a wide variety of accidents. In addition, such features permit simplification of many of the safety systems, with consequent cost savings. Therefore, these designs tend to meet both the safety concerns of the public and the utility need for systems of reasonable cost. We will examine the important concepts as well as the utility financial risk matter. As provided by 10 CFR 52 (§12.239), design certification procedures for several of these concepts are in progress.

Plant Size

15.7. Passive reactor plants have about one-half the electrical generating capacity of present large units. It is felt that such smaller units are more practical for medium-sized utilities that wish to increase their capacity than the large units. Generally, a generating unit should be no larger than about 15 percent of a utility system capacity. Therefore, worldwide market considerations favor a smaller standard plant. Since smaller units lend themselves to a greater degree of standardized factory manufacturing than large units, cost savings are likely. Another consideration is that smaller plants contain less radioactive material and are easier cooled. Some level of risk reduction therefore results.

Advanced Systems Common Design Features

15.8. During the years that have elapsed following the design and construction of present nuclear plants, much effort has been devoted to planning the features of new plants. In fact, a major Electric Power Research Institute (EPRI) program was devoted to the development of a comprehensive set of design requirements for advanced plants. Although there are several different advanced plant concepts, they all share features intended to simplify design, with an accompanying reduction in construction time and costs. Standardized design is emphasized for the entire plant, not just the nuclear portion. Also, all concepts utilize advanced instrumentation and control systems to improve safety and reduce costs.

15.9. Designers have specified standard components where possible and arranged these components into prefabricated modules. During recent years, the petroleum, shipbuilding, and power plant industries have developed successful construction practices featuring the manufacturing of very large component modules at a central location and then transporting them to the construction site. In this way, fabrication and inspection is more efficient and field labor costs are reduced. Clearly, cost benefits are increased if the manufacturing facility serves several plant sites. Modularization is carried an additional step with gas-cooled and liquid metal–cooled concepts in which several small steam generating units are coupled to a turbine-generator to form a *power block*. These may then be replicated at a given site to provide whatever station capacity needed.

15.10. The development of advanced instrumentation and control systems has been coordinated by the Electric Power Research Institute. Control rooms have been redesigned using human factors principles. Digital instrumentation and multiplex signal transmission have been featured. The level of automation of control tasks has been increased. Maintenance-related instrumentation throughout the plant has received attention, particularly that providing diagnostics to identify equipment that may be approaching failure although still operationally functional.

Types of Passive Systems

15.11. Passive systems based on light-water core designs benefit from the considerable design and operating experience of existing reactors and therefore are attractive to utilities. A PWR design, designated as AP600, has been developed by Westinghouse Electric Corp. in cooperation with the U.S. Department of Energy (DOE) and EPRI. Similarly, a 600-MW(el) BWR, known as the SBWR, has been developed by the General Electric Co., with DOE and EPRI cooperation. In this case, the smaller size permits natural recirculation of the coolant, a significant design simplification.

15.12. Gas-cooled reactors of various types have been used to generate electricity since the earliest days of the nuclear power industry, particularly in the United Kingdom. During more recent years, prototypes of the high-temperature, gas-cooled reactor (HTGR) concept have operated in the United States and in Germany. This type of helium-cooled, graphite-moderated system features fuel embedded in small spherical particles that retain fission products. The modular high-temperature gas-cooled reactor (MHTGR) is a passive, inherently safe concept which builds upon HTGR experience.

15.13. Sodium-cooled fast reactors have also received development attention for many years. Most notable among experimental reactors is EBR-II, which has been operating very satisfactorily since 1961 at a rated electrical output of 20 MW. Demonstrations of passive, inherent safety characteristics led to the development of the integral fast reactor (IFR) concept at Argonne National Laboratory. Subsequently, a team led by the General Electric Co. developed a modular reactor concept PRISM (Power Reactor, Innovative Small Module), which appears promising for future energy requirements.

15.14. The advanced passive PWR and BWR are likely to receive early favor by electrical utilities seeking a 600-MW(el)-size unit, particularly since they utilize technology familiar to them. The other candidates for commercialization are the MHTGR and PRISM concepts, which, should they receive utility support, are likely to come somewhat later. Finally, we will mention examples of other interesting concepts which have potential. One is the process inherent ultimate safety (PIUS), developed in Sweden, and the safe integral reactor (SIR), developed by a team led by Combustion Engineering based on earlier PWR designs intended for maritime use.

THE AP600 [2]

Introduction

15.15. The AP600 ("advanced, passive") reactor design features a two-loop Westinghouse PWR arrangement modified to have conservative safety margins and permit simplification of many supporting subsystems. The coolant loop arrangement and some of these features are shown in Fig. 15.1. In each loop, circulation is provided by two closely coupled canned motor pumps. As in the Combustion Engineering evolutionary PWR (§13.24), the steam generator has been enlarged to improve operating margins.

15.16. Some preliminary design specifications are summarized in Table 15.1. The core contains 145 fuel assemblies of the 17 × 17 lattice type with an active length of 3.66 m (12 ft), but similar to that described for the larger PWR in Table 13.1. However, the power density has been

Fig. 15.1. AP600 reactor coolant system [2, McIntyre and Beck].

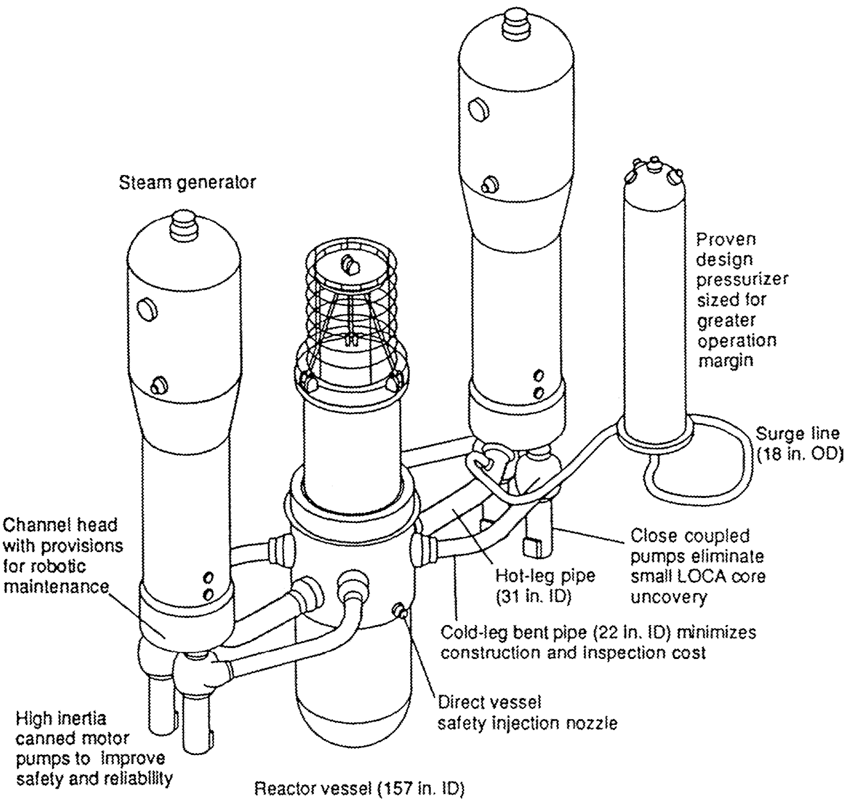


TABLE 15.1. Design Specifications for Advanced PWR (AP600)

General	
Power	
Thermal	1933 MW
Electrical	600 MW
Specific power	28.9 kW(th)/kg U
Power density	78.8 MW(th)/m^3
Core	
Length	3.66 m (12.0 ft)
Diameter (equil.)	2.92 m (9.58 ft)
Fuel	
Rod, OD	9.5 mm (0.374 in.)
Clad thickness	0.57 mm (0.0225 in.)
Pellet diameter	8.19 mm (0.3225 in.)
Rod lattice pitch	12.6 mm (0.496 in.)
Rods per assembly	264 (17 × 17 array)
Assembly pitch	215 mm (8.466 in.)
Assemblies	145
Fuel loading, UO_2	75.9 × 10^3 kg (1.67 × 10^5 lb)

Thermal-Hydraulic	
Coolant	
Pressure	15.5 MPa(a) (2250 psia)
Inlet temp.	277°C (529°F)
Outlet temp.	316°C (599°F)
Flow rate, core	9.19 mg/s (7.29 × 10^7 lb/hr)
Mass velocity	2.57 Mg/s · m^2 (1.89 × 10^6 lb/hr-ft^2)
Rod surface heat flux	
Ave.	0.451 MW/m^2 (1.43 × 10^5 Btu/hr-ft^2)
Max.	1.17 MW/m^2 (3.72 × 10^5 Btu/hr-ft^2)
Linear heat rate, ave.	13.4 kW/m (4.1 kW/ft)
Steam pressure	5.62 MPa(a) (815 psia)
Control	
Rod cluster elements	24 per assembly
Control assemblies	45
	16 gray rod clusters

reduced by about 30 percent to improve safety margins. Consistent with this reduction, a number of other specifications are less challenging than those listed in Table 13.1.

Passive Features

15.17. The passive features center on the operation of the safety injection systems and provisions for removing core decay heat. Passive systems are defined as those which are self-contained or self-supported. To perform their safety functions, they rely on gravity, as in natural circulation cooling, or stored energy, such as that in compressed gases. Emergency electrical needs must be met by battery power rather than by standby diesel generators as is current practice. Valves to initiate safety system operation should be check valves or those activated by stored energy. Passive systems are expected to perform their safety functions for up to 72 hours after the initiating event *independent of operator action* and off-site power. Thus, should there be an emergency shutdown, decay heat must be passively removed from the core. Also, coolant must be replaced if there is a loss-of-coolant accident.

