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# Descriptions of Selected Accidents that Have Occurred at Nuclear Reactor Facilities

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and
Members of the Staff of the
Nuclear Safety Information Center

NUCLEAR SAFETY INFORMATION CENTER NSIC

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# DESCRIPTIONS OF SELECTED ACCIDENTS THAT HAVE OCCURRED AT NUCLEAR REACTOR FACILITIES

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# FOREWORD

This report was prepared at the request of the President's Commission on the Accident at Three Mile Island in order to provide the members of the Commission with some insight into the nature and significance of accidents in nuclear facilities. However, the report thus conceived was recognized to be of interest to a wider audience; therefore, we are pleased to give it the broad distribution afforded by this Oak Ridge National Laboratory—Nuclear Safety Information Center report.

In selecting the accidents that are included in this compilation, we screened all those available in the computerized files of the Nuclear Safety Information Center. While we can state with some certainty that this file includes all accidents that have occurred at commercial nuclear facilities in the United States, we can also state with equal certainty that there must have been accidents in foreign nuclear power plants of which we have no knowledge. In fact, several of the foreign accidents of which we have heard (e.g., the sodium-water explosion in the Russian fast breeder reactor Beloyarsk 3 in 1975 and the release of $CO_2$ from the Czechoslovakian gas-cooled heavy-water-moderated reactor Bohunice 1A in 1976) are known only through sketchy informal accounts. Such accidents cannot be included here because so few details are known to us. On the other hand, this report does include six foreign accidents where the information was documented.

Although H. W. Bertini is principally responsible for the preparation of this document, he was assisted by several members of the staff of the Nuclear Safety Information Center, including J. R. Buchanan, W. R. Casto, Wm. B. Cottrell, R. B. Gallaher, and R. L. Scott, who participated in the development of the selection criteria (for the accidents reported), prepared the draft on a few of the accidents, and reviewed the resulting document. Chapters 1 and 6 were written by Wm. B. Cottrell.

Readers are encouraged to write to the Nuclear Safety Information Center (c/o Wm. B. Cottrell, P.O. Box Y, Oak Ridge, TN 37830) regarding documentation of foreign accidents which should have been included, criticism of the criteria used for selecting those accidents which were included, or for information on the events which were included. With regard to this last category, it is noted that the Nuclear Safety Information Center annually publishes a compilation of all Licensee Event Reports submitted to the Nuclear Regulatory Commission by U.S. commercial nuclear power plants (see the Bibliography).

The presentation of the material in this document is aimed primarily at the educated layperson. The use of acronyms is avoided where practical and, when used, they are spelled out the first time they appear. Following the Introduction is a brief discussion of the fundamental principles of nuclear reactors and a description of some of the reactor systems that are used in the production of electricity in the United States. In this brief presentation we did not attempt to describe all the different types of reactors — much less the special features of each. Although the information on the accidents included herein comes from a variety of sources, we have endeavored to standardize the presentations and to include identification of the facility involved, date of the accident, a brief description of the accident (including any unique circumstances), and a discussion of the accident consequences. In all cases the documentation pertaining to each accident is cited so that interested persons may go to more detailed source material for additional information.

Wm. B. Cottrell, Director
Nuclear Safety Information Center
Oak Ridge National Laboratory

## PREFACE

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Support is also provided by the Division of Reactor Research and Technology of the Department of Energy. NSIC is a focal point for the collection, storage, evaluation, and dissemination of safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. Although the most widely known product of NSIC is the technical progress review *Nuclear Safety*, the Center prepares reports and bibliographies as listed on the inside covers of this document. The Center has also developed a system of keywords to index the information which it catalogs. The title, author, installation, abstract, and keywords for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

1. General Safety Criteria
2. Siting of Nuclear Facilities
3. Transportation and Handling of Radioactive Materials
4. Aerospace Safety (inactive ~1970)
5. Heat Transfer and Thermal Hydraulics
6. Reactor Transients, Kinetics, and Stability
7. Fission Product Release, Transport, and Removal
8. Sources of Energy Release under Accident Conditions
9. Nuclear Instrumentation, Control, and Safety Systems
10. Electrical Power Systems
11. Containment of Nuclear Facilities
12. Plant Safety Features — Reactor
13. Plant Safety Features — Nonreactor
14. Radionuclide Release, Disposal, Treatment, and Management (inactive September 1973)
15. Environmental Surveys, Monitoring, and Radiation Dose Measurements (inactive September 1973)
16. Meteorological Considerations

17. Operational Safety and Experience
18. Design, Construction and Licensing
19. Internal Exposure Effects on Humans Due to Radioactivity in the Environment (inactive September 1973)
20. Effects of Thermal Modifications on Ecological Systems (inactive September 1973)
21. Radiation Effects on Ecological Systems (inactive September 1973)
22. Safeguards of Nuclear Materials

Computer programs have been developed that enable NSIC to (1) operate a program of selective dissemination of information (SDI) to individuals according to their particular profile of interest, (2) make retrospective searches of the stored references, and (3) produce topical indexed bibliographies. In addition, the Center Staff is available for consultation, and the document literature at NSIC offices is available for examination. NSIC reports (i.e., those with the ORNL/NSIC and ORNL/NUREG/NSIC numbers) may be purchased from the National Technical Information Service (see inside front cover). All of the above services are free to NRC and DOE personnel as well as their direct contractors. They are available to all others at a nominal cost as determined by the DOE Cost Recovery Policy. Persons interested in any of the services offered by NSIC should address inquiries to:

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# 1. INTRODUCTION

This report was prepared at the request of the President's Commission on the Accident at Three Mile Island to provide the members of the Commission with some insight into the nature and significance of accidents that have occurred at nuclear reactor facilities in the past. Toward that end, this report presents a brief description of 44 accidents which have occurred throughout the world and which meet at least one of the severity criteria which we established.

The accidents selected for inclusion fulfill at least one of the following conditions: (1) caused death or significant injury; (2) released a significant amount of radioactivity offsite (e.g., many times the maximum permissible concentration for extended periods of time); (3) resulted in core damage (melting and/or disruption), or core damage was suspected although it did not actually occur; (4) resulted in severe damage to major equipment; (5) caused inadvertent criticality; (6) was a precursor to a potentially serious accident; or (7) resulted in significant recovery cost (e.g., greater than half a million dollars).

These criteria are expected to encompass all significant accidents. At the same time it should be noted that they also encompass some accidents which are *not* unique to nuclear reactor facilities. However, for the sake of consistency, all those which meet the established criteria are included. Similarly, there is some subjective judgment involved in evaluating the severity of many accidents. When in doubt, we have chosen to include the accident in the compilation.

As noted above, the accidents selected for inclusion here occurred throughout the world. We believe that our knowledge of U.S. reactor experience is sufficiently comprehensive to ensure that all relevant accidents that have occurred in this country have been considered. However, we are well aware that our knowledge of reactor experience in the rest of the world (and particularly in the Eastern Bloc countries) is very sketchy. Hence, we feel that it is most probable that there have been reactor accidents abroad which meet the criteria given above, but which are not included here because of our lack of information. However, this may not detract significantly from the value of this document — first, because the U.S. experience (with power reactors at least) constitutes approximately half the world experience and, secondly, because the experience outside the United States is derived primarily from other reactor types. Furthermore, these foreign pressurized-water reactors and boiling-water reactors are not built to U.S. criteria and safety standards.

This report encompasses all types of reactor facilities, except critical facilities. Thus, the accidents included in this report involve (1) central station power plants, (2) production reactors, and (3) experimental and research reactors, and they are grouped accordingly. While the principal concern of this document is with accidents that occurred at central station power plant reactors, the experience with other types of reactors is also relevant, although primarily in a generic sense. However, because of the tremendous differences from one type of reactor to another (and sometimes even within a given reactor type), it is generally not possible to extrapolate the accident sequence (in detail) from one reactor type to another. Thus, the experience with critical facilities (the simplest reactor form) is so far removed from what could happen at a central station power plant reactor as to be completely irrelevant. Furthermore, good reviews of accidents in critical facilities already exist.[1,2]

In this report we identify the reactor involved (by type, designer, operator, location, and power level)[3] and then present a brief description of the accident itself, including a brief commentary on the causes and consequences — where such information was available. We have undertaken no investigative work on, nor analytical evaluations of, accident causes or consequences; we simply describe the events that took place and report the conclusions that were reached in the sources that are cited. In reading the accident descriptions, the reader should note that the word "operator" is used rather loosely and may refer to any of the operating personnel at the facility, including, in some instances, instrument mechanics, maintenance personnel, and/or nonnuclear operators.

# 2. NUCLEAR REACTORS: FUNDAMENTALS

## 2.1 Basic Theory

### 2.1.1 Atoms and nuclei

An atom of any element consists of a very small, heavy nucleus surrounded by a cloud of electrons, which are very light negatively charged particles. The dimensions of the electron cloud are much larger than those of the nucleus. If one were to scale a fluorine atom (nine electrons and a nucleus) to dimensions roughly equivalent to those of the solar system (nine planets and a sun), one would reduce the mass of the sun about ten times, reduce its size (diameter) by about one-half, and make the distance between the planets and sun about fifteen times greater. The reactions of concern in a nuclear reactor are only those involved with the nuclei of atoms.

A simple concept of the nucleus is that it is a tightly bound cluster of bits of matter called neutrons and protons. They are about the same size, but the proton has a single charge of positive electricity whereas the neutron has none. If a proton is added to a nucleus, the atom becomes a different chemical element with different chemical properties. If a neutron is added to a nucleus, the atom becomes a different isotope (i.e., an atom of slightly different weight or atomic mass) and acquires different *nuclear* properties, but the element, and hence its chemical properties, remains unchanged. For example, the isotope of uranium whose mass number* is 235 ($^{235}U$) is needed to make a nuclear reactor function, but the isotope of uranium whose mass number is 238 ($^{238}U$) cannot be used for this purpose because of its different nuclear properties. However, both isotopes have the same chemical properties. The same is true of plutonium: the plutonium isotope $^{239}Pu$ is a nuclear "fuel" whereas $^{240}Pu$ is not.

The simplest nucleus is that of hydrogen ($^{1}H$), for it consists of a single proton. By adding a neutron, one gets a different isotope ($^{2}H$), but the atom thus formed has the same chemical properties as that of

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*The mass number is the total number of neutrons and protons in the nucleus.

hydrogen. Unlike any other element, the isotopes of hydrogen have different names. This one, $^2H$, is called deuterium (D). Since deuterium combines with other elements in the same manner as ordinary hydrogen ($^1H$), it can combine with oxygen to form water. In addition, since its mass is twice that of hydrogen, the water formed by deuterium ($D_2O$) is called "heavy" water. It has different *nuclear* properties than that of ordinary, or "light," water.

### 2.1.2 Fission and the nuclear chain reaction

The energy that becomes available in a nuclear reactor is explained by Einstein's famous formula, $E = mc^2$, where $E$ is the total energy of the matter, $m$ is its mass (or weight), and $c$ is the velocity of light. The interpretation is that matter and energy are equivalent; i.e., if a certain amount of matter is made to disappear, an equivalent amount of energy will appear. The reverse is also true: if energy is made to disappear, then matter will appear.

The fission process that takes place in nuclear reactors is based on this principle. In this process a neutron is "captured" by the nucleus of a $^{235}U$ atom; that is, a neutron strikes and penetrates the nucleus, thus forming $^{236}U$. However, this new nucleus, when formed in this way, is highly unstable; it breaks apart (fissions) almost instantaneously into two fragments plus a few free neutrons. If one were to determine the weight of the debris (the two fragments plus the free neutrons) after the fission and compare this weight with that of the $^{236}U$ atom before the fission, one would find that matter had disappeared; that is, the debris would weigh less than the original atom of $^{236}U$. Since matter has disappeared, then, according to Einstein's equation, an equivalent amount of energy must have been created.

When fission occurs, the two fragments and the free neutrons move apart with considerable speed, propelled by the energy created from the disappearance of matter. The bulk of this energy is transferred to neighboring atoms when these atoms are struck by the fragments. The released neutrons also transfer most of their energy to the atoms of the surrounding medium by means of scattering collisions with them. The atoms that have been struck, either by the fission fragments or by the neutrons, recoil and vibrate and thus the medium is heated. Hence, energy in the form of heat is created by the disappearance of matter, which occurred in the fission process.

All fissions are *not* exactly alike; that is, the fragments formed in one fission are somewhat different from those formed in another, and the number of neutrons emitted in one fission may be different from the number emitted in another.

Recall that the fission described above was caused by one neutron and that a few neutrons were emitted when the $^{235}U$ atom fissioned. If enough $^{235}U$ is present, and if other material with the necessary properties is also present, then at least one neutron released by one fission will cause another fission, and the process is repeated continually. The reaction is thus self-sustaining because each neutron that is "captured" in causing a fission is replaced by other neutrons, some of which cause other fissions, etc. Since each fission generates heat, it appears that a continuing source of heat has been devised. This would be true, except that it takes a certain amount of $^{235}U$ "fuel" to support a self-sustaining chain reaction, and some of the fuel is destroyed, or consumed, in the fission process. The control of the process is described below.

The extra neutrons that are released by each fission (i.e., those neutrons that do *not* cause additional fissions) either escape from the reactor or are captured by nonfissionable nuclei. It is important that neutrons be husbanded so that there will be a sufficient number to sustain the chain reaction.

### 2.1.3 Criticality in a nuclear reactor

When the materials and their configuration (to be described below) in a nuclear reactor are just right, the fissioning process becomes self-sustaining; when this happens, the reactor is said to be "critical." The terminology is unfortunate because it implies that a condition of crisis exists, which is not true, as will be explained later. Those conditions which constitute a crisis will also be explained. More detailed background information must be presented before these explanations can be made.

Not all the neutrons that are released in a fission reaction are emitted instantaneously. A small percentage (approximately 0.73% in $^{235}U$ fission) are emitted later — about 0.1 sec later on the average. The neutrons emitted instantaneously are called "prompt" neutrons, and those emitted a short time later are called "delayed" neutrons. Both the prompt and delayed neutrons help to initiate and sustain the chain reaction occurring within the reactor.

The term "subcritical" is used to describe the reactor configuration when it is less than self-sustaining. The term "supercritical" is used to describe the configuration when the number of fissions is increasing over a period of time rather than remaining constant over time, as when the reactor is simply critical. This occurs when *more than one* of the neutrons that are emitted in each fission cause more fissions. For example, when the reactor is simply critical, a single neutron out of the two or three that are released in a single fission causes another fission, and a short time later one of the neutrons released in that fission causes another fission, and a short time after that one of the newly released neutrons causes still another fission, etc.; thus, the fission *rate* is constant over time and so is the neutron population in the reactor. When the reactor is supercritical and, say, two of the neutrons from each fission cause two other fissions, then the first fission would be followed by two fissions, and a short time later by four fissions, then eight fissions, etc.; thus, the rate at which fissions are taking place, as well as the neutron population, would be increasing with time.

It should be pointed out that the reactor can be critical with *any number* of neutrons present so long as the number causing fission remains constant over time. For example, if the reactor is critical and ten neutrons are causing fissions at a particular time, then at any time later there will still be only ten neutrons (different from the original ones) causing fissions. And this *number* will remain the same as long as the reactor remains critical. An enormous number of neutrons participate in the fissioning process, even at low power. For example, if only 10 million neutrons were causing fissions in a reactor which was critical, the heat generated would be so small that it would be difficult if not impossible to measure, even with the most sensitive instruments. When a reactor is operating at full power, there are about $10^{17}$ (one hundred billion million) neutrons in the core at any instant.

One more term must be defined before proceeding to the subsequent chapters, and this is "reactivity." The numerical value associated with the reactivity is a measure of the criticality. The criticality is a loose term which broadly defines the general nuclear condition of the reactor. The reactivity is a more precise measure of the criticality. For example, the reactivity is taken to be zero when the reactor is critical. If the reactivity is +0.00001, the reactor is barely supercritical; if it is +0.001, the reactor is more supercritical. If it is -0.00001, the reactor is barely subcritical; if the reactivity is -0.1, the reactor is highly subcritical.

If the reactivity is greater than +0.0073, the reactor is said to be prompt critical, and the rate of fissions will increase at a very rapid rate. Under these conditions, the chain reaction is more than self-sustaining by the prompt neutrons alone, without the need for the delayed neutrons. This situation in a nuclear reactor would probably lead to a damaged core because the power would increase so fast that it would be difficult to control. This is a crisis situation.

Although it is somewhat perplexing at first, a reactor can be critical and yet be at any desired power level. There is a very crude analogy with an automobile. A car can be driven at constant speed at 5 miles per hour, and it can be driven at constant speed at 55 miles per hour. When a car is driven at a constant speed of 5 miles per hour, the gas pedal is kept in one position. In order to increase the speed to 55 miles per hour, the gas pedal is depressed until that speed is reached, and then the pedal is kept in the same position to maintain the speed. A reactor operates in almost the same way. It can be critical at low power (reactivity equal to zero), and to get to a higher power, the reactor is made supercritical (reactivity greater than zero), at which time the power increases to the desired level. The reactivity is then brought to zero again (critical configuration), and the reactor remains at the higher power. The reactivity is made negative (subcritical configuration) to decrease the power, as one would raise the gas pedal to reduce the speed of a car.

Table 2.1 summarizes the above comments. Note that reactivity equal to zero is equivalent to the gas pedal being held at a constant position.

## 2.2 The Components of a Nuclear Reactor

The main component of a nuclear reactor is the "core." It is surrounded by a thick (8- to 10-in.) steel vessel called the pressure vessel, whose thickness is determined by the operating pressures of the system.

The core of a reactor is the region where nuclear fission takes place and consequently where the heat is generated. It consists of three major components: the fuel, the coolant, and the moderator. A fourth component, the reflector or blanket, is sometimes used. A reactor can be generally characterized by specifying these components.

The fuel used in power reactors in the United States is an oxide of uranium, $UO_2$, which is a tough ceramic that melts at a very high temperature [2865°C (5189°F)]. The fuel is "enriched" — that is, the amount of the fissionable isotope $^{235}U$ in the uranium is increased over that which is normally present in uranium ore. The percentage of $^{235}U$ in uranium as it is found in nature (natural uranium) is 0.72%, whereas the uranium used in power reactors is enriched to 2 to 3%. (Note that this enrichment is considerably less than that required in the uranium used in an atomic bomb.) The $UO_2$ fuel is surrounded by a thin metal sheath called the cladding. The purpose of the cladding is to protect the $UO_2$ and to prevent the escape of radioactive fission products (to be described below). The cladding is made of an alloy composed mainly of zirconium. Other metal, such as stainless steel or aluminum, has been used in reactors other than power reactors in the United States. A load of fuel in the core will last anywhere from 1 to 3 years before the supply of $^{235}U$ is sufficiently depleted to require replenishment.

Table 2.1. Nuclear reactor vs automobile

| | Constant low power or constant low speed | Increasing power or speed | Constant high power or constant high speed | Fast increase in power |
|---|---|---|---|---|
| Reactor | Critical (reactivity = 0) | Supercritical (reactivity greater than 0 but less than 0.0073) | Critical (reactivity = 0) | Prompt critical (reactivity greater than 0.0073) |
| Automobile | Gas pedal at constant position | Gas pedal being depressed | Gas pedal at constant position | Passing gear |

The coolant removes the heat that is generated in the core. In power reactors in the United States, light water is used as the coolant. Gases such as helium (He) or carbon dioxide ($CO_2$) are used in other reactors, and liquid sodium is used in fast breeder reactors.

The function of the moderator is to reduce the speed of the neutrons that are released in the fission process. Uranium-235 has a greater propensity to capture neutrons, and consequently to fission, when the speed of the neutrons is reduced. When they are released, the neutrons move at very high speed. Their speed is reduced to the most efficient levels for fissioning when they scatter off the nuclei of the moderator and slow down. In power reactors in the United States light water serves the dual function of coolant and moderator. Other types of reactors use heavy water or graphite as the moderator.

The material that surrounds the core is called the blanket or reflector. When this material is used as a reflector, its main purpose is to scatter the neutrons that might otherwise escape, deflecting them back into the core. Beryllium has frequently been used as a reflector material in experimental reactors, but not in power reactors. When the material surrounding the core is used as a blanket, the prime purpose is to permit the transmutation of the blanket material to a fissionable isotope. For example, natural uranium, which is composed almost entirely (99.27%) of the isotope $^{238}U$, is being used as a blanket in some U.S. power reactors. When $^{238}U$ captures a neutron, the fissionable isotope $^{239}Pu$ is formed. Thus, the blanket not only helps to prevent the escape of neutrons, but also serves as a "breeding ground" for the fissionable isotope $^{239}Pu$, which contributes to the power of the reactor when it, in turn, fissions by neutron capture.

Control of criticality or reactivity is achieved by long rods made of material that readily absorbs neutrons. These rods are called control rods (or poison rods), and they are made of boron carbide powder or mixtures of silver, indium, and cadmium. Each one is encased in a stainless steel sheath. They are interspersed throughout the core, and each is equipped with a drive mechanism so that the rods can be inserted to various depths within the core. In some reactors, namely pressurized-water reactors (to be described below), boric acid is added to the coolant to enhance the control because boron is a good neutron absorber. When the control rods are withdrawn from the core, the reactivity increases; this movement of the rods is referred to as the insertion of reactivity.

When the control rods are fully inserted, the reactor is substantially subcritical (reactivity less than zero). As the control rods are withdrawn, reactivity increases until the reactor is critical (reactivity = 0); this is the critical "configuration" referred to above. For reasons of efficiency, power reactors are designed to be critical at full power when all the control rods are almost completely withdrawn or completely withdrawn, depending on the reactor type. In these positions, further withdrawal will add little or no reactivity. The immediate and complete insertion of all the rods is sometimes called a "scram," and sometimes a reactor "trip," which, in the parlance of power plant engineers and operators, means to switch something off. A scram can be initiated automatically or it can be initiated by the reactor operator by pushing the scram button on the control console.

The mechanisms that move the control rods, called the control rod drives, are located on top of or underneath the pressure vessel in U.S. power reactors. The term "reactor" refers to the pressure vessel, the control rod drives, and the core, which includes the control rods.

## 2.3 Radioactivity

The source of radioactivity is an unstable nucleus. Such a nucleus will tend toward stability by transmutation (radioactive decay) to a nucleus that is stable. Many transmutations in succession may be required before a stable nucleus is obtained, and this entire sequence of decays is called a decay "chain." During each transmutation, or radioactive decay, a particle is given off from the nucleus, and a gamma ray usually accompanies this particle emission. The particles that are emitted are either electrons (beta particles), the nuclei of helium atoms (alpha particles), or neutrons. The type of particles given off depends on the type of nucleus that is decaying. Some give off betas, some alphas, and a few give off neutrons. The alpha particle is the least penetrating for it can be stopped by the skin; the electrons are more penetrating, followed by the neutrons and gamma rays. The gamma rays are high-energy x rays; both are electromagnetic radiation, as are radio waves. The nucleus contains no electrons; those that are emitted in radioactive decay come from the transmutation of a neutron to a proton within the nucleus.

A measure of the rate of decay of the radioactive nuclei is the "half-life." This is the time it takes for half of all of the nuclei that are present at any instant to decay. If the half-life is short, the level of radioactivity will be reduced quickly because most of the radioactive nuclei will be transformed to nuclei of other elements in a short time. If it is long, the radioactivity will remain for a correspondingly longer time.

The fragments produced by the fissioning of a fuel nucleus (called either fission fragments or fission products) consist of clusters of neutrons and protons and are, in fact, the nuclei of other elements. They are usually highly unstable when they are created by the fission process and hence are radioactive. A large variety of fission fragments are formed during the fission processes, and each has a different half-life.

The energy carried by the emitted particles and the gamma rays during the radioactive decay of the fission products constitutes about 6 1/2% of all the energy generated from each fission. These emitted particles are almost entirely absorbed within the core of the reactor, and hence their energy is transferred to the core where it contributes to the total heat that is generated by the reactor.

When a reactor is operating at power, fission products are being created continuously, but they are also decaying continuously. When the reactor is shut down, the fission process stops and thus the creation of fission products stops, but the decay of the fission products already created continues. Immediately after shutdown, these fission products, by their decay, are still producing about 6 1/2% of the power at which the reactor was operating. However, since no new fission products are being created during shutdown, the heat generated by the fission products already created is gradually reduced as they decay. The heat produced by the fission products after the reactor is shut down is called "decay heat." It is substantial, and steps must always be taken to ensure that this heat is removed after the reactor is shut down.

Of the particles that are given off during the radioactive decay of the fission products (alpha and beta particles, neutrons, and gamma rays), only the neutrons will cause other nuclei to become radioactive. The half-life of those fission products that do lead to the emission of neutrons is so short that they are reduced to negligible amounts in a few minutes; hence, they are of little concern. One can then say, in general, that the radioactivity of any substance that might come from a reactor will not cause nearby materials to become radioactive themselves.* Radioactive substances can "contaminate" other materials by clinging to them (e.g., as a deposit of dust, a water layer, etc.), but a nonradioactive material will remain nonradioactive even if it is immersed in radioactive material.

However, most of the material in the core of a reactor will become radioactive because the core contains an enormous number of neutrons when the reactor is operating at power. The cooling water that passes through the core becomes radioactive. Radioactive tritium is formed by neutron absorption in the small amounts of deuterium in the water, and radioactive nitrogen-16 ($^{16}N$) is formed by neutron absorption in oxygen. Also, corrosion products of the metal piping, which are produced in small quantities, become radioactive as they are transported through the core by the water. In addition, traces of fuel particles ($UO_2$), called "tramp" uranium, are found on the outside of the cladding. These traces come from the manufacturing process. Most of the fission products that

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*There is a minor exception to this statement, and it applies only when hydrogen or compounds of hydrogen are exposed to radiation. Some of the gamma rays emitted from fission products have sufficient energy to jar a neutron loose from the deuterium found in natural hydrogen. These neutrons, when absorbed, can cause a substance to become radioactive. But because the fraction of deuterium in natural hydrogen is so small (0.015%) and the probability that the reaction will occur is so small, the neutrons produced in this way represent an insignificant factor (neutron flux $\sim 10^{-7}$ or $10^{-8}$ of gamma flux) in making other materials radioactive, particularly since hydrogen is not even present in many materials.

are formed in this tramp fuel ordinarily remain imbedded in the fuel, but those that escape go directly into the water, since there is nothing to prevent them from doing so. The vast majority of fission products are formed in the fuel that is inside the cladding, and most are prevented from entering the water by the cladding. However, some of the fission products migrate through the cladding, and some of them escape through small defects in the cladding. (A certain number of defects are allowed by the Nuclear Regulatory Commission.) When all of these contributions to the radioactivity of the water are summed, the total level of radioactivity is still much less than that of the fuel, although it is sufficiently high to be of concern.

In the event of a severe breach or melting of the cladding, some of the radioactive fission products can escape into the cooling water. The water will then become highly contaminated. Many of the technical specifications that set limits on reactor operation are formulated to prevent the cladding and the fuel from melting. Molten fuel can slump against the cladding and interact with it, causing a breach. It was the escape of fission products from melted fuel into the cooling water that was the source of the high levels of radioactivity in the water in the accidents involving melted fuel that are described in the following chapter.

This section is concluded with a few definitions that are pertinent to the measurement of radioactivity.

Curie (Ci): The curie is the unit used in measuring the "activity" of a radioactive source, i.e., the number of disintegrations (or radioactive decays) occurring per second, where 1 Ci = $3.7 \times 10^{10}$ dis/sec. It approximately represents the number of disintegrations per second in 1 gram of radium. A millicurie (mCi) is one-thousandth of a curie.

Roentgen (R): The roentgen (R) is the unit used in measuring the ionization* capability in air (or, equivalently, the potential for depositing energy in air) of x rays or gamma rays. Biological damage in

*In this context, ionization is the process of knocking off one or more electrons from atoms or molecules, thereby creating ions. High temperatures, electrical discharges, or nuclear radiation can cause ionization.

tissue is related to the degree to which x rays or gamma rays will ionize air, or deposit energy in air. Measuring instruments determine the radiation level in roentgens per hour (R/hr) or milliroentgens per hour (mR/hr). The dose of radiation that one can expect is determined by multiplying the radiation level in an area by the time spent in that area. For example, if a person spends one-quarter of an hour in an area where the radiation level is 4 mR/hr, his/her total dose would be 1/4 hr x 4 mR/hr = 1 mR.

Rad (radiation absorbed dose): The rad is the unit used in measuring the degree to which energy from any kind of radioactive source is absorbed in any material. This unit is not associated with the roentgen. For a given level of radiation consisting of x rays and gamma rays, the dose measured in rads or in roentgens is about the same.

Rem (roentgen equivalent man): The rem is the unit used in measuring both the energy deposited in any material and the potential for biological damage. The rem takes account of the fact that the various kinds of radiation, (i.e., x rays, gamma rays, beta particles, alpha particles, and neutrons) damage tissue and body organs in different ways. For x rays and gamma rays, the dose received by soft tissue will be about the same if the level of radiation given in roentgens is the same as that given in rems.

The average person is exposed to about 200 mrems over a period of a year from cosmic rays, medical x rays, x rays from television, etc. A dose of 600 rems will kill most people.

If a person spent 10 hr in an area where the radiation level was 20 mrems/hr, he/she would receive the same dose in that 10 hr that the average person receives in 1 year. Thus, a person can spend a short time in an area where the radiation level is high and still get only a small dose. This should help explain the urgency expressed in the following chapter regarding the necessity for spending only short times in areas where the radiation levels are high.

## 2.4 Electric Power Plants

An electric power plant utilizes some form of fuel to generate heat and subsequently converts the heat to electricity. The electricity is distributed via power transmission lines and sold to customers.

A simplified schematic diagram of a typical electric power plant is shown in Fig. 2.1. In a conventional plant, coal or oil is burned and the heat generated turns the water in the boiler to steam. The steam passes through pipes to the turbine. A large shaft connects the turbine to the generator. The steam causes the turbine and the shaft to spin, and the spinning shaft in conjunction with the other components of the generator results in the production, or generation, of electricity. In other words, the heat energy of the steam is converted to mechanical energy in the turbine, and the generator then converts the mechanical energy into electrical energy, or electricity.[1]

The steam passes from the turbine to a condenser where the steam is condensed to water, which is then pumped back to the boiler. The condenser extracts heat from the steam by passing cool water through pipes over which the steam flows and condenses. The cool condenser water is thus heated. This heat is removed by passing the heated water through large cooling towers or by transporting it to holding ponds where it is air cooled.

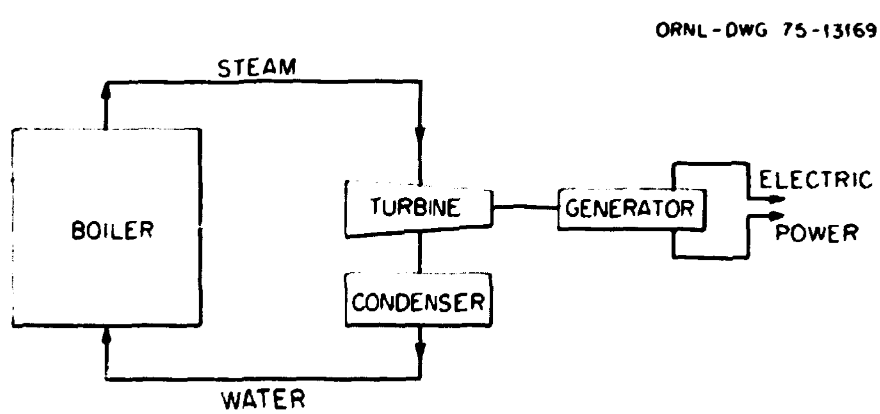


Fig. 2.1. Essential components of an electric power plant.

In a nuclear power plant, the nuclear reactor system simply replaces the boiler shown in Fig. 2.1. A vendor of reactors supplies all the equipment necessary to produce the steam that goes to the turbine. This equipment is called the nuclear steam supply system. The electric utility that owns and manages the power plant purchases the other equipment (i.e., the turbines, generators, condensers, etc.) from other sources. The present cost of the nuclear steam supply system is about $100 million out of a total plant cost of $1200 to $1400 million for a large [1000-MW(e)] plant.

The size of the plant is gauged by the electric power it produces, which is measured in megawatts (MW). One megawatt equals one million watts. The electric power produced by the plant is measured in megawatts of electric power [MW(e)], whereas the heat, or thermal power, that is generated by the source of heat within the plant to produce the electricity is measured in megawatts of thermal power [MW(t)]. Only about one-third of the thermal power that is produced by the heat source can be converted to electric power; thus, the megawatts of thermal power produced by a plant is about three times the megawatts of electrical power. In other words, power plants have an efficiency of about 30%.

A 1000-MW(e) plant is considered large; it will supply the electrical needs for a city with a population of about 600,000.

## 2.5 Classification of Reactors

Reactors are broadly classified according to the purpose for which they were built. However, various types of reactors can be used to satisfy the same purpose. Descriptions of the types of reactors that are included in this report are given below.

### 2.5.1 Reactors for central station electric power plants

A reactor that is used for the production of electricity falls into this classification. Light-water reactors, heavy-water reactors, liquid-metal fast breeder reactors, and gas-cooled reactors are all being used throughout the world in central station electric power plants.

Light-water reactors (i.e., reactors that are both cooled and moderated by light water) are the primary source of nuclear electricity in the United States and in many other countries of the world,[2] including Austria (1),* Belgium (7), Brazil (3), Bulgaria (4), Czechoslovakia (4), Finland (4), France (40), German Democratic Republic (7), Federal Republic of Germany (26), Hungary ([illegible]), Iran (4), Italy (7), Japan (25), Korea (4), Luxembourg (1), Mexico (2), Netherlands (2), Phillipines (2), Poland (1), South Africa (2), Spain (16), Sweden (12), Switzerland (7), Taiwan (6), and Yugoslavia (1). The United States has 71 light-water reactors in operation and 124 in various phases of construction.

