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# PASSIVE AND INHERENT SAFETY TECHNOLOGIES FOR LIGHT-WATER NUCLEAR REACTORS

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Abstract

Accidents at the Chernobyl and Three Mile Island (TMI) nuclear power plants have resulted in a reevaluation of safety philosophies for future light-water reactors (LWRs). One direction for development of improved nuclear power reactor safety is the use of passive safety systems and inherent safety characteristics. A four-step program has been initiated at Oak Ridge National Laboratory (ORNL) to investigate these options for LWRs. The steps are: definition of goals, determination of functional requirements for safety of LWRs, identification/characterization/invention/development of passive/inherent safety technology options (safety building blocks), and development of integrated reactor designs. A brief description of the progress at ORNL and elsewhere in each area is provided.

Two prerequisites are required before new approaches to safety can be developed – determination of safety goals, and identification of the functional requirements for safety. The overall objectives of this work is to achieve PRIME safety. PRIME is an acronym for Passive safety systems, Resilient safety, Inherent safety characteristics, Malevolence resistance, and Extended time safety. The basis, definition, and implications of PRIME safety are provided.

Throughout the past year, the major emphasis has involved identifying, describing, and evaluating both existing and proposed passive and inherent safety technologies applicable to LWRs. These technologies provide the building blocks (structures, systems, and components) upon which power plants can be designed. Over 70 classes of such technologies were identified, including those to regulate nuclear reactor power levels, ensure reactor cooling under all circumstances, control energy releases in accidents, and contain radionuclides. Many of these technologies are applicable to other types of nuclear

power plants and chemical plants. A description of this study, its conclusions, and applicability of these technologies to nuclear and chemical plant design is provided.

The final step is to develop advanced reactor options with PRIME safety. Various designs have been proposed by different organizations. The systematic approach to understanding safety goals and requirements establishes a basis for characterizing and categorizing LWR options. Preliminary observations of the status of development of new concepts are included.

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## I. Introduction

The Chernobyl and Three Mile Island (TMI) accidents have resulted in a reevaluation of safety philosophies for nuclear power reactors. From this reevaluation, several directions for safety have evolved: (1) improvement of the institutions associated with nuclear power, (2) improvement of existing technologies, and (3) design of reactors with passive/inherent safety. Passive/inherent safety implies a technical revolution in our approach to nuclear power safety. The examination of new directions in the nuclear industry is, in many ways, parallel to what occurred in the chemical industry following the Bhopal chemical disaster in India which killed thousands [Kletz, 1984; Corbett, 1988]. This direction is discussed herein for light-water reactors (LWRs) – the predominant type of power reactor used in the world today.

At Oak Ridge National Laboratory (ORNL), our approach to the development of passive/inherent safety for LWRs consists of four steps:

- identify and quantify safety requirements and goals;
- identify and quantify the technical functional requirements needed for safety;
- identify, invent, develop, and quantify technical options that meet both of the above requirements; and
- integrate safety systems into designs of economic and reliable nuclear power plants.

Significant progress has been achieved in the first three steps of this program. The last step involves primarily the reactor vendors. These activities, as well as related activities worldwide, are described here.

## 2. Safety Goals

Before research and development of any technology can begin, it is essential to define the goals – particularly, safety goals. This involves asking several questions such as:

1. What are the incentives for new safety goals?
2. What are the options?
3. What are the implications?

### 2.1 Why Consider New Approaches to Nuclear Power Safety?

A series of events, starting with the accident at the TMI Nuclear Power Plant, has led to a fundamental rethinking within the nuclear community about historical approaches to nuclear power plant safety. Public acceptance, economics, and environmental issues have all contributed to this reassessment.

Nuclear energy has become controversial, with issues such as safety and radioactive waste disposal receiving major attention. The TMI and Chernobyl accidents have contributed to the controversy, as have a variety of life-style issues.

Nuclear power costs have increased. This includes direct costs (capital and operating) and indirect costs (investment risk). Associated with these cost increases has been a wide variability of costs between different plants (Cook, 1985). It appears that 30 to 60% of the costs are associated with safety. This implies that if major improvements in economics are to be obtained, new approaches to safety are required. Several factors associated with economics are particularly noteworthy:

- The combined operating and fuel costs [UDI, 1988] of nuclear power plants [21.8 mills/kWh(e)] have grown rapidly until in 1987 they exceeded the average operating and fuel costs of coal-fired power plants [20.7 mills/kWh(e)] in the United States. A more detailed analysis shows that the growth of nuclear fuel costs [7.5 mills/kWh(e)] has been low and is low compared to equivalent coal fuel costs [16.4 mills/kWh(e)]; but, operating costs have risen rapidly. Many of the operating costs reflect tightening safety requirements. Safety systems require very high quality-assurance standards, thus implying high maintenance costs. Furthermore, other requirements that support safety, such as security, have expanded with corresponding cost increases. Nuclear power is now economical only in those parts of the country with high fossil fuel prices.

- The complexity of current nuclear power plants makes operation and maintenance difficult [Golay, 1988; Golay, 1990b]. Functional reliability and production costs are <u>highly dependent</u> on management and work-force skills. The nuclear reactor itself is simpler than a fossil fuel boiler, and the steam electric equipment is similar to that used in fossil plants. The complexity lies in the active safety systems and their interactions with the power plant. Some indicators of the impact of complexity on operations are the extremely wide variations in plant reliability and production costs. For example, there is over a factor of 5 difference in operating costs for similar plants. There are also large variations in plant reliability [Golay, 1990a]; some plants and utilities have consistently good records, while others have had both good and bad experiences.

In terms of complexity and resulting demands on management and work-force skills, the aircraft industry is the only commercial equivalent. It is, however, simpler in one important respect – the most complex technical operations, such as maintenance, are centralized in locations where a few specialized companies conduct these operations for the operating airlines. This option does not exist for nuclear power plants since they cannot be transported.

- Finally, the utilities have a concern for investment risk [Carnesale, 1981]. Historically, industrial organizations have understood that accidents could destroy industrial facilities. However, they did not fully appreciate until after the events at Bhopal, TMI, and Chernobyl, that the total financial risks for some types of accidents were far greater than the loss of the facility.

  The accident at TMI had many far-reaching effects. For example, the operation of the "sister" TMI-1 reactor was stopped for several years. In addition, the utilities that owned reactors similar in design to TMI had to undertake large retrofit programs, suffer extensive licensing delays, and receive much public criticism. The accident also contributed to the eventual shutdown of a similar reactor in California – Rancho Seco.

Recent environmental concerns – particularly the carbon-dioxide greenhouse effect – may imply expansion of nuclear power by an order of magnitude [Forsberg, 1990] and largescale use in undeveloped countries. This has major implications for long-term safety requirements and approaches to safety.