15.18. In the AP600, the safety injection systems are integrated with the reactor coolant makeup and residual heat removal systems. Decay heat is removed by a natural circulation system utilizing a heat exchanger submerged in a large refueling water tank within the containment as shown in Fig. 15.2. The tank volume is sufficient to absorb decay heat for about 2 hours, after which the water would start to boil. Then, the generated steam would condense on the inside of the containment and drain back into the refueling tank. Gravity-driven injection or pressurized accumulators replace water lost in the coolant system. As shown in Fig. 15.2, two core makeup tanks containing borated water at full cooling system operating pressure are available to inject water by gravity in the event of a small-break LOCA. For larger breaks, two accumulators pressurized to 4.93 MPa(a) (715 psia) by nitrogen are available for injection. Finally, additional water can be injected by gravity from the in-containment refueling water storage tank provided that the cooling system pressure has first been reduced to about 1.72 MPa(a) (25 psia). This passive safety injection system eliminates the need for various pumps and support components such as diesel generators, thus significantly reducing the number of plant components required.

15.19. Passive cooling is also provided for the containment. Figure 15.2 shows an inner steel containment vessel which serves to transfer heat to outside air moving by natural circulation between the steel shell and the outer concrete shield. Additional cooling is provided by water stored in a tank at the top of the concrete shield. This water is allowed to evaporate

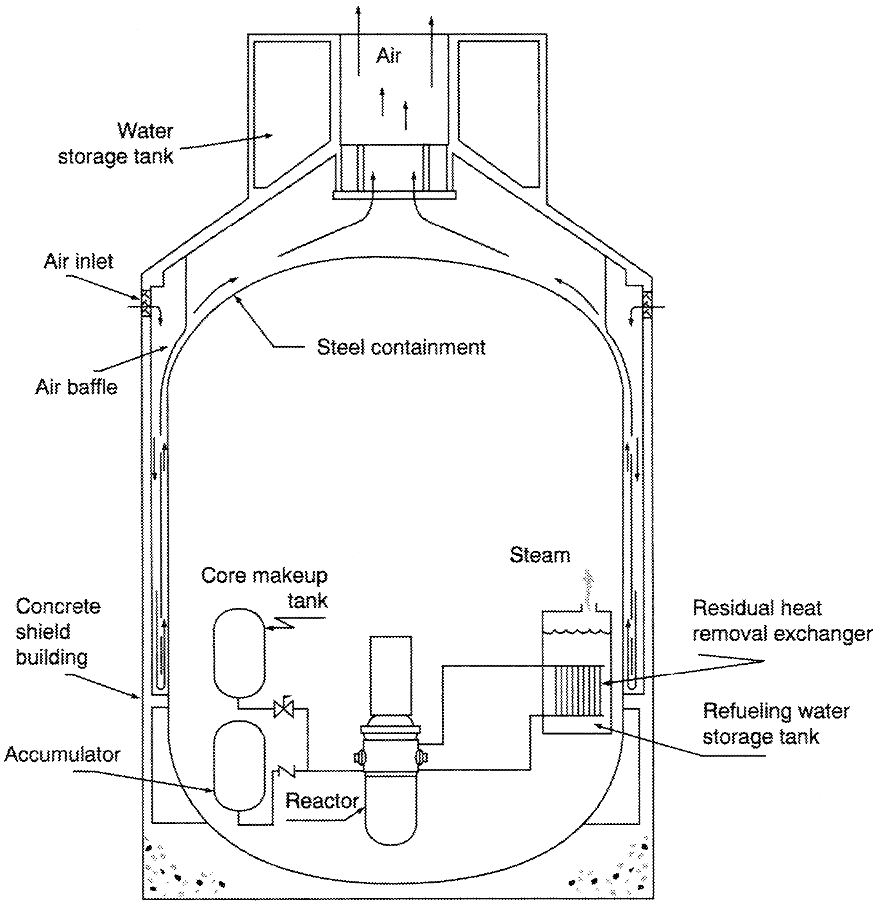


Fig. 15.2. AP600 passive cooling features (Adapted from Ref. 2, McIntyre and Beck).

on the outside of the steel vessel. This water tank is sized for 3 days of operation, after which refilling is expected. However, if the water supply is not replenished for some reason, the containment pressure will rise to a maximum in about 2 weeks, but will then reach only 90 percent of the design pressure.

Other Innovations

15.20. Probabilistic risk analysis was used extensively in the design process to select the design options that would minimize the predicted core melt frequency. This approach contrasts with PRA studies of already built plants for which remedial options are limited. Defense in depth is provided by various redundant safety systems which would be effective before the passive features are called upon to control an accident. The response of all safety systems to every known transient and accident scenario has received consideration during the design process.

15.21. The design includes the MSHIM load-following strategy, which utilizes "gray" rods, thus avoiding the need for boron concentration adjustment (§14.21). An advanced instrumentation and control system using multiplexing reduces the amount of wiring needed (§14.30). Plant layout has been planned to provide good access for construction and maintenance. Also, extensive use of prefabricated modules is expected to reduce construction time and improve economics.

15.22. The AP600 is designed for a 60-year operating life. Cost reduction is achieved by the use of passive safety features, general design simplicity, and complete plant standardization, which requires 100 percent engineering prior to the start of construction. Such savings tend to compensate for the unfavorable economy of scale of the 600-MW(el) size compared with large plants. An estimate of $1370 per kilowatt (1990 dollars) is within the $1500/kW EPRI goal for advanced plants.

SIMPLIFIED BOILING WATER REACTOR

Introduction

15.23. The Simplified boiling-water reactor (SBWR) is a 600-MW(el) reactor which features natural circulation and passive features to enhance safety and simplify system design. An assembly view of the vessel and internals is shown in Fig. 15.3. Noteworthy is a chimney section above the core to promote the natural circulation of the coolant. As a result, the reactor pressure vessel (RPV) is 2.6 m higher than that required for the 1356-MW(el) ABWR (Table 13.4).

15.24. A summary of some design specifications is given in Table 15.2. The power density has been reduced by about 25 percent compared with present BWRs to increase safety margins. The relatively short (2.44 m) core consists of 732 fuel assemblies of the standard 8 $\times$ 8 lattice type. However, advanced fuel assembly designs can be accommodated. A 7.0-m-i.d. vessel is then needed to accommodate this geometry. A large water inventory for a reactor of this power rating provides an inherent damping of transient disturbances.

Natural Circulation

15.25. Natural circulation, or more accurately, natural recirculation of the coolant, is the most interesting thermal-hydraulic feature of the SBWR.

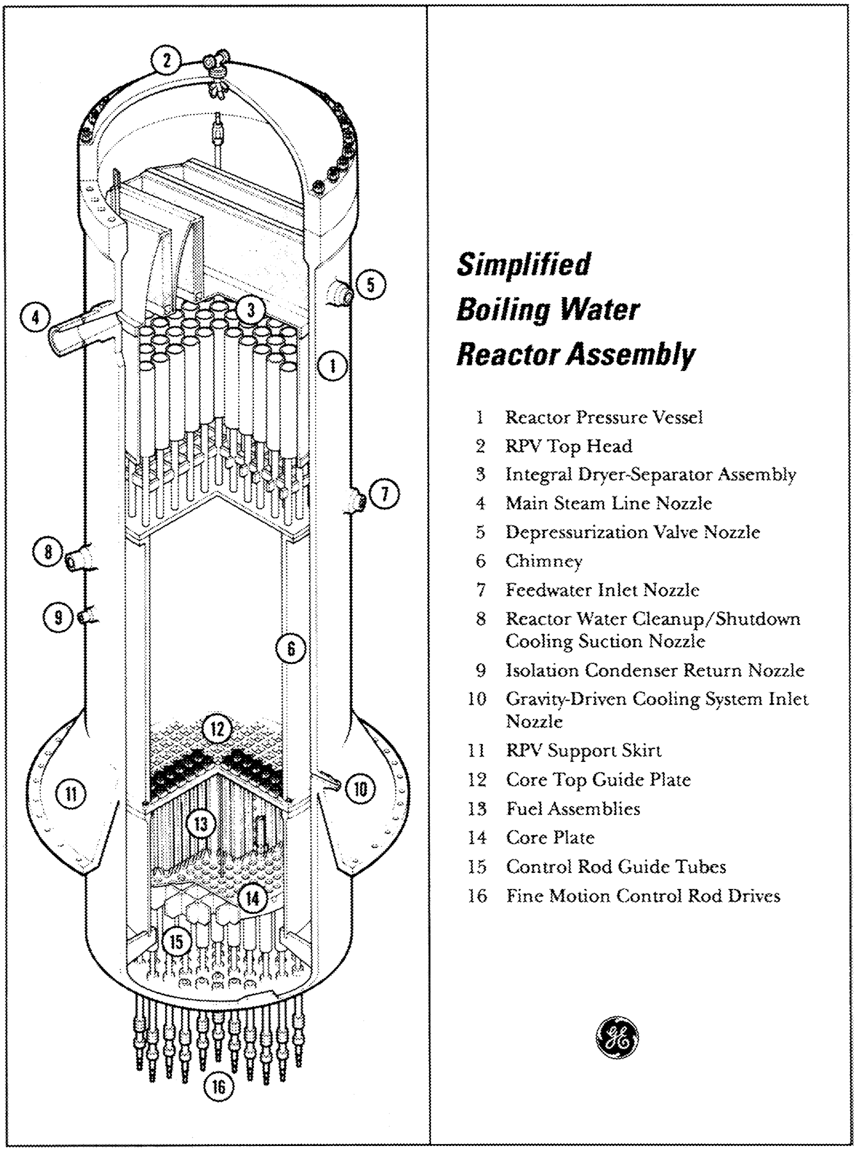


Fig. 15.3. SBWR assembly (General Electric Co.).