Although the U.S.S.R. has 12 light-water reactors, light-water-cooled graphite-moderated reactors, of which it has 21, are the main source of nuclear electricity in that country.

Heavy-water reactors are the primary source of nuclear electricity in Argentina (2), Canada (23), India (6), and Pakistan (1).

The United Kingdom has 37 gas-cooled reactors, which is the type they find most favorable.

Several countries find the liquid-metal fast breeder reactor (LMFBR) sufficiently promising to pursue on a large scale. France, the Federal Republic of Germany, Japan, the United Kingdom, and the U.S.S.R. have LMFBRs in operation or at various stages of construction.

Since the main emphasis of this report is on the reactors used in central station electric power plants in the United States, a more detailed description of these reactors, namely the light-water reactors, is given in Sect. 2.6. Descriptions of the other types of reactors used in central station power plants can be found in the literature listed in the bibliography.

### 2.5.2 Production reactors

Production reactors are used to produce the fissionable isotope $^{239}Pu$. It is produced by the absorption of a neutron in the nucleus of

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*The figures in parentheses indicate the number of light-water reactors in operation and in various phases of construction.

an atom of $^{238}U$. There are a variety of these reactors. They are moderated by graphite or heavy water and cooled by gas or light water. The fuel used is usually natural uranium. Detailed descriptions of these reactors are not generally available because they are classified.

### 2.5.3 Experimental and research reactors

Experimental and research reactors are grouped together in this report because they are small [less than about 30 MW(t)] and experimental in nature. However, they are different from each other.

A research reactor is designed for the purpose of conducting scientific research, mainly that involving the interaction of neutrons with the nuclei of matter. They are also used at universities as an experimental tool for instruction in nuclear engineering. They are generally cooled and moderated by light water, and the fuel cladding is usually aluminum.

An experimental reactor, sometimes called a proof-of-principle reactor, is the first step in the development of a full-scale central station electric power reactor of a particular concept. It is built primarily to determine whether the concept actually works or not, and if it does, to determine some of its characteristics. Since a variety of reactor concepts have been formulated, there are various kinds of experimental reactors.

## 2.6 Light-Water Reactors for the Production of Electricity

There are two types of reactors that are used for the central station generation of electricity in the United States. Both are light-water reactors; one is the pressurized-water reactor (PWR), and the other is the boiling-water reactor (BWR). Both are described in more detail below.

### 2.6.1 Pressurized-water reactors (PWRs)

The pressurized-water reactor is so called because the cooling water that circulates through the reactor is under high pressure (about

2250 lb/in.$^2$ or about 150 times the normal atmospheric pressure). A simplified schematic diagram of a PWR in conjunction with an electric power plant is shown in Fig. 2.2. The pipes and other equipment that handle the water that flows through the reactor constitute the primary system. The pipes and other equipment that handle the steam that goes to the turbine and also the condensed water that returns constitute the secondary system.

In a PWR, the primary system water passes through the core of the reactor where it is heated; then it is pumped through the steam generator and returned to the core. The heat that is picked up by the water of the primary system while it is in the core is transferred to the water of the secondary system in the steam generator. This transfer of heat

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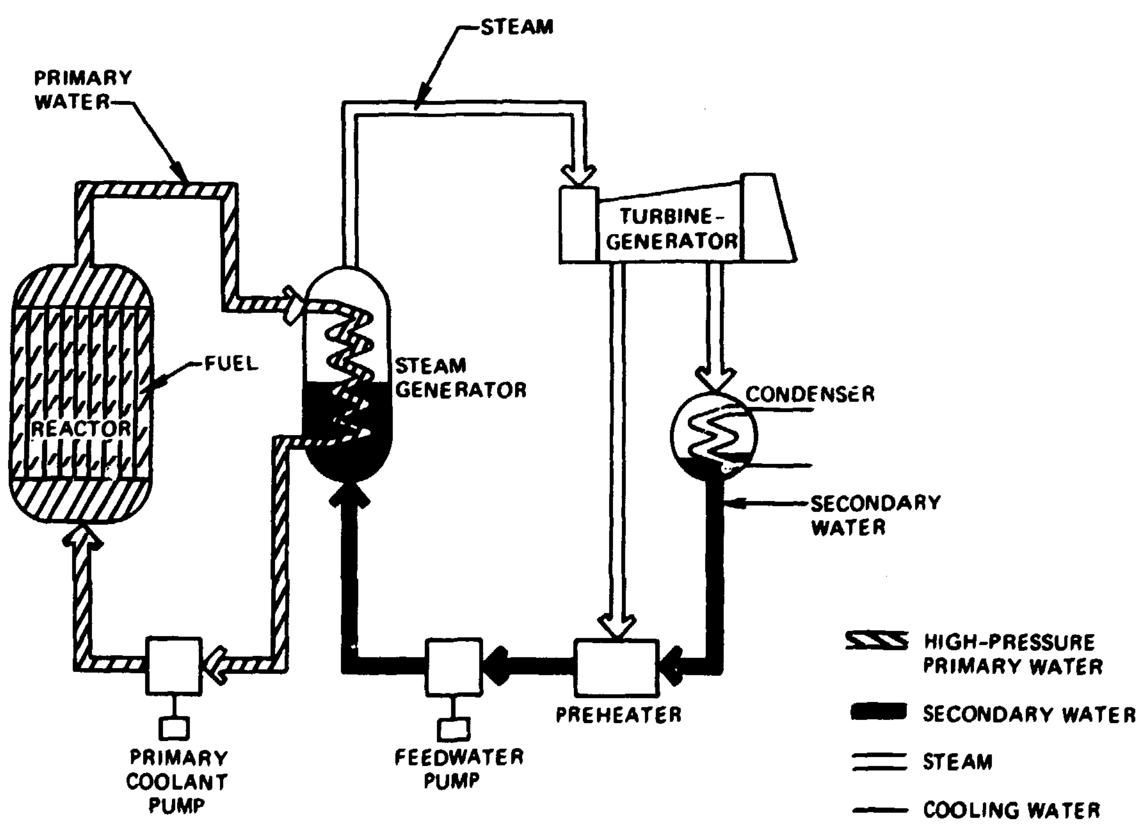


Fig. 2.2. The essential elements of a pressurized-water reactor system.

turns the secondary system water into steam and cools the primary system water.

Figure 2.3 is an illustration of a typical pressurized-water reactor. Primary water enters at the side of the reactor pressure vessel through the inlet nozzle and flows downward near the inside surface of the pressure vessel to the bottom of the vessel. Then it turns upward, passes through the core where it picks up heat, exits the vessel through the outlet nozzle at the other side, and flows to the steam generator (piping to steam generator not shown).

Figure 2.4 is an illustration of a typical steam generator. The hot water coming from the reactor enters at the bottom of the steam generator, passes through thousands of small tubes, exits from the bottom as cooler water, and is pumped back to the reactor. The tubes keep the water from the primary system separate from that of the secondary system. Water of the secondary system, which comes from the condenser of the turbine-generator, passes into the steam generator through the feedwater inlet and flows around the hot tubes where it is turned into steam. The steam flows out of the top of the steam generator and goes to the turbine-generator.

Figure 2.5 is an illustration of the nuclear steam supply system of a PWR. The system is very large — the main coolant pumps, for example, are about three to four stories high.

The function of the pressurizer is to maintain the pressure in the primary system. The pressurizer is connected directly to the primary system by a pipe. Figure 2.6 is an illustration of a pressurizer. The bottom half of the pressurizer is filled with water and the top with steam, which is under pressure and acts as a cushion for minor water or pressure surges. The pressure of the steam is transmitted to the water at the bottom of the pressurizer and, in turn, to the water of the primary system via the connecting pipe. If the pressure of the system gets too low, heaters at the bottom of the pressurizer turn on and boil some of the water; the steam generated is added to that at the top, which increases the pressure. If the pressure gets too high, cool water is sprayed through the steam; this condenses some of the steam and reduces the pressure. Nozzles (short nipple-shaped extensions formed

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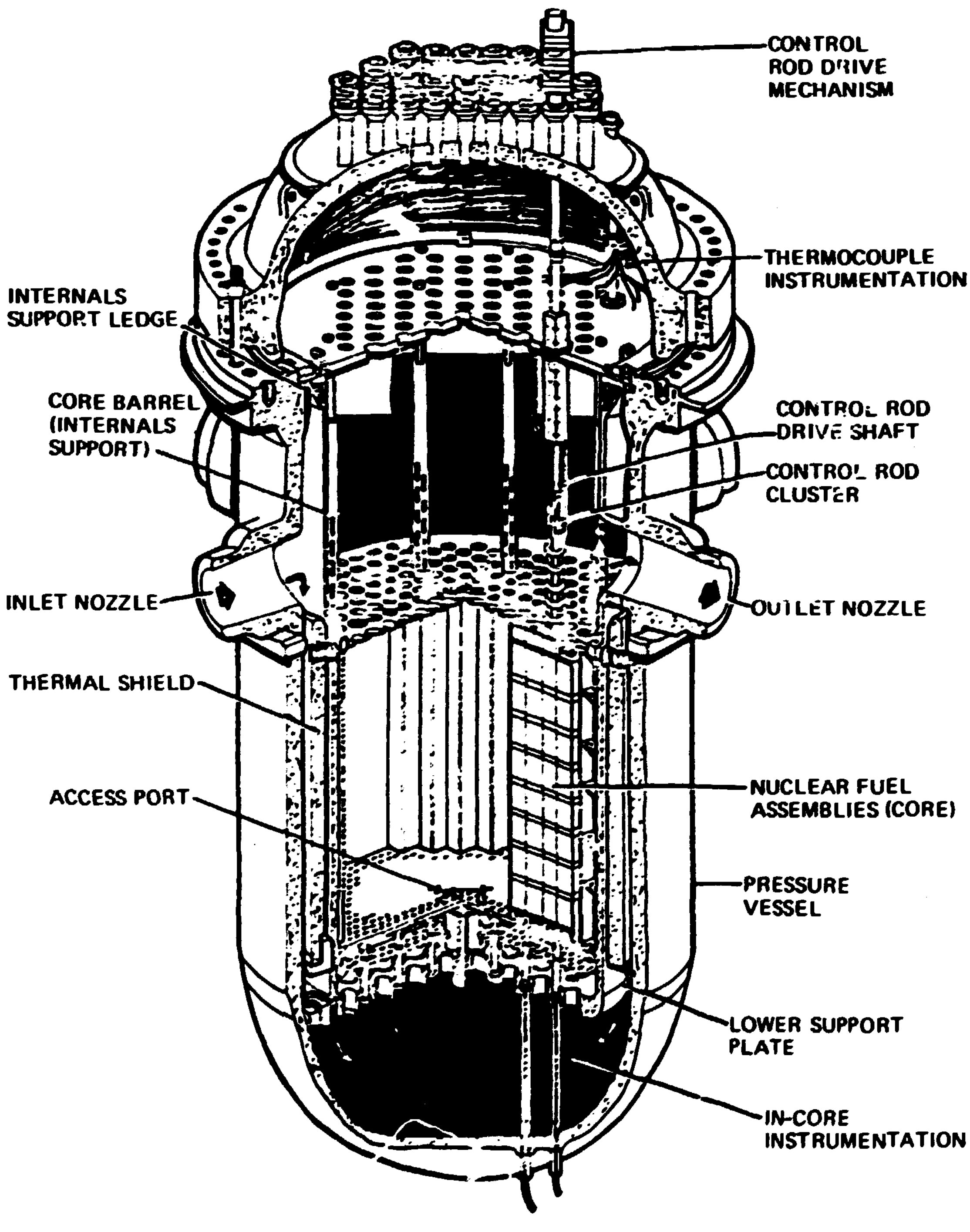


Fig. 2.3. Cutaway of pressure vessel and internals of a pressurized-water reactor.

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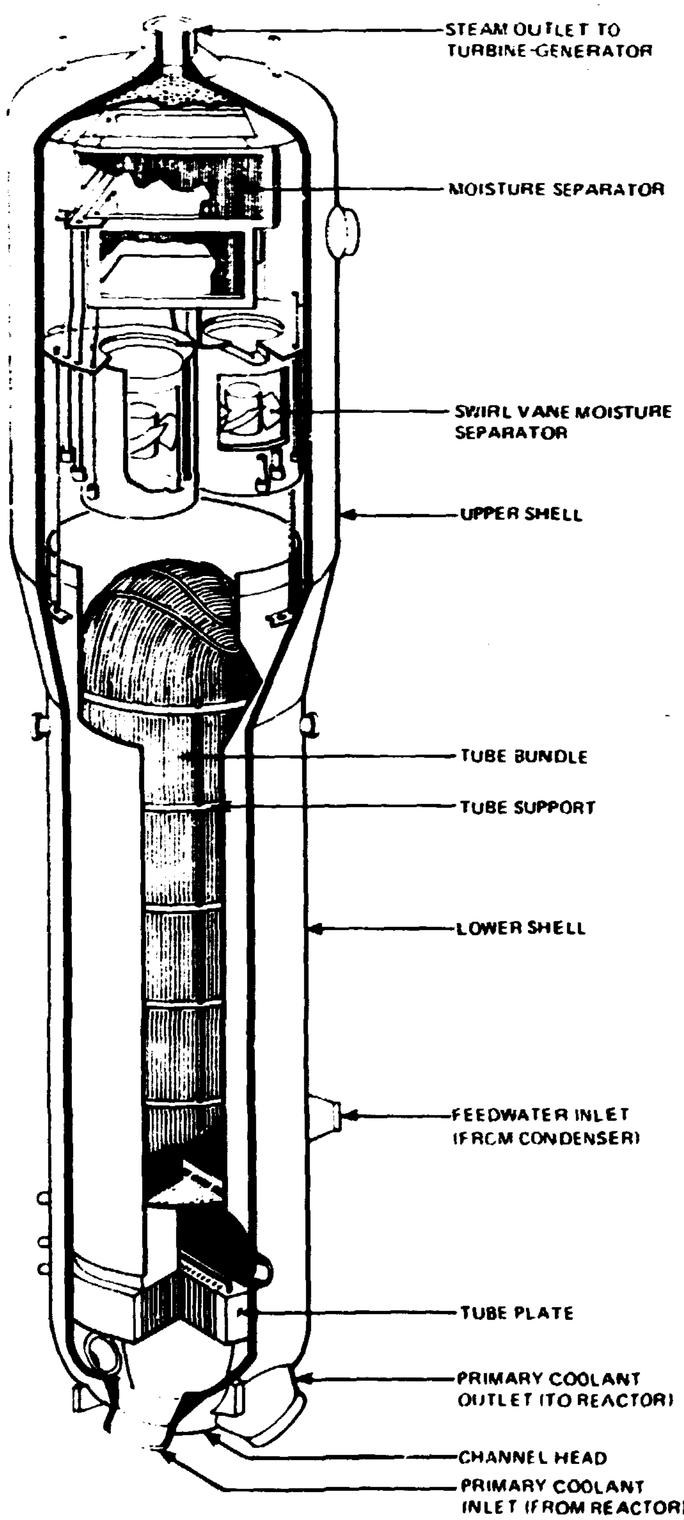


Fig. 2.4. Typical steam generator in the nuclear steam supply system of a pressurized-water reactor.

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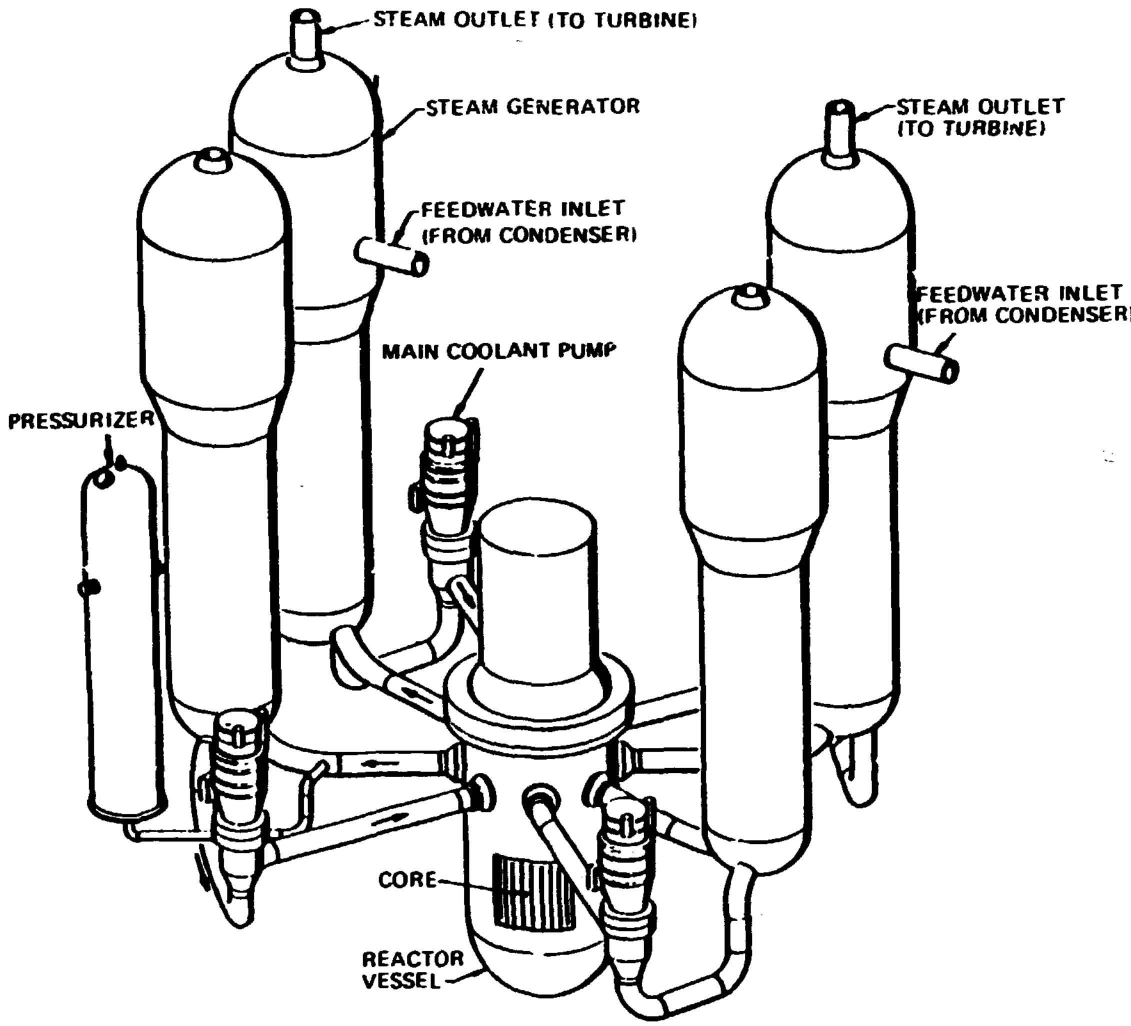


Fig. 2.5. Schematic arrangement of the nuclear steam supply system of a pressurized-water reactor.

from the outer steel shell) extend out at various points from the pressurizer. They are used for attaching additional piping called lines. Safety lines and pressure-relief lines are connected to the nozzles at the top. Safety valves and pressure-relief valves are installed in these lines, and their function is to relieve the pressure if it gets too high.

The entire nuclear steam supply system is enclosed in a containment building, as is illustrated in Fig. 2.7. The primary function of the building is to contain the radioactivity that might be released from the

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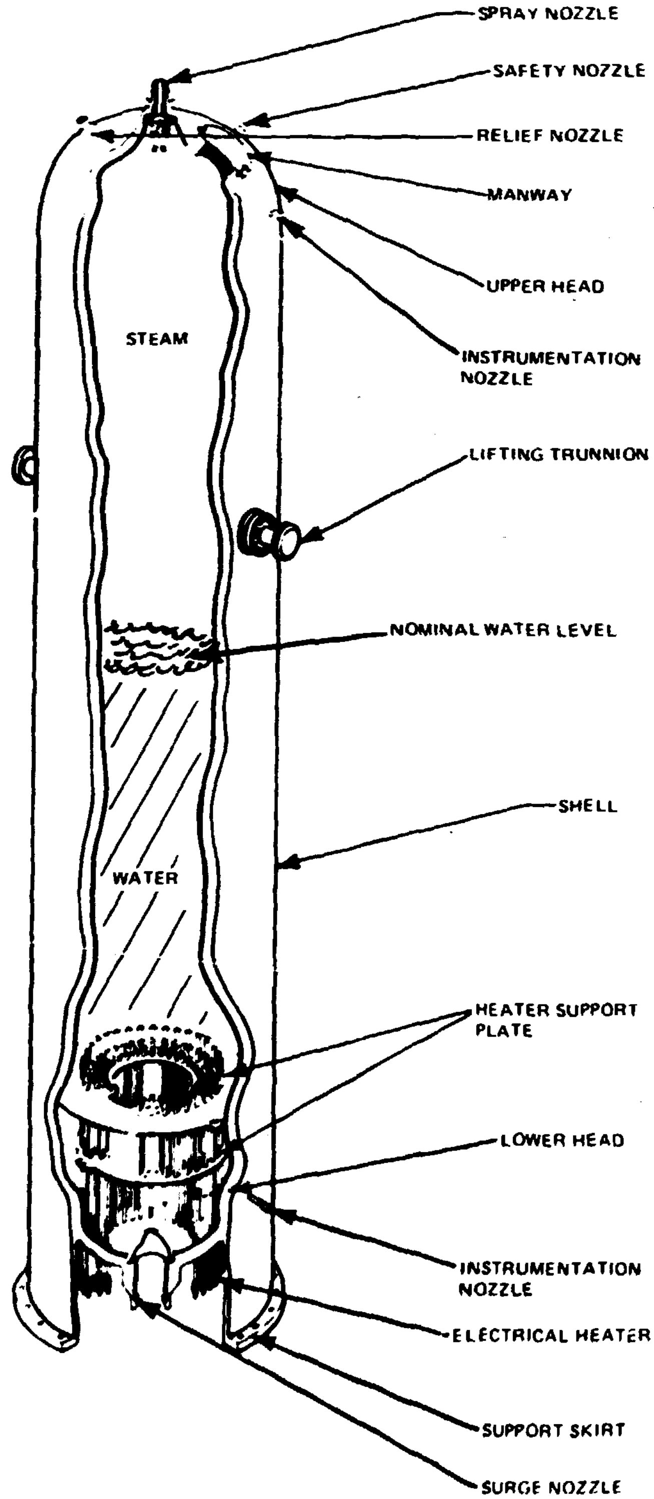


Fig. 2.6. Pressurized-water-reactor pressurizer.

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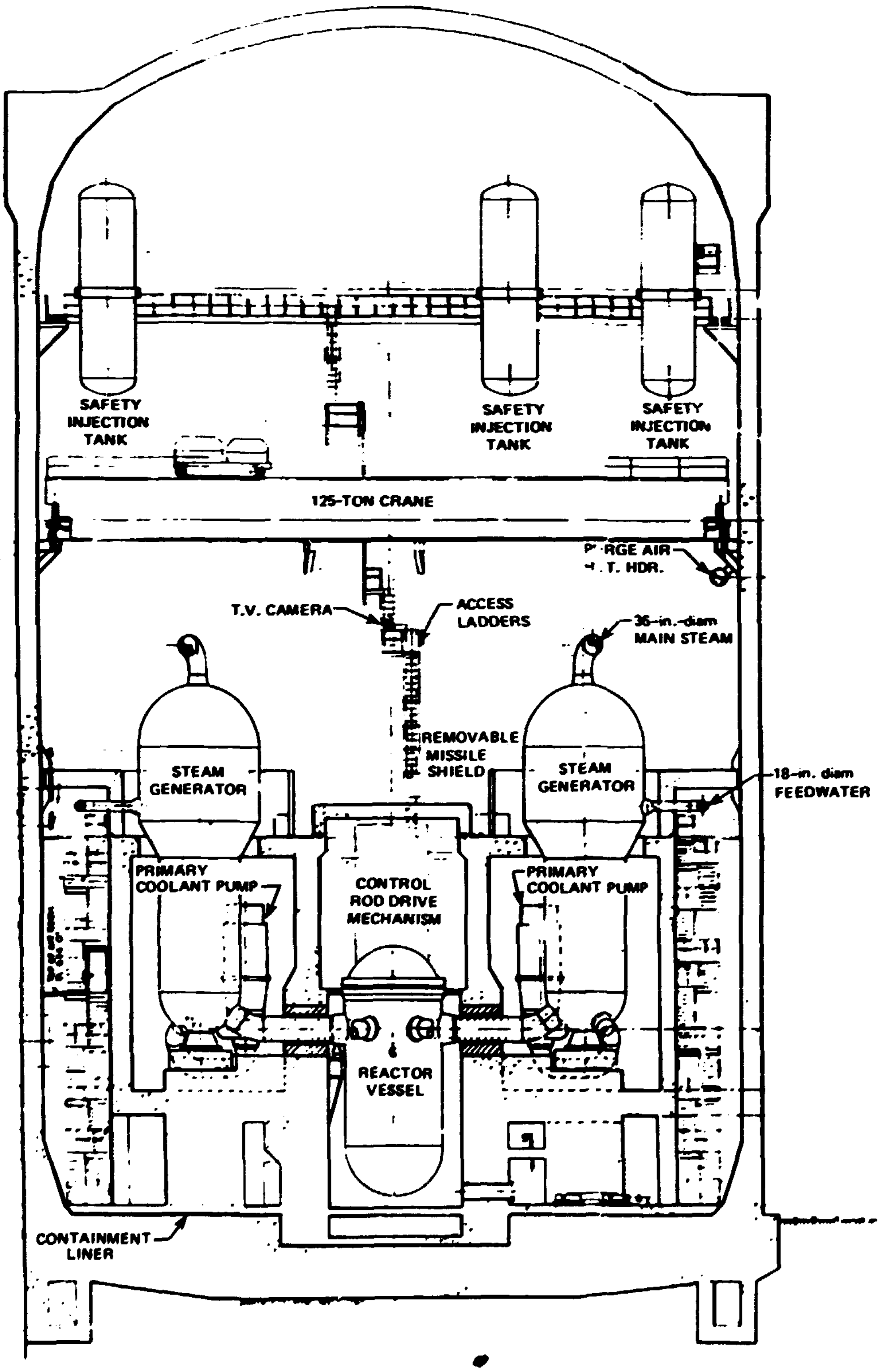


Fig. 2.7. Pressurized-water-reactor containment.

reactor. The structure is sufficiently strong to withstand the pressures that would be encountered in the event of a rupture of the pressure vessel, and it is sufficiently thick to shield the operating personnel from radiation. The air pressure within the containment building is kept at a lower level than that of the outside air so that airflow through cracks or leaks in the building will be from outside to inside. The air within the building is pumped out through a filtering system and is eventually released to the atmosphere via a tall stack.

It should be borne in mind that an actual PWR system is much more complex than the simplified description presented here, where only the basic features are mentioned. There are a host of pumps, lines, and various other equipment in the actual system, which have not been described.

### 2.6.2 Boiling-water reactors (BWRs)

The boiling-water reactor is so named because the water in the core is boiled. The steam thus generated passes directly to the turbine. The secondary system is thereby eliminated. A simple schematic diagram of an electric power plant powered by a BWR is shown in Fig. 2.8.

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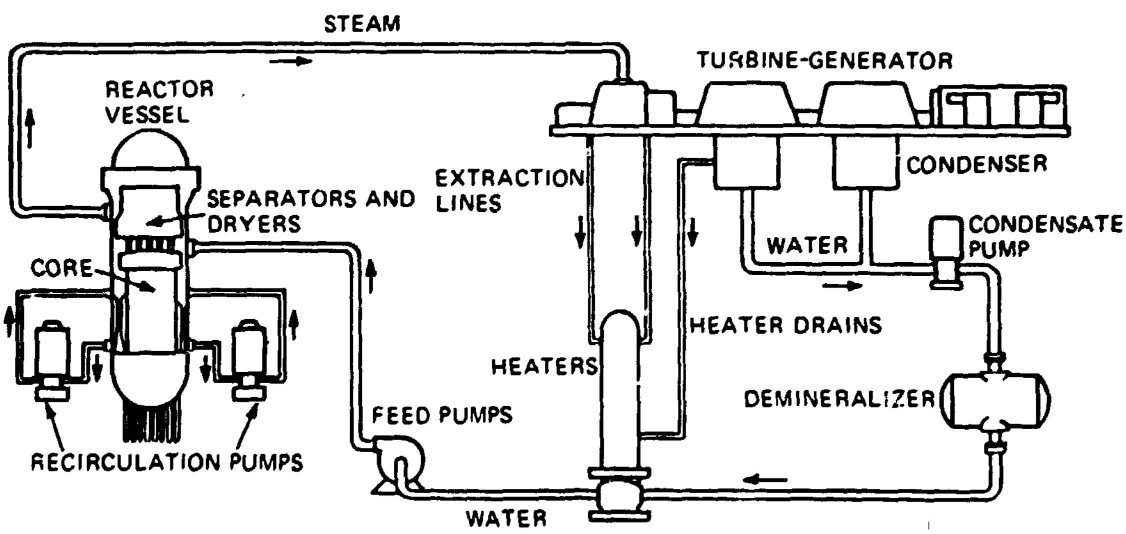


Fig. 2.8. Schematic diagram of a boiling-water-reactor electric power plant.

A BWR is diagrammatically illustrated in Fig. 2.9. Water from the turbine condenser enters from the side of the pressure vessel and is forced downward near the inside surface of the vessel. The jet pumps regulate the downward flow of this water. When the water reaches the bottom of the vessel, it turns and flows upward through the core. The heat frc . the core boils the water, turning it into steam. The steam-water mixture continues upward, whe.e it passes through steam generators and dryers that separate the steam from the water droplets, which would damage the turbine. The "dry" steam then passes to the turbine.

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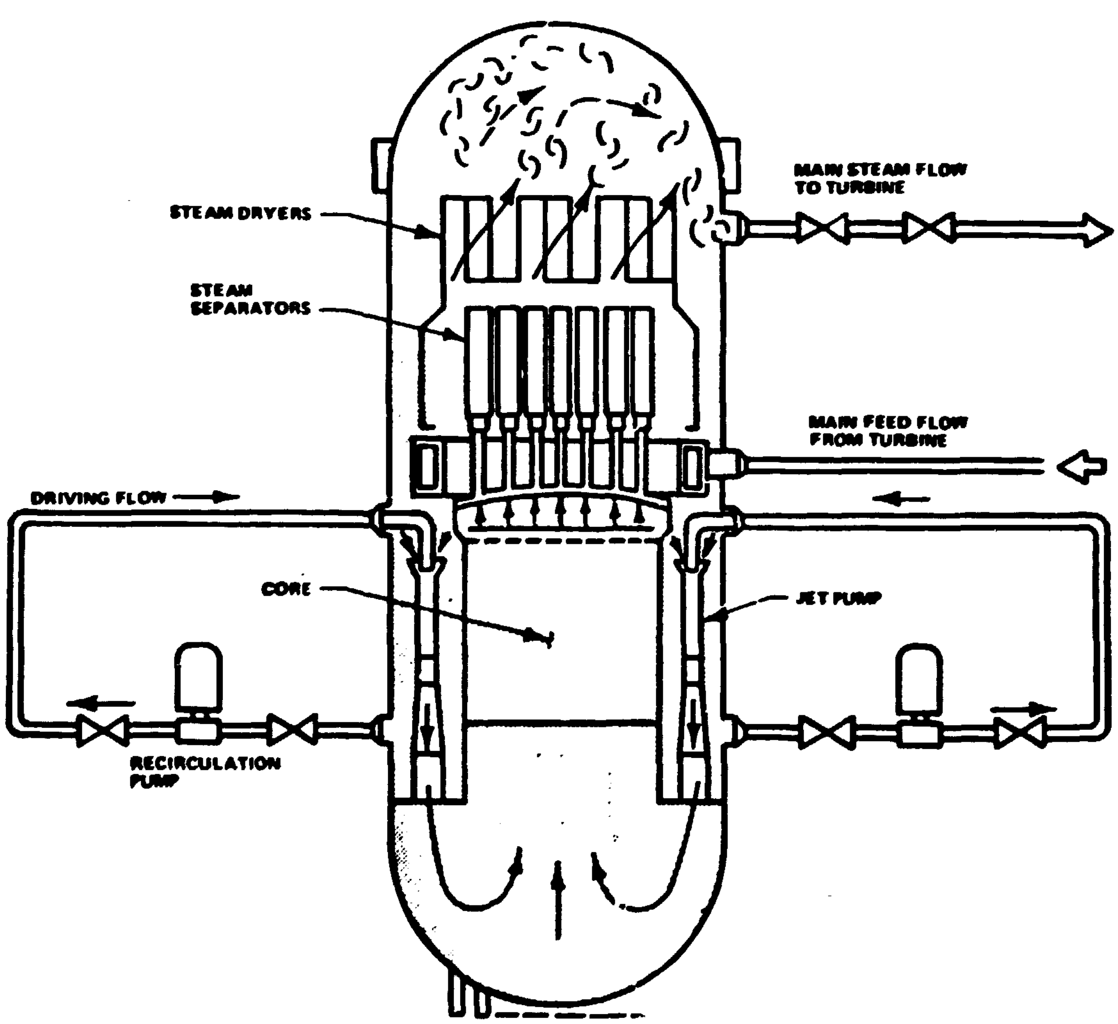


Fig. 2.9. Boiling-water reactor.