- The public acceptance of any technology partly depends on the absolute number of accidents, not the accident rate. This was first emphasized in 1963 in the aircraft industry by the Swedish engineer Bo K. O. Lundberg [Weinberg,1989; Lundberg, 1963]. Lundberg recognized that if the aircraft accident rate was constant and there was continued growth of the industry, the public acceptance of the industry and of flying would become a major problem due to the publicity of each accident. The experiences of the aviation industry on the institutional necessity for reducing accident rates is probably applicable to the nuclear industry.

- If nuclear power is used on a large scale in underdeveloped countries, there will be increased concerns about low skill levels, political instabilities, and limited resources to be applied to safety [Kessler, 1990; Goldman, 1990; Hibbs, 1990]. These factors may increase accident probabilities if passive and inherent safety technologies are not used.

## 2.2 Implications

An overall perspective of the technology shows that most of the technology is reliable and economical. The difficulties are concentrated in a small area – the high cost and controversy associated with safety systems. This implies that a research program with a relatively narrow focus has the potential for high payoffs. In many respects, the technical position is analogous to that of the commercial airline industry in the early 1950s. At that time, the industry was a regional, high-cost enterprise because aircraft performance, cost, and safety were limited by the high weight-to-power ratios of complex piston engines. However, the development of the jet engine simultaneously improved economics, performance, and safety. In a similar manner, existing LWR safety systems are the limiting factor for nuclear power.

## 2.3 Approaches to Safety

There are different approaches to safety. The designs of current nuclear safety systems have evolved from various experiences over a number of years. The LWR was originally developed for submarine service; therefore, the early design goals were for a very compact, high-power-density nuclear power plant. The requirement of small size placed a major emphasis on the use of active, mechanical equipment for operating and protection systems because such systems are more compact than other alternatives.

For power reactors, the submarine reactor was scaled up by almost a factor of 100, and safety systems for accident control and mitigation were added. Active, electromechanical systems (valves, motors, pumps, diesel generators) were included because they appeared inexpensive, were easy to design, and had already been tested. Due to a variety of outside factors, the basic design was fixed very early (Arthur, 1990). As described by Arthur, "The role of the U.S. Navy in early reactor-construction contracts, efforts by the National Security Council to get a reactor – any reactor – working on land in the wake of the 1957 Sputnik launch as well as the predilections of some key officials all acted to favor the early development of light-water reactors . . . by the mid-1960s, fixed the industry's path."

Some knowledge of nuclear power reactors is required in order to understand why these safety systems have become so expensive and controversial. Nuclear reactors are designed to produce heat by fissioning (splitting) of atoms of uranium into smaller pieces. This heat is then used, as in coal, oil, or natural gas power plants, to generate steam that, in turn, produces electricity. Unfortunately, the "fission products" formed in the heat generation process are very toxic. A safe reactor is one that does not allow these toxic fission products – the by product of nuclear reactions – to escape to the environment.

The concern about nuclear reactor accidents is that hazardous fission products are located in a high-temperature heat source. If the reactor should overheat, hazardous vapors and small particles will be generated and might escape to the atmosphere. The problem is similar to preventing a fire in a warehouse containing toxic materials. Abnormally high temperatures destroy containers and help spread toxic materials to the environment. For a nuclear reactor, there is a <u>complicating factor</u>. Immediately after reactor shutdown, the heat generation rate in the reactor does not drop to zero. Instead, the reactor continues to produce heat at a fraction of its full power level (tens of megawatts for a large reactor) for weeks from radioactive decay heat. This heat release <u>cannot</u> be stopped – it is a law of nature; thus, cooling must be continued for a considerable time after shutdown. This is not a problem for very small reactors, such as research reactors because the total heat is very small and will conduct through the walls of the machine to the atmosphere. For large reactors, however, the heat would be sufficient to melt the core unless there is cooling for months after shutdown. For typical LWRs, a loss of core cooling will lead to core destruction – such as occurred at TMI – within hours.

All U.S. power reactors are of the water-cooled type. The cooling method to remove decay heat is simply to add water. The active safety emergency core-cooling systems operate similar to those used by fire departments. Diesel engines provide power to operate pumps that deliver water onto the reactor core. Pumps and other equipment require many megawatts of electricity, which <u>must always be available</u>. It is very expensive to ensure, through multiple redundant equipment, that loss of cooling will never occur. For example, the installed cost of a set of safety-related emergency diesel generators exceeds $\$50 \times 10^6$.

The difficulties with active safety systems have resulted in new approaches to reactor safety, in particular, the use of passive and inherent systems. To avoid confusion of safety terms, the International Atomic Energy Agency (IAEA) held a meeting in Västerås, Sweden, in 1988, which drafted consensus definitions for safety terms (Table 1). These definitions are used in the discussions that follow.

## 2.4 PRIME Safety

The above concerns with nuclear power have resulted in development of a set of design goals which, if achieved, would eliminate safety as a public acceptance, environmental, or economic issue. Such design goals are independent of the technology and may be applicable to chemical plants.

Five characteristics for safety have been identified as necessary to eliminate major accidents. These are Passive safety systems, Resilient safety, Inherent safety characteristics, Malevolence resistance, and Extended safety. The term "PRIME safety" summarizes these characteristics. An understanding of PRIME will give us a good grasp of this revolution in safety philosophy.

PRIME safety implies using only passive safety systems and inherent safety characteristics in industrial plants versus the active safety systems used in today's plants. We can use examples from fire protection to explain these terms. A concrete warehouse full of pottery is inherently safe against fire. In other words, a fire cannot occur. Inherent safety implies no need for safety systems. An example of passive safety are water sprinklers. Active

Table 1. Draft IAEA descriptions of safety terms[a]

| Term | Description |
| --- | --- |
| Passive component | A component which does not need any external input to operate. It may experience a change in pressure, temperature, radiation, fluid level and flow in performing its function. The function is achieved by means of static or dormant unpowered or selfacting means. |
| Active component | Any component that is not passive is active. |
| Passive system | A system is composed of passive components and structures. |
| Passive safety function | Passive safety function is a function to be achieved by means of passive components or systems. |
| Inherent safety characteristic | A characteristic refers to the elimination of a specified hazard by means of the choice of material and design, through the laws of nature only. |
| Fail safe | The term refers to the behavior of a component or system, following a failure (either internal or external). If a given failure leads directly to a safe condition, the component or system is fail-safe with respect to that failure. |
| Grace period | Grace period relates to a period of time during which safety is ensured without the necessity of personnel action or attendance in the event of an incident/accident. |
| Walkaway safety | A plant in which safety is ensured for protracted period of time, without personnel action or attendance, may be termed Walkaway safe for that period. |
| Foolproof | Safe against human error or misguided human action. |
| Error tolerance | The term error-tolerance, often called forgivingness, is used to describe the degree to which human inaction (or erroneous action) can be tolerated. |
| Self-acting system | A system, which may contain active components, is considered to be self-acting, if it is able to function without action by, or inputs from, other equipment. |

[a] Taken from [IAEA, 1988].

safety, then, would be the fire department. Nuclear reactors cannot be made inherently safe because they contain hazardous radioactive materials, but reactors can be made inherently safe against specific types of accidents. U.S. power reactors are inherently safe against the type of accident that occurred at Chernobyl [Martinez, 1990]. While active safety works most of the time, plant operators (like fire departments) can make errors. Operator error was a major cause of the accidents at both TMI and Chernobyl.