TABLE 15.2. SBWR Design Specification Summary

General		Thermal-Hydraulic	
Power		System pressure	7.17 MPa(a) (1040 psia)
Thermal	1800 MW	Steam temp.	288°C (550°F)
Electrical	600 MW	Feedwater temp.	216°C (420°F)
Power density	42.0 MW/m^3	Recirculation flow	60 Mg/s
Core		**Control**	
Length	2.44 m	Control rod number	177
Diameter, equivalent	4.73 m	Neutron absorber	B_4C
Fuel		Control rod form	Cruciform
Assemblies in core	732	drive	Electro-hydraulic fine-motion
Lattice type	8 × 8		
Rod, OD	12.3 mm (0.484 in.)		

It is dependent on the difference in density between the cooler returning and feed fluid mix in the downcomer leg and the steam–water mixture in the core and chimney. As pointed out in §9.131, the flow rate depends on the relative influence of the density difference "driving force" and the frictional flow resistance. Hence, a short core of relatively low flow resistance is needed to assure an adequate recirculation rate. To assure reactor stability, the core flow must be maintained above certain levels which depend on the reactor power.

15.26. Reactor stability during normal operation and anticipated transients is an important design consideration for all BWRs. For example, reactivity feedback instability of the reactor core could result in power oscillations. Hydrodynamic channel instability could inpede heat transfer to the moderator and also drive the reactor into power oscillations. Finally, the total system stability depends on the basic process dynamics. Stability criteria are stated in terms of a *decay ratio*, which is defined as the ratio of the magnitude of the second oscillatory overshoot to that of the first overshoot resulting from a step perturbation. The lower the ratio, the more stable the system. SBWR stability decay ratios have been modeled and found to be at least a factor of 2 lower than those for currently operating BWRs [3].

Other Passive Features [11]

15.27. Engineered safety features rely on gravity or stored energy for core cooling and decay heat removal. In the event of a core overheating accident leading to excessive system pressure, relief valves automatically release the steam–reactor coolant mixture to a suppression pool similar to that in present BWRs. Nitrogen is used to inert the containment to reduce the potential for a hydrogen explosion. The system, now at low pressure, is then flooded by gravity with water stored in an elevated pool. Since there are no large pipes attached to the vessel at or below the core level, the core should remain covered for all design basis events. The consequences of a severe accident would be mitigated by gravity-driven drywell flooding.

15.28. Decay heat removal while the system is at full pressure is accomplished by nonpassive pumps and heat exchangers. However, after blowdown, when the reactor vessel is isolated from the turbine-condenser, decay heat is removed by three passive condensation loops. Following a loss-of-coolant accident, the steam–air–nitrogen mixture in the upper drywell flows to three condensers which are submerged in a separate elevated water pool which supplies the cooling. Condensate is drained to the gravity-driven injection pool and noncondensable gases are discharged below the surface of the suppression pool. Steam formed in the condenser pool as a result of the 30-MW(th) heat-transfer capacity is vented to the atmosphere. The pool has a capacity to provide 3 days of condenser cooling before water replenishment, which, of course, would normally be accomplished automatically.

15.29. A passive natural circulation air system is used to serve the control room area in the event of station blackout. Since the various core cooling features are also passive, the safety grade emergency diesel generators necessary for present systems can be eliminated.

Other Innovative Features

15.30. Similar to the ABWR (§13.49), the SBWR has electrical-hydraulic fine-motion control rod devices. The advanced control system makes wide use of multiplexing and microprocessor-based instrumentation.

15.31. The power generation system has been simplified. For example, an advanced tandem, double-flow turbine permits a more compact condenser and piping arrangement, reducing the building requirements. Also, as for other advanced plants, the design for the SBWR complete plant package includes numerous subsystem simplifications that utilize factory fabricated modular components which reduce costs and contribute to shorter construction schedules.

MODULAR HTGR (MHTGR)

Introduction

15.32. Gas-cooled reactors, such as the HTGR, are quite different from LWRs. The moderator is graphite, a solid, while the coolant, normally pressurized helium, is chemically inert and does not change phase. The graphite moderator remains stable up to very high temperatures, and has a high heat capacity, characteristics that permit the reactor to accommodate accidental transients safely. However, since power densities tend to be much lower than those for LWRs, the cores are relatively large, a disadvantage. By using the modular concept, in which several units of modest power rating and easily manageable core size collectively generate the desired electrical capacity, the major safety advantages of the HTGR concept can be practically realized.

15.33. Many commercial reactors cooled with carbon dioxide were built in the United Kingdom and in France starting in the 1950s. The early ones were fueled with metallic natural uranium clad with a magnesium-aluminum alloy called Magnox. Later plants, designated as advanced gas-cooled reactors (AGRs), used low-enriched uranium oxide clad with stainless steel as fuel and carbon dioxide as coolant. The AGRs are able to achieve high coolant temperatures, which provides a thermodynamic efficiency of over 40 percent. Newer AGRs built in the United Kingdom featured the use of a prestressed concrete reactor vessel (PCRV) for containing the whole primary system, consisting of the core, steam generators, and circulators [4].

15.34. While the AGR was being developed, attention was also given in various countries, including the United States, to the concept for an all-ceramic, helium-cooled, graphite-moderated, reactor using the thorium fuel cycle. This became the basis for the HTGR, the U.S. development of which was led by General Atomics. Other development efforts have been active in other countries, particularly in Germany. A 40-MW(el) demonstration plant operated successfully at Peach Bottom, Pennsylvania from 1967 to 1974 using a prismatic core concept (§15.41). This was superseded by a 330-MW(el) commercial-size demonstration plant at Fort St. Vrain, Colorado, which operated from 1979 to 1989. During this period, numerous mechanical problems were encountered, all of which were resolved or well understood prior to shutdown. Meanwhile, in Germany, a 300-MW(el) plant, the THTR-300, demonstrated the "pebble-bed" core concept from 1987 to 1989. During the early 1970s, a number of orders were placed by U.S. utilities for large HTGRs. However, as a result of economic pressures existing at the time, these orders were canceled.

Modular Concept

15.35. The low power density of gas-cooled necessitates very large cores for reactor plants having electrical generating ratings in the 600 MW or higher range. Prestressed concrete reactor vessels (PCRVs) to contain such systems are proven and safe, but are expensive. By dividing the desired capacity into smaller units, steel pressure vessels can be utilized. Also, the components and subsystems for such small modular units lend themselves to factory fabrication with resulting economies.

15.36. The plant capacity factor of the Fort St. Vrain plant was disappointing as a result of component problems, particularly with the water-lubricated circulator bearing seals. Although the modular HTGR (MHTGR) uses a proven magnetic bearing system for the circulator, which should be trouble-free, the Fort St. Vrain experience demonstrated the economic sensitivity of a large plant to excessive shutdowns. In a modular plant built up from small units, maintenance on individual unit components can be performed more easily than in a large plant. A multiunit plant also permits considerable fuel management flexibility.

15.37. A typical MHTGR plant consists of four identical modular reactor units housed in separate adjacent reinforced concrete structures located below grade, but under a common roof. The reactor units, each producing 350 MW(th) of steam, are paired to feed two turbine generators in a separate energy conversion area to generate a total of 538 MW(el).

Fuel Microspheres

15.38. The use of refractory-coated fuel microspheres is a key feature of the MHTGR as a result of their ability to retain fission products under severe conditions. There are two types of microspheres: fissile, containing a uranium oxycarbide (UCO) kernel with the uranium enriched to 20 percent, and fertile, containing a kernel of ThO_2. The fertile material enhances the conversion ratio.

15.39. Each fissile kernel, of about 350 μm in diameter, is first coated with a porous graphite buffer, followed by three successive layers of pyrolytic carbon, silicon carbide, and pyrolytic carbon, with an outer diameter of about 800 μm achieved. The fertile particles, are similarly coated, but slightly larger. In earlier reactors, only the fissile particles, designated as TRISO particles, contained a silicon carbide layer, to permit fuel recycling, which is not a current option. Both types of coated particles are mixed with an organic binder and graphite filler, then fired to form fuel rods, which are inserted into holes in hexagonal graphite core blocks (§15.41).

15.40. The inner layer of silicon carbide prevents the escape of most fission products, while the outer layer of dense pyrolytic carbon provides mechanical support and acts as backup containment. Extensive testing has confirmed the ability of the coated particles to maintain their integrity and retain fission products up to sustained fuel temperatures of 1760°C. Therefore the safety design philosophy of the MHTGR is that control of radionuclide releases can be accomplished by their retention within the fuel particles rather than by active features or operator actions [5].

Prismatic Core

15.41. The reactor core consists of stacks of hexagonal graphite prisms with holes for fuel rods, for coolant flow, and in some cases, for boron carbide control rods. As shown in Fig. 15.4, there are 10 prisms in a stacked column, arranged in an annulus. Unfueled graphite blocks serve as inner and outer radial and upper and lower axial reflectors. This reflected annular core arrangement is needed to permit passive decay heat removal while maintaining the maximum fuel temperature below 1600°C during a so-called conduction cooldown event during which there is no circulation of coolant. The core height is limited to 7.9 m to allow a maximum power rating while assuring axial power stability to xenon transients.