Boiling-water reactors are contained in a drywell, as shown in Fig. 2.10. The large doughnut-shaped tube connected to the drywell is called the suppression chamber. It is designed to quench, or suppress, the pressure surges that might result if all the water in the reactor were suddenly turned into steam (an event referred to as a "blowdown"). The drywell and suppression chamber serve the same purpose for BWRs as the containment building does for PWRs. Reactors of the early model shown in Fig. 2.10 were enclosed in a secondary containment building, as shown in Fig. 2.11. A modern system is shown in Fig. 2.12; note that the doughnut-shaped suppression chamber has been replaced by vaults within the building.

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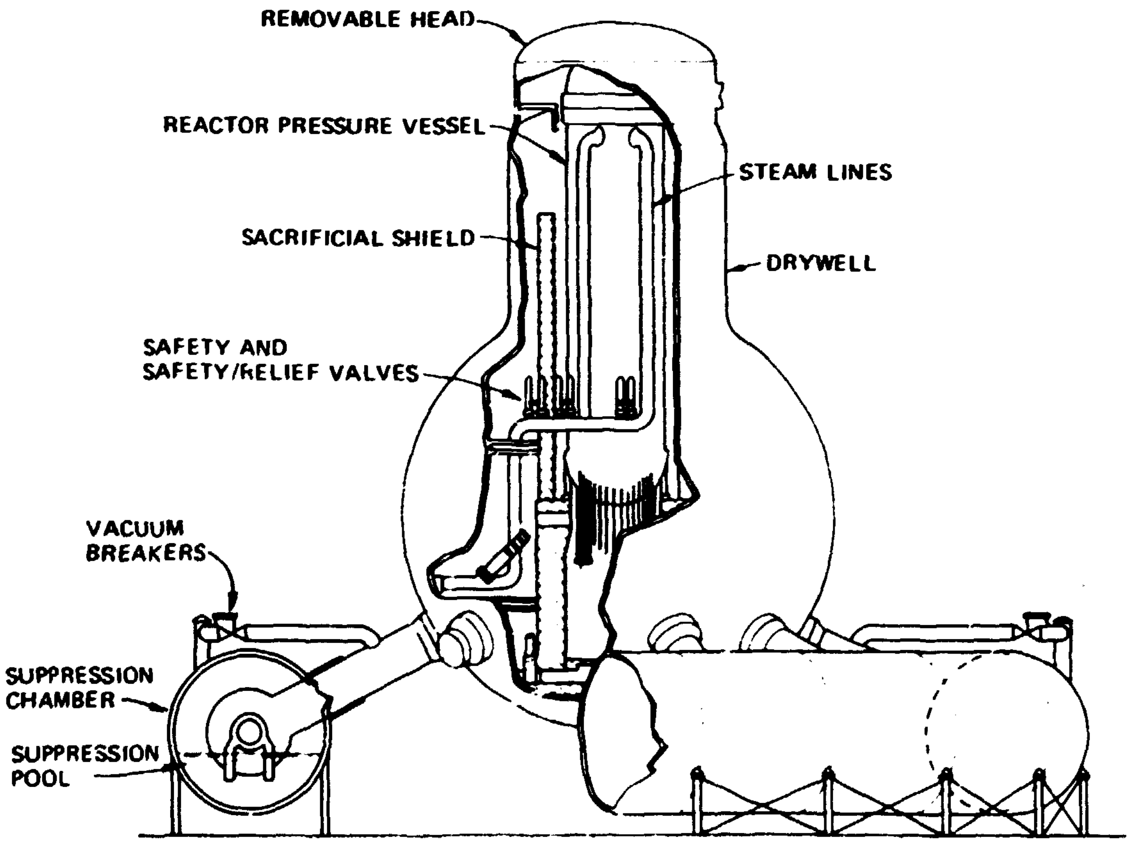


Fig. 2.10. Typical arrangement of an early version of a boiling-water reactor containment system.

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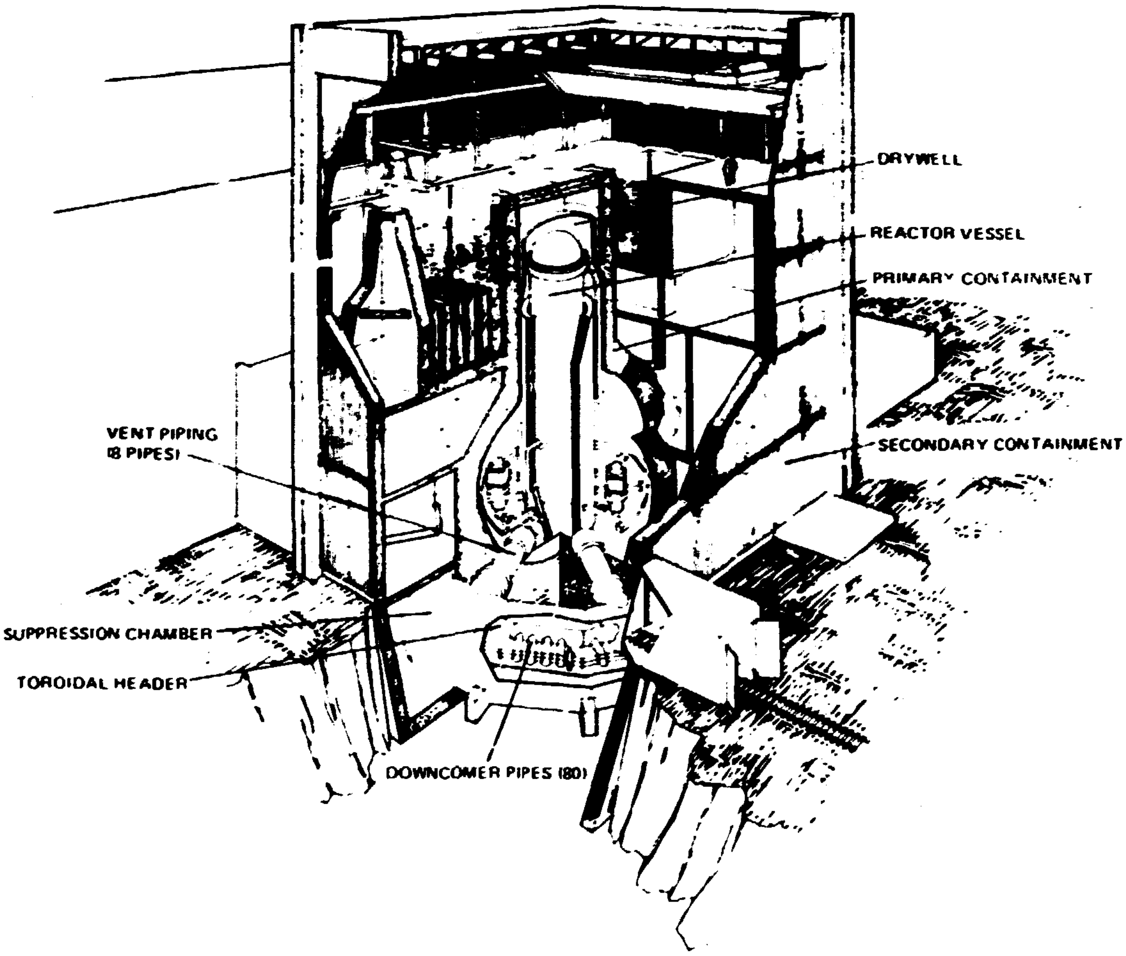


Fig. 2.11. Boiling-water reactor secondary containment building showing primary containment system enclosed.

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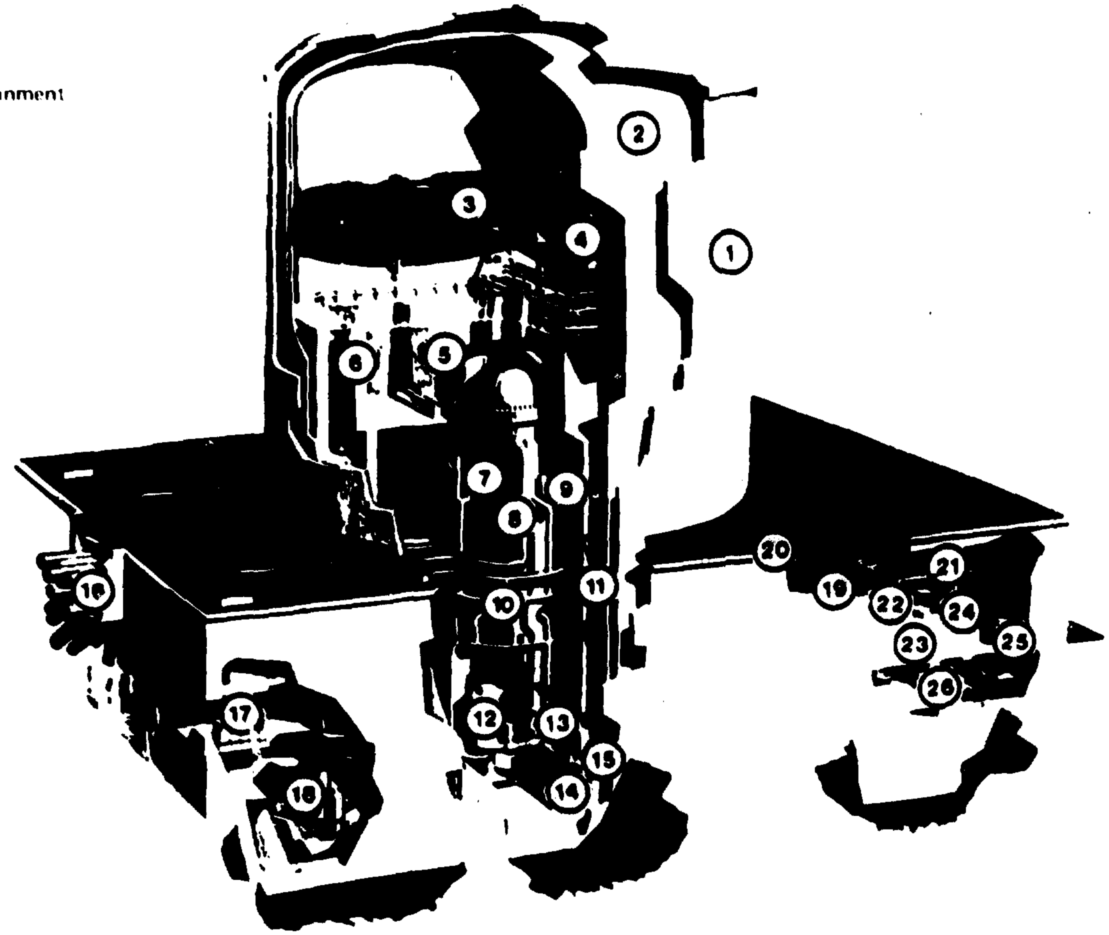


Fig. 2.12. Modern boiling-water reactor containment and internals. (Courtesy of General Electric Company.)

# 3. CENTRAL STATION POWER PLANTS

## 3.1 Fuel Melting Incident at the Fermi Reactor[1]

The reactor in the Enrico Fermi Atomic Power Plant, Unit 1 was a sodium-cooled fast breeder demonstration reactor. The capacity of Unit 1 was 200 MW(t) [61 MW(e)]. The plant was located near Lagoona Beach, Michigan, and was operated by the Power Reactor Development Company. It started up in 1963 and was shut down in 1973. The accident described below was included in this report because it resulted in core damage.

The Enrico Fermi Atomic Power Plant, Unit 1 went critical in 1963, and a series of tests were conducted over the following 2 years at power levels below 1 MW(t). During the next 6 months, power levels were increased in incremental steps up to 100 MW(t) (i.e., about half of full power). During this increase in power, it was noted that the coolant temperatures above 2 of the 155 fuel assemblies (i.e., clusters of fuel rods) were higher than normal and that the temperature above another assembly was lower than normal.

The reactor was shut down and the fuel assemblies were rearranged in the core in order to determine if these abnormal temperatures were dependent on their locations in the core or if they were characteristic of the fuel assemblies themselves. On the basis of previously observed temperature anomalies, it was determined that these tests should be run at 67 MW(t).

On Oct. 5, 1966, the rise to 67 MW(t) was begun. At about 20 MW(t), an erratic electrical signal was noted. It disappeared after a pause, and the rise to 67 MW(t) was resumed. At about 30 MW(t), the erratic signal appeared again. A check revealed that the positions of the control rods were not as expected and that the coolant temperature above two of the fuel assemblies was high. Shortly thereafter, radiation alarms sounded in the containment building. The operator scrammed the reactor. The bulk of the fuel in two fuel assemblies had melted.

Over the next year, many of the assemblies were removed and examined, and it was not until the end of that time that the cause of the accident was discovered. Metal (zirconium) sheets had been added to the coolant flow guide and vessel penetration barrier late in the construction phase as a barrier to molten fuel if a core meltdown were to occur. Segments of the zirconium sheets had torn loose and blocked the flow of coolant through some of the fuel assemblies. Without this flow of coolant, the assemblies had overheated and melted.

Damages were repaired, and the reactor reached full power output on Oct. 16, 1970, 4 years after the accident. It operated successfully for 3 years and was shut down in 1973 after completing all phases of its original mission.

There were no injuries, and there was no release of radioactivity.*

### 3.2 Electrical Cable Fires at San Onofre 1

The reactor in Unit 1 of the San Onofre Nuclear Generating Station is a pressurized-water reactor. The nuclear steam supply system was designed by Westinghouse Electrical Corporation. The station is located in San Clemente, California, and is operated by Southern California Edison and San Diego Gas and Electric Company. Unit 1, which began operation in 1967, has a capacity of 436 MW(e). The incident described below is included in this report because it was a precursor to a potentially more serious accident.

Two fires and a control rod malfunction due to related phenomena occurred within several weeks at San Onofre 1 early in its operating life. In the afternoon of Feb. 7, 1968, while the reactor was operating near full power, a fire was reported just outside the containment building.[2] Electrical cables that penetrated the building had overheated and caught fire and began to short-circuit and spark. The reactor was shut down without incident, and the fire was extinguished. Sixty-five cables at the site of the fire were damaged, and 11 cables in an adjacent penetration were slightly damaged. There was no damage to the cables located inside the containment building.

The reactor was returned to service after all damaged cables were replaced and after repairs were made to the penetration itself. The primary cause of the fire was determined to be overheating in an are of insufficient ventilation.[3]

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*Immediately after the accident, the highest dose measured in the area was 9 mR/hr at the outer surface of the containment building.

Three weeks later, while the reactor was operating at full power, it was discovered that a control rod had remained fully inserted in the core.[2] This situation is not particularly alarming, except that it indicated a lack of knowledge of the status of all systems. The problem was caused by incorrect wiring during the previous repairs. The reactor was shut down, the error corrected, and the reactor was restarted.

A short time later, in March, another fire broke out.[3] This one was located in a switchgear room outside the containment building, and the fire damaged the electrical cables in the room. The cause of the second fire was essentially the same as that of the first. The reactor was shut down; however, a significant problem developed, which was caused by a loss of power to several systems.

The problem involves an inherent characteristic of light-water reactors in that as the reactor cools it gains reactivity naturally. In reactors such as the one at San Onofre 1, this gain in reactivity is offset both by the control rods and by the addition of boron to the water that passes through the core. The boron is an efficient neutron absorber.

When the reactor is shut down by the insertion of control rods after operating at power, it is still very hot. Without boron, if it were allowed to cool to room temperature, it would gain reactivity and come close enough to becoming critical to violate the margin of safety required for cold-shutdown conditions.* Thus, boron is added to the water to offset this effect and to add a margin of safety.

When the San Onofre 1 reactor was shut down after the second fire, the usual method of adding boron was lost because of the fire. An auxiliary method was tried, and 4 hr later, as the reactor was cooling, it was discovered that boron was being removed rather than added. The error was corrected, and a cold shutdown was achieved with the correct safety margin.

There was no release of radioactivity nor were there any injuries in these incidents.

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*Cold shutdown is a condition in which no power is being produced, and the temperature of the cooling water, which is removing decay heat from the core, is about 200°F or less.

## 3.3 Fuel Meltdown at Saint Laurent[4]

> The reactor in Unit 1 of the Saint Laurent Plant is gas-cooled and graphite-moderated. The plant is rated for 500 MW(e) capacity and is operated by Electricité de France. The reactor was charged with its initial fuel in January 1969. The accident described below is included in this report because it resulted in core damage.

The reactor at Saint Laurent is different from U.S. power-producing reactors in that it allows for unloading of spent radioactive fuel and reloading of fresh fuel while the reactor is at full power. The heavy-water reactors that are in operation in Canada also have this capability, but none of the U.S. reactors are so designed.

The machine that unloads and loads the fuel at Saint Laurent is called a charging machine. It is a huge device that is computer-programmed to move about the top of the reactor and position itself properly over each access port to load and unload. It automatically latches on to the access port, unseals the port, loads or unloads through the port in such a way that the other concurrent seals are always airtight, reseals the port, unlatches itself, and moves to the next access port for which it is programmed. The operator cannot see what is being transferred. The fuel holes and other holes in the graphite of the core are directly beneath each access port. Figure 3.1 illustrates the configuration.

The charging machine itself has 24 separate storage chambers. Spent fuel, graphite plugs, etc., that are removed from the core are temporarily stored in chambers that are empty. The other chambers contain the fuel, etc., that is to be loaded into the core. The capacity of each chamber is only one-third that of each channel in the core, so that three chambers full of fuel in the charging machine are needed to completely load one fuel channel in the core with fuel.

The machine is programmed to stop if an incorrect command has been given. For example, if commanded to unload the spent fuel from a channel in the core and place it in a chamber of the machine which is not empty but already contains other material, the machine will stop just prior to

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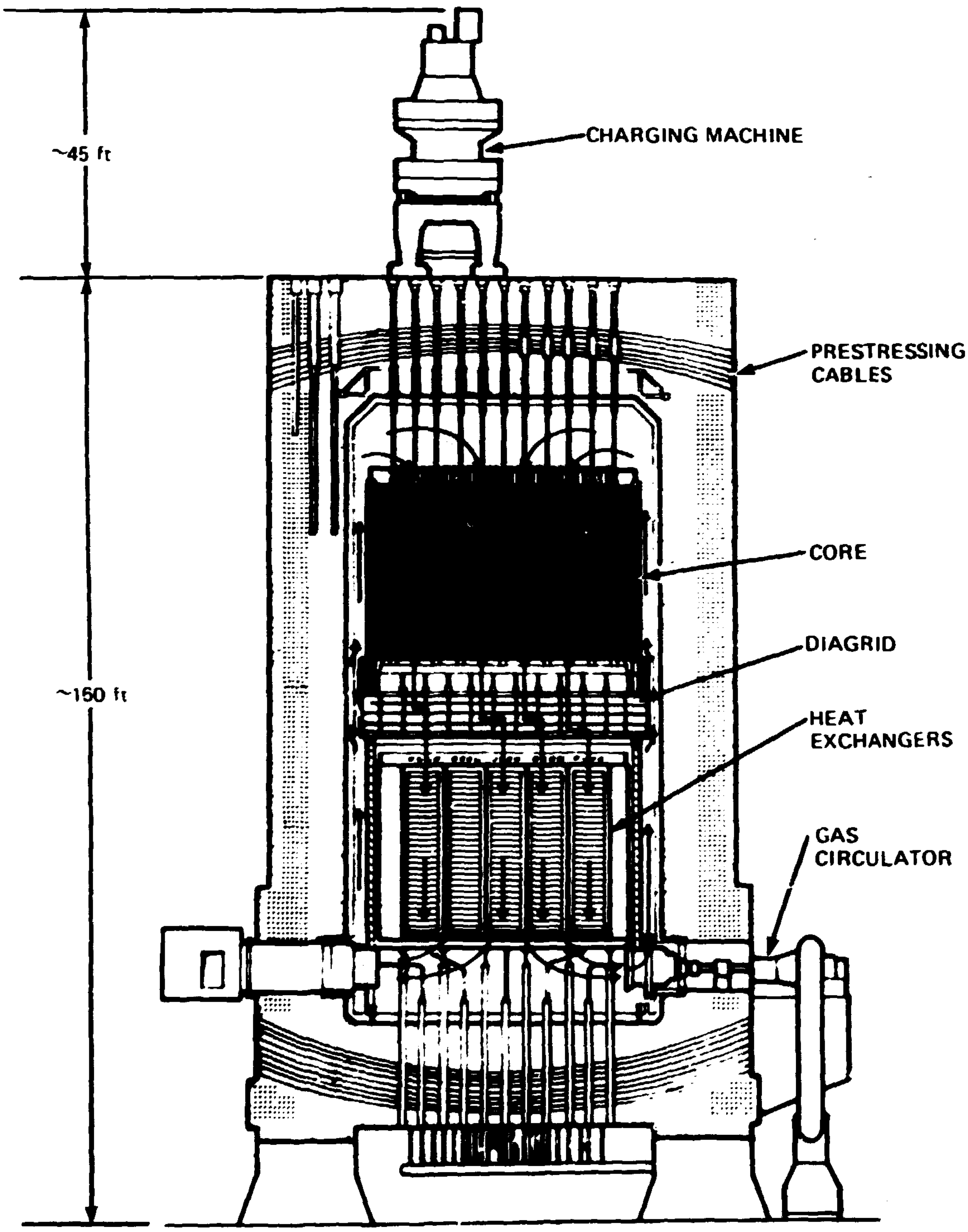


Fig. 3.1. Reactor at Saint Laurent.

fulfilling this order. However, all such stops can be overidden by the operator, and the charging machine can be operated by manual control.

During the midnight shift on Oct. 17, 1969, with the reactor near full power, a normal loading and unloading operation was in progress. Graphite plugs that had been placed temporarily in one of the fuel channels in the core were now being replaced by fuel. The charging machine had unloaded the graphite from the core into its empty storage chambers and had loaded fuel into the core from two of its full chambers, but then it stopped. The operator instructed the machine to probe the third chamber, and the machine started to obey the command but stopped again. The operator assumed (correctly) that the third chamber probably was empty and directed the machine to complete the loading from a fourth chamber which he ascertained to be full. After partially completing this command, the machine stopped again. The operator again overrode the stop and completed the loading by manual control.

A few minutes later, alarms sounded that were set off by high-radiation monitors within the core. The reactor scrammed automatically. A few of the fuel elements in the channel that had just been loaded had melted.

Subsequent investigation revealed that the charging machine had stopped the last time because a coolant flow restrictor rather than a fuel element was in position to be loaded. When the operator overrode the stop and loaded this restrictor into the top of the channel, the coolant flow to this channel was reduced to one-fourth the normal flow. Without proper cooling, some of the fuel elements and cladding heated up beyond their melting point and flowed out of the core onto the diagrid below (see Fig. 3.1), releasing radioactive fission products that set off the alarms and the scram. The melted fuel (about 110 lb)[5] was still well contained within the massive concrete structure (see Fig. 3.1); hence, little, if any, radioactivity was released outside of the structure, and there were no injuries. However, a year was needed to complete the cleanup operations and restart the reactor. Modifications to the machine have been made, and it is no longer so simple to override a charging machine stop.

## 3.4 Uncovering of the Core at La Crosse[6]

The reactor in the La Crosse Nuclear Generating Station is a boiling-water reactor. The station is located at La Crosse, Wisconsin, and has a capacity of 50 MW(e). The nuclear steam supply system was designed by Allis-Chalmers Manufacturing Company. The plant, which is operated by the Dairyland Power Cooperative, began operation in 1967. The incident described below is included in this report because core damage was suspected, although it did not actually occur.

In a boiling-water reactor the steam goes directly to the turbine; thus, there is no secondary system (see Fig. 3.2). On May 15, 1970, while the reactor at La Crosse was operating at 60% of full power, a malfunction occurred which closed a valve associated with the steam supply to the turbine (turbine main steam bypass valve). Noticing that the hydraulic system which operates this valve was in an abnormal state, the operator began normal reactor-shutdown procedures, but when the pressure in the reactor began to build up too fast, he scrammed the reactor.

At this point there are standard operating procedures which would vent the steam that is still being generated and also control the pressure and the water level in the reactor as it cools. However, the valve

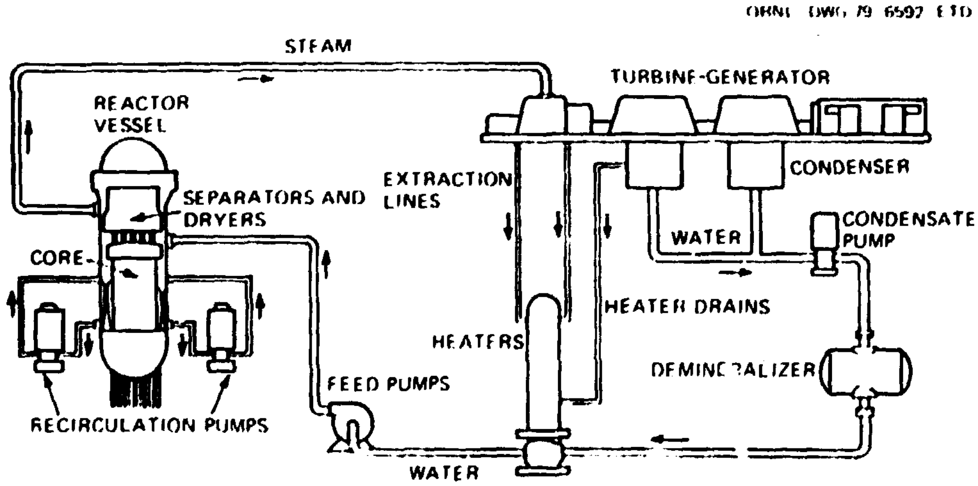


Fig. 3.2. Schematic diagram of steam system in a boiling-water reactor.

that closed initially began to open and close erratically, which upset the effects of the normal procedures. The reactor system pressure dropped precipitously, causing the water in the reactor to vaporize, and this, along with the intermittent shutting down of the feedwater pumps that supply water to the reactor, caused the water level to drop about 27 in. below the top of the core.

The valve causing the problem was manually closed, control over the system was restored, and the pressure and water level returned to normal. About 30 min had elapsed since the start of the incident. There were no indications of fuel damage or release of radioactivity; so the cause of the malfunction was repaired, and other systems were adjusted and instrumented to allow for better response. Two days later the reactor was restarted.

There were no injuries nor was there any release of radioactivity.

### 3.5 Seven Men Injured When Steam Nozzle Breaks at Robinson[7,8]

The reactor in Unit 2 of the H. B. Robinson Plant is a pressurized-water reactor. The nuclear steam supply system is a Westinghouse Electric Corporation design. The plant is located at Hartsville, South Carolina, and is operated by the Carolina Power & Light Company. Unit 2, which began operation in 1970, has a capacity of 700 MW(e). The accident described below is included in this report because it resulted in injuries.

There are two aspects of this accident that should be noted at the outset. One is that there was no potential for release of radioactivity because the nuclear fuel had not yet been loaded into the core at the time of the accident, and the other is that the accident occurred in the secondary system. About mid-1970, during the pre-startup pressure testing of the secondary system of Unit 2 of the H. B. Robinson Plant, a team of seven men were in the process of testing the safety valves. They were testing to be sure that the valves opened when the pressure in the steam lines became too high. There was no fuel in the core. The men had tested 8 of the 12 safety valves that were attached to 3 large steam pipes that came from the steam generators and were under pressure (see Figs. 3.3 and 3.4). As one of the men attached the testing equipment to valve 4, a jet of steam sliced out horizontally in a fan-like fashion, followed almost immediately by a blast as the entire valve tore loose and was propelled upward, followed by a vertical jet of steam that reached a height of 150 ft. The blast caused a shower of scaffolding, insulation, metal parts, and construction debris, and the men were either knocked to the floor or fell there intentionally to escape the steam.

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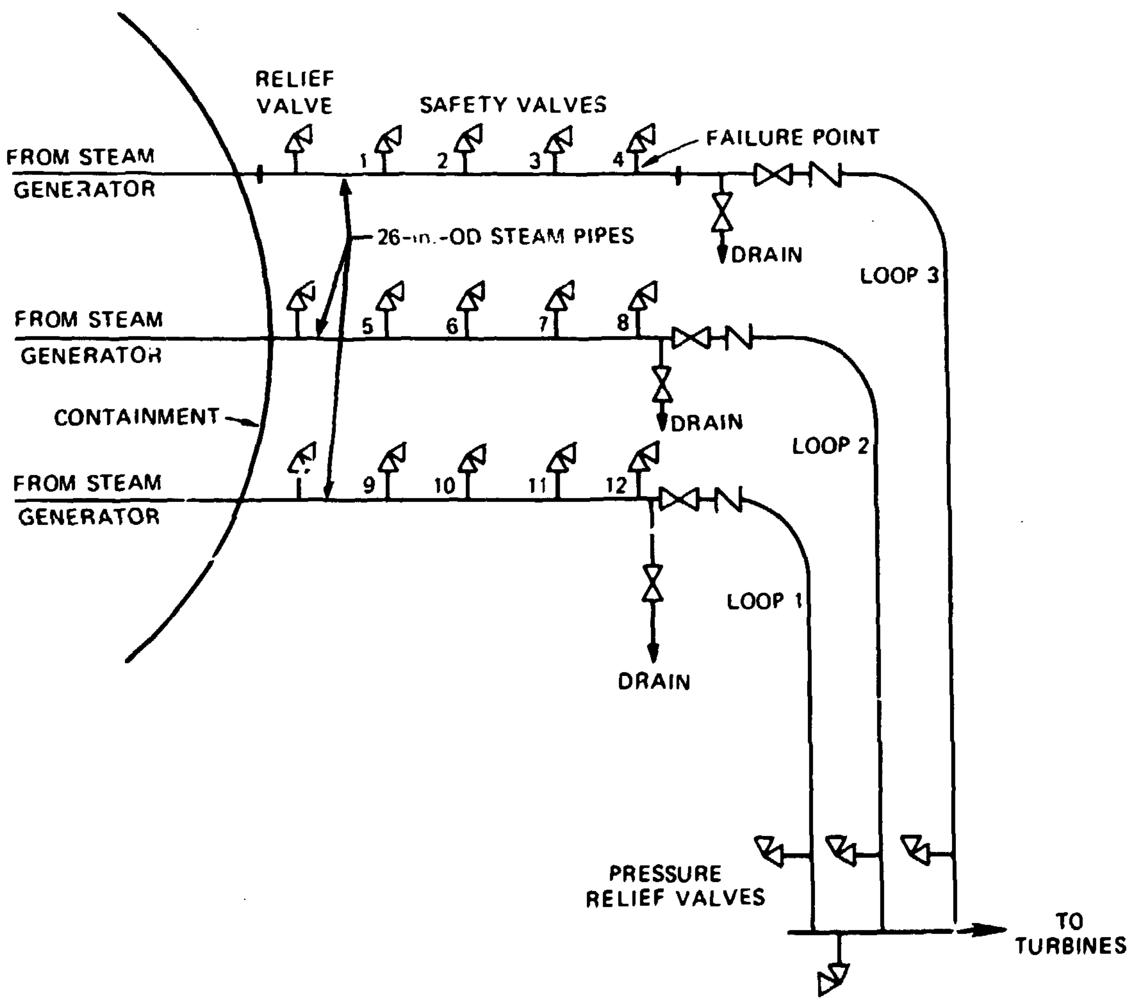


Fig. 3.3. Schematic diagram of the general area of the accident at Robinson 2.

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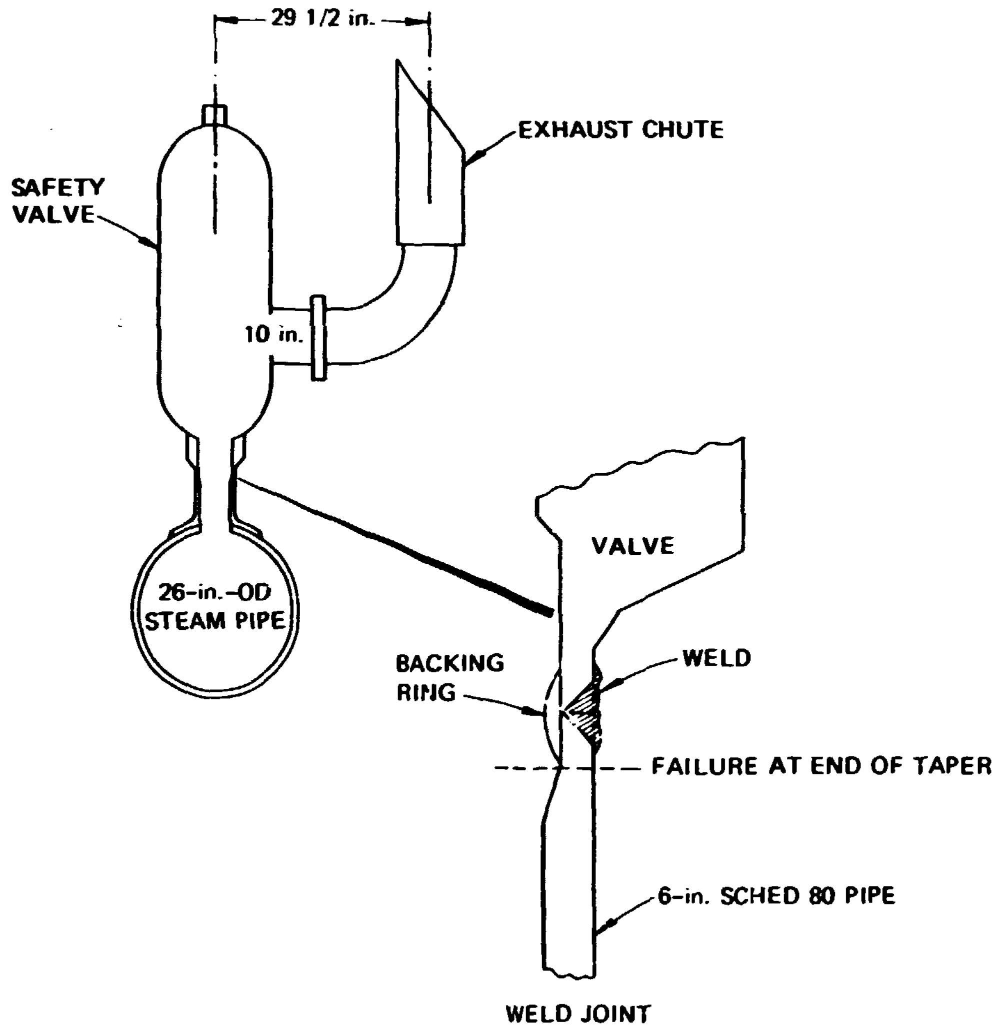


Fig. 3.4. Area of break in steam nozzle at Robinson 2.