Elaborate and expensive safety systems can be built; however, if they are not maintained, they may fail. Because safety systems sometimes complicate operations and accidents are rare events, there is often the incentive for an operator to bypass safety systems. To prevent this problem, safety systems must be resilient. The historical example of resilient safety is the railroad air brake – an active safety system that is very resilient. Railroad air brakes are designed to be on. To hold the brakes in the off position, the locomotive engineer must continuously supply high-pressure air to each railcar brake system. If either a brake line or the air pressure should fail, the brakes are immediately activated. In order for the train to function, the brake system – a resilient safety system – must work properly. In resilient systems, maintenance to ensure operation also ensures safety.

The fourth requirement for safety is malevolence resistance. Malevolence resistance protects against sabotage, terrorists, and off-the-shelf conventional military munitions. It is thought, by many, that the Bhopal chemical disaster was initiated by employee sabotage. In industries with high levels of safety, such as the aircraft, nuclear and chemical industries, sabotage may become a major accident initiator because other accident initiators have been eliminated. The "dark side" of man necessitates development of safety approaches that are not dependent on security forces. Malevolence resistance also provides protection against all types of operator error such as that which occurred at TMI or through shutdown of safety systems to improve plant availability – a problem in some parts of the world (Hibbs, 1990). Active safety systems (valves, computers, operators), which can be turned off, are sensitive to sabotage; therefore, malevolence resistance as a precondition requires both passive and inherent safety.

Finally, extended safety is required; that is, the plant must stay in a safe state for some defined period after an accident, sabotage, or attack without releasing hazardous materials. Typically, a period of 1 week is chosen to provide time for corrective actions.

This paper focuses on LWRs with PRIME safety characteristics. However, certain other reactor concepts with modifications may also meet these goals. The Modular High-Temperature Gas-Cooled Reactor (MHTGR) could potentially meet such goals with relatively minor modifications. In addition, proposed advanced heavy-water power reactor designs may be in accordance with these requirements.

## 2.5 Implications of PRIME Safety

PRIME safety has both public acceptance and organizational implications. Clearly, it has the potential to greatly improve the public's image of nuclear power. However, it also implies a shift in relative responsibilities for safety between operating and research/development/design organizations. Fundamental characteristics of the labor force support such a shift of responsibilities.

As a technology is used on a larger scale, the competence of the average operator decreases. New technologies are pioneered by very talented, dedicated individuals. In the nuclear business, most of the early reactor operators eventually received Nobel prizes. If a technology is to be widely used, it must evolve with time to match the lower knowledge and skill levels of the average operator. This evolution can be seen in both the aircraft and automobile industries over the past 80 years.

As described earlier, current power reactors depend on active safety systems. Loss of such systems will result in reactor core damage within hours. Ensuring safety implies ensuring competent operational organizations at every power reactor over the operating life of each reactor. PRIME safety implies that if the reactor is designed and built properly, there is a much reduced dependence on plant operations. Equally important, if a plant is inspected and meets standards, the safety is built in and cannot vanish with the next operating shift or degrade rapidly through neglect. PRIME reduces the required operator skill levels for any defined level of safety. It is much simpler to find a few good designers and inspectors than to staff, with assurance, potentially thousands of plants with highly skilled operators.

## 3. Functional Safety Requirements for LWRs

### 3.1 Uses of Functional Analysis

A nuclear power plant is composed of many structures, systems, and components (SSC)-engineered building blocks. Different designs of LWRs reflect differences in the designer's selection of building blocks. For example, a designer could use either active or passive cooling systems to cool the hot nuclear reactor core. Systematic identification and characterization of passive and inherent safety system building blocks are necessary steps in creating new reactor options and in identifying research needed to enhance the safety of reactors now being designed. However, before building blocks can be identified, it is first necessary to determine what functions the safety systems must accomplish.

Functional requirements analysis as a basis to assess safety options offers multiple benefits. First, it provides a clear, logical way to identify what each SSC accomplishes and to delineate the options that are available to achieve a particular requirement.

Second, organization of SSCs by functional requirements is one way to identify missing options. For example, options for heat sinks include internal plant heat capacity or dumping heat to the environment – ground, water, or air. This systematic approach leads to the identification of options; some of which have already been reported in the literature but, in other cases, have not been identified.

The last benefit of using functional requirements as a method of organization is that it provides a "checklist" of safety issues to address for new concepts. For example, accidents require energy to occur. Energy sources in a nuclear plant include radioactive decay heat, thermal mechanical energy, chemical energy, and fluid (pressure and temperature) energy.

## 3.2 Results of Functional Analysis

The functional breakdown of nuclear safety requirements is shown in Table 2. This functional requirements analysis was derived from earlier, similar efforts in the United States (HTGR, 1986; EPRI, 1990). There are four major requirements which prevent radionuclide release to the environment:

- *Maintain Core Integrity [prevent reactor core damage]*
- *Control Transport from Primary Circuit [first barrier to release of radionuclides]*
- *Control Transport from Containment Building [second barrier to release of radionuclides]*

- *Control Transport from Site [offsite emergency planning]*

Each of these requirements was, in turn, divided into lower-level, functional requirements as shown in Table 2. This type of functional analysis assists in identifying passive and inherent safety SSCs, clarifying what functional requirements a particular SSC fills (some SSCs fill multiple requirements), and provides a basis for judging SSCs.

### 3.3 Critical Safety Issues

As indicated in Table 2, there are many requirements for safety, but the fundamental characteristics of LWRs make some easy to accomplish and others more difficult. To prevent nuclear accidents, there are only two key requirements: Function 1.1 – Control Core Heat Generation Rate, and Function 1.2 – Remove Core Heat. The failure to accomplish Function 1.1 led to the accident at Chernobyl, while the failure to accomplish Function 1.2 resulted in the TMI accident. The characteristics of LWRs are such that the type of accident that occurred at Chernobyl is impossible in an LWR. Other types of loss of control of heat generation rate accidents are not impossible but very unlikely. In contrast, the loss of reactor core cooling (Function 1.2) is a very real concern, especially for a large reactor, which could experience core melting even when shut down.

## 4. Structures, Systems, and Components

Innovative industrial facilities may combine old SSCs (building blocks) in new ways or include new SSCs. Advanced nuclear power plants, like advanced chemical plants, include a mixture of old and new technology. The development of LWRs with PRIME safety initially requires identification, invention, understanding, and development of new SSCs. Only after component technologies are understood can integrated designs be developed.