15.42. A set of 24 control rods located in channels in the inner ring of the annular reflector blocks are used for normal control and for emergency

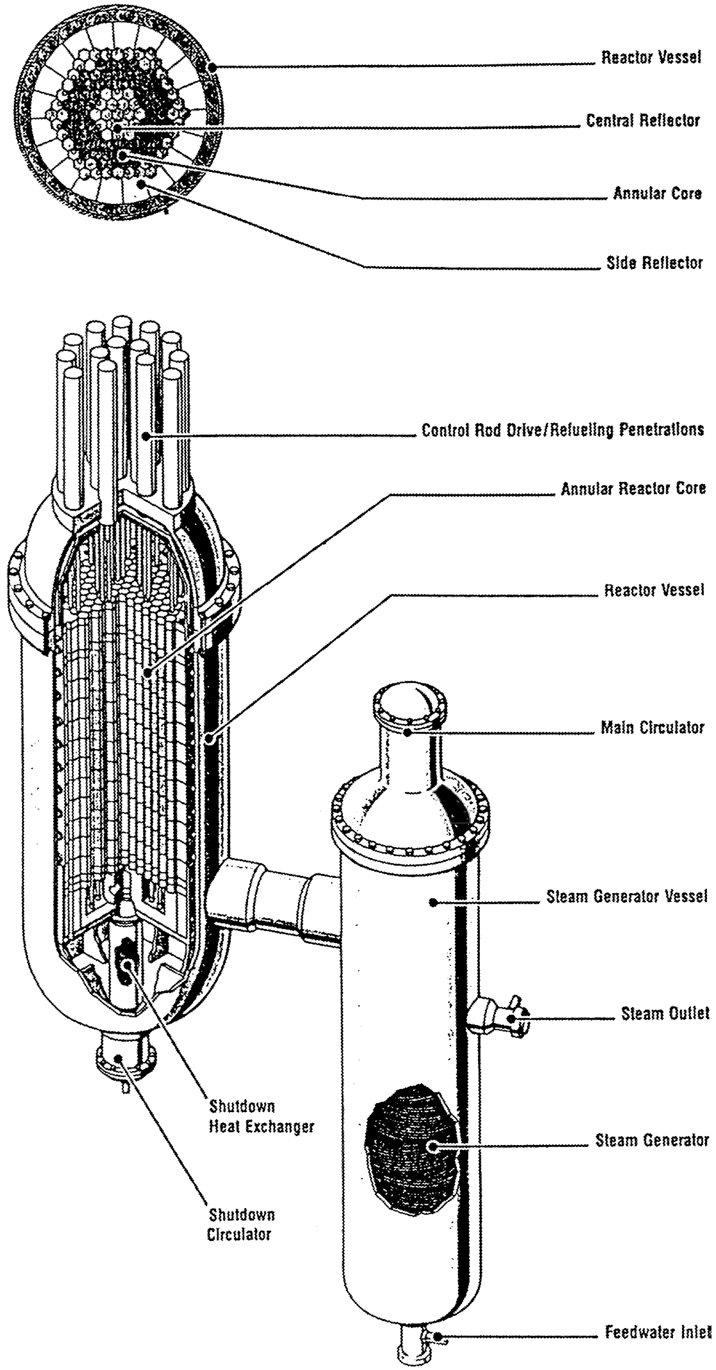


Fig. 15.4. MHTGR nuclear steam supply system (General Atomics).

shutdown. A set of six rods in the outer ring of the central reflector is inserted only during cold shutdown. A reserve shutdown capability is provided by boronated graphite pellets, contained in hoppers above the core, which can be dropped into channels in selected core fuel blocks, if needed.

Nuclear Steam Supply System

15.43. The reactor core, pressure vessel, and steam generator arrangement is shown in Fig. 15.4. Major design specifications are summarized in Table 15.3. In the vessel, the helium coolant flows downward through the core, where it is heated to an average temperature of 687°C, then flows through an inner portion of a coaxial cross duct to the steam generator. The gas flows downward through a helical heat exchange bundle in which steam is produced in the tubes. This cooled gas then flows upward in an annulus compartment to the main circulator integral to the top of the steam generator vessel. Compressed cool gas returns to the reactor vessel through an outer annulus in the cross duct, then flows upward through another annulus between the core and vessel wall to the top of the core to complete the circuit.

15.44. Decay heat during shutdown may be removed either using the main heat transport loop or through a separate shutdown cooling system driven by a circulator at the bottom of the reactor vessel. In this system, heat is transferred to service water in a separate helium to water exchanger.

Gas-Turbine Option

15.45. A design option is to utilize the helium heated in the reactor as the working fluid in a closed-cycle gas turbine power conversion system. The resulting elimination of the steam generator and associated components leads to system simplification and potential cost savings. However, somewhat more development is required than would be necessary for the steam-generating approach.

Passive Features and System Safety

15.46. The inherent characteristics of the HTGR concept provide the safety basis for the MHTGR system. First of all, graphite is stable at high temperatures and has a high heat capacity, which assures that core temperature transients will be slow and readily controllable. Helium is inert both chemically and neutronically. Therefore, coolant interactions with materials during the course of an accident are not possible and coolant

TABLE 15.3. MHTGR Design Specification Summary (Preliminary)

General (4 module basis)		Thermal-Hydraulic	
Power		Coolant (per module)	
Thermal	1400 MW	Pressure	6.39 MPa (925 psia)
Electrical, net	538 MW	Inlet temp.	259°C (498°F)
Power density	5.9 MW/m^3	Outlet temp.	687°C (1268°F)
Core (per module)		Flow rate	157 kg/s
		Pressure drop	34.5 kPa (5.0 psi)
Length	7.9 m	Steam pressure	16.6 MPa (2415 psia)
Diameter		Temp.	538°C (1000°F)
Inner	1.65 m	**Control**	
Outer	3.5 m	Control channels located in reflector blocks, (1 channel/block)	
Fuel		6 inner	
Fuel element	Prismatic hex-block, 0.36 m across flats × 0.0793 m high	24 outer	
		Material	Clad natural boron
Stack	10 elements/column, 66 - Column annulus		
Fuel holes	210, 0.0127 m diameter		
Equilibrium core loading	kg: U/Th 1030/706		
Ave. enrichment	19.9% U-235		

changes do not affect reactivity. Thorium loading in the fuel enhances the negative temperature coefficient of reactivity. These characteristics cause the reactor inherently to shut down.

15.47. In the MHTGR, should both the primary and secondary shutdown cooling systems become inoperative, decay heat would be removed by the passive reactor cavity cooling system (RCCS). The decay heat path is from the core to the surrounding reflector, then to the walls of the uninsulated reactor vessel. Heat would then be transferred by radiation and convection to RCCS cooling panels placed around the vessel. Outside air moved by natural circulation then cools the panels. The annular core geometry limits local heat generation so that this passive system prevents the fuel from reaching a temperature that would damage the refractory coatings on the microspheres.

15.48. The response of the MHTGR to a wide variety of accident scenarios has been studied extensively, but space does not permit even summary coverage here. However, a common question concerns the possibility of chemical reaction of graphite with water and air. The water–graphite reaction is endothermic and normally would have minor impact with no fission product release likely. A worst-case scenario involving a combination of failures results in about a 5-Ci iodine release to the environment, which would result in an acceptable offsite dosage. Air ingress would result in only a small amount of oxidation, primarily because of the large resistance to flow that the cooling tubes provide.

15.49. Another scenario worthy of mention is a depressurization accident, with failure of all forced cooling systems. The reactor would trip, but decay heat would be removed only by the RCCS. A peak core temperature of about 1600°C would occur after about 80 hours. A release to the environment of only about 1 Ci of iodine is predicted.

Economic Potential

15.50. The MHTGR concept benefits from the same advantages of standardization, factory fabrication, and system simplification that apply to other advanced designs having about the same electrical capacity. Although estimated capital cost requirements are slightly higher than for the other designs, they are probably within the error range associated with all such estimates. It should be noted that the MHTGR produces steam at the same conditions as used in fossil-fueled power plants, an advantage. A busbar cost of electricity of about 10 percent less than that from an equivalent-sized modern coal-fired plant has been claimed which appears to justify further attention being given to the potential of the MHTGR.

ADVANCED LIQUID-METAL-COOLED REACTOR

Introduction

15.51. The advanced liquid-metal-cooled reactor (ALMR) design is based on the PRISM (Power Reactor, Innovative Small Module) concept developed by the General Electric Co., while the fuel system is based on the IFR (integral fast reactor) concept developed by Argonne National Laboratory [6,7]. Liquid-metal-cooled fast breeder reactors have attracted the attention of reactor designers since the earliest days of the nuclear power industry because of their ability to transform fertile uranium-238 into fissile plutonium-239 efficiently using fast neutrons (§10.65). Furthermore, sodium cooling allows the reactor to be at atmospheric pressure. However, commercial development of such systems was deferred in the early 1980s as a result of proliferation concerns and economic conditions at the time.

15.52. During recent years, the outstanding performance of new metallic fuels in EBR-II, a 20-MW(el) fast reactor that has been operating very satisfactorily since 1961, resulted in the development of the IFR concept, which features an inherently safe core with passive features and proliferation-resistant on-site fuel recycling. The modular approach offers the flexibility, standardization, and construction economies described previously for other advanced passive systems. Although funding for continued development appeared uncertain in 1994, the concept is described briefly here as an example of a fast-reactor option with future potential.

Plant Description

15.53. Should an electrical generating capacity of 1395 MW be desired, a plant would consist of nine reactor modules arranged in three identical 465-MW(el) power blocks, each with its own turbine generator. Thus, smaller plant sizes of 465 MW(el) and 930 MW(el) can be provided by using one or two of the standard power blocks. Steam generators, one for each module, are in a separate building. Thus, only the reactor modules themselves need be constructed to nuclear safety grade standards.

15.54. A summary of important system characteristics is given in Table 15.4. A reactor module is shown in Fig. 15.5. The core immersed in a pool configuration provides large thermal inertia. Primary sodium coolant is circulated by four electromagnetic pumps from an outer compartment of the vessel receiving cold discharge from two intermediate heat exchangers (IHXs) to a high-pressure plenum at the bottom of the core. Heated sodium leaving the core enters the surrounding pool, from which the IHXs are fed. Secondary sodium flow to the IHXs is provided by an electromagnetic pump in the steam generator facility.