The men immediately made their way, unassisted, out of the area and down a stairway away from the accident. They were transported by ambulance to a hospital where they were treated for burns and injuries.

All the connections (nozzles) between the 26-in. steam pipes and the safety and relief valves were replaced by nozzles of larger diameter and thicker steel.

In this accident there were seven injuries, but there was no release of radioactivity.

## 3.6 Discharge of Primary System into Drywell at Dresden 2 (Ref. 9)

The reactor in Unit 2 of the Dresden Nuclear Power Station is a boiling-water reactor. It was designed by General Electric Company. The station is located at Morris, Illinois, and is operated by the Commonwealth Edison Company. Unit 2, which has a capacity of 794 MW(e), began operation in 1970. The incident described below is included in this report because it resulted in significant recovery costs.

On the evening of June 5, 1970, while the reactor at Dresden 2 was undergoing initial startup power tests and was at about 75% of full power, a spurious signal led to the opening of valves associated with the turbine steam supply (turbine bypass valves; see Fig. 3.5). This

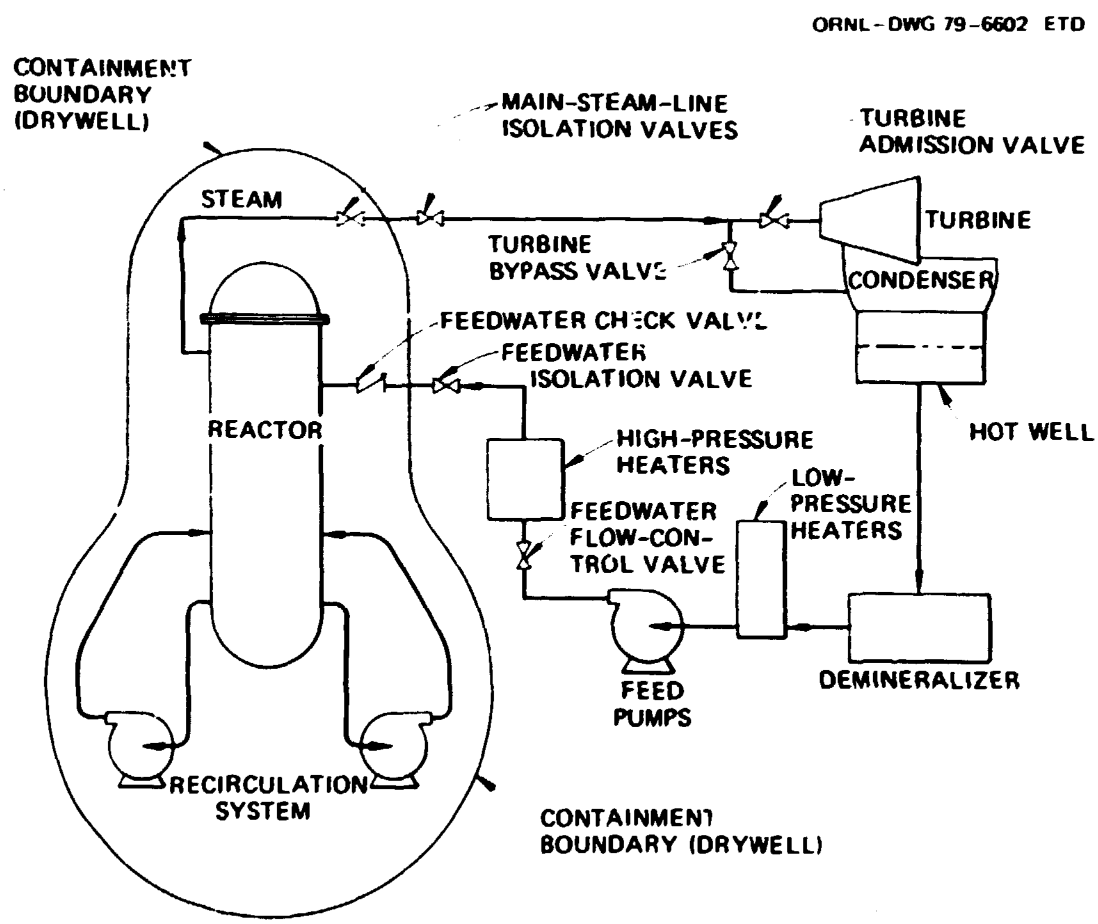


Fig. 3.5. Diagram of a typical steam system in a boiling-water reactor.

shut down the turbine, which, in turn, caused an automatic scram of the reactor. The reactor remained shut down throughout the subsequent events. With the reactor shut down, the pressure began to drop, causing water in the reactor to turn to steam. This caused a drop in the water level of the reactor. The pumps that return the water to the reactor (feedwater pumps) began to operate intermittently, and the water level in the reactor varied considerably. Then the bypass valves that started the incident closed apparently due to the disappearance of the spurious signal, and other valves that allow steam to pass from the reactor to the turbine (main-steam-line isolation valves; see Fig. 3.5) closed because the pressure in the reactor continued to fall. All of this happened within 33 sec. The water level in the reactor then began to rise steadily, but the pen on the water-level-indicator chart in the control room stuck at a low value. The operator, believing that the reactor needed more water, increased the flow rate to the reactor. He discovered about a minute later that the pen was stuck, but by this time the water level had risen so high that it flooded the main lines that normally contain steam. He tried to shut off the flow of water completely, but the shut-off valves began leaking, allowing water to continue to enter the reactor.

Because of the flooded condition of the reactor and the residual heat in the reactor, the pressure began to rise. A few methods were tried to reduce the pressure, but they failed. The operator then manually opened a pressure-relief valve that normally allows steam to escape to the suppression pool (see Fig. 3.6). The escaping water and steam jarred the adjacent safety valves partially open, but at least the pressure began to fall in the reactor vessel. However, the pressure and temperature in the drywell began to rise. Unfortunately, the temperature recorder ran out of paper at this time, and so the highest temperature reached could not be recorded. To reduce the pressure in the drywell (which was above 19 psi), the operator vented the containment through the standby gas-filtering system and thence through the off-gas stack to the atmosphere. The radioactivity released was about 2 1/2 times the normal release for a half hour, and then it returned to normal. By various manipulations, the reactor pressure and water level came under

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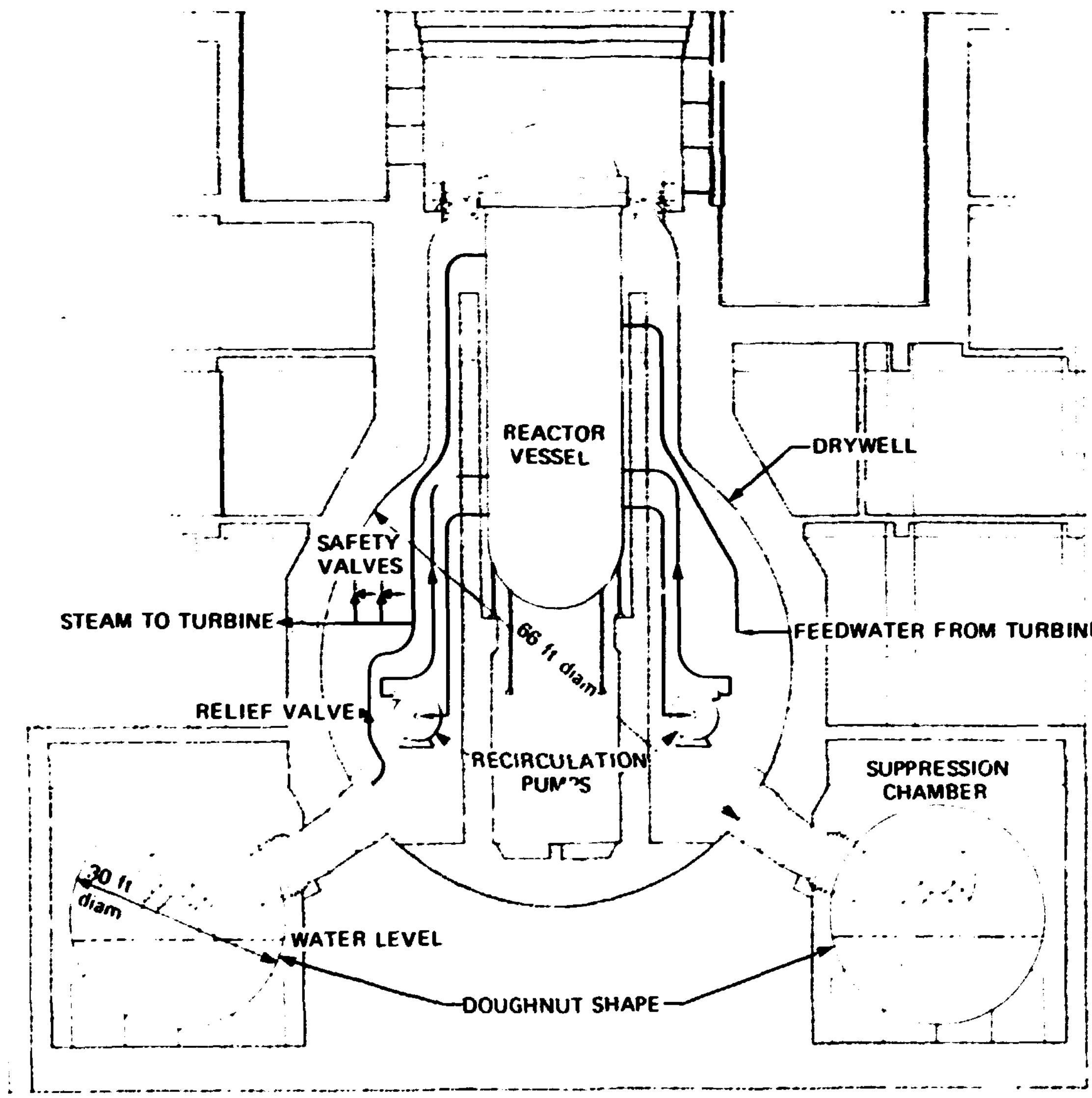


Fig. 3.6. Primary system and containment of a boiling-water reactor.

control, and so did the pressure and temperature in the drywell. About 2 hr had elapsed since the incident began.

The drywell was contaminated, and there was moderate damage to the wire cables and other electrical equipment in the drywell. Environmental radiation survey samples collected downwind after the event showed no difference from those taken upwind, and both offsite and onsite radiation detection meters showed normal readings.[13] The damaged equipment was repaired and replaced, instrumentation was improved, some of the operating procedures were modified, and the reactor returned to service in about 2 months.

There were no injuries, and no significant amount of radioactivity was released as a result of this accident.

### 3.7 Turbine Damage Caused by Human Error at Robinson 2 (Refs. 11, 12)

The reactor in Unit 2 of the H. B. Robinson Plant is a pressurized-water reactor. The nuclear steam supply system is a Westinghouse Electric Corporation design. The plant is located in Hartsville, South Carolina, and is operated by the Carolina Power & Light Company. Unit 2, which began operation in 1970, has a capacity of 700 MW(e). The incident described below is included in this report because it resulted in damage to major equipment.

In addition to the standard ac supply of electricity to the H. B. Robinson Plant, there is a dc supply from banks of batteries. The dc supply is used to activate circuit breakers, valves, emergency pumps, etc., including the reactor scram system.

On Mar. 14, 1971, with the reactor operating near full power, a workman turned on an emergency oil pump, powered by the dc supply, for a weekly 2-hr test run and then forgot to turn it off. The emergency oil pump is sufficiently large so that the batteries will eventually discharge if the pump is left running too long, even with the battery charger being turned on.

About 4 hr after the oil pump had been turned on, the reactor scrammed. However, by this time the voltage from the dc electrical supply system had become too low, which caused several auxiliary electrical circuits to fail. One of these failed circuits was associated with the lubrication of the bearings in the turbines, and another was associated with the cooling of the shafts of the primary reactor coolant pumps. The result was that all eight turbine and generator bearings suffered some damage. One of them became so hot that the adjacent metal became molten and flowed through the bearings. Also, a wheel shroud cracked, and there was damage to some steam seals. In addition, a primary-coolant-pump shaft warped and had to be replaced.

It took about 2 months to repair the damage and resume operation. There were no injuries, and no radioactivity was released.

### 3.8 Construction Fire at Indian Point[13]

The reactor in Unit 2 at Indian Point Station is a pressurized-water reactor. It was designed by Westinghouse Electric Corporation. The station is located in Buchanan, New York, and is operated by the Consolidated Edison Company of New York, Inc. Unit 2, which began operation in 1973, has a capacity of 873 MW(e). The incident described below is included in this report because it resulted in significant recovery costs.

On Nov. 14, 1971, Consolidated Edison Company of New York reported that a fire occurred Nov. 4 at the Indian Point 2 nuclear plant, which was *nearing completion.*[14] The fire started in a wooden shed, which was temporarily located in the Primary Auxiliary Building (about 150 ft from the reactor containment) and was being used by construction forces as a combination storeroom and office facility. The reactor core had not yet been loaded with fuel; there was no nuclear safety problem, nor was the public endangered as a result of the fire.

The first report of the fire was at 7:00 PM, and by 7:50 PM the control rooms of both Unit 2 and Unit 1 contained smoke to the extent that breathing apparatus was deemed advisable. In addition, portable fans were installed to clear the air. By 8:40 PM the breathing apparatus was no longer needed. During the entire emergency period, Unit 1 remained operational at 80% of its licensed power level of 270 MW(e). The fire was extinguished by 9:00 PM.

The fire, which lasted 2 hr and utilized the services of five fire companies, resulted in considerable damage to the Primary Auxiliary Building and equipment located there. Electrical cables, as well as motor-control centers, were damaged. An estimate of the damage has been placed at less than $5 million. The building and equipment were restored to the condition that existed before the fire, and Unit 2 became fully operational in 1973.

### 3.9 Valve Separations at Turkey Point 3 (Ref. 15)

> The reactor in Unit 3 of the Turkey Point Plant is a pressurized-water reactor. The nuclear steam supply system is a Westinghouse Electric Corporation design. The plant is located in Florida City, Florida, and is operated by the Florida Power & Light Company. Unit 3, which began operation in 1972, has a capacity of 693 MW(e). The incident described below is included in this report because it was a precursor to a potentially serious accident.

During hot functional testing (prior to fuel loading) of the Florida Power & Light Company's Turkey Point Plant, Unit 3, three of four safety valves suddenly separated from two headers of a main steam line in the secondary system.[16] At the time of the failure, the secondary system's pressure and temperature were 990 psig and 545°F, respectively, with the pressure and temperature of the primary system at 2232 psig and 547°F. These systems had been at these conditions for 9 days, and there had been no pressure or temperature transients in either system until the time of failure.

The failed header assemblies each consisted of two valves mounted vertically in a dead-end 12-in.-diam pipe that projected horizontally at a 90° angle from the main steam line. The two headers were mounted 180° from each other on opposite sides of the main steam line. Prior to the hot functional testing, the system had been hydrostatically tested at 1356 psig under cold conditions. Each header had two safety-valve nozzles which made up a weldolet and a reducing flange. The headers failed in the pipe material just below and outside the nozzle-to-pipe weld. On one side a valve, complete with nozzle, broke free, leaving the pipe wide open, whereas on the other side both valves with nozzles were blown off the pipes.

A similar incident was reported in *Nuclear Safety*[17] in 1970. The cause of the failures was determined to be an insufficient number of tie-downs. At Turkey Point a new design for the headers was adopted. Licensees of other operating power reactors were informed of details of the failure and were requested by the Atomic Energy Commission (by telegram) to supply data concerning installation of similar header assemblies. Substantial rule changes were adopted as a result of this incident.

The nuclear system was not involved in this incident; consequently, there was no release of radioactivity. There were no injuries.

### 3.10 Turbine Basement Flooded at Quad Cities[18]

> The reactors in Unit 1 and Unit 2 of the Quad Cities Station are boiling-water reactors. The nuclear steam supply systems were designed by General Electric Company. The station, which is located in Cordova, Illinois, is operated by the Commonwealth Edison Company and the Iowa-Illinois Gas & Electric Company. Both units started operation in 1972. Each unit has a capacity of 789 MW(e) each. The incident described below is included in this report because it resulted in significant recovery costs.

After passing through the turbine, the steam in the primary system is condensed back to water and returned to the reactor. There, the condensed water is converted to steam again by the heat generated in the reactor. (See Fig. 3.7.)

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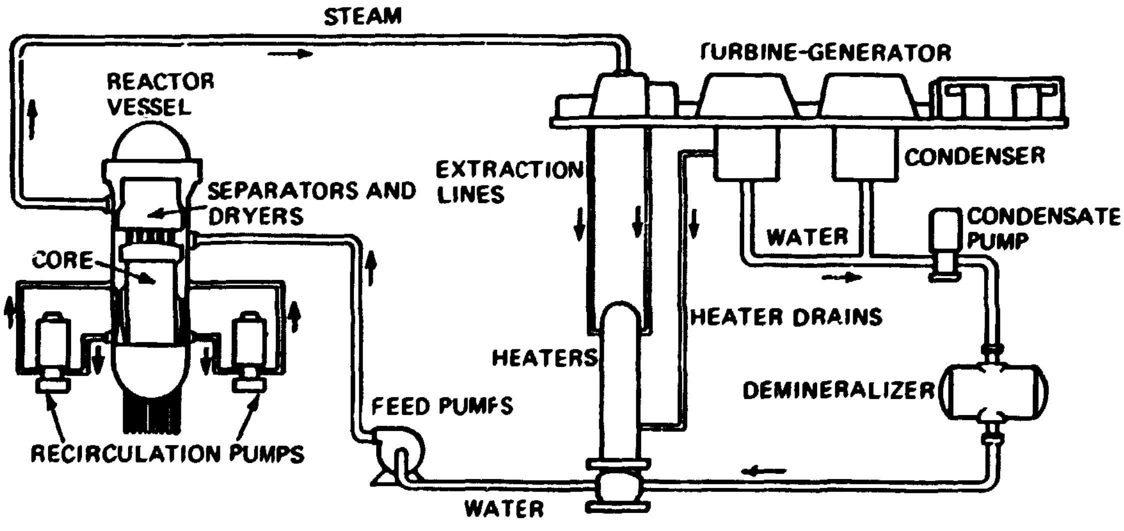


Fig. 3.7. Schematic diagram of the steam system in a boiling-water reactor.

Heat must be extracted from the steam after it passes through the turbines in order to condense it to water. Since a considerable amount of heat must be extracted and returned to the environment without undue disturbance to the environment, a large system must be used. In the case of Quad Cities, water from the Mississippi River is used to condense the steam from the turbines.

Some problems had been experienced at Quad Cities with valves associated with the condensers in the area of the turbines, and attempts were being made to repair them. On June 9, 1972, Unit 1 had been shut down and Unit 2 was operating at less than 1% of power. While workmen were in the basement of the turbine buildings modifying the valves, an adjacent 10-ft butterfly valve slammed shut. The shock ruptured a seal in the circulation system for the river water, and water poured into the basement. The workmen evacuated the basement and notified the reactor operator, who shut Unit 2 down completely. Before the leak could be plugged, 15 1/2 ft of river water had accumulated in the basement of the turbine building. However, in 3 days both units were ready to begin operation.

There were no injuries, and there was no release of radioactivity.

### 3.11 Steam Generator Damaged in Hot Tests at Oconee 1 (Ref. 19)

The reactor in Unit 1 of Oconee Nuclear Station is a pressurized-water reactor. The nuclear steam supply system was designed by Babcock & Wilcox Company. The plant, which is operated by the Duke Power & Light Company, is located in Seneca, South Carolina. Unit 1, which began operation in 1973, has a capacity of 887 MW(e). The incident described below is included in this report because it was a precursor to a potentially serious accident.

In March 1972 following the first phase of the hot functional testing program at Oconee Nuclear Station, Unit 1, an inspection of the reactor coolant system revealed extensive damage to the tube ends and to the tube-sheet welds in the upper head of one of the two steam generators. Only minor damage was observed in the second unit.

The cause of the damage was found to be loose parts from failed reactor-vessel internal components, primarily 3/4-in.-diam in-core instrument nozzles. These nozzles penetrate the bottom of the reactor vessel and allow insertion of flux instrumentation into the reactor core.

Of the 52 nozzles, 21 had broken off and 14 had cracks in the region of the weld. In addition, 4 in-core instrument-guide-tube extensions were broken and 4 were cracked. The remaining 44 extensions were intact.

Examination revealed that, in addition to the damaged tube ends and tube-sheet welds, the inside surface of the steam generator head had nicks and scratches. The other steam generator was protected from severe damage by some temporary thermocouple instrumentation in its upper head. About half of the tube ends in the second steam generator had been damaged by the loose parts, but only about 10% of these required minor weld repairs. Nicks and scratches were found on the flow-guide vanes and on the bottom of the reactor vessel, as well as at the edge of the mating surfaces between the thermal shield and the lower grid assembly. The retention weld on each of the eight dowels at the lower edge of the thermal shield was broken; one of the dowels had even backed out about 3/4 in. Other minor scratches were found in the system, such as on the impellers of the reactor coolant pumps.[20,21]

The failures were found to be primarily due to hydraulically induced forces. Redesign involved increasing the natural frequency of the in-core instrument nozzles and guide tubes. A more rigid attachment of the thermal shield to the core barrel was also effected.[22]

### 3.12 Two Fatalities in Steam Line Accident at Surry 1 (Ref. 23)

The reactor in Unit 1 of the Surry Power Station is a pressurized-water reactor. The nuclear steam supply system was designed by Westinghouse Electric Corporation. The plant is located in Gravel Neck, Virginia, and is operated by the Virginia Electric & Power Company. Unit 1, which started operation in 1972, has a capacity of 882 MW(e). The accident described below is included in this report because it resulted in two fatalities.

The reactor in Unit 1 of the Surry Power Station had been shut down on July 26, 1972, but the system was kept in "hot-standby." In this condition, the only heat generated in the system is from the decay of radioactive fission products in the reactor and from the operation of the primary coolant pumps, with the latter being the main source of heat. Even with no other heat added to the primary system, the operation of the huge primary coolant pumps can bring the primary system up to full operating temperatures and pressures. In order to remove this heat, the secondary system must be in operation, but the steam that is generated is made to bypass the turbines and is sent directly to the condensers. Thus, even in the hot-standby condition with the reactor shut down, steam is generated in the secondary system.

The next day, July 27, the valves that were usually used to bypass the turbines (main turbine bypass valves) were scheduled for maintenance. The control room operator tried to get rid of the steam in the secondary system by opening the auxiliary valves that dump the steam into the atmosphere. However, the dump valves did not work, so three men went out to the building where the auxiliary valves were located to investigate the problem. The men made adjustments to three of these valves to no avail. The operator tried a fourth valve (decay-heat release valve), but because of the noise in the building, the men could not tell if it was open. When open, steam would pass through the valve and into a vent pipe that went through the roof of the building. To determine if the valve was open, one of the men went outside to see if steam was escaping from the pipe through the roof. It was not. However, as he was returning to the building, he heard and saw the steam suddenly blast out of the pipe. When he reentered the building, he saw that the upper level, where the valves and the two other men were located, was full of steam. He called for help on the nearest intercom, and the control room operator quickly closed the valve. But it was too late, as the two men had been badly scalded by the escaping steam. They somehow managed to get down to the lower level and were rushed to the hospital. Both men died 4 days later.

The valve that caused the accident had been operated successfully about 20 times previously. It was not the valve itself that was at fault, but the manner in which it vented the steam (see Fig. 3.8). When steam passed through the valve, it came out vertically through a 4-in.-diam extension. Inserted over this extension (but not attached) was the 8-in.-diam vent pipe that passed through the roof. There was a 4-1/8-in. overlap between the pipe and the extension; that is, the valve extension extended into the vent pipe a distance of 4-1/8 in. Escaping steam, under normal conditions, forced the valve back down so that the overlap was only 1/4 in. At the time of the accident, it was believed that the valve "hung up" and then suddenly opened, which caused it to move down further than usual for a few seconds. Thus, the valve extension was completely out of the vent pipe momentarily, which allowed steam to escape into the room.

In this incident there was no release of radioactivity, but there were two fatalities.

## 3.13 Seawater Intrusion into Primary System at Millstone 1 (Refs. 24–26)

The reactor in Unit 1 of the Millstone Nuclear Power Station is a boiling-water reactor. The nuclear steam supply system was designed by the General Electric Company. The station is located in Waterford, Connecticut, and is operated by the Northeast Nuclear Energy Company. Unit 1, which started operation in 1970, has a capacity of 660 MW(e). The incident described below is included in this report because it was a precursor to a potentially serious accident.

The steam from a boiling-water reactor goes directly to the turbine (see Fig. 3.9). As in any steam plant, the steam is condensed back to water when it leaves the turbine, and in a boiling-water reactor the condensed water is pumped back into the core of the reactor. As was mentioned previously, a large system must be used to carry away the heat involved in the process of condensation without creating environmental problems, and in the case of Millstone 1 water from the Atlantic Ocean is used for this purpose.

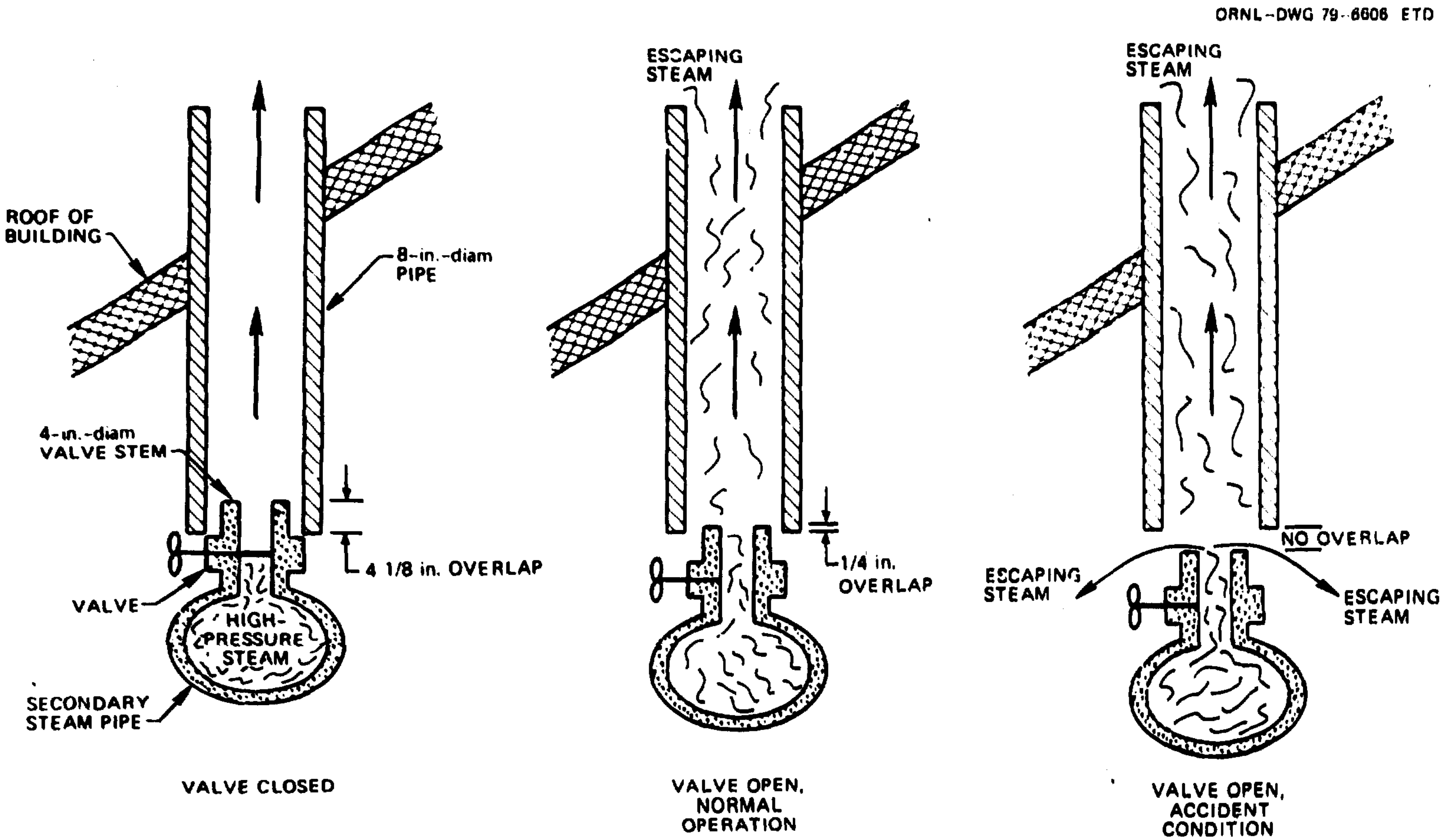


Fig. 3.8. Decay-heat release valve under various conditions.

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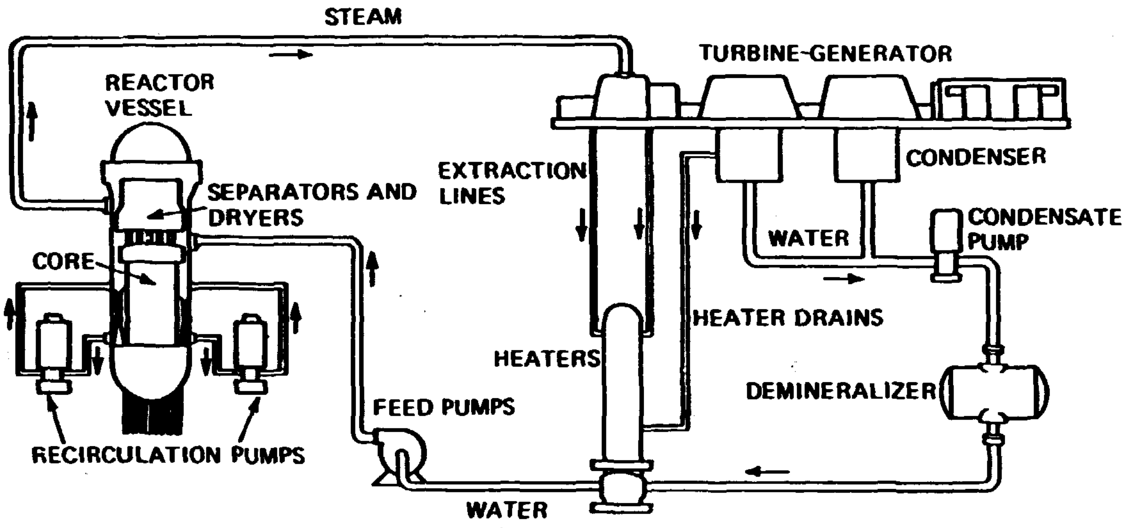


Fig. 3.9. Schematic diagram of steam system in a boiling-water reactor.

On Sept. 1, 1972, the Millstone 1 reactor was undergoing a routine startup. The reactor had achieved less than 0.1% of full power when the operator noted a problem with a demineralizer (see Fig. 3.9; the condensed water passes through a demineralizer before going back to the core). He switched to a second demineralizer and proceeded with the startup. Half an hour later it became apparent that there was also a failure in the second demineralizer, so the operator began a deliberate reactor-shutdown procedure. An hour later when excessive chlorides were noted in the primary water, the deliberate shutdown procedures were abandoned and the reactor was scrammed. Seawater had entered the primary system.

An intensive investigation revealed that the tubes in the condenser had corroded, allowing seawater to enter the primary system. The main damage was to the instruments that measure the power of the reactor (local power-range monitors). All 120 of them had failed.

All of the aluminum-brass tubes in the condenser were replaced by copper-nickel tubes, which have a higher resistance to seawater corrosion. The instruments were replaced, and the reactor eventually resumed operation.

There were no injuries, and no radioactivity was released.

### 3.14 Fracture of Shaft of Main Reactor Coolant Pump at Surry 1 (Ref. 27)

The reactor in Unit 1 of the Surry Power Station is a pressurized-water reactor. The nuclear steam supply system is a Westinghouse Electric Corporation design. The station is located in Gravel Neck, Virginia, and is operated by the Virginia Electric & Power Company. Unit 1, which began operation in 1972, has a capacity of 882 MW(e). The incident described below is included in this report because it resulted in damage to major equipment.

In November 1973, with the reactor in Surry 1 operating at 95% of full power, the operator noted a reduction of water flow from one of the primary coolant pumps, accompanied by excessive vibration in the same pump. He initiated a controlled shutdown of the reactor.

When the power level had dropped to 40% of power, an automatic scram was set off by the complete loss of flow from the pump. The pump shaft had broken.

An examination of the disassembled pump revealed that the break had occurred at a place where the shaft diameter was reduced. The radius machined in this area was too small, resulting in excess stress. The shafts of all main reactor coolant pumps were replaced with shafts of a different design, and the reactor was returned to service.

There were no injuries, and no radioactivity was released.

### 3.15 Inadvertent Criticality During Refueling at Vermont Yankee[28]

The reactor in the Vermont Yankee Nuclear Power Station is a boiling-water reactor. The nuclear steam supply system was designed by General Electric Company. The station, which has a capacity of 514 MW(e), is located in Vernon, Vermont, and is operated by the Vermont Yankee Nuclear Power Corporation. It began operation in 1972. The incident described below is included in this report because it was an inadvertent criticality.