Table 2. Functional safety requirements for LWRs

| Number | Functional requirements |
|---|---|
| 1 | Maintain Nuclear Reactor Core Integrity |
| | 1.1 Control Core Heat Generation Rate (Reactivity Control) |
| | 1.1.1 Provide Process Shutdown Mechanism |
| | 1.1.2 Limit Excess Reactivity (Excess Power Levels) |
| | 1.1.3 Control Reactivity with Inherent Feedback |
| | 1.2 Remove Core Heat |
| | 1.2.1 Maintain Core Coolant Boundary Integrity |
| | 1.2.2 Maintain Core Coolant Makeup |
| | 1.2.3 Transport Heat to Ultimate Heat Sink |
| | 1.2.3.1 Transport Heat |
| | 1.2.3.2 Heat Sink |
| | 1.3 Control Chemical Attack of Clad |
| 2 | Control Transport of Radionuclides from Primary Circuit |
| | 2.1 Maintain Pressure Boundary Integrity |
| | 2.2 Control Primary Circuit Pressure |
| | 2.3 Isolate Primary Circuit from Balance of Process |
| 3 | Control Transport from Containment |
| | 3.1 Maintain Containment Boundary Integrity |
| | 3.1.1 Effect Containment Isolation |
| | 3.1.1.1 Containment Structure |
| | 3.1.1.2 Containment Isolation |
| | 3.1.1.3 Pressure Control |
| | 3.1.2 Control Energy in Containment |
| | 3.1.2.1 Reduce Energy Sources |
| | 3.1.2.2 Heat Removal |
| | 3.2 Immobilize Releases |
| 4 | Control Transport from Site |

## 4.1 Identification of Passive and Inherent SSCs for LWRs

A major effort at ORNL has been initiated to identify passive and inherent SSCs (building blocks) applicable to LWRs. This effort [Forsberg, 1989] has included worldwide literature and patent searches, reviews of safety analysis reports of current reactors, discussions with universities and reactor vendors, and review of other reactor concepts (High Temperature Gas-Cooled Reactor, Liquid Metal Reactor) to identify technologies useful for LWRs.

Structures, systems, and components that were identified in this study have been analyzed and described using the standard format shown in Table 3. The descriptions of each technology include several technical judgments:

- The functional requirements that a particular SSC could potentially accomplish are identified.
- The current status of the technology is estimated. This can vary from standard commercial practice to a speculative concept for which no detailed engineering analysis has been done and the technical feasibility is uncertain.
- Additional functional requirements to support the proposed SSC are identified.

If several related SSCs were found to operate on the same principles, they were included in a single description. A good example of this is a suppression pool, which is a device designed to condense steam, remove radionuclides, and reduce pressure in the containment building over the reactor after a reactor accident. This suppression pool is a type of off-gas scrubber that operates by bubbling air and steam mixtures from a damaged reactor core through baths of water. In the United States, BWRs have a single large suppression pool

Table 3. Description of Structures, Systems, and Components Using Standard Format

| | |
|---|---|
| Title | Title of SSC (building block) |
| Functional requirements | List of the functional requirements achieved by SSC |
| Safety type | Inherent, passive, or active |
| Developmental status | Current status of this technology on a scale of 1 to 6:<br><br><u>Status</u> — <u>Definition of status</u><br>1 — In commercial application in multiple LWRs<br>6 — Speculation |
| Reactor type | Types of reactors to which the technology could be applied: LWR, BWR, PWR |
| Examples of implementation | Examples may include non-LWR reactors |
| Description | This will generally include a very simple nonmathematical description of the physics of the SSC, followed by a summary of an actual design using a summary table or a figure |
| Alternative versions | Alternative versions or use of the technology |
| Status of technology | Current status of technology (discussion) |
| Advantages | Potential advantages of particular SSC; may include safety, cost, and operational factors |
| Added requirements | Some safety systems may have additional functional requirements beyond those needed for LWR safety. Such requirements raise questions about meeting safety goals. |
| Comments | Self-explanatory |
| References/contacts | References or contacts for additional information |
| Update date/compiler | Last update of description/compiler of this information sheet |

[Cochrell 1988]. In the Union of Soviet Socialist Republics, bubbler-condenser towers have been reported to be installed to condense steam in PWR accidents [Bukrinksy, 1977]. These towers consist of several hundred small suppression pools. The mechanical designs of these two types of devices are very different, with different performance requirements; however, the underlying operating principles are identical.

Table 4 shows a partial list of SSCs that we have identified. Over 70 such SSCs were identified. These descriptions were organized by functional requirement.

## 4.2 Observations

### 4.2.1 Passive and Inherent SSCs

Multiple passive and inherent SSCs exist for six of the eight second-level functional requirements for nuclear reactor safety (Table 2). The number of options is very large. Many of these have been demonstrated either in other types of reactors or in other industries. Furthermore, many of the SSCs are relatively new ideals that suggest the potential for future discoveries and advances.

### 4.2.2 Limits to Passive and Inherent Safety

The functional requirements to maintain core integrity can be accomplished by a variety of passive and inherent SSCs. In contrast, there were no literature citations of passive or inherent safety SSCs for the following two second-level safety functional requirements:

- *Isolate Primary Circuit from Balance of Process (Function 2.3)*
- *Effect Containment Isolation (Function 3.1.1)*

Table 4. Partial list of structures, systems, and components

| Description title | Function |
|---|---|
| **1.1 Reactivity Control** | |
| Process Inherent Ultimate Safety (PIUS) Reactor Technology | 1.1, 1.2 |
| Passively Safe Pressurized-Water Reactor with Gas Bubble | 1.1, 1.2 |
| Passive Control of Power Levels by Variable Boron Concentration in Pressurized-Water Reactor (GEYSER) | 1.1, 1.2 |
| Use of Hydrides Which Reversibly Adsorb and Desorb Hydrogen to Control Reactivity in LWRs | 1.1 |
| Hydraulic Control Rod System with Passive Shutdown Mechanisms for Whole Core Disturbances (Low Water Level, Rapid Pressure Changes) in a Boiling-Water Reactor | 1.1.1 |
| Slow-Withdrawal Control Rods | 1.1.1 |
| Fluidized-Bed Control Rods | 1.1.1 |
| Self-Actuating and Locking Shutoff Valve for Hydraulic/Fluidic Control Systems Initiated on Low Water Flow or High Temperature | 1.1.1 |
| Neutron/Gamma/Coolant Thermal Fuse Control Rods/Devices | 1.1.2 |
| In-Reactor Core Power Fuses | 1.1.2, 1.2.2, 2.2 |
| Use of Select Material Phase Changes to Enhance Doppler Reactivity Feedback in LWRs | 1.1.3 |
| Large-Prompt Negative-Moderator Coefficient of Reactivity from Hydrogen Bond Structure | 1.1.3 |
| High-Temperature Ceramics, Clads, and Fuels for LWRs | 1.1.3, 1.2 |
| Graphite-Disk $UO_2$ Fuel Elements for Enhanced Thermal Margin and Reduced Excess Reactivity | 1.1.2 |