TABLE 15.4. ALMR Design Specification Summary (Preliminary)

General	
Power	
Thermal	1413 MW
Electrical	465 MW
Reactor Vessel	
Outer diameter	5.74 m
Length	16.94 m
Design pressure	0.24 MPa
Core Assembly Number	
Fuel	42
Internal blanket	25
Radial blanket	48
Radial shield	42
Control	6
Fuel	
Composition, %, 27 Pu, 63 U, 10 Zr	
Assembly, 217 rods in hex. bundle	
Rod, O.D.	7.2 mm (0.283 in)
Clad material, HT-9 ferritic alloy	

Thermal-Hydraulic	
Primary Sodium	
Inlet temp.	318°C (604°F)
Outlet temp.	485°C (905°F)
Flow rate	2.54 Mg/s
Intermediate sodium	
Inlet temp.	282°C (540°F)
Outlet temp.	443°C (830°F)
Flow rate	2.27 Mg/s
Peak linear heat rate	34.5 kW/m (10.5 kW/ft)
Control	
System requirements (dollars)	
Burnup reactivity swing	1.3
Temp. defect	3.6
Shutdown margin	1.0
Uncertainties	1.3
Total	6.2

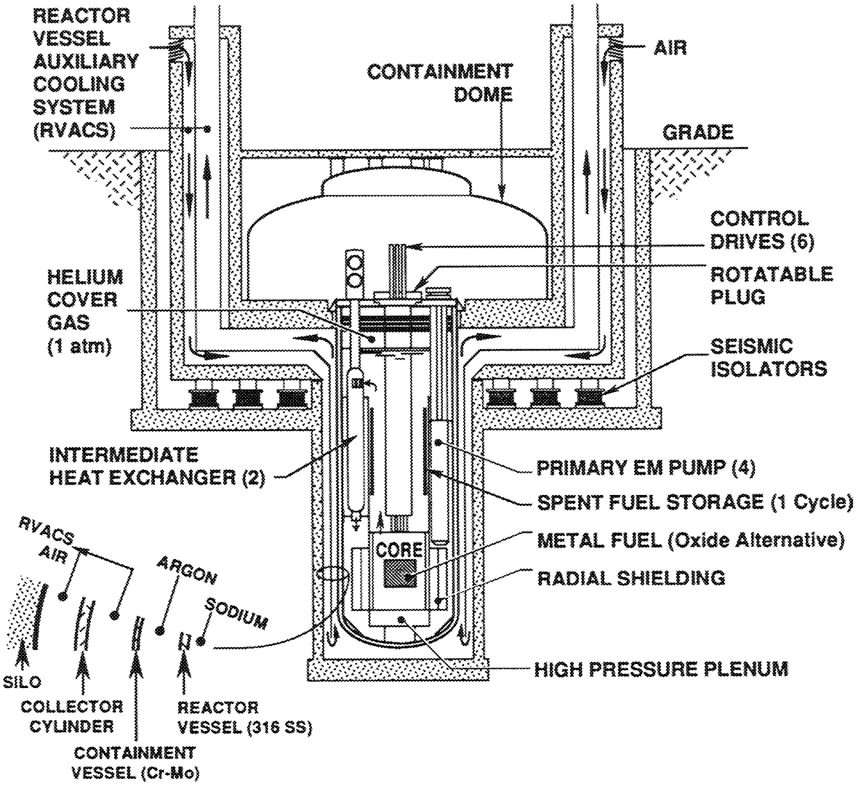


Fig. 15.5. Advanced liquid-metal-cooled reactor module (General Electric Co.).

15.55. Decay heat is normally removed by sending steam directly to the condenser. Natural circulation of secondary sodium to the steam generator is one passive backup. A second backup is by natural circulation of atmospheric air around the containment (guard) vessel in the silo as shown in Fig. 15.5 for the reactor vessel auxiliary cooling system (RVACS).

15.56. The reactor modules, located in underground silos, rest on seismic isolators which decouple the systems from horizontal accelerations. Containment is provided by a guard vessel which backs up the primary vessel and an upper containment dome which backs up the head closure. During power operation, all sodium and cover gas service lines are closed with double isolation valves. Although there are no penetrations in the reactor vessel, the reactor vessel and containment vessel are sized so that if a leak should occur, the core and IHX inlets will always remain covered.

Fuel System

15.57. Prior to about 1970, metallic fuel was thought to be unacceptable because of its poor irradiation behavior. This picture was reversed completely by new alloy designs and very favorable irradiation experience at EBR-II during the 1970s. Metal fuel that can achieve a peak burnup of 12.9 TJ/kg (150,000 MW · d/t) now appears feasible.

15.58. The thermal expansion and high thermal conductivity of metallic fuel provides important safety advantages. For example, in a loss-of-flow-without-scram (LOFWS) event, the exit coolant temperature will increase during flow coast-down. The resulting thermal expansion of the fuel assemblies provides a negative reactivity feedback which reduces the power. Although during power reduction, the stored Doppler reactivity effect appears as a positive contribution, it is not significant because of the high thermal conductivity of the fuel. This behavior was demonstrated dramatically at EBR-II in 1986 when a deliberate LOFWS experiment resulted in automatic reactor shutdown within about 7 minutes, with a maximum coolant temperature rise of about 200°C, which occurred within the first minute [8].

15.59. The reference fuel for the ALMR is an alloy consisting of uranium with 27 percent Pu and 10 percent Zr. Blanket assemblies contain an alloy of depleted uranium with 10 percent zirconium. A heterogeneous core configuration includes 199 assemblies of various types as shown in Fig. 15.6 sodium-cooled fast reactors generally use hexagonal geometry for the assembly ducts instead of the square design employed in LWRs. Since the coolant is not a moderator, the tighter triangular rod pitch can be used to maximize the fuel volume fraction and improve heat transfer characteristics.

15.60. For the equilibrium fuel loading cycle, one-third of the core is replaced after each 18 months of operation. Since the metallic fuel yields a breeding ratio of about 1.2, more fissile material is produced than consumed. In a planned on-site facility, electrorefining using a liquid cadmium anode and a fused chloride salt will be used to extract the discharged fuel uranium–plutonium mixture from the dissolved mixture of fuel and fission products. New fuel rods would then be prepared by a casting process.

Concept Potential

15.61. Reactor behavior, as inferred from EBR-II tests and extensive analyses, indicates an ability to safety accommodate a wide variety of accident scenarios. However, traditional sodium-cooled reactor concerns regarding the positive sodium void reactivity effect and the possibility of fuel failure propagation in the event of fuel assembly blockage may have

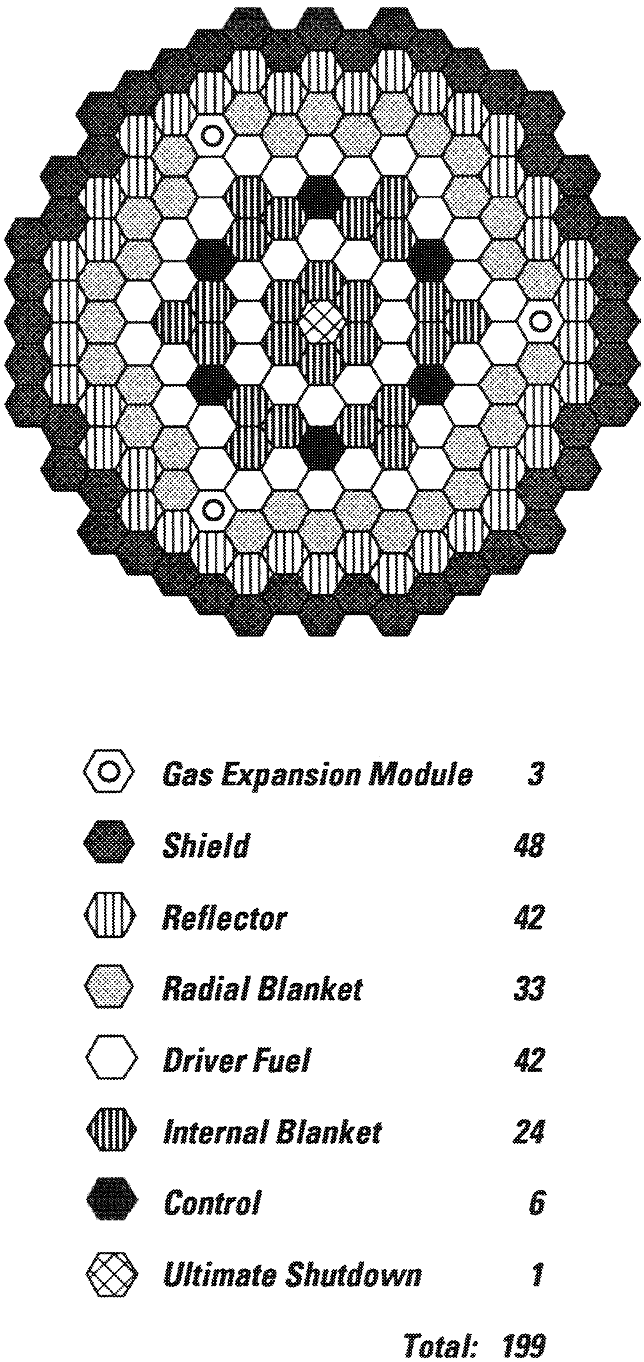


Fig. 15.6. ALMR reference metal core (General Electric Co.).

to be addressed. Also, an additional containment structure enclosing the primary system may be required.

15.62. The advantages of modular design yield cost estimates that are competitive with those for other advanced reactors and fossil-fueled systems after about four full-size power stations are built and operated. Since the need to produce fissile material by breeding appears to be at least several decades away, there is time for a prototype plant test program to test further passive safety features and refine the system design. This could be done by first building a single module or as an alternative, a single commercial power block, if necessary funds are made available.