On Nov. 7, 1973, the reactor in the Vermont Yankee Nuclear Power Station was undergoing final tests following a normal refueling. The head of the pressure vessel was off. Two of the tests involved were core-verification tests and control rod friction-timing tests. In order to perform both tests concurrently, jumpers were used on some of the electrical control systems. This was contrary to approved procedures. Although their use was authorized for the core-verification tests, they were not authorized for the friction-timing tests because the jumpers nullified an interlock system that prevented the simultaneous withdrawal of two or more control rods. Apparently, the operators were not aware of the full scope of the nullification.

During the friction-timing tests, a control rod was withdrawn, but it was inadvertently left in the fully withdrawn position. The operator selected a control rod for testing that was adjacent to the one that was fully withdrawn. As the selected rod was withdrawn, the operator noted a rapid increase in power. He immediately inserted the rod and, simultaneously, a full scram occurred automatically. The reactor had gone critical, but it was quickly shut down by the inserted rods.

Workers in the area were examined for radiation exposure but none was found. An examination revealed no damage anywhere. The reactor power had barely begun to rise when the scram occurred.

There were no injuries, and no radioactivity was released in this incident.

### 3.16 Operator Sucked Through Manhole into Containment at Surry 2 (Ref. 29)

The reactor in Unit 2 of the Surry Power Station is a pressurized-water reactor. The nuclear steam supply system is a Westinghouse Electric Corporation design. The station is located in Gravel Neck, Virginia, and is operated by the Virginia Electric & Power Company. Unit 2, which began operation in 1973, has a capacity of 882 MW(e). The incident described below is included in this report because it resulted in an injury.

The reactor containment building which houses the primary system in a pressurized-water reactor is maintained at an air pressure less than that of the outside atmosphere. Airflow through cracks or leaks in the building is therefore always from outside to inside. This prevents the escape to the atmosphere of any gaseous radioactivity that might leak from the reactor through these cracks or leaks. Air that is pumped out of the containment building must pass through a filtering system, and this system prevents the escape of radioactive particles in the air that is pumped out.

An air lock at Surry 2 allows personnel to enter the containment building for maintenance. It consists of an outer door and an inner door. In order to enter the containment building through the air lock, one pressurizes the passageway between the two doors up to normal atmospheric pressure, opens the outer door, and enters the passageway; next, one closes the outer door, depressurizes the passageway to the low-air-pressure level of the containment building, opens the inner door, and enters the building. The inner door contains an 18-in.-diam manhole-type emergency-exit hatch.

On the evening of Dec. 10, 1973, with the reactor operating at about 85% of full power, a signal indicating a slight increase in pressure in the containment building was received in the control room. An operator was sent to the air lock to check for leaks. About an hour later an alarm signaled that the pressure in the containment building had increased further and that the pressure was continuing to rise.

An orderly shutdown of the reactor was started, and two more operators were sent to investigate. They found the outer door to the air lock wide open and the emergency hatch cover on the inner door also open. They could not see the missing operator through the hatch, so they assumed that he was in the containment building doing maintenance and had simply been careless in leaving the outer door and inner hatch open. They resealed the air lock and waited. About a half hour later they observed that the missing operator had opened and crawled through the emergency hatch on the inner door, but that he could do no more because of injuries. The reactor was scrammed, but with the hatch on the inner door now open, the air lock passageway could not be pressurized to allow the opening of the outer door. The air pressure in the passageway was at the low-level pressure of the containment building. Workmen could not even pry the outer door open against the forces caused by the difference in pressure between the outside and inside (~4 psi). The entire containment building had to be pressurized in order to equalize the pressure, and this procedure took about a half hour. The injured operator was rescued and rushed to the hospital by helicopter. He had sustained serious injuries, and his condition was critical.

What had happened was that he had entered the air lock but forgot to close the outer door and depressurize the passageway. While he was checking the 18-in.-diam escape hatch on the second door for leaks, it had opened suddenly, whereupon he was sucked through it and flew 20 ft through the air and crashed against a crane. Although his injuries were serious, he recovered and returned to work. This accident resulted in one injury, but there was no release of radioactivity.

### 3.17 Malfunction of Pressurizer Relief Valve at Beznau 1 (Ref. 30)

The reactor in Unit 1 of the NOK Nuclear Electric Generating Station (known as Beznau 1) is a pressurized-water reactor. The nuclear steam supply system was designed by the Westinghouse Electric Corporation. The station is located in Beznau, Switzerland, and is operated by the Nordostschweizerische Kraftwerke AG (NOK). Unit 1, which started up in 1970, has a capacity of 350 MW(e). The accident described below involved a pressurizer relief valve that stuck in the open position and led to a misleading indication to the operators that the primary coolant system was full of water. The description is included in this report because this accident is considered to be a precursor to a more serious accident, such as that which occurred at Three Mile Island in March 1979 (see *Three Mile Island: A Report to the Commissioners and to the Public*, Vol. 1, by the Nuclear Regulatory Commission Special Inquiry Group, Mitchell Rogovin, Director).

The pressurizer in a pressurized-water reactor is a tank that is connected directly to the primary water piping system by a pipe attached to the bottom of the pressurizer. Under normal conditions, the pressurizer contains water in the bottom (and also in the connecting pipe) and steam in the top. The trapped steam at the top is under pressure, and this pressure is transmitted through the water at the bottom of the pressurizer and through the connecting pipe to the primary system. There is a pipe (line) connected to the top of the pressurizer that curves down and is connected to a drain tank. There are two valves in this line: a "block valve," which is normally open, and a pressure-relief valve, which is normally closed. The block valve serves as a backup in case the pressure-relief valve fails. If the pressure in the pressurizer gets too high, the relief valve opens and relieves the pressure. The steam or water-steam mixture that comes out of the pressurizer passes through the open block valve and through the opened relief valve and into the drain tank. It normally takes only a few seconds to relieve the pressure. If the relief valve fails to close, the block valve can be used to close (or open) the line. If it, too, should fail to open, safety valves, installed in a parallel pipe, should open. In some systems, there are two separate pressure-relief lines that are connected to the top of the pressurizer. Each has a block valve and a pressure-relief valve (four valves in all, not including safety valves). If the pressure gets too high, both relief valves open. This is the design, more or less, at Beznau 1.

On Aug. 20, 1974, with the reactor in Beznau 1 operating at full power, a disturbance occurred in the external electrical grid network. One of the two turbines shut down, but a valve that should have bypassed this turbine (turbine bypass valve) did not function. This malfunction caused the pressure in the steam generator to increase, which reduced its cooling capability, whereupon the temperature and pressure in the primary system began to rise. Both relief valves in the pressurizer opened, but a few seconds later only one of them closed, whereas both should have closed. With one valve open, steam and fluid began to flow through the open valve and into the drain tank, and the pressure in the primary system began to drop. The reactor scrammed, and the second turbine shut down. It was less than 1 min into the incident.

The pressure in the primary system continued to fall until saturation was reached, at which point some of the water turned to steam. This caused the pressurizer to fill completely with water, and now water and steam poured out of the pressurizer through the stuck-open relief valve and began to fill the drain tank. The operator now realized that the relief valve was stuck open, so he closed the block valve. It was then 3 min since the incident began. It should be noted that *under normal conditions* if the water-level indicator for the pressurizer shows that the pressurizer is full of water, this indicates that the entire primary system is full of *water* (a "solid system"). However, *at saturation* (i.e., when the water in the system is turning into steam), this is not the case; instead, when the indicator shows that the pressurizer is full of water, this means that the entire primary system is full of a *mixture of steam and water*. Therefore, the same reading indicates two different conditions and could be misleading.

With the block valve closed, the pressurizer was now sealed, whereupon the pressure in the primary system increased and the steam bubbles therein collapsed. At about this time, the rupture seal on the drain tank blew, allowing primary water to spill onto the floor of the containment building. This spillage soon stopped because the closed block valve prevented further escape of water from the pressurizer. An auxiliary system (high-pressure safety injection system) turned on and pumped water into the primary system. A few minutes later, this system was turned off manually because conditions had stabilized. It was then 12 min since the incident began. Procedures to bring the reactor to a cold shutdown were initiated.

Subsequent investigation revealed that the relief valve which had stuck open had broken loose from the line. There was also minor damage to the line and some radioactive contamination in the containment building that had come from the discharge of primary water through the rupture disk of the drain tank.

There were no injuries, and no radioactivity was released.

### 3.18 Electrical Cable Fire at Browns Ferry 1 (Refs. 31, 32)

The reactor in Unit 1 of the Browns Ferry Nuclear Power Plant is a boiling-water reactor. The nuclear steam supply system was designed by the General Electric Company. The plant is located near Decatur, Alabama, and is operated by the Tennessee Valley Authority. Unit 1, which began operation in 1973, has a capacity of 1065 MW(e). The incident described below is included in this report because it resulted in damage to major equipment.

A lighted candle that was used to test for leakage of air around penetrations through a concrete wall at the Browns Ferry Nuclear Power Plant started a fire that caused damages estimated at $10 million and shut the plant down for over a year.

As background to this event, a brief description of the pertinent components is given here. In any nuclear plant, there are thousands of electrical wires that are used to transmit the information required to determine the status of the plant and to exercise control. In order to create some order and to aid in the identification of so many wires, they are grouped together into small bundles that form cables, each of which is labeled. The cables themselves are grouped together (for the same purpose) and laid in long, shallow metal troughs called cable trays. Each tray is labeled. There are approximately 30 cables to a tray. The trays pass over each other in parallel or criss-cross fashion, but in accordance with specific design criteria.

The fire at Browns Ferry started at a point where several trays passed through a concrete wall of the containment building. The air pressure on the inside of the wall is kept lower than that on the outside so that air leaking through any small cracks in the wall would pass from outside to inside. Therefore, any release of airborne radioactivity inside the containment building from the reactor would not be able to escape, through small leaks, to the outside atmosphere, since all airflow through these leaks would be from outside to inside.

The cable tray penetrations through the concrete were from a room called the cable-spreading room. It was crowded with cable trays as shown in Fig. 3.10. Each cable tray penetration through the concrete is sealed with polyurethane foam to minimize inleakage of air. Several new penetrations had been made but were inadequately sealed, and two workmen were crawling around inside the cable spreading room looking for leaks. A candle was used because a leak allows air to swish through from this room to the reactor containment, and the rushing airs bends the flame of the candle toward the leak.

Such a leak was found in a corner of the room by a workman who was lying on his side using a candle and a flashlight, since lighting was

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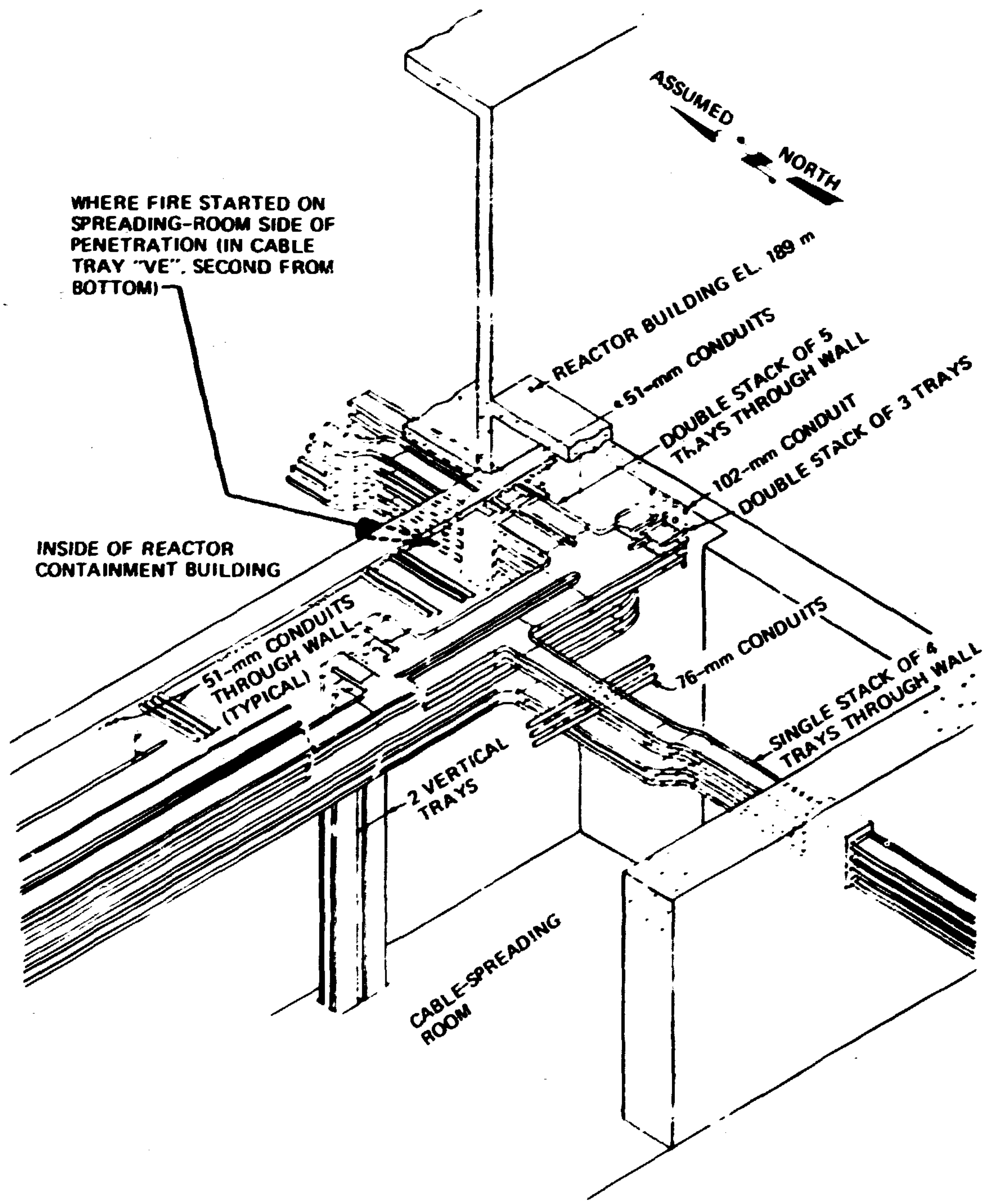


Fig. 3.10. Area of fire at Browns Ferry Nuclear Power Station (51 mm = 2.04 in.; 76 mm = 3.04 in; 102 mm = 4.08 in.; 189 m = 620 ft).

poor. The time was 12:15 PM on Mar. 22, 1975. The workman had stuffed some sheet polyurethane (which is more flammable than the foam) into the hole and again tested for leaks with the candle. The leak persisted, but this time the horizontal flame of the candle ignited the sheet polyurethane, and the burning sheet polyurethane, in turn, ignited the foam. The air rushing into the hole spread the fire into the hole away from the workmen so that they could not extinguish it with hand-held dry-chemical extinguishers. An alarm was sounded and the cable-spreading room was evacuated. The fire was smouldering and beginning to spread slowly by burning the insulation on the cables on both sides of the wall. Workmen went into the reactor containment building to try to extinguish the fire and found that the penetration on the inside of the wall was about 20 ft above the floor level. Using a ladder to reach the fire, a workman discharged a dry-chemical fire extinguisher on the smouldering tray, but he was forced to withdraw because of smoke. Smoke made visibility difficult, and it curtailed fire-fighting efforts even after airspace breathing apparatus was obtained.

In the meantime, both Unit 1 and Unit 2 were shut down, but it was still necessary to cool the reactors because of the residual decay heat. After both reactors were shut down, supplying makeup cooling water to Unit 1 was complicated because the fire in the electrical cables had caused a number of pieces of equipment to lose some or all of their capabilities. There was an adequate supply of high-pressure makeup water available at all times to keep the fuel covered in Unit 2. Heat removal throughout the course of the fire was not critical because relief valves to transfer the decay heat to the suppression pool were available.

The fire damaged the control arrangement for the main-steam-line isolation valves in Unit 1, and the valves closed and could not be reopened. The decay heat was removed for a time, using automatic operation of the relief valves with the reactor remaining at high pressure. However, the fire had also affected the two primary high-pressure makeup water systems provided for maintaining the water level in an emergency. Therefore, the operator chose to depressurize the reactor by remote control of the relief valves and to use the low-pressure makeup water systems that were still available.

During the period of depressurization, the water level in the core dropped to 4 ft above the fuel but never fell below this level. (Normal level is about 17 ft; the 4-ft level is still 2.5 ft above the level at which additional emergency cooling systems are actuated.) Once the reactor pressure was reduced below 350 psi, a condensate booster pump provided an adequate source of makeup water, and the normal water level was attained. However, when stable low-pressure operation was attained, the operability of the relief valves being used to maintain low pressure was lost as a result of the loss of control air. The relief valves closed, pressure increased, and the availability of the low-pressure makeup water systems was lost. After about 3 1/2 hr, operability of the relief valves was reestablished and low-pressure operation restored.

With low-pressure operation restored, adequate makeup water could be supplied by one of the condensate pumps. Also, two additional condensate booster pumps and two additional condensate pumps were available to the operator.

Another alternative would have been to use a nonstandard system configuration and manual valve alignment. Two residual-heat-removal pumps in Unit 2 could have been aligned to the Unit 1 reactor through a cross-tie pipe, and, as an additional backup, river water could have been used from either of two available service-water pumps.

The point is that an adequate supply of cooling water was provided throughout the incident, and additional alternatives could have been used to provide makeup water with the reactor at either high or low pressure.

The fire in the cable-spreading room was finally extinguished about 4:20 PM by dry chemicals. The fire in the reactor containment was extinguished at about 7:45 PM by water. The use of water had been suggested much earlier by the chief of the Athens, Alabama, fire department, which had been called to assist in fighting the fire. It was not used earlier because of the danger of electrocution to fire-fighting personnel and the probability of causing additional short circuits, which might have further impaired the cooling of the reactor.

The fire was declared out at 7:45 PM. When the smoke began to clear away and reliance on breathing apparatus decreased, a more orderly approach to obtaining shutdown cooling could be taken. The actual valve conditions (opened or closed) were determined, and control power to motor operators, pump controls, etc., was established, using temporary jumpers. At 4:10 the next morning, normal shutdown cooling was established.

Although the fire was costly in time and money, there was no release of radioactivity, and the core was covered at all times.

### 3.19 Seal Failure in Main Coolant Pumps at Robinson 2 (Refs. 33, 34)

The reactor in Unit 2 of the H. B. Robinson Plant is a pressurized-water reactor. The nuclear steam supply system is a Westinghouse Electric Corporation design. The plant is located in Hartsville, South Carolina, and is operated by the Carolina Power & Light Company. Unit 2, which began operation in 1970, has a capacity of 700 MW(e). The incident described below is included in this report because it resulted in significant recovery costs.

The primary coolant pumps that are used in pressurized-water reactors are huge affairs, over three stories high. The operation of the pumps alone generates a great deal of heat. There are multiple-stage seals around the main shafts of these pumps in order to prevent them from leaking. These seals must be kept from becoming too hot, and so a series of cooling-water lines is passed around each seal. Some of the associated plumbing for these lines are interconnected.

In May 1975, while the reactor was at full power, an alarm was set off by the leakage of the first-stage seal of one of the pumps. The operators started a controlled shutdown of the reactor. When the power level had been reduced to 36% of full power, the operators shut off the troublesome pump. Shortly thereafter the reactor scrammed automatically on a signal from one of the steam generators. The reason for this scram is not clear. In the meantime, the broken first-stage seal on the first primary coolant pump led to the loss of coolant water to the seals of the remaining primary coolant pumps. They were then shut down also.

At this stage it was necessary to maintain circulation of primary water through the reactor. Since the coolant water to the seals of the remaining primary coolant pumps was blocked, the operator restarted the first pump. It still had two seals which appeared to be intact, and coolant water was available to them. It operated for about 2 hr, but then the remaining seals collapsed, and primary coolant water began to spill into the containment building at a high rate. Circulation of water through the primary system was maintained by auxiliary means until cold shutdown was achieved and the leak in the primary pump was plugged. By this time about 135,000 gal of primary water had escaped, covering the floor of the containment building to about a foot in depth.

Escape of radioactivity to the atmosphere was contained within prescribed limits by the filtering system for the containment building. There was no radioactivity released by the fuel. The systems which had failed were eventually repaired, and the reactor was returned to service.

There were no injuries, and the release of radioactivity was within prescribed limits.

### 3.20 Hydrogen Explosion at Cooper Injures Two[35,36]

The reactor in the Cooper Nuclear station is a boiling-water reactor. The nuclear steam supply system is a General Electric Company design. The station, which has a capacity of 778 MW(e), is located in Brownville, Nebraska, and is operated by the Nebraska Public Power District and the Iowa Power & Light Company. It began operation in 1974. The accident described below is included in this report because it resulted in injuries.

On Nov. 5, 1975, with the reactor at the Cooper Nuclear Station operating at 60% of power, the pressurization of a sump* located in an auxiliary building near the off-gas stack was noticed. Men were sent to

*A sump is a depressed area or a basement room that is used to receive drainage.

investigate. While one of them was lifting the manhole cover leading to the sump, another flicked on an air sampler. Apparently a small spark occurred, for there was an explosion which seriously injured one man and caused minor burns to another. The operators initiated a controlled shutdown of the reactor. The clothing of the injured man had become slightly contaminated with radioactivity (~3 mrems/hr at contact), and six other men in the vicinity received minor exposure (~15 mrems).

An investigation revealed that a valve in the air stream leading from the steam-jet sparger system at the main condenser (described in more detail in Sect. 3.24) had been left closed. This diverted the hydrogen, oxygen, and steam in this air stream through the sump and then back through the off-gas system. The control light on the control room console showed this valve to be open. Wiring changes that were to be made at a future date had been made without the knowledge of the operators. In addition, the workman who had done the wiring had set this valve open, as indicated by a notch on the stem of the valve. Although from the exterior the valve appeared to be open, it was in fact closed. The butterfly valve gate inside the valve was not in line with the notch on the stem of the valve.

Although there was some ground-level release of radioactivity at the site of the explosion, there was no indication of abnormal conditions outside the site boundary. Repairs were made, the valve and rewiring were corrected, and the reactor was returned to operation.

There were two injuries and a release of some radioactivity at ground level in the vicinity of the explosion.

### 3.21 Explosion Destroys Off-Gas Building at Cooper[37,38]

The reactor in the Cooper Nuclear Station is a boiling-water reactor. The nuclear steam supply system is a General Electric Company design. The station, which has a capacity of 778 MW(e), is located in Brownville, Nebraska, and is operated by the Nebraska Public Power District and the Iowa Power & Light Company. It began operation in 1974. The incident described below is included in this report because it resulted in damage to major equipment.

On Jan. 7, 1976, two months after the hydrogen explosion that injured two men, the reactor at the Cooper Nuclear Station was operating at 83% of full power when an alarm indicated that the airflow from the off-gas stack was reduced. A dilution fan that helped force the air up through the stack did not appear to be operating properly. A second fan was turned on automatically because of the reduced airflow, and the first one was shut down by the operators, but the airflow did not increase.

The shift supervisor and an operator went out to examine the auxiliary off-gas building and noted that the radioactivity in the building was higher than normal and that the air pressure within the building was not at its usual low level. The air pressure inside the building is kept lower than the normal atmospheric pressure for the same reason that the air pressure is kept low in the reactor containment building.

The supervisor and operator then went up to the top of the off-gas stack but found no indication of a problem. They returned to the off-gas building. When they entered, they noticed an unusual odor; they also noted that the radioactivity meters were reading off scale. They evacuated the building forthwith, and shortly thereafter the building exploded. It was completely demolished. The reactor was shut down immediately.

An investigation revealed that an ice plug had formed inside the 325-ft-high off-gas stack at the top, producing a back-pressure which the systems below were not designed to cope with and which permitted the release of hydrogen in the off-gas building. A spark from the machinery inside the building probably set off the explosion. Although there was some release of radioactivity in the vicinity of the explosion, there was no indication that it had reached the site boundary.

The off-gas building was rebuilt, the upper portions of the stack were lined with heaters, other modifications were made, and the reactor was started up 11 days later.

No injuries were incurred, but there was some release of radioactivity in the vicinity of the explosion. Subsequent calculations indicated that the levels of radioactivity at the site boundary were below maximum permissible concentrations throughout the accident.[39]

## 3.22 Unplanned Criticality During Refueling at Millstone 1 (Ref. 40)

The reactor in Unit 1 of the Millstone Nuclear Power Station is a boiling-water reactor. The nuclear steam supply system was designed by the General Electric Company. The station is located in Waterford, Connecticut, and is operated by the Northeast Nuclear Energy Company. Unit 1, which started up in 1970, has a capacity of 660 MW(e). The incident described below is included in this report because it was an inadvertent criticality.

In the latter part of 1976 a shutdown-margin test was being performed in the midst of a normal refueling at Millstone 1. This test is mandated to ensure that the control rods can shut down the reactor with an adequate margin of safety at all stages while fuel is being loaded in the core. Among other things, it consists of withdrawing a specified control rod part way and then completely withdrawing the control rod which is diametrically opposite it in the core. Under these conditions, the reactor should remain subcritical, and this ensures the required margin of safety.

In performing this test, the operator withdrew the wrong rod part way and started to withdraw the correct rod, which happened to be adjacent to the wrong rod, all the way. The reactor went critical but was scrammed immediately and automatically. The supervisor was called, and the whole procedure repeated, mistake and all, but this time the operator hastily reinserted the correct rod before the reactor could scram automatically. The error was discovered, and reloading continued normally. A plant feature (rod worth minimizer) that would have automatically prevented the withdrawal of control rods out of sequence was available, but it had not been programmed for these tests and therefore was not in use.

Investigation revealed no damage to the fuel nor exposure to the men. There were no injuries, and there was no release of radioactivity.

## 3.23 Short Causes Instrument Failures at Rancho Seco 1 (Ref. 41)

The reactor in Unit 1 of the Rancho Seco Nuclear Generating Station is a pressurized-water reactor. The nuclear steam supply system was designed by the Babcock & Wilcox Company. The station is located in Clay Station, California, and is operated by the Sacramento Municipal Utility District. Unit 1, which began operation in 1974, has a capacity of 918 MW(e). The incident described below is included in this report because it was a precursor to a potentially more serious accident.

The initial stages of the incident at Rancho Seco had characteristics that were similar to those of the accident at Three Mile Island 2 on Mar. 28, 1979. The most important difference was that the pressure-relief valve at the pressurizer did not stick in the open position, as it did at Three Mile Island.

The main components of the system are illustrated in Fig. 3.11. Note that only one of the two steam generators in the system is illustrated. The description below is based on a reconstruction that was made following the incident, because the loss of power during the incident caused various degrees of instrument failure, which invalidated the record of the time sequence of events.

On Mar. 20, 1977, with the reactor operating at about 70% of power, an operator was in the process of replacing a burned-out light bulb on a control console. The bulb was in a mounting bracket behind the front panel of the console. The bracket contained several other bulbs. When the mounting bracket was loosened and turned over to facilitate the replacement, a loose bulb dropped into the electrical circuitry below. A short circuit was created that cut off the power to about two-thirds of the nonnuclear instruments. This was a common-mode failure. The nonnuclear instruments indicate pressures, temperatures, water levels, water flow, etc., in the system (nuclear instruments indicate reactor power, control rod position, etc.). The loss of power to these nonnuclear instruments causes them to read zero, read incorrectly, or read erratically. Furthermore, the plant has an integrated control system (ICS) which operates valves, etc., automatically to match instrument readings to plant requirements and which will not operate correctly when incorrect signals are received.

On the basis of erroneous signals produced by the short circuit, the ICS shut off the secondary feedwater supply to the steam generators

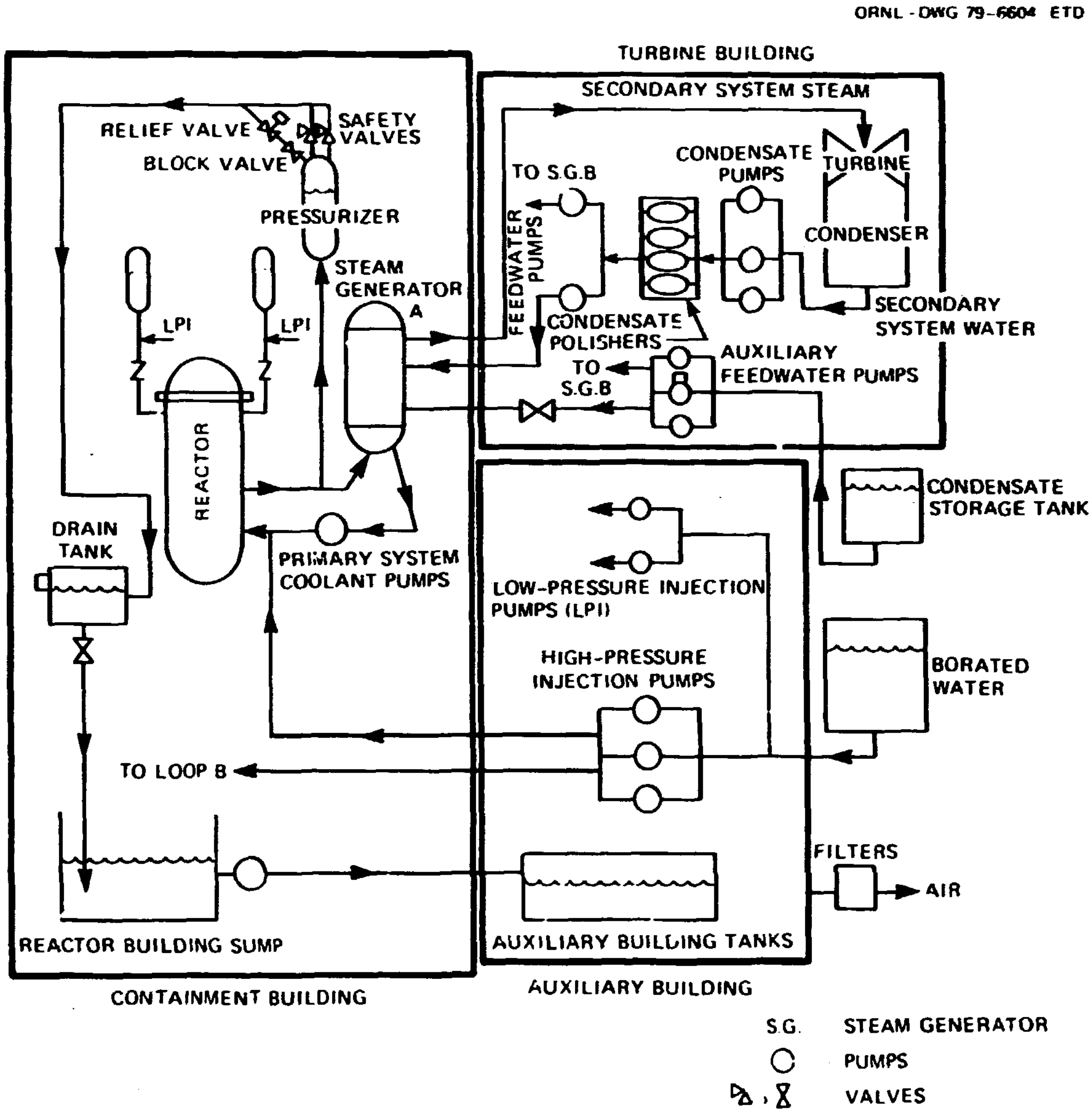


Fig. 3.11. Typical flow system for pressurized-water reactor. (Courtesy of the American Institute of Physics.)

(Fig. 3.11, turbine building). In pressurized-water reactors, the primary system water, heated by its passage through the reactor, is cooled by its passage through the steam generators. Essentially, the heat generated by the reactor is transferred to the secondary system via the primary system. The heat transfer takes place in the steam generators as the primary system water and secondary system water, separated by piping, flow past each other. In the process, the secondary system water turns to steam.

With no water flowing through the steam generators, the water temperature and pressure in the primary system began to rise, and the reactor was scrammed automatically. The pressure remained stable because the relief valve at the top of the pressurizer operated properly; that is, it opened to relieve the pressure, then closed. In fact, it activated at a lower pressure than it should have, and this helped to keep the pressure within reasonable limits. In the meantime, auxiliary feedwater pumps had turned on (Fig. 3.11, turbine building) automatically to supply circulating water to the steam generators. However, the valves between these pumps and the steam generators were closed by the ICS because of incorrect signals. This closure prevented the steam generators from receiving the much needed water, and they soon dried up. Apparently, the temperature of the primary system water was kept at reasonable values by the normal and continuous injection of cool makeup water, the system for which is not shown in the figure, and also by the fact that the pressure of the primary system was being controlled by the relief valve that was actuating at lower pressure than was intended.

Secondary-water-level indicators for each steam generator began drifting in opposite directions: one indicated that one steam generator was becoming full, and the other indicated that the second steam generator was emptying. As one of the water-level indicators drifted down, it activated a system which opened one of the valves, previously closed by the ICS system, and allowed water to flow into one of the steam generators. About 9 min had elapsed since the incident started.

With water beginning to enter the steam generator, the water in the primary system began to cool and the pressure began to drop. When the pressure reached 1600 psig, two things happened automatically (and correctly): (1) water from an auxiliary system (high-pressure injection system) began to flow into the primary system, and (2) the other valve, previously closed by the ICS system, was opened and permitted water to flow into the other steam generator.

The temperature indications for the water in the primary system did not seem to be reliable, but the operator maintained pressure as well as possible by manipulating the pressure-relief valve at the pressurizer and by controlling the flow of cold water into the primary system by means of the auxiliary system (high-pressure injection system).

Seventy minutes later, all of the electrical circuits were restored, and the instruments began to respond correctly. It turned out that the water-level controls in the steam generators had failed, and the steam generators had been flooded up into the steam lines. This created a large heat sink for the water in the primary system, and it cooled beyond expectations. With the instruments reading properly, control over the system was readily obtained.