Table 4. (continued)

| Description title | Function |
|---|---|
| Graphite-Disk $UO_2$ Fuel Elements for Enhanced Thermal Margin and Reduced Excess Reactivity | 1.1.3, 1.2 |
| Adsorber (Poison) Pill for Reactor Shutdown on High Temperatures | 1.1.3 |
| 1.2 Remove Core Heat | |
| Fluidic In-Vessel Emergency Core-Cooling System | 1.2, 1.1.1 |
| Low-Water-Level-Initiated, Hydraulic-Valve Core-Operated, Emergency Core Cooling, and Shutdown Systems | 1.2, 1.1.1 |
| Main-Recirculation-Pump-Failure-Initiated Cooling and Shutdown System | 1.2, 1.1.1 |
| Passive Safety and Shutdown System | 1.2, 1.1.1 |
| Fluidized-Bed Pressurized-Water Reactor | 1.2, 1.1 |
| Prestressed Concrete Reactor Vessel (PCRV) for Light-Water Reactors | 1.2.1 |
| Metal Pressure Vessel in Pool | 1.2, 1.2.1, 3.2 |
| Passive Reactor Core-Cooling System Which Can Operate with Primary Pressure Vessel Leak | 1.2.1 |
| Reactor Pressure Vessel with no Bottom Penetrations | 1.2.1 |
| Integral PWR with no Primary System Piping | 1.2.1 |
| Fluidic Diodes | 1.2.1, 1.2.3 |
| High Water Inventory | 1.2.2 |
| Integral Safety Injection System | 1.2.2, 1.1.1 |
| Jet Injector Decay-Heat Core Cooling System | 1.2.2 |
| Natural Circulation of Water | 1.2.3 |
| Decay Heat Removal by Natural-Air-Circulation Steam Condensers | 1.2.3 |
| Cooling Ponds | 1.2.3, 3.2 |
| Water Quench Pool with Air Cooler for Reactor Accident and Decay Heat Sink | 1.2.3, 3.1.2 |

Table 4. (continued)

| Description title | Function |
|---|---|
| Decay Heat Removal with Seawater | 1.2.3 |
| Condensation of Pressurized Steam or Cooling Hot Water with Boiling Water Bath | 1.2.3 |
| **1.3 Chemical Attack of Clad** | |
| Reduction of Coolant/Clad Chemical Reactions Under Severe Accident Conditions | 1.3, 1.2.3, 3.1.2 |
| Chemical Getters to Protect Interior of Fuel Pins | 1.3 |
| **2.1 Maintain Coolant Boundary Integrity** | |
| Double Pressure Vessel | 2.1, 1.2.1 |
| **2.2 Control Primary Circuit Pressure** | |
| Safety and Relief Valves | 2.2 |
| Continuous/Simultaneous Pump Pressurizer | 2.2, 1.2.2 |
| Passive Pressurizer Spray | 2.2 |
| **2.3 Isolate Primary Circuit from Balance of Plant** | |
| None Found | |
| **3.1 Maintain Containment Boundary Integrity** | |
| Reactor Containment Buildings | 3.1.1 |
| Self-Sealing Ice Containment Structure | 3.1.1 |
| Filtered, Vented Containment | 3.1 |
| Vacuum Containment | 3.1.2, 3.2 |
| Passive Hard Vacuum Containment System | 3.1.2, 3.2 |

Table 4. (continued)

| Description title | Function |
|---|---|
| Containment-Pressure-Control Suppression Pool and Bubbler Condensers | 3.1.2, 3.2 |
| Steam Condensation and Vacuum Creation by Suppression Pools | 3.1.2 |
| Water-Coolable Core Catcher | 3.1.2, 3.2 |
| Containment Pressure Control by Pressure-Activated Water Sprinklers | 3.1.2, 3.2 |
| Temperature-Activated Water Sprinklers for Steam Condensation and Accident Mitigation | 3.1.2, 3.2 |
| Containment Pressure Control: Ice Condensers | 3.1.2 |
| Containment Cooling with Existing and Augmented Heat Capacity of Containment | 3.1.2 |
| Heat Pipes for Reactor Containment Cooling or Reactor Core Decay Heat Removal | 3.1.2, 1.2.3 |
| High Heat Conductivity Reactor Containment for Containment Cooling | 3.1.2 |
| 3.2 Trap Radionuclides | |
| Water/Steam in Post-Accident Environment | 3.2 |
| In-Containment Post-Accident Water Collection | 3.2, 1.2.2, 3.1.2 |
| Core Melt Source Term Reduction System (COMSORS) | 3.2 |

These functional requirements – closing of the reactor steam isolation valves – are only called upon if the reactor core is at risk or is damaged and releasing radioactive materials. In theory, a passive device could be used to isolate the reactor or containment building if it could be initiated (triggered) by the radioactivity in the fluid stream. In practice, current isolation devices (large mechanical valves) need external signals to determine whether they should be activated.

### 4.2.3 Sources of Technology

The analysis provides a basis to determine where work on advanced LWRs is being done, as shown in Table 5. The dominance of the United States in Table 5 reflects early work on nuclear reactors. In recent years, the dominant source of new technology has been Sweden. This reflects the presence of the research laboratories of Asea Brown Boveri (ABB) in Västerås, Sweden, which is the largest electromechanical company in the world. In the United States, new concepts have primarily originated from ORNL and ABB Combustion – the U.S. subsidiary of ABB. Japan is becoming an important source of advanced technology in this field because of the very rapid growth of nuclear research in that country during the past 5 years.

## 4.3 Chemical Technology

The emphasis on passive and inherent safety has resulted in examination of chemical systems for nuclear safety. A nonproprietary example is the proposed COre Melt SOurce Reduction System (COMSORS). COMSORS is an inherent safety system to limit the releases of radionuclides after a reactor core melt accident such as occurred at TMI. The safety problem in a reactor core melt accident is the generation of small radioactive particles

Table 5. Sources of passive/inherent technology by country and organizational type

| Type of Organization | Percent of worldwide total | | | | |
|---|---|---|---|---|---|
| | United States | Sweden | Japan | Other | Total |
| Vendor | 17 | 17 | 4.3 | 15 | 53.3 |
| Industrial firm | 9.8 | 0 | 0 | 0 | 9.8 |
| National laboratory | 6.5 | 0 | 4.3 | 9.8 | 20.6 |
| University | 8.7 | 0 | 1.1 | 6.5 | 16.3 |
| Total | 42 | 17 | 9.7 | 31.3 | 100 |

and vaporization of radioactive materials which can disperse to the atmosphere. In such an accident, chemical and thermal reactions of some reactor core materials with some aggregates – such as limestone – in concrete reactor structures generate large gas volumes which bubble out molten core materials creating radioactive aerosols.