OTHER PASSIVE SYSTEMS

Introduction

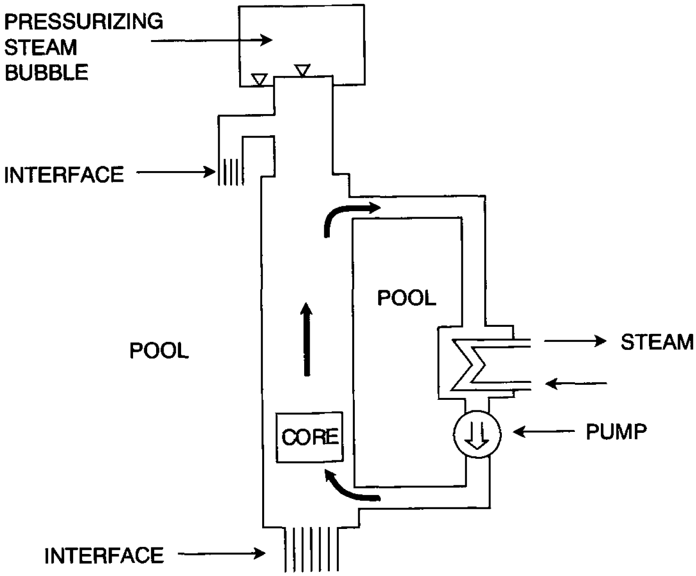
15.63. The four concepts previously described incorporate passive features in basically proven systems that have been converted to a smaller size amenable to modular construction. For the two LWRs, the passive features permit substantial system simplification with accompanying economic benefits. An improvement in public acceptance is also expected.

15.64. Other concepts have been proposed which have an improvement in public acceptance as a major objective. Probably the most interesting of these is the process inherent safety (PIUS) reactor developed in Sweden. Also of interest is the safe integral reactor (SIR), which has evolved from a system originally proposed for maritime use. We will briefly describe each of these.

The PIUS Reactor [9]

15.65. The PIUS system features a very simple PWR design that is claimed to be so inherently safe that it does not require injection by passive systems to prevent a serious accident. A major feature is the submergence of the entire primary system, including the steam generator, in a very large pool of relatively cool, borated water. This pool is inside a large prestressed concrete vessel which serves as both the pressure boundary and containment.

15.66. As shown schematically in Fig. 15.7, during normal operation, primary coolant moves up through the core, through a tall "chimney," and into a hydraulic interface region, from which it is pumped (recirculated) to the steam generator. On leaving, the coolant passes through a downcomer duct to a plenum under the core, below which is another hydraulic interface region.



15.67. The recirculation pumps are operated so that the pressure drops of the water flowing through the core results in a hydraulic balance preventing the pool water from mixing with the coolant water through the lower interface. If the circulating rate drops or the core temperature increases, the pressure balance will be upset and the borated pool water will flow automatically up into the core, shutting down the reactor and cooling it. Decay heat is removed by allowing the pool water to boil. Sufficient water is initially present so that the core will remain covered for about a week.

15.68. Since the design is still being developed, it is premature to provide even preliminary specifications. Several design options are under consideration, including one in which three core modules are submerged in a single pool contained within a prestressed concrete vessel. Questions of cost, licensability, maintainability, and operability require resolution. For example, concern has been expressed regarding the effect on plant availability factor of the time required to restart the plant after inadvertent shutdowns. A prototype plant would be needed to prove the feasibility of the concept.

The Safe Integral Reactor

15.69. In the safe integral reactor (SIR), the entire PWR primary system, including the steam generator, is enclosed within the steel reactor vessel. Thus, we have a unit steam generating package with no possibility of a LOCA. In a sense, the characteristics of a PWR are combined with some BWR technology. Passive safety systems are included, such as safety injection accumulators and blowdown pressure suppression tanks. The reference SIR design has an electrical generating capacity of 320 MW, although an extended SIR of 400 MW(el) is an option. Construction of a lead unit at Winfrith in the United Kingdom has been proposed.

COMMERCIALIZATION ISSUES

Introduction

15.70. Within the nuclear power industry, the term *commercialization* is used somewhat loosely. A power plant is said to be in commercial service when, after a suitable startup period, it furnishes electricity on a regular basis to a utility's distribution grid. On the other hand, a new reactor type reaches commercialized status only after an unsubsidized plant of a size appropriate for a given utility grid operates reliably and economically for sufficient time to convince utilities that another plant should be built with reasonable financial risk. Some feel that the demonstration can be accomplished by a subsidized plant.

15.71. The willingness of a utility to accept the financial risk of a new nuclear plant depends on several considerations in addition to the technical and economic merits of the concept. Clearly, a need must exist for the new capacity. A stable regulatory picture over the lifetime of the plant is essential. This applies to both the NRC and state level. We have discussed the major responsibility of the NRC in the licensing and operation of the plant. State regulators are responsible for a rate structure that will permit a utility to recover the expenses of energy production. Public acceptance and associated political pressures can play a significant role in this picture. However, we will not consider such considerations further here.

15.72. The competitive cost of a fossil-fueled or other type of power plant would also logically be studied by an electric utility planning new generation capacity. A comparison with a nuclear option is likely to be complicated with environmental, financial risk, and regulatory factors relevant (§15.81). Therefore, for our purposes, we will assume that advanced nuclear plants and other types of generating plants will be "competitive," and concentrate on the relative potential of advanced nuclear plant designs.

15.73. Now that we have described the evolutionary and important advanced nuclear plant passive designs, it is helpful to examine them from the viewpoint of their commercialization potential. For example, the evolutionary designs are large while the advanced passive concepts are smaller.

The question of reactor size is one important issue that is relevant to their commercialization.

The Size Issue

15.74. Compared with present large reactors having electrical generating capacities in the 1000-MW range, we have seen that smaller passive reactors in the 600-MW(el) range have a number of attractive features. The new designs are simpler since the passive features eliminate the need for some active safety systems. They generally are more forgiving to operator and system failures and are designed to have longer plant lifetimes. They have generally received favorable public attention since press descriptions have emphasized that the industry has something new to offer in safety.

15.75. The smaller plants can be built faster, an attractive feature that reduces the indirect construction costs. Also, for a utility considering a restart of its nuclear program, a smaller reactor would require less financial exposure than a large plant and might be more commensurate with load growth requirements. Siting for a smaller plant is likely to be easier.

15.76. Opinions differ regarding whether or not the inherent economy of scale advantage of the large plant can be counteracted by the system simplification and manufacturing advantages of the smaller plant. Although cost estimates for the large evolutionary plants yield about the same results as those for the smaller passive plants, the former estimates are based on proven systems and are therefore likely to be more reliable. Furthermore, it has been pointed out that the large fraction of the capital cost required for the nonnuclear part of the plant is subject to the scaling factor, leading to the conclusion that on a unit energy basis, generating costs for the large plant are likely to be less than for the smaller plant. However, until more information becomes available, it appears that a size preference should be based on considerations other than the economy-of-scale issue.

Other Issues

15.77. Although it has been possible to eliminate most of the active safety subsystems in the passive plants, the entire plant remains a very sophisticated complex system. Therefore, plant reliability and its effect on the plant availability factor is an issue. There are differences among the concepts described regarding the amount of design innovation involved. Proponents of the AP600 and the SBWR claim that the technology is sufficiently proven so that a "demonstration plant" is not required. On the other hand, there appears to be agreement that a "lead plant" to gain experience before other plants are constructed would be desirable. Since the MHTGR and the ALMR are more innovative, but are modular, the road to commercialization would clearly start with a "demonstration" module. Also, other designs are being proposed. For example, a hybrid system uses active systems to cope with non-LOCA events, with passive features available as a backup for serious accidents. This approach is claimed to result in a higher plant availability than in an all-passive design, since shutdowns would be shorter [12].

15.78. As a result of experience with the Fort St. Vrain plant, the MHTGR is claimed to be in an advanced state of development. However, as a result of various problems, the operation of the plant was uneconomical. Therefore, the operability of a modular plant with some advanced features still requires some demonstrations to assure utilities that their investment risk will be reasonable. As mentioned in §15.62, there is ample time for an ALMR demonstration, both to permit further development and provide a firmer basis for cost estimation.

THE FUTURE

15.79. In this chapter we have looked forward to the rebirth of nuclear power and described some of the new technology that is being made available to create a favorable climate for new plant construction. However, we cannot predict the future. Therefore, it is well to summarize here some of the nontechnical challenges that matter [10].

15.80. In the United States, a decision to build a new plant is made by an individual utility at the local level to meet the energy needs of its customers. National electricity usage growth forecasts do not necessarily apply at the local level. Nonnuclear options, including the upgrading of older units, are also available. Another consideration is the role of nonutility energy providers in developing generating facilities (§10.108).

15.81. Four topics are interrelated: investment risk, licensing stability, rate regulation, and public opinion. A utility can recover its investment only if initial cost projections are not changed by new licensing requirements during construction and operation. Also, unreasonable limitations on rates charged customers must be prevented. In each of these areas, public opinion can play a role through political pressure on regulators. Thus, acceptance by local government and the local populance is very desirable as part of risk management strategy as well as good citizenship.

15.82. To conclude on a positive note, we should look ahead some years into the next century. Likely global population and economic growth will lead to a substantial increase in the demand for electricity. Solar power and other alternative energy sources may contribute to meet this demand, but only to a minor extent, leaving the burden to be shared by fossil and nuclear fuels. Assuming that unfounded public sensitivity to the risks associated with high-level radioactive waste can finally be resolved and the environmental damage caused by fossil plants is generally realized, a general preference for nuclear power plants is likely to result [13].

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APPENDIX

The International System of Units (SI)*

The SI units are a coherent and consistent set of units that can be used in calculations without the need for conversion factors. For the present purpose two classes of SI units may be distinguished: base units and derived units.

There are seven dimensionally independent base units, but only the following five are used in this book:

*For further information, see "The International System of Units (SI)," National Bureau of Standards Special Publication 330; "Standard for Metric Practice," American Society for Testing and Materials.