The primary and secondary systems were thoroughly examined for damage, but none was found. Four days after the incident the rise to full power was started.

There were no injuries, and no radioactivity was released.

### 3.24 Fire in Off-Gas System at Browns Ferry 3 (Ref. 42)

The reactor in Unit 3 of the Browns Ferry Nuclear Power Plant is a boiling-water reactor. The nuclear steam supply system is a General Electric Company design. The plant is located near Decatur, Alabama, and is operated by the Tennessee Valley Authority. Unit 3, which began operation in 1976, has a capacity of 0.65 MW(e). The incident described below is included in this report because it was a precursor to a potentially more serious accident.

In a boiling-water reactor, the air-filtering system in the pathways that lead to the stack (off-gas stack) has two main functions: the first is to reduce the level of radioactivity that might be released, and the second is to prevent the accumulation of hydrogen beyond its explosive concentration.

The first is accomplished by passing the air through a 6-hr-holdup piping system, which permits the short-lived radioactive gases to decay. Then it is passed through a high-efficiency particulate air prefilter and through large beds of charcoal, which adsorb and further delay the radioactive gases. Next, the air passes through an "after filter," after which it is routed out through the off-gas stack for release to the atmosphere. The charcoal beds are kept in an auxiliary building called a vault.

The hydrogen that is of concern comes from the radiolytic decomposition of the water of the primary system. Steam-jet spargers are used in the main condensers below the turbines to remove the hydrogen, the oxygen, and any radioactive gases that are dissolved in the primary system water. The hydrogen and oxygen are produced from the radiation-induced disassociation of the water as it passes through the reactor. This stream of hydrogen, oxygen, any radioactive gases, and steam is preheated and then passed through a catalytic recombiner. The stream is then chilled (to remove the condensable gases), reheated, and passed into the system described above, where it is delayed, etc.

On July 15, 1977, a startup sequence was begun at Unit 3 of the Browns Ferry Nuclear Power Plant. Over the following 2-day period, problems developed in the off-gas stream filtering system, and the concentration of hydrogen became greater than its combustible limit. Various attempts were made to stabilize the system, but they were unsuccessful. The temperature of the charcoal beds and the vault increased throughout the second day, and finally the air stream was diverted from the vault to a bypass system. A stream of nitrogen was passed through the vault to cool the charcoal and to extinguish a fire if one were present.

Subsequent examination revealed that a blocked drain in the air system had prevented the air stream from being preheated prior to entering the recombiner. As a result, the catalyst in the recombiner had become ineffective, and a high concentration of hydrogen passed downstream to the vault. It apparently exploded there and ignited the charcoal beds. The purging of the vault by nitrogen had extinguished the fire.

There were no injuries, and there was no release of radioactivity.

### 3.25 Stuck Pressurizer Relief Valve at Davis-Besse 1 (Refs. 43, 44)

Unit 1 of the Davis-Besse Nuclear Power Station is a pressurized-water reactor. The station is located at Oak Harbor, Ohio, and is operated by Toledo Edison Company. Commercial operation began in 1977. The system was designed by Babcock & Wilcox Company. Unit 1 has a capacity of 906 MW(e). The accident described below involved a pressurizer relief valve that stuck in the open position and led to a misleading indication to the operators that the primary coolant system was full of water. The description is included in this report because this accident is considered to be a precursor to a more serious accident, such as that which occurred at Three Mile Island in March 1979 (see *Three Mile Island: A Report to the Commissioners and to the Public*, Vol. 1, by the Nuclear Regulatory Commission Special Inquiry Group, Mitchell Rogovin, Director).

On Sept. 24, 1977, with the reactor in Davis-Besse 1 about 10% of its full power (but not generating electricity), a system associated with the feedwater to the steam generator partially shut down. The cause of the initiating event is unknown. This "half trip" caused a valve to close the water supply from the condenser to one of the steam generators. Figure 3.12 is a schematic of the system.

The water in the steam generator began to boil away and eventually the steam generator went dry because its replacement supply of water was cut off. As the water level in the steam generator dropped, its capacity to remove heat from the primary system water, which passes through and cools the reactor, was reduced. As a result, the temperature and pressure of the water in the primary system began to rise. When the pressure reached the value for which the pressure-relief valve in the pressurizer was set to open, it opened but did so in an oscillatory manner; that is, it opened and closed nine times and finally stuck in the open position. (A similar valve on the pressurizer at Three Mile Island stuck open but not necessarily for the same reasons.) The oscillations had damaged the valve. Steam escaped from the pressurizer to the quench tank, but, with the pressurizer relief valve stuck open, the pressure in the quench tank became too high, and its rupture disk blew out, allowing steam to escape into the containment building. At some time during this pressure excursion, the operator shut the reactor down.

Six minutes after the reactor was shut down and began to cool and lose pressure, the water reached the saturation condition, and, as at Beznau (Sect. 3.17), there was a surge of steam in the primary system,

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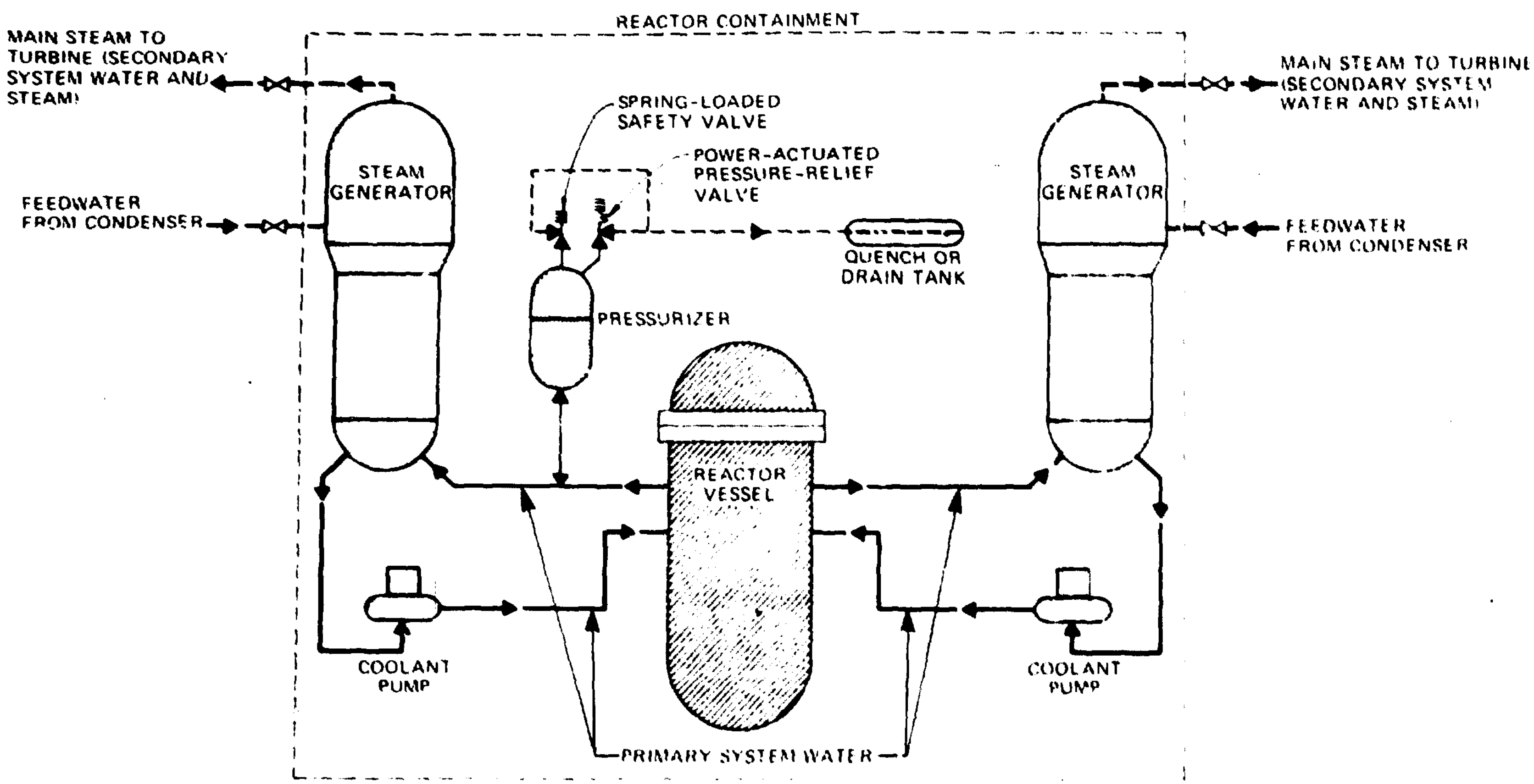


Fig. 3.12. Schematic diagram of a typical nuclear steam supply system in a pressurized-water reactor.

which sent the water level in the pressurizer to its maximum value.* The primary system continued to lose pressure, and about 20 min after the onset of the event the operators determined that the pressurizer relief valve was stuck open. They isolated it by closing a block valve, which collapsed the steam bubbles in the primary system, and then the pressure stabilized. Subsequently, they were able to achieve cold shutdown.

During the investigation that followed, it was determined that a relay which is used to hold the pressurizer relief valve open until there is a 50-psi drop in pressure was missing. The use of this relay prevents oscillations of the kind that had damaged the valve.

The relief valve in the pressurizer was repaired, the missing relay was installed, other equipment that had not operated properly was modified, and the reactor was returned to service. There were no injuries, and no radioactivity was released.

### 3.26 One Injured in Hydrogen Explosion at Millstone 1 (Ref. 45)

> The reactor in Unit 1 of the Millstone Nuclear Power Station is a boiling-water reactor. The nuclear steam supply system was designed by the General Electric Company. The plant is located in Waterford, Connecticut, and is operated by the Northeast Nuclear Energy Company. Unit 1, which started operation in 1970, has a capacity of 660 MW(e). The accident described below is included in this report because it resulted in an injury.

At 9:30 AM on Dec. 13, 1977, with the reactor at Millstone 1 operating at 89% of full power, a hydrogen explosion took place in an auxiliary building containing some of the equipment and lines of the off-gas system. The damage was relatively minor, but, among other things, some

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*It should be noted that *under normal conditions* if the water-level for the pressurizer shows that the pressurizer is full of water, this indicates that the entire primary system is full of *water* (a "solid system"). However, *at saturation* (i.e., when the water in the system is turning into steam), this is not the case; instead, when the indicator shows that the pressurizer is full of water, this means that the entire primary system is full of a *mixture of steam and water*. Therefore, the same reading indicates two different conditions and could be misleading.

of the water seals* in the off-gas system lines blew out. It appeared that the damage could be readily repaired and that safe operation of the reactor had not been jeopardized; consequently, the reactor was not shut down.

At 1:00 PM of the same day another explosion occurred in a building at the base of the off-gas stack, and the operators scrammed the reactor. The second explosion was considerably larger than the first. It blew a door of the building into a warehouse located about 180 ft away. Besides doing extensive damage to the building and the equipment within, it cracked the base of the stack. However, these cracks (~2-mm maximum thickness) did not reduce the strength of the stack.

One man was injured and was hospitalized for 4 days.

The cause of the first explosion is uncertain. It was deduced that the second explosion was the result of the loss of the seals which occurred in the first explosion but which had not been adequately refilled. The off-gas flow system was so designed that the loss of the seals permitted the release of hydrogen, which accumulated in the building where the second explosion occurred. The explosion was set off by a spark from electrical equipment.

Calculations indicated that the maximum ground-level release that could have occurred was 54 Ci. That release consisted of a normal mixture of radioactive noble gases and iodine which would otherwise have been discharged from the top of the stack.[46]

All damage was repaired, and the reactor was returned to operation.

There was one injury and some radioactivity was released in the vicinity of the explosions.

### 3.27 Disassembly of Burnable-Poison-Rod Assembly at Crystal River 3 (Ref. 47)

The reactor in Unit 3 of the Crystal River Nuclear Plant is a pressurized-water reactor. The nuclear steam supply system was designed by the Babcock & Wilcox Company. The

*A water seal consists of a U-shaped bend in a pipe, which is partially filled with water. The water blocks the flow of gases through the pipe.

plant is located in Red Level, Florida, and is operated by the Florida Power Corporation. Unit 3, which began operation in 1977, has a capacity of 825 MW(e). The incident described below is included in this report because it resulted in damage to major equipment.

A nuclear reactor operates most efficiently if the heat produced in each part of the reactor is the same as that in every other part. This is an ideal condition, which cannot be achieved in practice. However, attempts are constantly made to approach this ideal both in the design and in the operation of the reactor. Control rods that are partially inserted in one end of the reactor depress the heat that is generated in their immediate vicinity; hence, they distort the heat distribution from the ideal, particularly in a pressurized-water reactor. In the early stages of the life of this type of reactor, burnable poison rods are used to augment the control of the reactor instead of partially inserted control rods (boron dissolved in the water of the primary system is also used). The burnable poison rods* extend the entire length of the reactor; although they also depress the heat generated in their immediate vicinity, the distortion from the ideal uniform distribution is less than that for rods that are partially inserted at one end of the core. The rods are called burnable poison rods because their effectiveness diminishes during the operation of the reactor: they are "burned up," in effect.

Although the distribution of heat generated in a reactor cannot be completely uniform, if it becomes too far from uniform (power tilt), warning alarms will be sounded.

On Dec. 12, 1977, and on Jan. 1, 1978, power-tilt alarms sounded at Unit 3 of the Crystal River Nuclear Plant. Such conditions are corrected by control rod manipulation. There were sporadic alarms triggered by the loose-parts monitoring system in the days following, and on Feb. 17, 1978, a continuous alarm was set off by the loose-parts system in one of the steam generators. Reactor power was reduced, and one of the coolant pumps associated with that steam generator was shut down.

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*A burnable poison rod is a rod that contains a material that readily absorbs neutrons. This material is depleted with time because it is transmuted by neutron absorption into another material which has a lesser propensity for neutron absorption.

On Mar. 3, 1978, the reactor was shut down so that the steam generator could be inspected. Several parts of a burnable-poison-rod assembly were found in the steam generator, which had sustained damage. Other parts of the assembly were found in the core and in various regions of the pressure vessel. A failure of the assembly latch appeared to be the cause of the problem.

It is possible that the first power-tilt alarms were set off by the initial stages in the disintegration of the burnable-poison-rod assemblies. All other similar assemblies were removed (they were no longer needed), and the steam generator was repaired. However, these repairs took 7 months.

There were no injuries, and there was no release of radioactivity.

## 3.28 The Accident at Three Mile Island 2

The reactor in Unit 2 of the Three Mile Island Nuclear Station is a pressurized-water reactor. The nuclear steam supply system was designed by the Babcock & Wilcox Company. The station is located in Middletown, Pennsylvania, and is operated by the Metropolitan Edison Company. Unit 2, which began operation in 1979, has a capacity of 906 MW(e). The accident described below is included in this report because it resulted in core damage.

No other accident has had such a profound impact on the public, the regulatory agency, and the industry as the one which occurred at Three Mile Island 2. There are at least 14 committees investigating this accident.[48] The description below is based on Refs. 48–51.

The systems of interest are illustrated in Fig. 3.13. There are two steam generator systems, A and B; only system A is illustrated in the figure.

At about 4:00 AM on Mar. 28, 1979, a condensate pump in the turbine building (Fig. 3.13) stopped. This led to the shutdown of other pumps downstream (steam generator feedwater pumps in the turbine building), which in turn shut down the turbines. The reactor was scrammed automatically. When the pumps stopped, there was no supply of secondary system water to the steam generators. Secondary system water must circulate through the steam generators in order to pick up and carry away

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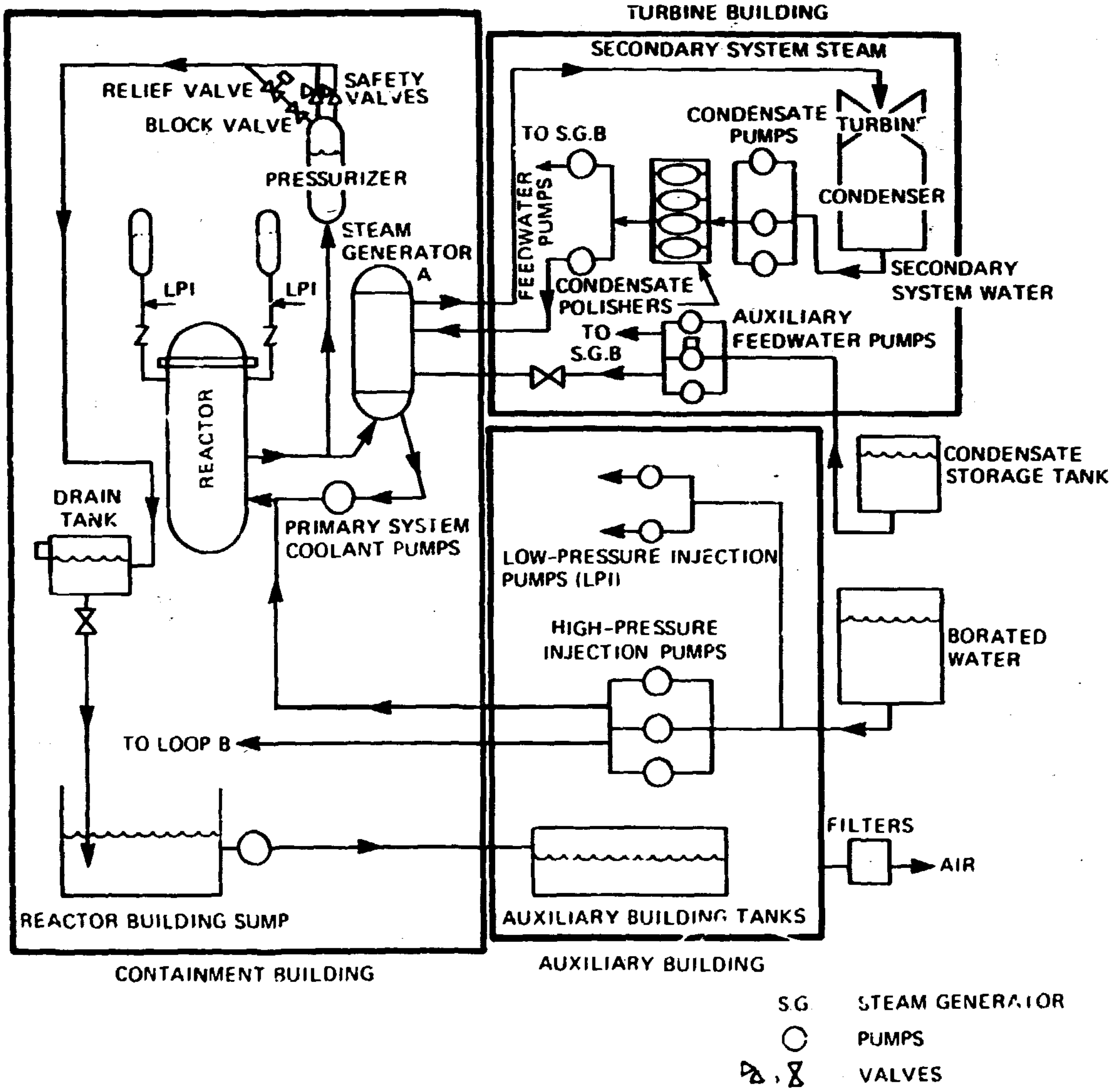


Fig. 3.13. Typical flow system for pressurized-water reactor. (Courtesy of American Institute of Physics.)

the heat from the water in the primary system. Even with the reactor shut down, the decay heat in the reactor and also the heat generated by the primary coolant pumps must be removed from the primary system. It is normally removed and transferred to the secondary system in the steam generators. Almost simultaneously with the shutting down of the pumps that feed the steam generators, other pumps (auxiliary feedwater pumps in the turbine building) which would have added the needed water to the steam generators started up. However, valves that blocked the flow of this water had been incorrectly left closed following maintenance. Without secondary system water flowing through the steam generators, they soon dried out. Without heat removal provided by the secondary system water, the primary system water began to heat up, and the pressure in the primary system began to rise. The pressure-relief valve in the pressurizer opened and later failed to close. The pressure began to plummet as water and steam poured out of the stuck-open valve into the drain tank. At about midway in this drop in pressure, two pumps (high-pressure injection system pumps in the auxiliary building) turned on to feed water into the primary system and stabilize the pressure. At this point, a meter on the pressurizer indicated that the pressurizer was completely full of water,* so the operator turned one of the pumps off and then about 10 min later he turned off the other one. Actually, primary system water continued to escape through the stuck-open valve. The pressure in the primary system continued to drop until it reached the saturation point, at which time some of the water began to change to steam.

At about this time the operator opened the valves that had inadvertently been left closed during maintenance, and secondary system water began to circulate through the steam generators and cool the primary system water. The operator turned on the high-pressure injection pumps (which he had previously turned off) to further stabilize the system and to add water to the primary system. About 12 min had elapsed since the reactor scram.

A few minutes later, the rupture disk in the drain tank blew. The water coming out of the pressurizer through the valve that had stuck open was being fed to the drain tank. The pressure in the drain tank became too high for the tank because water from the pressurizer had been entering the tank without letup. This water was radioactive because it came from the primary system. It spilled into the bottom of the containment building where it was pumped from the building sump into tanks in the

*The pressurizer itself may have been full of water, but there were probably voids or bubbles in the rest of the primary system.

auxiliary building. There was so much water that it exceeded the capacity of the tanks and overflowed and covered the floor of the auxiliary building. The transfer of this water to the auxiliary building was stopped about 20 min later.

In spite of the blowing of the rupture disk, the systems continued to stabilize and remained relatively stable for about 3/4 hr. The primary cooling pumps then began to vibrate due to steam that had been generated when some of the primary system coolant water turned to steam during the initial drop in pressure. The operator turned them off. Since the water in the primary system was thus no longer circulating, it began to heat up from the decay heat generated in the reactor, and the pressure began to rise. In the meantime, the operator had ascertained that the relief valve on the pressurizer was stuck open, and he activated the "block" valve, thus sealing the pressurizer. (The block valve serves as a backup for the pressure-relief valve; see Fig. 3.13).

In a short time the pressure in the primary system was high enough to require that the block valve be opened. It worked properly and the pressure in the system stabilized. About 3 hr had now elapsed since the reactor scram.

For the next 10 hr, the operator attempted to reduce the pressure in the primary system by opening and closing the block valve.* [An auxiliary system which is designed to remove the decay heat from the reactor was available, but it is only operable at lower primary system pressure ($\sim$400 psi).] These attempts failed, but the effort probably resulted in the venting of most of the noncondensable gases from the primary system. He closed the block valve to allow the pressure to build up sufficiently to collapse the steam bubbles that had made the primary coolant

*The fuel apparently had become so hot that there was a reaction between the primary system water and the Zircaloy metal (cladding) that surrounds the uranium fuel. This reaction generates hydrogen from the primary system water. The hydrogen was released to the containment building by the subsequent openings of the block valve in the pressurizer. The hydrogen flowed out of the pressurizer, through the block valve, into the drain tank, and then passed into the containment building via the blown rupture disk in the drain tank. Small, sharp pressure surges that occurred in the containment building are interpreted to be small hydrogen explosions.

pumps vibrate. After about 3 hr, he managed to start one of these pumps, and the primary coolant system began stable operation at a moderate pressure (1000 psi) and remained in that condition for the rest of the first day.

However, the damage had been done. At some time after the vibrating coolant pumps had been shut off, a large bubble of steam and hydrogen had formed, and the top of the core had become uncovered. Without water to cool it, some of the fuel probably melted and released some noncondensable fission products. The existence of the bubble precluded the depressurization of the system to permit the use of the auxiliary system for removing residual decay heat, which the operator had tried to utilize previously. Depressurization would only permit the bubble to grow and further expose the core.

The next day while the reactor system remained stable, the significance of the bubble was debated. There was some fear that an explosion might take place if the bubble contained a sufficient concentration of oxygen.

It was decided to eliminate the bubble by two methods, which proved to be successful. The first method employed the normal purification system used for the primary system water. Its equipment, not shown in the figure, is housed in the auxiliary building. The method worked as follows: The gas in the bubble was being absorbed by the water in the primary system, which was at moderate pressure ($\sim$1000 psi). Some of this water was bled (as is done continually even under normal operating conditions) into a "let-down" tank, which is kept at essentially atmospheric pressure, where the absorbed gas fizzed out as when a soda bottle is opened. The gas was sent to a system which delays the release of the gas for 30 days; then it passed through filters and vented out of the off-gas stack to the atmosphere. The remaining water was passed through cleanup columns to a "makeup" tank, again as is the normal operating procedure, where hydrogen can be forced into the water or allowed to escape out of it, depending on the concentration levels required. Normally, it is desired to keep the concentration level of hydrogen in the secondary system water greater than that of the oxygen in order to "get" the oxygen and thereby reduce corrosion in the system. From the makeup tank, the now purified water was pumped back to the primary system.

In the second method, heaters in the pressurizer were turned on, which forced the dissolved gas out of the primary system water in the bottom of the pressurizer into the gas space at the top. The block valve at the top of the pressurizer was then opened to permit the gas to escape.

The bubble was eliminated by these two methods, and on Apr. 28 cooling by natural circulation was achieved.

About 3 hr after the start of the accident, radiation levels in the containment and auxiliary buildings had become so high that a site emergency was declared. Apparently, the Zircaloy cladding that surrounds the fuel had ruptured, exposing the uranium fuel. A significant amount of volatile radioactivity was released into the water of the primary system, and wherever the water went, so did the radioactivity.

The containment building became contaminated when radioactive water escaped via the stuck-open pressure-relief valve in the pressurizer and then through the block valve each time this valve was opened. The radiation levels in the auxiliary building became high because of the radioactivity in the purification system used for the primary system water, as described above. Higher than normal releases were made to the environment via the auxiliary building ventilation system. The maximum offsite dose measured[52] was 50 mR/hr at 3:48 PM on the first day; the measurement was made ~1500 ft south of North Gate. Other measurements in surrounding areas were not above background until 10:38 PM when the maximum dose measured 13 mR/hr (5.56 miles NNW). Several other measurements in this vicinity showed radiation levels above background before midnight. The maximum dose measured the second day[52] was 30 mR/hr in Goldsboro at 6:00 AM. The remainder of the day the levels were less than 1 mR/hr until 3 mR/hr was measured in Royalton at 11:53 PM. The following day[52] the measurements were less than 1 mR/hr with the exception of 1 mile south of the plant where the level reached 15 mR/hr at 9:06 AM.

The Ad Hoc Population Dose Assessment Group, composed of members of of the Nuclear Regulatory Commission, the Department of Health, Education and Welfare, and the Environmental Protection Agency, has examined the available data for a period following the accident (March 28 through April 7) and concluded that the off-site collective dose associated with the radioactive material represents minimal risks of additional health effects to the off-site population (e.g., an increase of one cancer death over the 325,000 which would otherwise be expected).[53] Furthermore, the collective dose will not be significantly increased by extending the period past April 7.

There were no known injuries, but there was release of radioactivity. In addition, the core sustained significant damage, and an enormous clean-up operation lies ahead.

### 3.29 Loss of Coolant Inventory at Oyster Creek 1 (Ref. 54)

The reactor in Unit 1 of the Oyster Creek Nuclear Power Plant is a boiling-water reactor whose nuclear steam supply system is a General Electric Company design. The plant, which is located in Toms River, New Jersey, is operated by the Jersey Central Power & Light Company. It began operation in 1969. Unit 1 has a capacity of 650 MW(e). The incident described below is included in this report because core damage was suspected, although it did not actually occur.

The design of the reactor in Unit 1 of the Oyster Creek Nuclear Power Plant is somewhat different from those illustrated elsewhere in this report. It has an external recirculation loop rather than a circulation system contained within the pressure vessel, which is enhanced by jet pumps (see Fig. 2.9, Chap. 2). A schematic diagram of this reactor system is shown in Fig. 3.14, which should be referred to when reading the following description.

The core is covered by the core shroud. The shroud is positioned inside the pressure vessel (which is about three-quarters full of water) in the same manner as an inverted glass might be positioned in a bucket of water with air trapped inside it. However, instead of trapped air, the shroud contains "wet" steam. The water level in the pressure vessel is normally above the top of the shroud, and water cannot enter the shroud directly. Pipes connected to the top of the shroud allow the steam within the shroud to pass out into the region above the water level

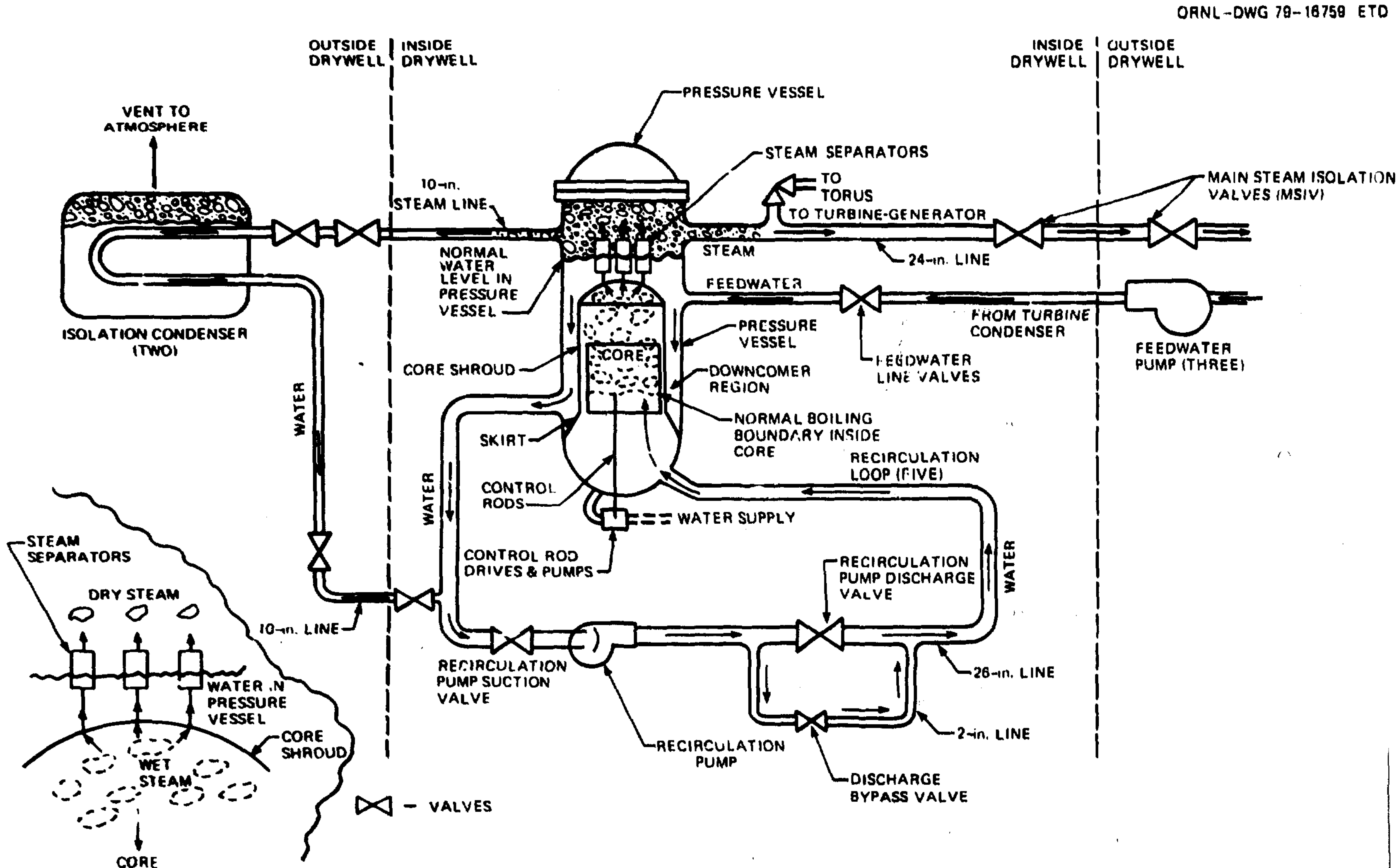


Fig. 3.14. Schematic diagram of reactor circulation system at Oyster Creek.

in the pressure vessel (see the inset in Fig. 3.14). This steam must first pass through steam separators that "dry" the steam (i.e., remove the water droplets which would ruin the turbine). This dry steam is piped to the turbine-generator and then to the condenser where it is condensed. The water from the condenser is pumped back to the pressure vessel via feedwater lines and feedwater pumps, and it is now referred to as feedwater.

The feedwater enters below the normal water level of the pressure vessel and flows over and around the core shroud but cannot flow directly into the shroud. It is prevented from doing so by the skirt at the bottom of the shroud. Instead, it flows downward in the downcomer region (i.e., the space between the outside of the shroud and the pressure vessel) and enters the bottom of the shroud by means of a recirculating loop (there are five recirculation loops at Oyster Creek). From the bottom of the shroud it passes upward through the core where the heat generated in the core boils the water and turns it into "wet" steam. This steam passes from the shroud into the top of the pressure vessel via the steam separators, and the cycle continues. Within all of this piping are many valves that can be closed to block or divert the normal flow of fluid.

If the main steam isolation valves in the steam lines and the valves in the feedwater lines are closed, which blocks the flow through the lines (see Fig. 3.14), then all the valves leading to and from the isolation condenser are usually opened. The steam that has collected at the top of the pressure vessel now passes to the isolation condensers where the steam is condensed and returned to the recirculation pump in the recirculation loop. This water is pumped into the bottom of the shroud, where it is turned to steam by the residual decay heat in the core, and the steam passes through the steam separators into the top of the pressure vessel.

When the reactor is at power, the water in the core is boiling furiously, and the boundary between the steam and the water in the core (referred to as the "boiling boundary") is rather poorly defined. The boiling boundary is about one-third of the way up from the bottom of the core, and the upper two-thirds of the core is immersed in a cloud of wet steam.