Physically, COMSORS involves adding special aggregates – such as borosilicate glass – to the concrete under the reactor core. If there is an accident, the molten reactor core reacts with the concrete. The chemical composition of the aggregate is chosen to: (1) lower the melting point of the concrete-reactor core mixture, thus reduce vaporization of radioactive materials, (2) generate no gaseous chemical reaction products which would create aerosols, (3) depress vaporization of key radioactive elements by formation of nonvolatile high-temperature compounds, (4) adsorb radioactive decay heat in the melting process, and (5) create an insoluble waste matrix when solidified. COMSORS is an insitu high-level waste solidification system to prevent large radionuclide releases in an accident.

## 4.4 Technologies from Other Fields

Many possible SSCs, which are commercial technologies in other fields, have been identified but have not been applied to LWR safety. In most cases, these are relatively new technologies. An example is the heat pipe, which is a sealed tube containing a liquid that boils and transfers heat by a vaporization-condensation cycle from the hottest to the coldest parts of the sealed pipe. Heat pipes could be used for emergency core cooling, containment cooling, or other purposes. They have been considered for, and used for, special reactors (e.g., space reactors) but only one early study was identified in which heat pipes were considered for LWRs [Ahmad, 1983]. The particular application was passive containment building cooling.

Heat pipes are a good example of changing conditions that could make a previously bypassed passive technology potentially attractive. The following remarks should be helpful in such considerations:

1. Recent probabilistic accident risk assessments show that loss of partial containment building cooling may be a significant accident initiator [Garrick, 1989] in newer plants. In newer plants, the containment building is divided into more zones to isolate equipment against fire and other risks; this makes equipment cooling more difficult and increases the probability of local ventilation system failures. With ventilation failures, higher equipment-room temperatures increase equipment failure rates for both normal operating and emergency equipment. Heat pipes provide one high-reliability technical solution for cooling isolated zones in containment buildings.

2. In the past 15 years, the experience base for the technology has increased because heat pipes are being used for control of building temperature, solar heating, and temperature control of foundation structures for the Alaska Pipeline. Over 100,000 heat pipes are in use in this last application with a very high record of reliability.

3. The technology has advanced. For example, heat pipes can be designed to begin operation at a preset temperature. This type of feature would prevent freezing temperatures in containment for certain reactor applications such as containment cooling in cold climates.

Whether these changes are sufficient to make the technology the preferred choice for future LWRs is not known. However, it is clear that changing technologies and changing requirements can make the reevaluation of previously unused technologies worthwhile.

### 4.5 Applicability of Building Blocks to the Chemical Industry

The fundamental safety and environmental goals for nuclear and chemical plants have a common objective – to avoid the release of highly toxic materials to the environment. As such, much of the technology for passive and inherent safety in LWRs is applicable to chemical plants. In particular:

- Technologies for passively cooling nuclear power reactors are applicable to cooling chemical reactors where exothermic chemical reactions raise concerns about runaway chemical reactions.

- Technologies to contain or confine toxic materials after an accident are essentially identical for chemical or nuclear plants.

## 5. Advanced Reactors

Experience and new technology are resulting in new power plant designs. A brief summary of worldwide activity in this area shows trends toward passive and inherent safety in reactor design. The new generation of LWRs may be divided into three classes when defined by the goals of the designers: *evolutionary* plant LWRs, *evolutionary* design LWRs, and PRIME reactors. Both classes of evolutionary reactors have the requirement that the first reactor be sold as a commercial unit with no prototype or pilot plant being required. This limits adoption of new technology. All three classes have similar steam cycles, reactor core designs, and thermal efficiencies. Table 6 summarizes some of the differences in these reactor designs.

## 5.1 Evolutionary Plant LWRs

The *evolutionary* plant LWR designs exemplified by the seven reactors listed in Table 7 are similar to the overall plant design of existing LWRs, except that certain refinements and modernized features have been included. Their safety, like that of their predecessors, depends on a variety of active safety systems with power supplied by diesel generators or equivalent power sources. In the event of an accident, the safety systems must start up and continue to operate to prevent reactor core damage. LWRs have been built with various types of pumps, valves, motors, control-rod drives, containments, and other components/systems. There is now sufficient operating experience to use as a basis for judging which variations in design work the best. These designs reflect the rapidly increasing experience base and are the nuclear plant equivalents to evolutionary designs in cars and aircraft.

## 5.2 Evolutionary Technology LWRs

Evolutionary technology LWRs (Table 7) are proposed advanced LWRs that use the technology of current LWRs (components and systems) but have significant changes in plant design, particularly the safety systems. The major, government-supported LWR development programs in the United States today are those at General Electric and Westinghouse. Most of the proposed safety systems for these reactors require power to initiate operation of safety systems (such as to open a valve) but do not require power for continued function. Thus, operation of safety systems after initiation is passive. This is

Table 6. Summary of passive/inherent safety characteristics and limitations for various Light-Water Reactors

| LWR generation | Emergency power supply | Need pumps?[a] | Need valves?[a] | Need operator action? | Resists errors/ malice? |
|---|---|---|---|---|---|
| Present | Reliable AC | Yes | Yes | Yes (but not immediately) | No |
| Evolutionary Plant | Reliable AC | Yes | Yes | Yes (but not immediately) | No |
| Evolutionary Design | Batteries only | No | Few, small active valves | No (Automated control system) | No |
| PRIME[b] | NONE | No | NO | No[c] | YES |

[a]For safety functions.
[b]There are other reactor types which may meet PRIME safety goals. The Modular High Temperature Gas-Cooled Reactor and the advanced CANDU Project Reactor may be able to meet these goals.
[c]No control system used for safety, falls to safe state. See Appendix A.