Base Units

Quantity	Unit	Symbol
Length	meter (or metre)	m
Mass	kilogram	kg
Time	second	s
Thermodynamic temperature	kelvin*	K
Amount of substance	mole†	mol

*For convenience, *but not for calculations*, temperatures may be expressed on the Celsius (formerly called the centigrade) scale, where t°C is equal to t + 273.15 K. The unit degree Celsius is equal to the unit kelvin.

†The mole is the amount of substance in a system that contains as many elementary entities (atoms or molecules) as there are atoms in 0.012 kg of carbon-12, i.e., the Avogadro number of entities.

Derived units are formed by the combination of base units or the combination of other derived units with base units. The derived units of special interest here, which have special names and symbols, are given below:

Derived Units

Quantity	Name	Symbol	Component Units
Force	newton	N	$kg \cdot m/s^2$
Pressure, stress	pascal	Pa	N/m^2 (or $kg/m \cdot s^2$)
Energy, work, heat	joule	J	$N \cdot m$ (or $kg \cdot m^2/s^2$)
Power	watt	W	J/s (or $kg \cdot m^2/s^3$)
Activity (radionuclide)	becquerel	Bq	1/s
Absorbed dose (radiation)	gray	Gy	J/kg
Dose equivalent (radiation)	sievert	Sv	J/kg

The following units are not SI but are accepted temporarily because they are still used extensively in the specialized literature:

Other Units

Name	Value
barn	10^{-28} m^2
curie	3.7×10^{10} Bq
electron volt	1.602×10^{-19} J
rad	10^{-2} Gy
rem	10^{-2} Sv

Multiples and submultiples of SI units may be expressed by the addition of the following tabulated prefixes. (Note that the factors are in steps of 1000, so that the centimeter is not an acceptable unit.) In computations, quantities must always be stated in terms of the actual base or derived units with appropriate powers of ten.

Recommended Prefixes

Prefix	Factor	Symbol
exa	10^{18}	E
peta	10^{15}	P
tera	10^{12}	T
giga	10^{9}	G
mega	10^{6}	M
kilo	10^{3}	k
milli	10^{-3}	m
micro	10^{-6}	μ
nano	10^{-9}	n
pico	10^{-12}	p
femto	10^{-15}	f
atto	10^{-18}	a

Some Universal Constants in SI Units

Acceleration of gravity, g	9.807 m/s^2
Avogadro number, N_A	6.022×10^{23} atoms/mol
Atomic mass unit, u	1.661×10^{-27} kg
Boltzmann constant, k	1.381×10^{-23} J/K
Gas constant, R	8.314 J/K · mol (or Pa · m^3/K · mol)
Planck constant, h	6.626×10^{-34} J · s

TABLE A.1. Factors For Conversion to SI Units*

Multiply:	By:	To Obtain SI Unit:	Symbol
Length			
foot	3.048 E − 01	meter	m
inch	2.540 E − 02		
Area			
foot^2	9.290 E − 02	meter^2	m^2
inch^2	6.452 E − 04		
barn	1.000 E − 28		
Volume			
centimeter^3	1.000 E − 06	meter^3	m^3
foot^3	2.832 E − 02		
inch^3	1.639 E − 03		
gallon	3.785 E − 03		
liter	1.000 E − 03		
Mass			
atomic mass unit	1.661 E − 27	kilogram	kg
pound	4.536 E − 01		
ton, metric (tonne)	1.000 E + 03		
Density			
gram/cm^3	1.000 E + 03	kilogram/meter^3	kg/m^3
pound/ft^3	1.602 E + 01		
pound/gallon	1.198 E + 02		
Velocity			
foot/s	3.048 E − 01	meter/s	m/s
Acceleration			
foot/s^2	3.048 E − 01	meter/s^2	m/s^2

Force			
dyne	1.000 E − 05	newton	N (kg · m/s^2)
pound-force	4.448 E + 00		
kilogram-force	9.807 E + 00		
Pressure			
atmosphere	1.013 E + 05	pascal	Pa (N/m^2)
bar	1.000 E + 05		
dyne/cm^2	1.000 E − 01		
pound/inch^2	6.895 E + 03		
torr (mm Hg)	1.333 E + 02		
kg-force/mm^2	9.807 E + 06		
Energy			
British thermal unit	1.055 E + 03	joule (watt · s)	J
calorie	4.187 E + 00		
electron volt	1.602 E − 19		
erg	1.000 E − 07		
kilowatt · hour	3.600 E + 06		
foot · pound-force	1.356 E + 00		
Power			
Btu/hr	2.931 E − 01	watt	W
calorie/s	4.187 E + 00		
horsepower	7.460 E + 02		
Specific heat			
Btu/lb$_m$ · °F	4.187 E + 03	joule/kg · kelvin	J/kg · K
cal/g · °C	4.187 E + 03		
Heat flux			
Btu/ft^2 · hr	3.155 E + 00	watt/meter^2	W/m^2
cal/cm^2 · s	4.187 E − 04		

TABLE A.1. (*Continued*)

Multiply:	By:	To Obtain SI Unit:	Symbol
Thermal conductivity			
Btu/ft · hr · °F	1.731 E + 00	watt/meter · kelvin	W/m · K
cal/cm · s · °C	4.187 E + 02		
Heat-transfer coefficient			
But/ft^2 · hr · °F	5.678 E + 00	watt/meter^2 · kelvin	W/m^2 · K
cal/cm^2 · s · °C	4.187 E + 04		
Power density			
Btu/hr · ft^3	1.035 E + 01	watt/meter^3	W/m^3
cal/sec · cm^3	4.187 E + 06		
Fuel burnup			
MW (thermal) days/1000 kg	8.650 E + 07	joule/kg	J/kg
Viscosity, dynamic			
lb$_m$/s · ft	1.488 E + 00	pascal · s	Pa · s (kg/m · s)
poise (g/s · cm)	1.000 E − 01		
Viscosity, kinematic			
stokes (cm^2/sec)	1.000 E − 04	meter^2/s	m^2/s
Radiation			
absorbed dose (rad)	1.000 E − 02	gray	Gy (J/kg)
dose equivalent (rem)	1.000 E − 02	sievert	Sv (J/kg)
radiation exposure (roentgen)	2.580 E − 04	coulomb/kg	C/kg
activity (curie)	3.700 E + 10	becquerel	Bq

*The letter E (for exponent) in a factor followed by plus or minor two-digit number indicates the power of ten by which the number given must be multiplied; thus, E − 03 represents multiplication by 10^{-3} and E + 03 multiplication by 10^3.

TABLE A.2. Cross Sections For The Elements and Certain Molecules (0.0253-eV Neutrons)

Atomic No.	Element or Compound	Atomic or Mol. Wt.	Density kg/m^3 × 10^{-3}	Nuclei per m^3 × 10^{-28}	Microscopic Cross Section, barns (10^{-28} m^2) σ_a	σ_s
1	H	1.008	8.9*	0.0053	0.332	38
	H_2O	18.015	1.000	3.34†	0.664	103
	D_2O	20.028	1.105	3.32†	0.0013	13.6
2	He	4.003	17.8*	0.0026	<0.05	0.76
3	Li	6.939	0.534	4.6	70.7	1.4
4	Be	9.012	1.848	12.36	0.0092	6.14
5	B	10.811	2.35	12.81	759	3.6
6	C	12.011	~1.6	8.03	0.0034	4.75
7	N	14.007	0.0013	0.0053	1.85	10.6
8	O	15.999	0.0014	0.0053	27*	3.76
9	F	18.998	0.0017	0.0053	0.0095	4.0
10	Ne	20.183	0.0009	0.0026	0.038	2.42
11	Na	23.000	0.97	2.54	0.530	3.2
12	Mg	24.312	1.74	4.31	0.063	3.42
13	Al	26.982	2.699	6.02	0.230	1.49
14	Si	28.086	2.33	5.00	0.16	2.2
15	P	30.974	1.82	3.54	0.180	~5
16	S	32.064	2.07	3.89	0.52	0.98
17	Cl	35.453	0.0032	0.0053	33.2	~16
18	Ar	39.948	0.0018	0.0026	0.678	0.64
19	K	39.102	0.862	1.32	2.10	1.5
20	Ca	40.08	1.55	2.33	0.43	~3
21	Sc	44.956	2.5	3.35	26.5	24

TABLE A.2. (*Continued*)

Atomic No.	Element or Compound	Atomic or Mol. Wt.	Density kg/m^3 × 10^{-3}	Nuclei per m^3 × 10^{-28}	Microscopic Cross Section, barns (10^{-28} m^2) σ_a	σ_s
22	Ti	47.90	4.51	5.67	6.1	4.0
23	V	50.942	6.11	7.21	5.04	4.9
24	Cr	51.996	7.19	8.33	3.1	3.8
25	Mn	54.938	7.43	8.15	13.3	2.1
26	Fe	55.847	7.87	8.49	2.55	10.9
27	Co	58.933	8.9	8.99	37.2	6.7
28	Ni	58.71	8.90	9.13	4.43	17.3
29	Cu	63.54	8.96	8.49	3.79	7.9
30	Zn	65.37	7.13	6.57	1.10	4.2
31	Ga	69.72	5.91	5.11	2.9	6.5
32	Ge	72.59	5.36	4.45	2.3	7.5
33	As	74.922	5.73	4.61	4.3	7
34	Se	78.96	4.8	3.67	11.7	9.7
35	Br	79.909	3.12	2.35	6.8	6.1
36	Kr	83.80	0.0037	0.0026	25	7.5
37	Rb	85.47	1.53	1.08	0.37	6.2
38	Sr	87.62	2.54	1.75	1.21	10
39	Y	88.905	5.51	3.73	1.28	7.6
40	Zr	91.22	6.51	4.29	0.185	6.4
41	Nb	92.906	8.57	5.56	1.15	~5
42	Mo	95.94	10.2	6.40	2.65	5.8
43	Tc	99	~11.5	~7.0	19	—
44	Ru	101.07	12.2	7.27	2.56	~6