At shutdown following full power operation the intensity of the boiling is greatly reduced, since the heat being generated is only decay heat (approximately 6 1/2% immediately after full power operation, dropping to less than 2% after about 10 sec). The boundary between the steam and the water is now more clearly defined, and it moves above the top of the core.

When the plant is in normal operation, part of the electricity it generates is diverted to drive many, but not all, of the pumps (feedwater recirculation, etc.) within the plant. When the reactor is shut down, those pumps must still operate. The needed electricity comes from the outside electrical grid and must be adjusted in voltage to suit that required by the pumps. This is accomplished, at Oyster Creek, by means of two startup transformers. All of the above is pertinent to the incident.

On May 2, 1979, the plant was operating at 98% of full power. One of the two startup transformers was out of service, as was one of the five recirculation loops. The operator was performing a routine test (done electrically) of some switches when a spurious electrical signal generated by these tests simultaneously scrammed the reactor and turned off the recirculation pumps (which coasted down before stopping completely). With the reactor power reduced to decay-heat levels, the water temperature began to drop and less steam was generated, which reduced the pressure in the system. The boiling boundary inside the shroud rose above the core. After 13 sec, the turbine-generator automatically shut down, and this was followed by the shutdown of the feedwater pumps and condensate pumps (located upstream from the feedwater pumps but not shown in the figure).

With the turbine-generator shut down, the plant was no longer generating electricity, and the supply of electric power to these pumps was cut off. At this point electricity from the outside grid network should have switched in automatically in order to keep the pumps in operation. But the single transformer in operation could not handle the electrical load and could not supply all the pumps with electricity at the required voltage. So they started coasting down to a stop.

The reactor was therefore without a supply of feedwater. Water in the form of steam was being lost from the reactor and little was being supplied. The water level in the reactor vessel and in the shroud began to drop. The operator closed the main steam isolation valve to block the flow of steam to the main condenser and to conserve the supply of water. It was 43 sec into the incident.

Water had been flowing from the downcomer region into the bottom of the shroud via the five discharge bypass lines, and it continued doing so.

One of the two isolation condensers was placed into service in order to remove the decay heat from the reactor. The discharge valves for the recirculation pumps were closed, and the reason given for this, in essence, was to reduce the cooling rate of the core.[54] However, water was being lost from the core (by being converted to steam) at a faster rate than it was being returned. Although it was being returned to the recirculation loop (see the figure) as fast as it was lost from the core, it was being returned to the bottom of the core (shroud) through the stopped recirculation pumps at a much slower rate because the passageway back to the bottom of the core was via the small 2-in.-diam bypass lines, which limited the rate of flow. The excess water that was not passing through the restricting 2-in.-diam pipes was passing upward through the downcomer region and filling the pressure vessel. The core low-water-level alarm was sounded (low-low-low-water-level trip). It was about 3 min since the start of the event.

About a minute later, the operator closed the valves to the isolation condensers, thereby shutting off the flow to and from the isolation condenser. This increased the pressure in the cooling system and reduced the boiling rate in the core and hence reduced the rate at which the core was losing water. For about the next 25 min, the isolation condensers were alternately brought into service and removed from service. The core was cooled when the condensers were used because they introduced cold water into the system. The temperature in the system dropped continuously, the pressure in the core decreased, and the water level increased according to predictions. When it was felt that the core temperatures were sufficiently low, one of the recirculation pumps was started and its discharge valve was opened. The single available transformer was capable of handling the electrical load from one of the pumps. The increase in flow rate raised the water level in the core above the alarm setting, and steps were taken to bring the reactor to a cold-shutdown condition. It was then 45 min since the start of the incident.

Theoretical analysis of the water level in the core indicated that the water level was always above the core. Measured radioactivity levels in the water and in the off-gas system were never above normal.

There were no injuries and there was no release of radioactivity as a result of this incident.

# 4. PRODUCTION REACTORS

## 4.1 Blockage of Coolant Tube in Hanford KW Reactor[1]

The Hanford KW Reactor was a graphite-moderated, water-cooled production reactor located at the Hanford site in Washington. It was designed by General Electric Company. It started up in 1955 and was shut down in 1970. The incident described below is included in this report because it resulted in significant recovery costs.

Prior to the startup of the Hanford KW Reactor, a series of tests was to be run to determine its characteristics. One of the tests involved the blocking of several hundred coolant tubes with neoprene disks. After completion of the tests, the disks were removed and counted. One of them was overlooked and remained in the system. Although a pressure indicator definitely recorded a blockage, the supervisor failed to notice it. An instrument technician, believing the gauge was reading incorrectly, adjusted it to give the reading that he thought was correct but was in fact false.

On Jan. 4, 1955, the initial startup of this new reactor began. The next day, as startup procedures continued and power increased gradually, the reactor experienced a sharp decrease in power. The control rods were withdrawn to maintain power at that level. Twelve minutes later an alarm sounded. Suspecting a leak in a process tube, the supervisor initiated a high-speed scan of the outlet water temperatures in order to find the leaking tube. Before the scan was completed, the reactor was scrammed automatically by a high-pressure reading. Other instruments indicated that a fuel element had ruptured.

All standard methods that were tried to alleviate the problem failed. After 11 days of effort, it was decided to cut a hole in the rear concrete shield wall of the reactor and remove the entire graphite channel. It took 6 days to drill and cut through the concrete and 8 more days to remove the ruptured fuel element and the neoprene disk and to clean up and complete the repair.

Although there was no release of radioactivity and there were no injuries, the recovery cost was $550,000.

### 4.2 Fuel Fire at Windscale, No. 1 Pile[2,3]

The No. 1 pile (reactor) at Windscale, England, was a natural-uranium, air-cooled, graphite-moderated production reactor. The accident described below is included in this report because a significant amount of radioactivity was released offsite.

Most solids exposed to neutron radiation of high intensity will undergo a change in their physical properties. This change is called radiation damage. In many cases this damage can be "repaired" (i.e., the normal physical properties can be restored) by simply heating or annealing the material.

Graphite, the moderating material used in the No. 1 pile at Windscale, undergoes a peculiar type of damage when irradiated with neutrons. It grows or swells, its thermal and electrical conductivity decreases, and it tends to store thermal energy. This stored energy is called "Wigner energy," since this property was first suggested by Dr. Eugene Wigner. If, after being irradiated, the graphite is heated slowly, it will release its Wigner energy; these small additions of heat to radiation-damaged graphite will trigger the release of even more heat. The process of attempting to remove the Wigner energy from the graphite led to the Windscale accident.

It was standard procedure at Windscale to release the Wigner energy from the graphite after a normal reactor shutdown in order to restore the original properties in the graphite. Eight such releases had been successfully carried out previously. It is a rather slow and time-consuming process.

On Oct. 7, 1957, following a normal reactor shutdown, procedures to release the Wigner energy were initiated. The procedure was to restart the reactor and operate it at low power with the cooling blowers shut off. This would add heat to the graphite and thus trigger the release under controlled conditions. Following the first addition of heat, the operators observed that the graphite temperatures were falling rather than rising. There is some confusion on this point because a Committee of Inquiry that was set up after the accident examined the records and found that, although the temperatures in some parts of the core were decreasing, a substantial number were increasing. At this stage, a decreasing temperature would have indicated no release of heat while an increasing temperature would have indicated a release of heat from the graphite. Apparently, there was some release in substantial parts of the core but not in others.

The operators proceeded as though there had been no release of Wigner energy, and the next day they added more power, but because of a faulty power meter they added it too quickly. Also, the temperature instruments were located in the reactor at the positions of maximum temperature *for the reactor operating at full power*, but these were not the positions of maximum temperature for the reactor operating at low power when the Wigner energy was being released. Hence, although the operators were observing temperatures that were well within operating limits, other parts of the core became so hot, because of the release of Wigner energy, that the uranium fuel caught fire the following day and subsequently so did the graphite. Even then there was no indication of the combustion except that the temperatures showed considerable variation. Steps were taken to cool and stabilize the reactor. These steps appeared to help temporarily, but on the morning of the fourth day there were indications of release of radioactivity through the off-gas stack. Also, the graphite temperatures began to rise again. Suspecting a ruptured fuel cartridge,* the operators attempted to use remote-scanning gear to locate it but found that the gear was jammed. Donning special clothing, workmen opened a plug at the front of the reactor and found that the fuel was red hot. This was their first indication of a fire, which had been smouldering for 2 days. Various attempts to smother and contain the fire were tried to no avail; finally, the reactor was flooded with water on the fifth day, and the fire was extinguished.

The reactor was ruined, and there had been widespread release of volatile radioactivity, primarily iodine and noble gases. Over a period of many hours and under varying meteorological conditions, an

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*Cylindrical natural-uranium fuel element about 1/2 in. in diameter and about 12 in. long with a finned aluminum cladding. There were 20 cartridges laid end to end in each fuel channel. The channels were horizontal.

estimated 20,000 Ci of $^{131}I$ was released into the atmosphere of the countryside from the 405-ft stack.

Surveys of radioactivity in the surrounding countryside indicated that the highest level of gamma radiation was 4 mR/hr. Vegetation sampling indicated that the stack filter had removed almost all the radioactive particulates while permitting the radioactive gases to be released. Accordingly, radioactive gases such as $^{131}I$ were transported directly to animal feed, which resulted in subsequent contamination of milk. Thus, the only health hazard to the public as a result of the accident was an accumulation of radioactive iodine in the milk supply.

Radiochemical analysis of milk taken over a larger area at a later date showed that the ban on milk distribution had to be extended to a total area of about 200 sq miles, beginning 2 or 3 miles north of the plant and extending over a strip 7 to 10 miles wide to a distance of about 30 miles south of the plant. The use of milk by the population in the restricted area was prohibited for 25 days; for the most highly contaminated locations, this prohibition was maintained for 44 days.

The Medical Research Council Committee concluded "that it is in the highest degree unlikely that any harm has been done to the health of anybody, whether a worker in the Windscale plant or a member of the general public." Except for the confiscation of milk, no other environmental action was required.

### 4.3 Failure of Primary Scram System in the N Reactor at Hanford[4]

The N Reactor, located at the Hanford site at Richland, Washington, is owned by the Department of Energy. It is a dual-purpose facility; that is, it is primarily a production reactor, but it also provides a total of 860 MW(e) locally. The electrical generation plant is operated by the Washington Public Power Supply System. The reactor is water-cooled and graphite-moderated. The reactor started up in 1963. It produced 800 MW(e) in 1966 and reached full electric power output in 1972. Although the incident described below does not meet any of the severity criteria we established, we included it in this report because it illustrates the importance and usefulness of redundant safety systems.

The incident that occurred at the N Reactor at Hanford is an example of a common-mode failure, that is, the failure of multiple systems from a single cause. It also illustrates the need for independent and redundant safety systems.

The primary reactor control system is such that the control rod mechanisms must have electrical energy in order to withdraw the control rods. The control rods go in from the front and also from the back of the reactor. When the control rods are withdrawn, they are withdrawn against the pressure of individual hydraulic systems, thereby "pressurizing" each system. In the event of a complete electric power failure, the hydraulic system acts as a spring and drives each control rod back into the reactor. The hydraulic system is thus independent of the availability of electric power.

In addition to the primary system, there is an independent auxiliary scram system, which is situated on top of the reactor. In the event that the primary system control rods do not enter the reactor within 1.5 sec after a scram signal, the auxiliary system is activated. This system simply drops small balls of "poison" material (samarium oxide), that is, a material that readily absorbs neutrons, into the reactor and shuts it down. Figure 4.1 is an illustration of the mechanism.

Returning to the primary control system, each control rod drive mechanism has a five-position switch, and each position serves a different function. In addition, there are two independent sources of electrical energy to the system. Unfortunately, they were wired in such a way that if the five-position switch on a single control rod mechanism was in a particular position (an uncommonly used position) and if also that same control mechanism had a serious short circuit, then a scram signal would not release any of the rods. The scram signal ordinarily cuts off the supply of electrical energy to the rods, but under the conditions noted above, the auxiliary electrical system continued to supply electrical energy, thus holding the rods out. This exact condition prevailed at the time of the incident.

Three days before the incident the reactor was shut down so that some water leaks could be located. While it was shut down, the time-to-

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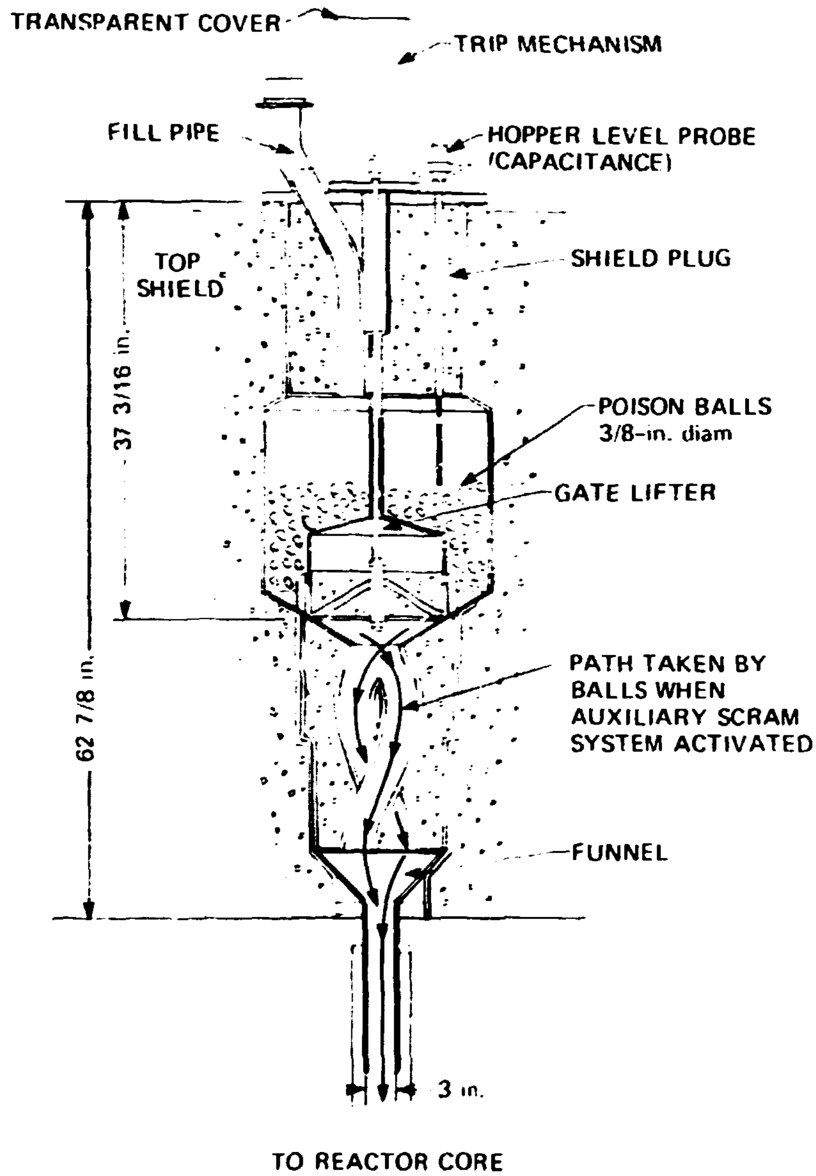


Fig. 4.1. Auxiliary scram system.

scram was checked for each individual rod. On the morning of Sept. 30, 1970, the reactor startup sequence began. One of the rods failed to function; so it was removed from the system and the five-position switch on its drive mechanism was set appropriately. This was the uncommonly used setting mentioned above.

When the reactor reached about 10% of full power, with all other systems stable, a pump filter became clogged with foreign material. This initiated a scram signal. The primary scram system failed to activate, but the auxiliary scram system shut down the reactor.

Subsequent investigation showed that the drive system of the control rod that had failed to function had a serious short. The common-mode failure of the combination of short circuit and switch setting, which caused the reactor scram system failure, was also discovered at this time. The electrical circuitry was altered to eliminate this common-mode failure, and the reactor was returned to operation.

There were no injuries, nor was there damage to the system. There was no release of radiation.

# 5. EXPERIMENTAL AND RESEARCH REACTORS

## 5.1 Core Damage in the NRX Reactor at Chalk River[1]

The experimental NRX reactor located in Chalk River, Canada, is light-water-cooled and heavy-water-moderated. It was designed to operate at a full power of 30 MW(t). It achieved criticality in 1947 and reached full power in 1948. The core was ruined in 1952 and was replaced with an improved version 14 months later. The accident described below is included in this report because it resulted in core damage.

The nuclear characteristics of the NRX reactor were such that a loss of light-water coolant would make the reactor more critical, whereas a loss of heavy-water moderator would make it less critical. A cross-section view of a typical fuel channel of the NRX reactor is shown in Fig. 5.1.

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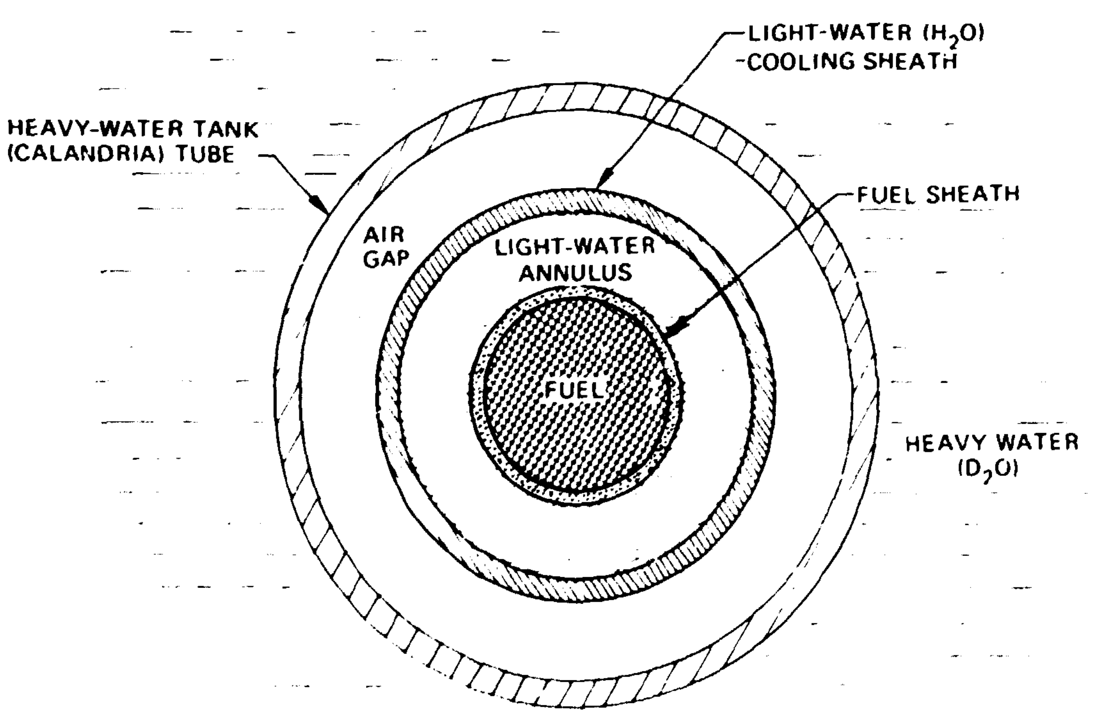


Fig. 5.1. Cross section of NRX fuel tube.

Groups of control rods (banks) could be withdrawn by pressing various numbered buttons on the control panel. Red lights above these numbers indicated that the banks were in their fully withdrawn positions. The banks were inserted by increasing the air pressure above them, which was accomplished by pushing button 4. Button 3 activated a solenoid which ensured that the seal for this increased air pressure would not leak. Therefore, to drive the control rods into the reactors, the operator had to press buttons 3 and 4 at the same time; in order to facilitate this action, the buttons were located near each other on the control panel. The withdrawal buttons for the control rod banks were spaced an arm's length away.

On Dec. 12, 1952, the reactor was undergoing tests at low power. The circulation flow of the light-water coolant was reduced in many of the rods, since there was not much heat being generated in the fuel. The supervisor noted that several of the red lights suddenly came on. He went to the basement and found that an operator was opening valves that caused the control rod banks to rise to their fully withdrawn positions. Horrified, he immediately closed all of the incorrectly opened valves, after which the rods should have dropped back in. Some of them did, but for unexplained reasons others dropped in only enough to cause the red lights to turn off. These latter rods were almost completely withdrawn.

From the basement, he phoned his assistant in the control room intending to tell him to start the test over and to insert all the control rods by pushing buttons 3 and 4. A slip of the tong e caused him to say: "Push 4 and 1." (Button 1 was a control rod withdrawal button.) The assistant laid down the phone, because it required outstretched arms to push 4 and 1 simultaneously; hence, he could not be recalled immediately to rectify the error. Since button 3 had not been pushed, the air seal was not secured; thus, the air rushing into the chambers, where it should have compressed and forced the control rods in, rushed out through the seal instead. Control rod bank No. 1 was withdrawn.

The operator in the control room soon realized that the reactor power was rising rapidly, and he pressed the scram button. Even without compressed air, the rods should have dropped in by gravity, but again for unexplained reasons many of them did not, and the power continued to climb. He phoned his supervisor in the basement and asked him to do something to increase the air pressure. After a hurried consultation with physicists and the assistant superintendent, who were present in the control room, it was decided to dump the heavy-water moderator. This succeeded in shutting down the reactor but not instantaneously, since it took some time to drain. The reactor power had peaked between 60 and 90 MW(t).

More was to come. Water began to pour into the basement. It was light-water coolant. Radiation alarms sounded, both inside and outside the building, and a plant evacuation procedure was ordered. Conversations in gas masks, donned by the staff who were inside the building, proved to be too difficult to carry on, so they moved into an adjoining building. However, except for sealing the vents, there really was not much more that could be done.

The metal sheaths containing the cooling-water annuli for about 25 fuel rods ruptured, there was some fuel melting, and the heavy-water tank (calandria) was punctured in several places. The initial power surge caused the cooling water around the rods to boil, which increased the internal pressure in the rods and ruptured the metal sheaths. This boiling and loss of water increased the reactivity of the reactor, which enhanced the power surge. About 1 million gallons of water containing about 10,000 Ci of radioactive fission products had been dumped into the basement of the building.

The core and the calandria, which were damaged beyond repair, were removed and buried, and the site was decontaminated. An improved calandria and core were installed about 14 months after the incident.

There were no injuries, but there was some release of radioactivity.*

## 5.2 Operator Error Causes Fuel Melting in EBR-1 (Ref. 2)

The Experimental Breeder Reactor (EBR-1) was a sodium-cooled, unmoderated experimental reactor located at the National

*The level of radioactivity released was not given in the reference.

Reactor Testing Station* in Idaho, about 40 miles from Idaho Falls. It was designed to operate at 1.4 MW(t) and 0.2 MW(e). It started up in 1951. The accident described below is included in this report because it resulted in core damage.

The EBR-1 was an experimental reactor principally designed to study the breeding capabilities and the time-response characteristics of reactors of this type. It was the first reactor from which electricity was produced. Successful experiments had been performed in the 4 years of operation before the accident.

The reactor had two control methods: one was the insertion (or withdrawal) of control rods whose motor drives did not move the rods very quickly (slow control), and the other was the dropping (or raising) of a natural uranium blanket that surrounded the core (fast control).

Because the data generated from a previous experiment had not been sufficiently conclusive with regard to some of the characteristics under study, it was decided to repeat the experiment with improved instrumentation and to modify the experiment somewhat.

The experiment from which the necessary information was to be obtained consisted of bringing the reactor above critical so that the power would rise over a certain period of time and then making it even more reactive and allowing the power to increase at an even faster rate over a short period of time. The experiment was designed to terminate with the maximum fuel temperature about 450°F below that at which the fuel would interact (form a eutectic) with its surrounding metal covering (cladding). The signal to terminate the experiment by scramming the reactor was to be a spoken word from the staff scientist who would be watching the instruments to the technician who was operating the reactor controls. (Looking back today, with plenty of hindsight and also with knowledge of the modern, fast-responding, automatic instruments that are now available, such a procedure is almost inconceivable.)

The experiment was begun on Nov. 29, 1955, and it proceeded as planned. With the reactor power in the fast-rise state in the final

*Now the Idaho National Engineering Laboratory.

stage of the experiment, the spoken word was given. The technician mistakenly pressed the insertion button for the slow control system instead of the scram button for the fast control system. (In the previous experiment, the slow control method had been used.) As soon as the staff scientist realized the situation, he reached over and pressed the button for the fast-responding scram system, and the reactor shut down. But it was too late. During the few seconds that had elapsed between the spoken word and the pressing of the scram button, the fuel had become so hot that about 40 to 50% of the core had melted. About 15 min later, radioactivity was measured in the reactor cooling system and in the ventilation exhaust ducts from the building. The building was evacuated and a thorough survey made. The level of contamination was so low that the building was reoccupied without further incident.

There were no injuries, and only a trivial amount of radioactivity was released.[3]

### 5.3 Ruptured Fuel Element Causes Contamination of Reactor Building at NRU (Ref. 4)

The NRU reactor is a heavy-water-cooled and -moderated engineering and research reactor. It was designed to operate at 200 MW, and it is located in Chalk River, Canada. The incident described below is included in this report because it resulted in significant recovery costs.

The decay heat from fission products in the fuel of a reactor is sufficient to require that the fuel be cooled for some time after shutdown. It was lack of such cooling that caused the accident at the NRU reactor.

Problems had developed in some of the fuel elements that were being used. It was found that leaks had developed in the cladding that surrounds the fuel; this allowed some of the radioactive fission products to escape the cladding and enter the reactor tank. These leaky fuel assemblies were replaced when their condition was discovered, but the tank had become somewhat contaminated, and this background radioactivity obscured the presence of other leaking fuel elements.

For the week prior to May 23, 1958, the reactor had been in steady operation, but it shut down automatically that day on a signal indicating that the power was rising too fast. No reason could be found for the shutdown, so the operator attempted to restart the reactor four times. Each time spurious signals kept him from doing so.

On the fifth try, he was successful, and the power was brought up by setting a switch which governed the rate of rise of power. Five minutes later, the reactor was shut down on an excessive-rate-of-rise signal again. This time the shutdown was accompanied by alarm signals indicating that a fuel element had ruptured, resulting in contamination of the coolant.

It is postulated that the switch which set the rate of power increase was faulty, allowing the power to rise faster than desired. This, in turn, had caused the violent failure of an undetected leaky fuel element. The pressure shock from this transient had, in turn, caused a spurious signal to be generated, which caused the control rods to be withdrawn, thereby creating a rapid power transient. This transient shut down the reactor.

Two fuel elements had been damaged, one severely and one moderately. The moderately damaged element was withdrawn from the reactor without incident.

When a fuel element that has been in use is withdrawn from this reactor, a flask containing circulating cooling water is attached to the reactor, and the fuel element is drawn up into the flask, after which the flask is moved to a storage tank by a large crane. The flask has its own pump to circulate the water and keep the enclosed fuel element cool from the time it is withdrawn from the reactor until it is deposited in the storage tank.

It was decided to remove a guide tube along with the more severely damaged fuel element. When the fuel element was drawn up into the flask, it was found that the guide tube had prevented cooling water from entering the flask. It was further found that the fuel element could not be reinserted into the reactor, where it could be cooled by the heavy water. Attempts to do so had caused it to become stuck in an improper position in the flask. The fuel element began to get hot, and time became important.

There was an emergency water hose available at the side of the reactor tank, so attempts were made to move the flask to this position with the crane. Since the fuel element was now stuck in an improper position within the flask and the flask was not in its normal state, a series of safety interlocks came into play that were designed to prevent the movement of the cask unless a specific operational routine was followed. Each interlock had to be overcome by time-consuming efforts. In the meantime, the fuel element began to disintegrate and burn. A piece of it fell on top of the reactor, and a much larger piece fell into a maintenance pit. Sand was dumped on the pieces to quench the burning uranium.

The building was severely contaminated, and 600 men participated in its cleanup. The highest dose received by anyone involved in the incident was 19 R to one man. The area of detectable contamination outside the building was in the 100 acres adjacent to the building.

There were no apparent injuries, but there was some release of radioactivity, which appeared to be confined to the area of the building.

### 5.4 Improper Instrumentation Results in Fuel Melt in HTRE-3 (Refs. 5, 6)

The core that was installed in the Heat Transfer Reactor Experiment (HTRE) Facility at the time of the accident was air-cooled and was moderated by hydrized zirconium. The facility was located in the National Reactor Testing Station* in Idaho. The accident described below is included in this report because it resulted in core damage.

The HTRE facility was designed to test high-temperature reactor cores. Two cores had been tested in the facility, and on Nov. 18, 1958, a third core was undergoing tests. Previously, the instrumentation had been altered in order to reduced background noise, and the instruments

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had been checked out. Also, manual control had been used to raise the power to the specified levels. In the experiment that was under way, the power was to be increased manually to 90% of full power, and then control would be transferred to a servomechanism, which would automatically increase the power to the maximum level desired (0.12C MW) at the rate of increase that was specified by the test. The servomechanism was connnected to an instrument (linear flux recorder) that indicated the reactor power, and the actions of the servomechanism were governed by this instrument. As it turned out, the addition of the electronic circuitry for the servomechanism to the existing circuitry caused the linear flux recorder, which governed the actions of the servomechanism, to read incorrectly at the higher reactor power levels.

The experiment proceeded according to plan until control was switched to the servomechanism. Reactor power began to rise as planned, but then the linear flux recorder began to show (incorrectly) a decrease in reactor power. In response, the servomechanism pulled out the control rods, which caused the power to increase at a much faster rate, and then the reactor was scrammed automatically. Extensive damage to the core had been done in the 20 sec that had elapsed between the time that the servomechanism had been turned on and the scram. It was a case where the operator switched control to the automatic system and sat back astonished as it carried the reactor through the accident.

There were no injuries, but small amounts of radioactivity were released within the facility[7] and some was released downwind.[6]

### 5.5 Leakage of Organic Seal Coolant Causes Fuel Damage in SRE (Refs. 8, 9)

The reactor for the Sodium Reactor Experiment (SRE) was graphite-moderated and sodium-cooled. It was owned and operated by the Atomic Energy Commission and was designed to produce nominal electric power from a total heat production rate of 20 MW(t). It went critical in July 1957. The facility was located in Santa Susanna, California. The accident described below is included in this report because it resulted in core damage.

In the Sodium Reactor Experiment (SRE), an organic material (tetralin) had been used as an auxiliary coolant to seal the pumps that were used to circulate the primary coolant (liquid sodium). Over a series of runs between Nov. 29, 1958, and July 26, 1959, it was found that tetralin had leaked into the primary coolant. The decomposition products of the tetralin had coated the fuel elements. Since this coating would inhibit the transfer of heat from the fuel elements to the primary coolant, intermittent attempts were made to purge the coolant and clean up the fuel elements.

This series of runs, including the one in which the reactor was finally shut down, was plagued with equipment failures, spurious scrams, unexplained transients, erratic temperature readings, and, in the last run, the release of radioactivity into the primary coolant.

Attempts to reach power and to generate electricity during the final run (July 12-26, 1959) resulted in ten scrams and four forced shutdowns, including the last. The reactor was shut down in order to determine the cause or causes of the troubles. An examination of the core revealed that 10 of the 43 fuel assemblies had undergone severe damage. This had apparently occurred during a brief, small power excursion.

The primary cause of the accident has been attributed to the tetralin that leaked into the primary system and decomposed. The decomposition products not only prevented the fuel assemblies from being cooled properly, but they had also blocked some of the coolant passages. The temperature readings available to the operators were those of the sodium coolant and not of the fuel. So, although the coolant temperatures were high at times, they were not inordinately so. However, the fuel temperatures had become extremely high because the decomposition products had formed a barrier to the removal of heat from the fuel.

No injuries were incurred, but some radioactivity was released from the stack.*

*The level of radioactivity released was not given in the references, but it was insufficient to require the shutting down of the reactor.

## 5.6 Melting of Fuel Element WTR (Refs. 10, 11)

The Westinghouse Testing Reactor (WTR) was cooled and moderated by light water. It was designed and operated by the Westinghouse Electric Corporation and was located in Waltz Mills, Pennsylvania. It began operation in 1959 and was dismantled in 1962. The accident described below is included in this report because it resulted in core damage.

Before the date of the accident (Apr. 3, 1960), the Westinghouse Testing Reactor (WTR) had operated successfully at 20 MW. The accident occurred in the process of raising the power to the 60-MW rating of the reactor. In this process, the reactor power was being raised in small incremental steps, and checks were made at each increment. Among other things, the checks included the reduction of the coolant flow rate until boiling of the coolant water within the reactor began. This flow rate was then checked against theoretical or calculated flow rates at which boiling would occur; if the measured and calculated flow rates agreed, the reactor power would be increased by the next increment.

The point of all this was to ensure that the coolant flow rate would be sufficient when and if the reactor reached the rated power. Boiling is an indication that the flow rate is marginally adequate for proper cooling of the fuel. The flow rate cannot be less or the fuel will be damaged. Theoretical predictions indicated that the flow rate would be adequate at full power, but these predictions were being checked at each step in the rise to power.

After some minor difficulties, the reactor power had reached 34 MW and was being allowed to settle when a drop in power occurred. The operator manually withdrew the control rods, and the power increased to slightly above 34 MW. Shortly thereafter radiation alarms sounded, and the reactor was manually scrammed.

Examination revealed that a fuel element had melted. It is believed that the element was faulty, since the conditions were not such as to have caused a normal fuel element to fail. The probable cause of the failure was a separation in the bonding between the metal (cladding) surrounding the fuel and the fuel itself. A separation between the fuel and cladding would act as a barrier to the heat being removed from the fuel and allow the fuel to heat up beyond its melting point.