Table 7. Advanced light-water reactors

| Name | Type | Size/(MWe) | Countries | Lead organizations | Status |
|---|---|---|---|---|---|
| Evolutionary Plant Designs | | | | | |
| N4 | PWR | 1400 | France | Framatome | Construction |
| Sizewell | PWR | 1250 | Great Britian | Central Electric Generating Board | Construction |
| Advanced Boiling-Water Reactor | BWR | 1356 | Japan/U.S. | Hitachi/Toshiba/General Electric | Design/construction |
| Advanced Pressurized-Water Reactor | PWR | 1350 | Japan/U.S. | Mitsubishi/Westinghouse | Design |
| Advanced BWR 90 | BWR | 1050 | Sweden/Switzerland | ABB | Design |
| Combustion 80Plus | PWR | 1280 | United States | ABB-Combustion | Design |
| VVER 88/92 | PWR | ~1000 | U.S.S.R. | | Design |
| Evolutionary Technology Designs | | | | | |
| Simplified Boiling-Water Reactor | BWR | 600 | United States | General Electric[a] | Development |
| Safe Integral Reactor | PWR | 300 | U.S./Great Britian | Combustion/Rolls Royce | Development |
| Advanced Passive-600 | PWR | 600 | United States | Westinghouse | Development |
| Hitachi Simplified Boiling-Water Reactor | BWR | 600 | Japan | Hitachi[a] | Development |
| Toshiba 900 | BWR | 310 | Japan | Toshiba[a] | Development |
| Simplified Pressurized Water-Reactor | PWR | 350 | Japan | JAERI[b] | Development |
| PRIME Reactors[c] | | | | | |
| PIUS (Secure P®) | PWR | 640 | Sweden/Italy, Korea/U.S. | ABB | Development |
| ISER | PWR | 210 | Japan | University of Tokyo | Research |
| PIUS/BWR | BWR | 750 | United States | Oak Ridge National Lab. | Research |

[a]There are long standing technical agreements between these three organizations with much commonality of design.
[b]JAERI = Japan Atomic Energy Research Institute
[c]There are other reactor types which may meet PRIME safety goals. The Modular High-Temperature Gas-Cooled Reactor (MHTGR) and the advanced CANDU Project Reactor may be able to meet these goals.

the key distinction between these designs and the evolutionary *plant* designs and is a significant advance in safety technology. Such changes in design reflect two experiences. First, all of these designs were initiated after the TMI accident and take into account the technical lessons learned. Second, they reflect the operating experiences of current plants that have shown which features tend to make a plant difficult to operate or expensive. These proposed reactors have the following common features:

- All water required for heat removal in the primary system drains by gravity to the reactor core, which is located at the lowest elevation in the plant. In the TMI accident, the plant layout did not permit the water in the steam generators (heat sink) to flow by gravity to the overheated reactor core. Such water flow would have cooled the reactor core by boiloff and prevented damage to the core.

- Large ac power sources (diesel generators) to run emergency equipment have been eliminated. Experience shows that emergency equipment that consumes large amounts of electric power and associated power supplies are expensive to build, maintain, and operate. Furthermore, the complexity of the equipment increases the probability of operator error in an emergency. The elimination of emergency diesel generators has necessitated major changes in those emergency systems that consumed electric power – the emergency core cooling systems and the containment cooling systems. The evolutionary technology LWRs do require battery power in an emergency to operate valves and instruments.

- Existing and proposed evolutionary plant nuclear power plants are designed to pump water into the reactor core for emergency cooling in the event of a major pipe break or other accident . This function requires large pumps and, hence, diesel generators to provide power. The proposed evolutionary technology

LWRs use a different approach. In all these designs, large volumes of water are stored above the reactor core. In an accident, first the reactor is depressurized by opening valves, and second, water flows by gravity from overhead tanks into the reactor vessel. Typically, there is sufficient water to flood the reactor containment and reactor system above the level of any pipe failure in the primary system. This technology was first developed by ABB and later adopted by U.S. vendors [Hannerz, 1990].

- Lastly, a major effort has been made to simplify the design. The complexity of existing plants implies high cost and the possibility of operator/maintenance error. Plant simplification is possible because the designs are new and not simply modifications of existing plant designs.

5.3 PRIME Reactors

In PRIME reactors, the third class of reactors under development, the goals of the designers are radical improvements in safety and public acceptance with the potential for major improvements in economics. Key characteristics include high resistance to assault, sabotage, and misoperation. This, in part, follows from the TMI and Chernobyl experiences where automated safety systems were shut down for what were thought to be good reasons at the time. PRIME implies no off switches for safety systems. Because the goals are aggressive, new technologies are required for these reactor designs. Appendix A provides a technical description of these concepts [Hannerz, 1983; Forsberg, 1986].

The largest effort to develop a PRIME light-water reactor is that by ABB. Their particular concept [Hannerz, 1983] is called the Process Inherent Ultimate Safety Reactor (PIUS). Key new technologies for PIUS have been demonstrated in the laboratory. Because of the early start of this program and ABB's excellent historical track record of building LWRs in Europe and in the United States through their subsidiary – ABB Combustion – it is considered the leading program worldwide. The major uncertainties, as with any innovative technology, are costs and plant reliability. A demonstration plant will be required to show economics and reliability.

## 6. Conclusions

Fifteen years ago, the concept of a nuclear power reactor with only passive and inherent safety systems or a power reactor with PRIME safety was thought to be technically impossible. As a result of a series of technical developments and inventions, the technical community now agrees that such reactors can be built. The remaining questions are cost and reliability. The analysis to date have been highly favorable but much work remains to be done.

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# APPENDIX A: TECHNICAL DESCRIPTION OF PIUS LWRs

## A.1 INTRODUCTION

This appendix provides brief descriptions of the two PIUS concepts with PRIME safety goals that have proposed: the PIUS-PWR and the PIUS-BWR. The PIUS-PWR was invented earlier and thus, is further along in development. Each reactor has particular advantages and disadvantages. In each case, various derivative concepts have been developed.

The two concepts have many features in common but differ in their basic safety mechanisms. The description of the PIUS-PWR is more complete and reflects its advanced development. The common features of both reactors are included in the description of the PIUS-PWR.

## A.2 PIUS-PWR

The PIUS-PWR, which was invented by K. Hannerz of ABB, is also referred to as PIUS and Secure P® in the literature.

The PIUS-PWR is a modified "swimming pool" PWR; the pool is at full reactor pressure and contains high concentrations of cool, borated water. The reactor normally operates in a second volume of hot, low-boron reactor water within the pool. In the event of an accident, the cool, borated (neutron poisoned) water enters the reactor core. The boron in the water shuts down the reactor. The reactor core is cooled by boiloff of the borated water. The period during which this Emergency Core Cooling System (ECCS) works in a passive mode depends on the volume of borated water available to be boiled off. Current proposed designs provide 1 week of passive heat removal.

This reactor has two unique features: (1) a very large pressure vessel that includes the reactor core and all key safety systems, and (2) the safety system that puts cool, borated water in direct contact with the hot, low-boron reactor coolant water. The cool, borated water does not enter the reactor core during normal operations because of a hydraulic balance maintained by the main recirculation pumps.

The pressure vessel is a prestressed-concrete reactor vessel (PCRV). Key characteristics include the following:

1. The PCRV contains sufficient borated water to cool the reactor core for 1 week after reactor shutdown. To accomplish this goal, the internal vessel diameter is 12 to 13m.

2. The PCRV is large enough to allow spent fuel storage in the vessel for the reactor lifetime.

3. The PCRV provides very high levels of protection against external threats. The wall thickness is 12 to 13m.

The PCRV has several unique design features:

1. It contains both steel reinforcing bars and prestressed steel tendons. The redundant design allows for failure of either reinforcing bars or tendons without catastrophic vessel failure.