45	Rh	102.905	12.41	7.26	150	~5
46	Pd	106.4	12.02	6.79	6.9	5.0
47	Ag	107.87	10.50	5.86	63.6	~6
48	Cd	112.40	8.65	4.64	2450	5.6
49	In	114.82	7.31	3.83	194	~2
50	Sn	118.69	~7.0	3.4	0.63	~4
51	Sb	121.75	6.691	3.31	5.4	4.2
52	Te	127.60	6.24	2.95	4.7	~5
53	I	126.904	4.93	2.34	6.2	~4
54	Xe	131.30	0.0059	0.0027	24.5	4.30
55	Cs	132.905	1.873	0.85	29	~20
56	Ba	137.34	3.5	1.54	1.2	~8
57	La	138.91	6.19	2.68	9.0	9.3
58	Ce	140.12	6.78	2.91	0.63	4.7
59	Pr	140.91	6.78	2.90	11.5	3.3
60	Nd	144.24	6.95	2.90	50.5	16
61	Pm	~145	—	—	~60	—
62	Sm	150.35	7.52	3.01	5800	~5
63	Eu	151.96	5.22	2.07	4600	8.0
64	Gd	157.25	7.95	3.05	49000	—
65	Tb	158.93	8.33	3.16	25.5	20
66	Dy	162.50	8.55	3.17	930	100
67	Ho	164.93	8.76	3.20	66.5	9.4
68	Er	167.26	9.16	3.20	162	11.0
69	Tm	168.93	9.35	3.31	103	12
70	Yb	173.04	7.01	2.44	36.6	25
71	Lu	174.97	9.74	3.35	77	8
72	Hf	178.49	13.31	4.49	102	8
73	Ta	180.95	16.6	5.53	21	6.2
74	W	183.85	19.3	6.32	18.5	~5

TABLE A.2. (*Continued*)

Atomic No.	Element or Compound	Atomic or Mol. Wt.	Density kg/m^3 × 10^{-3}	Nuclei per m^3 × 10^{-28}	Microscopic Cross Section, barns (10^{-28} m^2)	
					σ_a	σ_s
75	Re	186.2	21.02	6.80	88	11.3
76	Os	190.2	22.57	7.15	15.3	~11
77	Ir	192.2	22.42	7.03	426	14
78	Pt	195.09	21.45	6.62	10	11.2
79	Au	197.00	19.32	5.91	98.8	~9.3
80	Hg	200.59	13.55	4.07	375	~20
81	Tl	204.37	11.85	3.49	3.4	9.7
82	Pb	207.19	11.35	3.30	0.17	11.4
83	Bi	208.98	9.75	2.81	0.033	~9
84	Po	210.05	9.32	2.67	—	—
85	At	~211	—	—	—	—
86	Rn	~222	0.0097	0.0026	~0.7	—
87	Fr	~223	—	—	—	—
88	Ra	226.03	~5	1.33	11.5	—
89	Ac	227	10.1	2.68	515	—
90	Th	232.04	11.72	3.04	7.4	12.7
91	Pa	231.04	15.37	4.01	210	—
92	U	238.03	19.05	4.82	7.53	8.9
	UO_2	270.03	10.96	2.44†	7.53	~18
93	Np	237.05	20.25	5.0	169	—
94	Pu	239.13	19.84	5.0	1011	7.7

*Value has been multiplied by 10^5.
†Molecules/m^3.

TABLE A.3. Cross Sections for Individual Isotopes of Special Interest (0.0253-eV Neutrons)

Isotope	Cross section (barns)	Reaction
Lithium-6	940	n, α
Lithium-7	0.037	n, γ
Boron-10	3837	n, α
Boron-11	0.0055	n, γ
Oxygen-16	0.00018	n, γ
Oxygen-17	0.235	n, α
Xenon-135	2.65×10^6	n, γ
Samarium-149	41,000	n, γ
Thorium-232	7.40	n, γ
Thorium-233	1500	n, γ
Protactinium-233	41.5	n, γ
Uranium-233	48	n, γ
	531	n, f
Uranium-234	100	n, γ
Uranium-235	99	n, γ
	582	n, f
Uranium-236	5.2	n, γ
Uranium-238	2.70	n, γ
Neptunium-239	~46	n, γ
Plutonium-239	269	n, γ
	742.5	n, f
Plutonium-240	289.5	n, γ
	0.03	n, f
Plutonium-241	368	n, γ
	1009	n, f
Plutonium-242	18.5	n, γ
	0.2	n, f

TABLE A.4. Properties of Moderators at 20°C

Material	Ordinary Water	Heavy Water (99.75% D_2O)	Beryllium	Graphite (Reactor Grade)
Atomic (or molecular) weight	18	20	9	12
Density (kg/m^3 × 10^{-3})	1.00	1.10	1.85	1.70
N (atoms or molecules/m^3) × 10^{-28}	3.34	3.32	12.4	8.55
Thermal (0.0253-eV)				
σ_a (barns)	0.66	0.003	0.0092	0.0034
σ_s (barns)	103	13.6	6.1	4.8
Σ_a (m^{-1})	2.2	0.0085	0.114	0.029
Σ_s (m^{-1})	345	45	76	41
D (m)	0.0016	0.0085	0.0054	0.0086
L (m)	0.0275	1.00	0.21	0.54
Epithermal				
ξ	0.93	0.51	0.207	0.158
σ_s (barns)	42	10.5	6.1	4.8
Σ_s (m^{-1})	140	35	75	41
$\xi\Sigma_s$ (m^{-1})	128	18	16	6.5
$\xi\Sigma_s/\Sigma_a$	62	21,000	126	216
Age to thermal				
τ (m^2)	0.0027	0.013	0.010	0.037

TABLE A.5. Properties of Reactor Coolants

Temperature (K)	Density (kg/m^3)	Specific Heat ($J/kg \cdot K \times 10^{-3}$)	Viscosity ($Pa \cdot s \times 10^5$)	Thermal Conductivity ($W/m \cdot K$)	Prandtl Number
		Gases			
Helium*					
273	0.178	5.20	1.86	0.141	0.68
500	0.0973	5.20	2.80	0.211	0.69
700	0.0703	5.20	3.48	0.278	0.65
900	0.0529	5.20	4.14	0.335	0.64
1100	0.0432	5.20	4.60	0.389	0.61
Steam*					
400	0.554	2.014	1.34	0.0264	1.02
500	0.441	1.985	1.70	0.0357	0.95
600	0.365	2.026	2.07	0.0464	0.90
700	0.314	2.085	2.43	0.0572	0.89
800	0.274	2.152	2.79	0.068	0.88
		Liquids			
Water					
293	1000	4.18	101	0.60	7.0
333	985	4.18	47.1	0.65	3.0
373	961	4.22	28.2	0.68	1.7
473	867	4.51	13.9	0.67	0.94
573	714	5.73	9.6	0.54	1.02
Sodium					
500	897	1.33	43	80	0.0071
600	874	1.29	32	75	0.0056
700	850	1.27	26	70	0.0048
800	826	1.26	23	66	0.0045
900	802	1.25	20	61	0.0041
1000	778	1.26	18	57	0.0040

*At a pressure of 1.013×10^5 Pa (1 atm).

TABLE A.6. Physical Properties of Some Reactor Materials
(Average values for preliminary calculations only; not to be used for design purposes)

Material	Temperature (K)	Density (kg/m^3)	Coefficient of Linear Thermal Expansion* (per K × 10^6)	Specific Heat (J/kg·K × 10^{-3})	Thermal Conductivity (W/m·K)	Ultimate Tensile Strength (MPa)	Yield Strength or Compressive Strength (C) (MPa)	Young's Modulus (10^9 Pa)	Poisson's Ratio
Graphite	300	1700	—	0.71	156	13.8	58 (C)	—	—
(average	500	—	3.6†	1.25	118	15.9	—	9.0	—
nuclear	800	—	5.0†	1.67	73	17.2	—	—	—
grade)	1400	—	6.3†	1.88	36	19.3	—	—	—
	2500	—	8.5†	—	31	27.6	—	—	—
Steel, carbon	300	7860	—	0.50	52	550	340	207	0.28
(A 533-B)	600	—	10.2	0.59	43	530	280	182	—
	750	—	10.4	0.63	38	450	240	172	—
	800	—	10.4	0.67	35	390	200	169	—
Steel,	300	7950	—	0.50	14	520	210	—	—
stainless	500	7860	16.9	0.52	17	420	—	173	0.30
(type 347)	700	7710	17.4	0.55	20	400	150	166	0.31
	800	—	18.5	0.57	22	390	—	157	0.32
Uranium	300	13,630	—	0.15	—	—	372 (C)	214	—
carbide	800	—	—	—	23	—	—	—	—
(UC)	1250	—	10.3	—	23	—	—	—	—
Uranium	300	10,980	—	0.23	8.0	—	960 (C)	183	—
dioxide	500	—	9.0	0.28	6.1	—	—	—	—
	800	—	10.1	0.30	4.1	—	—	165	—
	1100	—	—	0.31	2.6	—	—	—	—
	1400	—	12.8	0.32	2.2	—	—	—	—
	2300	—	—	0.42	2.3	—	—	—	—
Zircaloy-2	300	6560	—	0.28	12.7	490	300	95	0.43
	500	—	—	0.31	15.2	280	170	90	0.38
	600	—	6.5	0.33	16.5	210	117	78	—
	800	—	—	0.35	18.9	—	—	—	—
	1000	—	—	0.37	21.6	—	—	—	—

*Average values from 20°C to indicated temperatures. †Average of longitudinal and transverse properties.

Index

A

G

H

I

K

L

Q

R

T