There were no injuries, but there was some release of radioactivity within the plant and into the atmosphere. A survey of radioactivity levels outside the plant was made by the Nuclear Science and Engineering Corporation. It showed no appreciable increase in radioactivity above background values that had been measured the preceding 3 years. However, it is not clear when this survey was made.

### 5.7 Three Fatalities at SL-1 (Refs. 3, 12, 13)

The reactor in the Stationary Low-Power Plant No. 1 (SL-1) was a natural-circulation (no pumps) boiling-water reactor with a 3 MW(t) capacity. The plant was designed as a prototype for those whose mission would be to supply both heat and electricity at remote military installations. It was located at the Atomic Energy Commission's National Reactor Testing Station* (NRTS) in Idaho. The reactor was made critical on Aug. 11, 1958, and was disassembled following the accident described below, which occurred on Jan. 3, 1961. The accident described below is included in this report because it resulted in fatalities.

The accident at SL-1 was the first in an operating reactor that caused fatalities.

The SL-1 facility had been used to gain operating experience, develop plant performance tests, obtain core burnup data, train military personnel in operations and maintenance, and test components that might subsequently be used in improved versions. It was made critical in August 1958 and had successfully logged 500 hr of full-power operation by December 1958.

It was shut down for maintenance purposes in December of 1960 with the intention of starting up again on Jan. 4, 1961. The additional instrumentation that had been installed prior to the planned startup required the disconnnecting of the control rods from their drives. The installation of the instruments had been completed during the day shift on Jan. 3, and it was the job of the crew of the 4:00 to 12:00 PM shift

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to reconnect the control rods. The crew consisted of three men in the military service: two of them were experienced and qualified reactor operators, and the third was a trainee.

When disconnected, the control rods could be lifted completely out of the reactor manually. The justification for this was that maintenance in remote areas should be as simple as possible. However, lifting the central control rod about 16 in. was sufficient to make the reactor critical.

The first indication of an accident was the sounding of alarms at 9:01 PM in the Fire Stations and Security Headquarters for the NRTS located some distance from the SL-1 facility. Since the alarms could have been set off either by fire, radiation, or pressure surges in the facility, members of the fire department and the plant security force, as well as a health physicist responded. They searched for the three men in the building adjacent to the reactor building and also the ground floor of the reactor building, but the radiation levels were greater than the limits of their meters (25 R/hr), so they withdrew. There were no indications of smoke or fire. Calls were placed to other facilities at the NRTS, but the missing men were not there, so it was concluded that they must still be in the reactor building. Other health physicists and support and military personnel began to arrive.

Wearing protective clothing, two men went up the stairs of the reactor building, but when they encountered radiation levels of 200 R/hr, they withdrew. They were followed by another pair who reached the top of the stairs and looked into the basement of the reactor building and in doing so encountered fields of 500 R/hr, so they quickly withdrew also. They could see no one, but they did see some evidence of damage. It was then about 10:30 PM.

Two others reached the basement and saw two men, one of whom was moving. Five others then went in, placed the man who was moving on a stretcher, and raced out. They had ascertained that the second man inside was dead. The man on the stretcher was placed in an ambulance, but he died before it traveled very far. The ambulance then returned to the SL-1 area.

Four more men entered the basement in search of the third missing man. Upon looking up, they found him pinioned to the ceiling by a control rod. Assuming that he was dead, they did not try to remove the body. Since both men in the building were assumed to be dead, rescue operations were suspended temporarily.

About 6:00 AM the following morning, the dead man in the ambulance was taken out of it for decontamination purposes; lead shielding had to be used in removing his clothing. The radiation levels of his body measured about 300 R/hr (upon subsequent removal, the bodies of the other two men measured about the same). At about 7:30 PM the second body was recovered by men working in teams; one team carried it part way out of the high-radiation area, and other teams completed the removal. It took six such teams to remove the body of the third man. The recovery operation was completed on Jan. 9.

The recording instruments had been turned off while the control rods were being reattached to their drives, and there were no survivors; thus, the cause of the accident is conjecture only. Based on a careful examination of the remnants of the core and the vessel during the cleanup phase, it is generally concluded that the central control rod was withdrawn manually and withdrawn quickly. Examination revealed that it had been withdrawn about 20 in. at the time of the excursion, sufficient for a large increase in reactivity ($\sim$2.4% $\Delta$ k/k). It is believed that the resulting short power surge, which reached a peak of $\sim$20,000 MW ($\sim$130 MWsec energy release), created a sudden volume of steam in the core, causing the water above it to rise with such force that when it hit the lid of the pressure vessel, the vessel itself rose 9 ft in the air and then dropped back to its approximate original position.

Monitoring of the area for radioactivity began shortly after the accident. An aerial survey early the next day revealed no increase except in the immediate vicinity of SL-1. Four flights were made in the next 9 days, and some air samples taken revealed a radioactivity level about 50% above background. Sagebrush samples downwind indicated maximum levels about 40 times greater than background. Even though the radioactivity was high, it was apparent that almost all (99.99%) of the radioactive fission products were contained in the reactor building in spite of the fact that the building had no air locks, airtight seals, reduced pressure, etc.

The reactor vessel and core were removed, the building razed, the area decontaminated, and the site made suitable for other purposes by July 1962.

There were three fatalities, and some radioactivity was released.

### 5.8 Pressurizer Failure in SPERT-3 (Ref. 14)

The reactor in the Special Power Excursion Reactor Test No. 3 (SPERT-3) was a pressurized-water reactor, which was designed for transient power tests. It was designed and operated for the Atomic Energy Commission by the Phillips Petroleum Company and was located at the Idaho National Engineering Laboratory. It began operation in 1958 and was shut down in 1968. The incident described below is included in this report because it resulted in damage to major equipment.

The SPERT-3 reactor was designed for experiments on the effects of rapid power increase (excursions) in pressurized-water reactors (PWRs). Although it was not designed to operate continuously at the high power levels at which PWRs normally operate, it was designed to duplicate, as nearly as economically feasible, the nuclear and other conditions (temperatures, pressures, coolant flow rates, etc.) normally found in PWRs. The reactor system contained coolant lines, coolant pumps, and a pressurizer, as in a normal nuclear steam supply system for a PWR, but it contained heat exchangers rather than steam generators as does a standard PWR system. The heat exchangers removed the heat from the water of the primary system, but the heat was dumped rather than used to generate electricity.

On Oct. 26, 1961, the reactor was not in operation. Its loading of fuel was insufficient to make it critical. Nonnuclear tests were being performed on some of the equipment, and all plant instruments indicated normal values. The insulation on the pressurizer began to smoke followed by the escape of steam from the pressurizer, even though the indications from the pressurizer instruments indicated normal values. The tests were stopped, and the pressure and temperature in the pressurizer were reduced.

An examination of the pressurizer revealed a crack along a welded seam. The conclusion was reached that the water-level indicator for the pressurizer was in error, and that the water level had dropped far enough to expose the pressurizer heaters. Thus exposed, they had heated the steam above the water level to temperatures sufficiently high to crack the vessel.

The pressurizer was replaced by one of improved manufacture; it was fitted with more reliable instrumentation; and the operation at SPERT-3 were resumed.

There were no injuries, and there was no release of radioactivity.

## 5.9 Hydrogen Fire at PM-3A (Refs. 15, 16)

The reactor in the PM-3A Nuclear Power Plant was a pressurized-water reactor with a single primary loop. The plant was designed and built by the Martin Company, Nuclear Division and was located in McMurdo Sound, Antarctica. It was designed for a plant power of 1.5 MW(e). It started up in 1962 and was shut down in 1973. The accident described below is included in this report because it was a precursor to a potentially serious accident.

The Portable Medium Power Plant 3A (PM-3A) was designed to provide electrical power at remote military installations. Its design was such that it could be readily transported and then assembled with tools that are available at a remote site. Figure 5.2 is a diagram of the plant. The four interconnected containment vessels (spent-fuel storage, reactor, steam generator, and void tank) also serve as shipping containers for the major reactor components. Note that the pressurizer is housed in the steam generator containment vessel.

The reactor had operated successfully for the 3 months prior to the incident. On the morning of Oct. 7, 1962, an automatic scram shut the reactor down. The reason for the scram could not be determined, but instrument problems caused by excess humidity had been experienced previously. Spurious signals resulting from humidity twice prevented the reactor from being restarted after the scram. On the third attempt, a scram initiated by a different signal occurred. Investigation revealed

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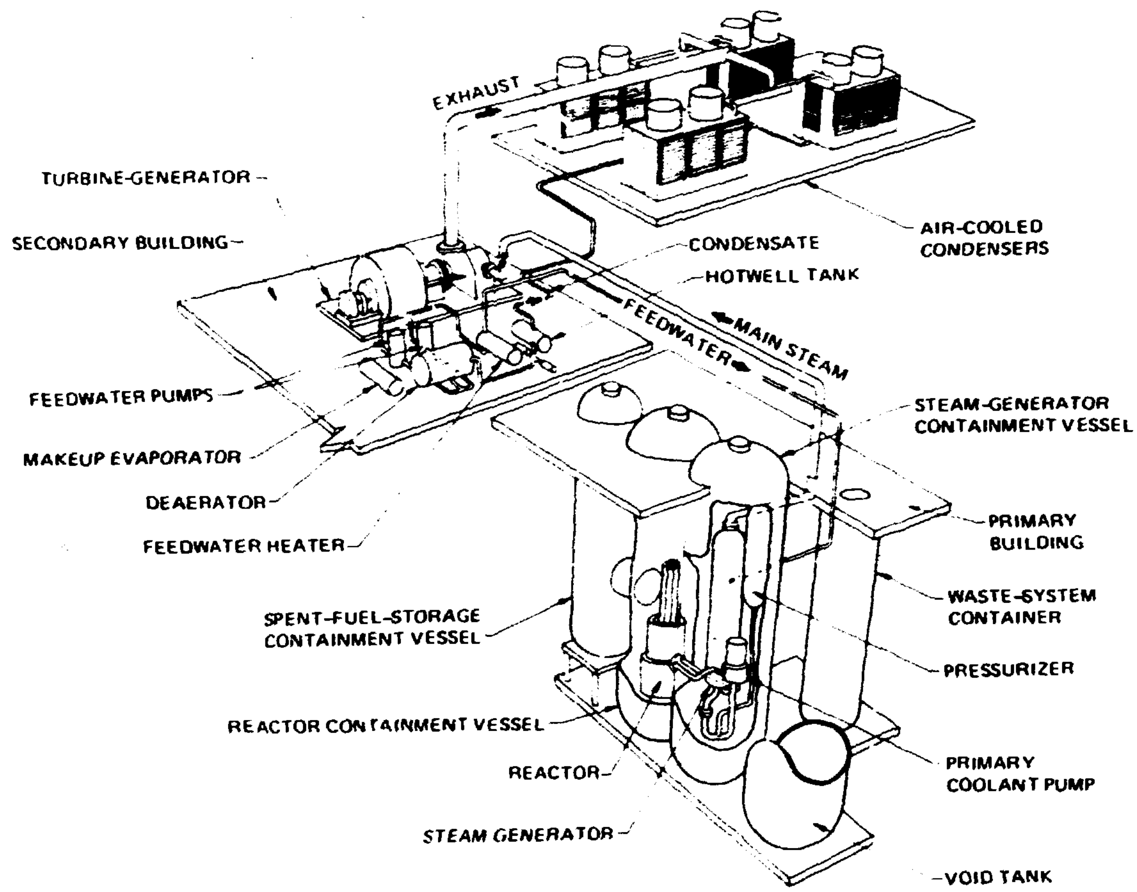


Fig. 5.2. Basic flow and orientation diagram of PM-3A Nuclear Power Plant.

that the signal had been generated by a short circuit in the electrical wires that passed through the containment vessel for the steam generator.

The system was allowed to cool for several hours, and then the containment vessel for the steam generator was entered. It was found that a fire of short duration, which had caused some superficial damage, had occurred in the upper portion of the vessel. It was ascertained that the seal for a valve on the pressurizer had allowed hydrogen gas to escape the system and accumulate at the top of the vessel, where it was ignited by a spark from the short circuit. Shocks or other manifestations of an explosion had not been noticed by the personnel. A combustible-gas detector located in the vicinity of the fire was not operating properly at the time.

Repairs were made and the following corrective actions were taken:

1. Fans and ducts were installed to prevent stagnant pockets of hydrogen from forming.

2. A second hydrogen detector was installed in the containment volume.

3. Air samplers were placed in the containment tanks.

4. Changes in the operating procedures were made to ensure that significant hydrogen leaks are prevented and to ensure that the operator is aware at all times of the hydrogen distribution in the system.

5. A hydrogen recombiner was added and a hydrogen detector was included in the recombiner line.

6. An alternate method for hydrogen detection was added to confirm hydrogen readings.

The plant resumed operation following these actions.

There were no injuries, and there was no abnormal release of radioactivity.

### 5.10 Fuel Element Melting at Oak Ridge Research Reactor[17,18]

The Oak Ridge Research Reactor is cooled and moderated by light water and is contained in a large aluminum tank. It has a capacity of 30 MW(t). It is owned by the Department of Energy and is located at the Oak Ridge National Laboratory in Oak Ridge, Tennessee. It started up in 1958. The accident described below is included in this report because it resulted in core damage.

The fuel in the Oak Ridge Research Reactor consists of a layer of uranium-aluminum alloy sandwiched between curved aluminum plates. The plates are grouped together in a box-like array (see Fig. 5.3) to form a 19-plate fuel element. Light water of the primary system is pumped through the core, where it picks up the heat that is generated by the reactor; then it is piped outside the reactor tank to a cooling tower, where it gives up this heat; after it is cooled, it is returned to the

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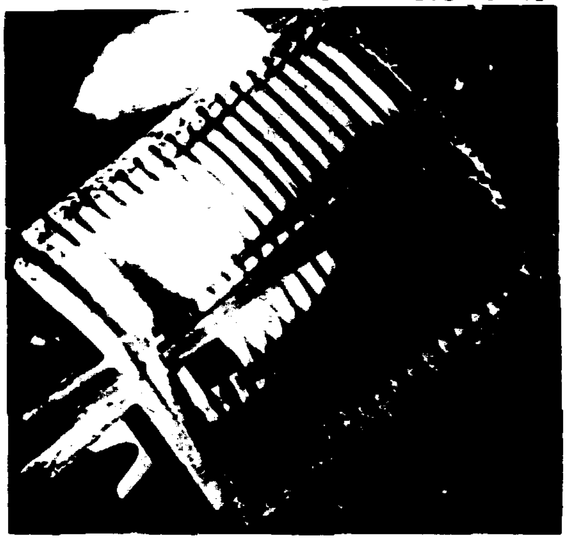

Fig. 5.3. End view of fuel element where melting occurred (Oak Ridge Research Reactor).

reactor. The reactor and the reactor tank are immersed in a pool of water, whose primary function is to provide shielding.

A decay tank (to allow radioactive contaminants to decay) and a degassifier (to remove gaseous contaminants) are provided. "Bleed" lines from the primary system pass some of the primary system water through a demineralizer (see Fig. 5.4), through a surge tank, and then back to the reactor tank.

The gases from the degassifier pass through filters and a caustic scrubber and are then released to the atmosphere via an off-gas stack.

In the early morning of July 1, 1963, the reactor was being brought to full power after being shut down for maintenance and installation of a new experiment. The reactor power was increased in small increments, and the systems were allowed to stabilize before the next increase. At 6 MW, a visual inspection was made of the top of the core through a viewing port to search for obstructions (at 6 MW, the Čerenkov radiation*

*Čerenkov radiation (named for the Russian scientist P. A. Čerenkov) is the light emitted when charged particles pass through a transparent material at a velocity greater than that of light in that material. It can be seen as a blue glow in the water around the fuel elements of pool reactors.

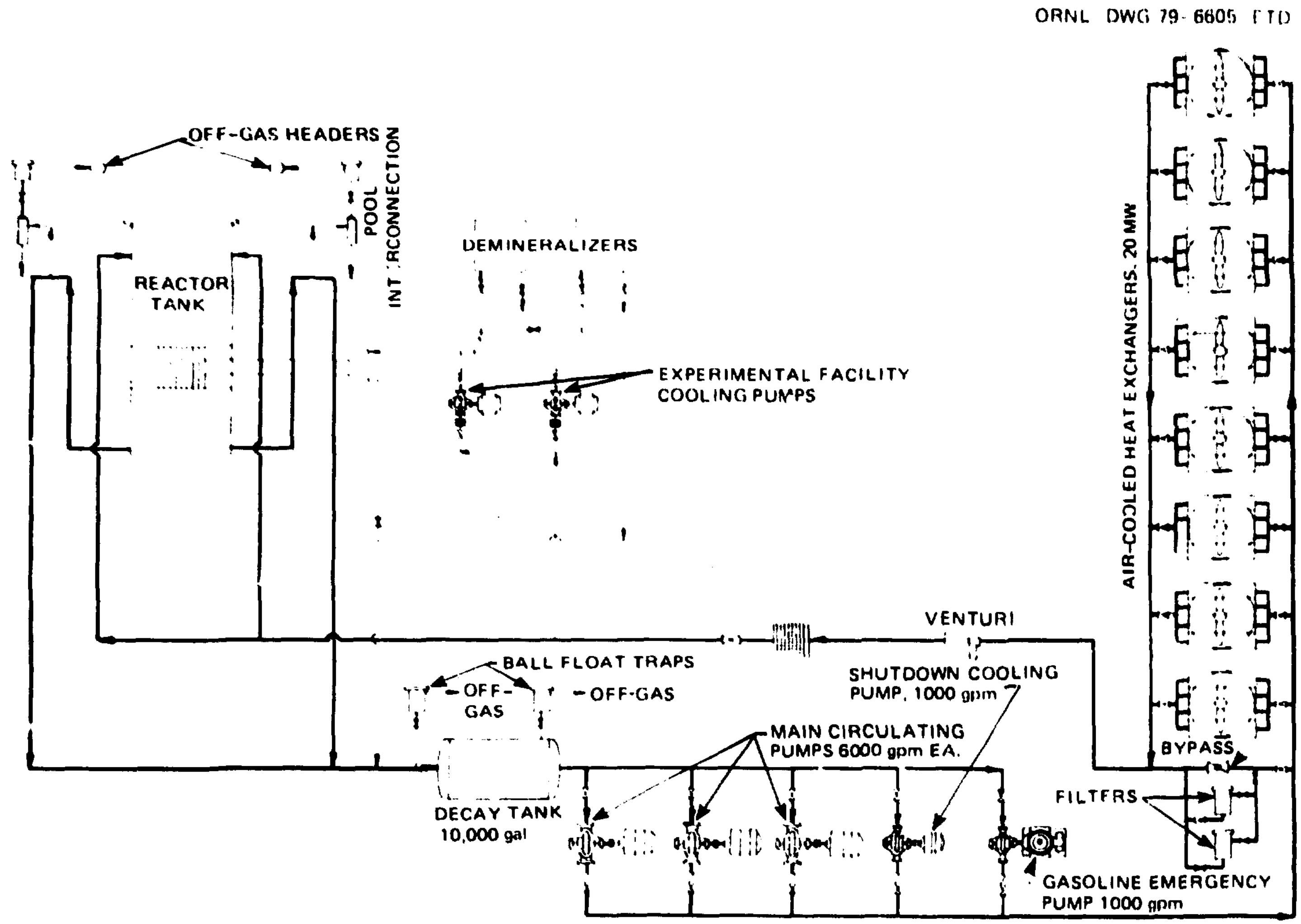


Fig. 5.4. Schematic diagram of the coolant system in the Oak Ridge Research Reactor.

was sufficiently bright that an obstruction would cast a shadow). Nothing was observed in the field of view, so the power increases were continued. Small fluctuations were observed in some of the instruments when the power reached 12 MW, but these were attributed to malfunctions of the servomechanism for the control rods. Actually, this was probably an indication of boiling in one of the fuel plates. When the power level reached 24 MW, radiation alarms associated with the primary cooling system sounded, and the reactor was shut down.

The levels of airborne radioactivity inside the building reached the evacuation level, and the building was evacuated. As a precaution, the reactor containment system was manually actuated before the evacuation. Activation of the containment system caused various intake and exhaust vents in the building to close; it reduced the pressure in the building so that all airflow through cracks and leaks would be from outside to inside; and it routed the exhaust air from the building through the filtering system. Volatile radioactive fission gases had been released from the primary system water into the building via the pool.

The highest radiation level measured outside the building was 2 R/hr (about 10 to 20 times higher than levels reached at full-power operation) in the vicinity of the components of the cooling system. These levels decayed by a factor of 10 in 4 hr. The highest level measured inside the building was 20 R/hr (about 100 times the levels reached at full-power operation) in the vicinity of the demineralizer for the water of the primary system. Estimates indicated that about 1000 Ci in fission products had been released into the water of the primary system. The release of radioactive iodine to the atmosphere was between 0.2 and 0.4% of that normally released during the year.

About 20 hr after the shutdown, the radiation levels had been reduced to normal and the building was reentered. Examination of the core revealed that a neoprene gasket had slipped off a fixture on the inside of the reactor tank and had become lodged in the top of one of the fuel elements, blocking the flow of cooling water. One of the fuel plates in this element had melted. Figures 5.5 and 5.6 illustrate the gasket wedged in the fuel element and the melted fuel plate. This fuel element was outside the field of view of the viewing port, and, therefore, the

ORNL PHOTO 4365-79

Fig. 5.5. Upper end of fuel element showing gasket that obstructed flow (Oak Ridge Research Reactor).

ORNL PHOTO 4363-79

Fig. 5.6. Melted fuel plate and adjacent plates (Oak Ridge Research Reactor).

blockage was not observed during the visual inspection that had taken place when the reactor was at 6 MW.

The faulty fuel element was removed and replaced, and normal operation was resumed on the day following the incident.

There were no injuries, but there was minor release of radioactivity to the atmosphere.

## 5.11 Rupture-Loop Failure in PRTR[19,20]

The Plutonium Recycle Test Reactor (PRTR) was moderated and cooled by heavy water. The reactor was designed to operate with various fuels (i.e., natural $UO_2$, plutonium-aluminum alloys, $PuO_2$-$UO_2$ mixtures, etc.) under diverse operating conditions. It was rated for a maximum power of 70 MW(t). No electricity was produced. It was designed and operated by the Pacific Northwest Laboratories (PNL), which is operated by the Battelle Memorial Institute. The reactor began operation in 1960 and shut down in 1969. It was located near Richland, Washington. The incident described below is included in this report because it resulted in significant recovery costs.

The PRTR had a special vertical channel in the center of the core to facilitate the testing of fuel elements whose cladding had been deliberately failed. The fuel elements that were placed in this channel were cooled with light water. The pumps and piping system that handled this water were separate from those that were required for normal operation of the reactor. This separate system was called the rupture loop. Since the deliberate failures in the cladding of the fuel elements that were tested in this loop would permit the escape of radioactive fission products into the light-water coolant, the facility was designed to safely handle all aspects of the treatment of this water. The rooms containing the pumps, cleanup equipment, etc., for the light water were located underground, as were the reactor and its surrounding containment vessel.

On Sept. 29, 1965, while a fuel element in the rupture loop was undergoing high-temperature tests that were designed to melt the central portions of the fuel, the reactor was scrammed on a low-pressure signal from the loop. The metal sheath within the channel that contained the light-water coolant for the fuel element being tested had ruptured. This permitted radioactive water and gases to escape into the containment vessel.

The radiation in the control room reached 35 mR/hr, and nonessential personnel were evacuated. Construction work on nearby facilities was temporarily suspended as a precautionary measure.

The contaminated light water managed to enter the tank (collandria) containing the heavy-water moderator for the reactor through penetrations that had been poorly sealed and thereby contaminated the heavy-water also.

The radiation level in the reactor hall, a passageway above the reactor that was accessible to personnel during reactor operation, reached a maximum of 20 R/hr and was reduced to 1 mR/hr following cleanup. The radiation levels in the access space below the reactor were up to 35 R/hr and were reduced to about one-third that level after cleanup. Two hours after the accident the measured radiation level 300 ft from the reactor was 20 to 50 mR/hr.

The maximum personnel exposure to the exterior of the body was 100 mR. The maximum internal exposure (due to inhalation) was 105 mrads. The cost for repair and decontamination was $895,000 (Ref. 20).

There were no injuries, and there was no appreciable release of radioactivity to the environment.[21]

## 5.12 Loss of Coolant Damages Core at Lucens[22]

The experimental reactor at the Lucens Experimental Nuclear Power Station at Lucens Vaud, Switzerland, is deuterium-moderated and $CO_2$-cooled. It is rated for a power of 30 MW(t). The accident described below is included in this report because it resulted in core damage.

The accident at the Lucens Experimental Nuclear Power Station occurred on Jan. 21, 1969, during startup procedures. It had operated at nearly full power during the previous 6 months and had been shut down for routine maintenance. Figure 5.7 is a schematic diagram of the essential features of the core of the Lucens reactor. It is located in an underground rock cavern.

ORNL DWG 79-6610 ETD

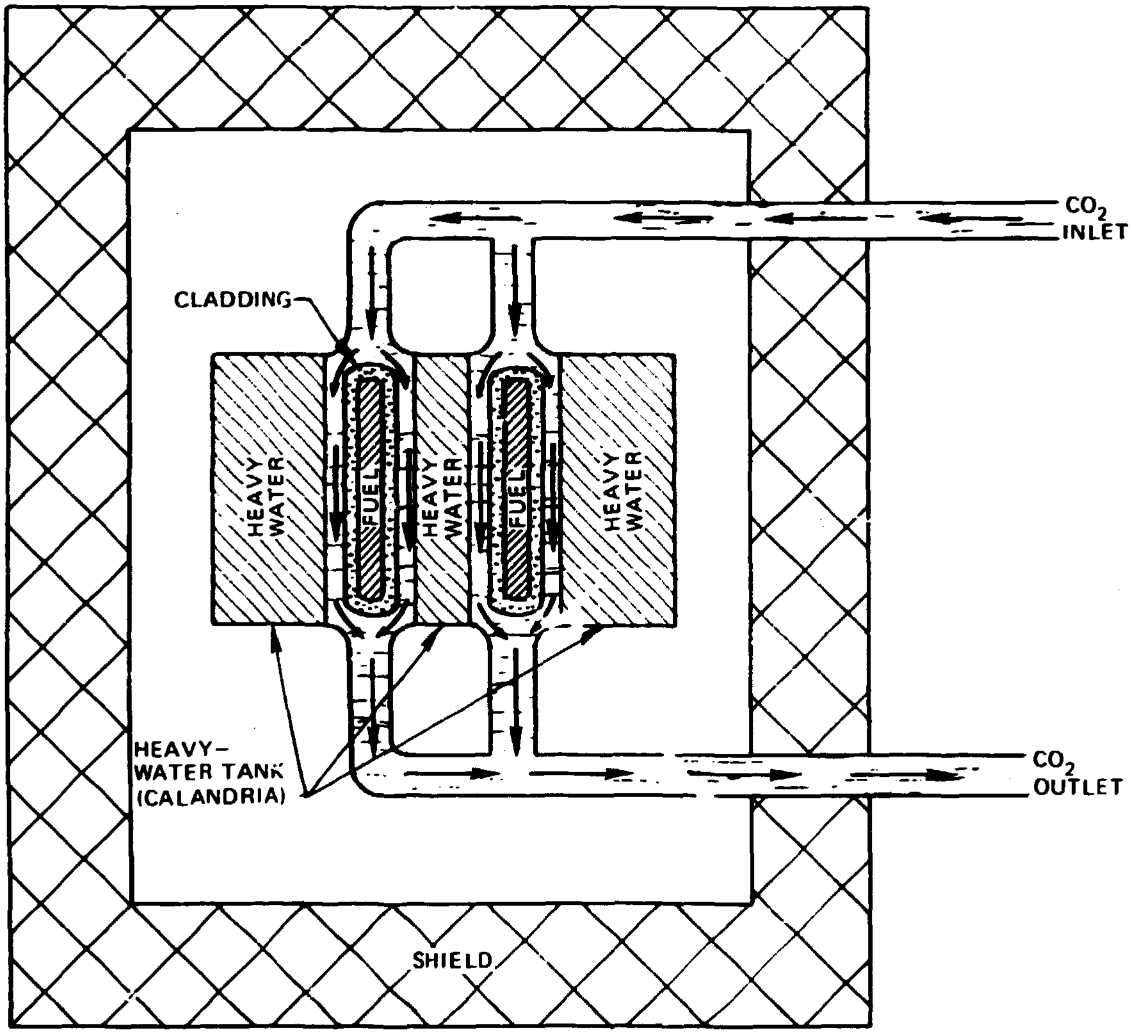


Fig. 5.7. Schematic diagram of the main features of the core of the Lucens reactor.

When the power level reached 12 MW(t), several things happened within 10 min. The $CO_2$ coolant pressure dropped to atmospheric pressure (it normally is about 60 times atmospheric pressure), and the reactor was scrammed. Then radioactivity levels increased in the cavern, whereupon the cavern was sealed. Finally, there were indications of the leakage of large amounts of heavy water. The moderator tank had ruptured.

The cause of the accident was a breach in the seal of the pressurized-$CO_2$ system. The cooling capability of any gas is decreased with decreasing pressure. Thus, when the $CO_2$ pressure dropped to atmospheric, the fuel had heated up, and one fuel element had melted along with its cladding. The melted fuel was the source of the radioactivity that was released into the cavern. The channels in the moderator tank that contain the fuel had collapsed onto the fuel with the loss of $CO_2$ pressure.

The radiation level in the cavern reached a level of a few hundred rems per hour and dropped by a factor of 1000 to a few hundred millirems per hour the first 44 hr. Four days after the accident, the cavern was vented to the atmosphere through filters, and the level of radioactivity released was negligible compared to natural background. At no time was the radioactivity released beyond that permitted under normal operation.

There were no injuries, and there was no release of radioactivity beyond prescribed limits, but the reactor was severely damaged.

# 6. DISCUSSION AND CONCLUSIONS

This report is a compilation of all significant accidents which have occurred in both U.S. and foreign nuclear reactors (excluding criticality facilities) and which have been publicly documented. In assessing this compilation, the reader should be aware that the foreign reactor accidents presented are limited to those reported in the open literature. We are aware of foreign reactor accidents of unknown severity which have not been publicly documented. On the other hand, the U.S. experience was gleaned from a total of over 20,000 Licensee Event Reports from commercial nuclear power facilities in this country and other documentation on all types of U.S. reactors available to the Nuclear Safety Information Center. However, this documentation does not cover naval reactors (such military information is not publicly available). Events occurring at commercial reactors in the United States — now reported in Licensee Event Reports (LERs) — include all conceivable operational occurrences of possible concern to the U.S. Nuclear Regulatory Commission. These LERs are publicly available (at the NRC Public Document Room) as soon as they are submitted to the Nuclear Regulatory Commission and are compiled and published annually by the Nuclear Safety Information Center (see the list of such reports in the bibliography appended to this report). In addition to these LERs, the Nuclear Safety Information Center collects documentation on accidents of any type occurring at all types of reactors and thus has rather complete files on all reactor accidents (exclusive of naval reactors) that have occurred in the United States. As previously noted, information on accidents at critical facilities is also readily available, but is not included here because of its lack of relevance.

The fact that the accidents included in this compilation are such a small fraction of the total number of recorded events is not a criticism of the selection criteria used, but rather an indication of the extent of reporting required by the Nuclear Regulatory Commission. These reporting requirements are defined in NRC Regulatory Guide 1.16 (Ref. 1).

The occupational and public health effects from the reactor accidents summarized in this report are presented in Table 6.1. It should be borne

Table 6.1. Number of deaths, injuries, and radiation exposures resulting from nuclear reactor accidents

| | At central station power plants | | At all other reactor facilities* | |
|---|---|---|---|---|
| | United States | Foreign | United States | Foreign |
| Occupational death | 2 (Surry, 1972) | | 3 (SL-1, 1961) | |
| Occupational injury (exclusive of exposure) | 7 (Robinson, 1970)<br>1 (Surry, 1973)<br>2 (Cooper, 1975)<br>1 (Millstone, 1977) | | | |
| Occupational exposure to radiation (due to accidents) | | | ? (HTRE-3, 1958) | ? (NRX, 1952)<br>? (NRU, 1958) |
| Public death | | | | |
| Public injury | | | | |
| Members of the public exposed to radiation | Three Mile Island, 1979 | | | ? (Windscale, 1957)<br>? (NRX, 1952) |

*Question mark indicates insufficient data.

in mind that Table 6.1 does not list those events which resulted in some increase in release of radioactivity but not enough to exceed allowable limits, nor does it include those occupational exposures which occur in various accident control and recovery activities. Furthermore, the fact that there are no entries under foreign central station power plants is believed to indicate a lack of information on accidents at these facilities rather than superior performance of foreign reactors.

As previously noted, the U.S. accident experience is believed to be fully reflected in this report and in the summary results presented in Table 6.1. These data indicate that there have been only 2 deaths and 11 injuries at commercial nuclear power plants in the United States. Furthermore, these deaths and injuries were not due to radiation, but to more conventional occupational hazards (although they occurred in unconventional environments). These statistics may be converted into death and injury rates by observing that, by mid-July 1979, the United States exceeded 500 reactor-years of commercial nuclear power plant operation.[2] While these data represent only one portion of the overall nuclear fuel cycle, they do represent that portion over which the most uncertainty exists and regarding which the most concern had been expressed.

Comparing these data with the data on other fuel cycles is frought with many difficulties. However, it is clear that the U.S. reactor accident experience to date does not make a large contribution to the total number of deaths and injuries resulting from the reactor fuel cycle. Thus, this favorable U.S. accident experience is in part responsible for the 20 (or greater) to 1 advantage which the nuclear fuel cycle enjoys over the coal fuel cycle in terms of either public or occupational health effects.[3-12]

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