2. It contains a double internal steel liner to prevent leakage of water. From the inside to the outside, the vessel includes an inner liner, 1-m-thick concrete, a secondary liner, and the main PCRV.

The second unique feature of the PIUS PWR is the hydraulic emergency core cooling system. The operating principles of this system are shown in Fig. A.1.

Figure A.1(a) shows a natural-circulation PWR reactor core (C) inside a very large pressure vessel (A). The reactor core is in a zone of low-boron water (D) at the bottom of the riser. The riser incorporates a pressurizer (I) to maintain reactor vessel pressure at desired levels. The pressure vessel is primarily filled with cool, borated water (B). The low boron concentration of the water allows the reactor to be critical and produce heat. In this configuration, the reactor would be shut down quickly by the natural circulation of borated water into the core from below (J) and out through the top of the riser (K).

In Fig. A.1(b), the hot reactor water is returned from point M near the top of the riser to point N below the core by the addition of a recirculation pump (E).

In Fig. A.1(c), a steam generator (F) has been added to the circulating water flow to keep the temperature constant. The steam generator and pump can be located either inside or outside the pressure vessel. The reactor is a natural-circulation reactor dependent on

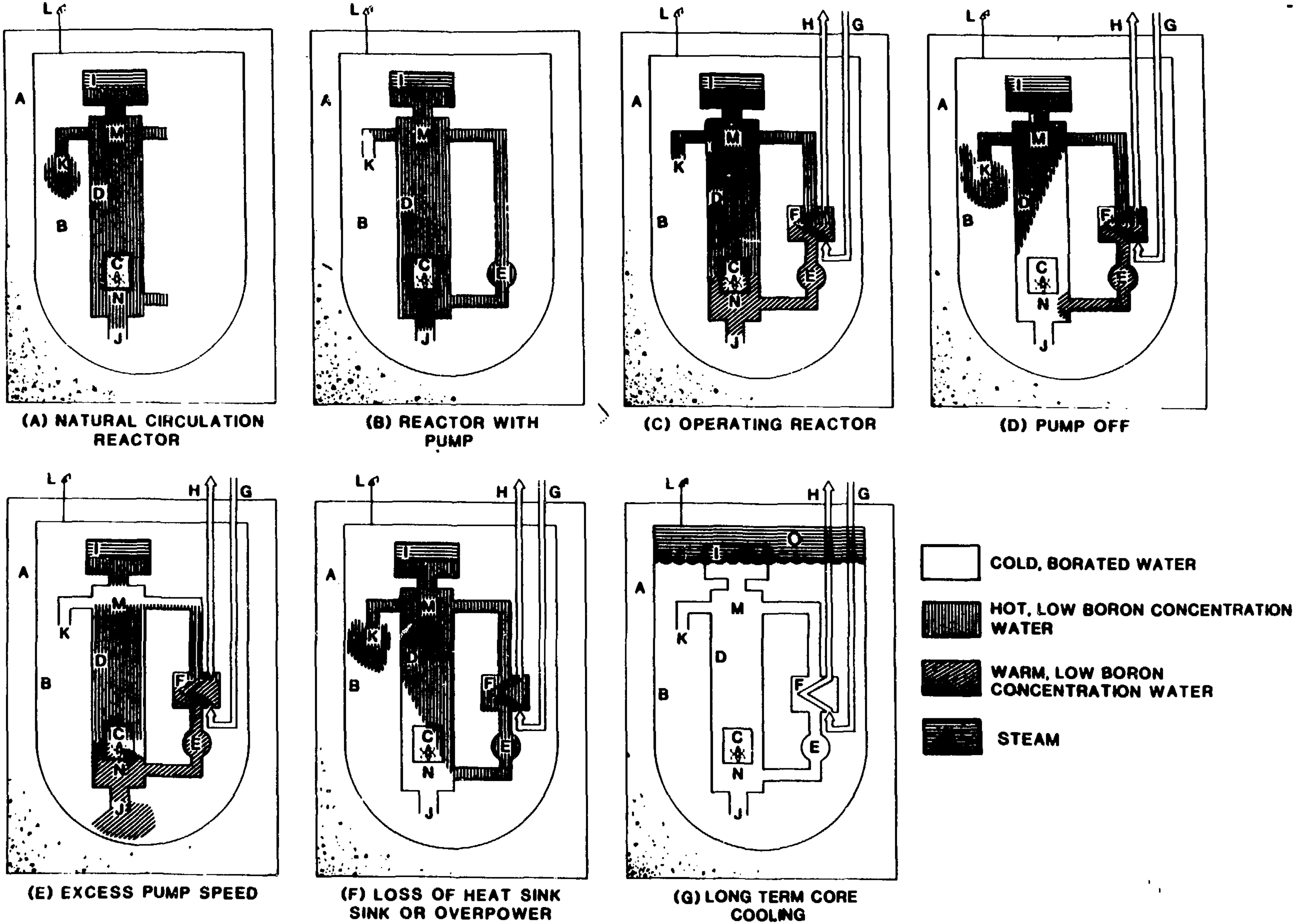


Fig. A.1. Operating principles of PIUS.

differences in water densities of the high-temperature, low-boron-concentration water in the riser and the low-temperature, high-boron-concentration water in the pool. The pump simply overcomes pressure drops in the steam generator and associated piping between points M and N. It pulls the full flow of hot water from the reactor at point M and delivers it to point N.

There are two flow paths for the water from above the reactor core (point M) to back below the reactor core (point N). The first is through the steam generator and pump (M, F, E, N). The second is through the cold, borated water zone (M, K, B, J, N). If the cool, highly borated water flows into the core, the reactor will be shut down. This does not happen in operation because of a careful hydraulic balance generated by the pump.

If the rate of the recirculation pump slows to less than that of the natural water circulation [Fig. A.1(d)] through the core, then cold, borated water will enter the core from point J and shut the reactor down. If the pump operates too rapidly, pump suction will draw cold, borated water into the system near point M and through the steam generator and pump [Fig. A.1(e)]. The pump discharge will push some highly borated water into the core near point N and the remaining water into the cold, borated water zone below point N. In effect, the hot, low-borated water zone that allows the reactor to produce power is stable against the ingress of cold, borated water at only one pump speed for each set of operating conditions.

The hot reactor water is separated from the cold, borated water by interface zones (J, K). The large density differences between the two waste zones make the interface very stable. Instruments sense whether the hot/cold interface zone is moving up or down and will adjust the pump speed accordingly.

Power levels in the core are controlled by varying the boron concentrations in the hot reactor water. The hydraulic balancing also protects against reactor overpower conditions or loss of feedwater to the steam generators. In either case, boiling will eventually occur in the reactor core [Fig. A.1(f)]. Boiling causes major increases in natural circulation flows through the core. The recirculation pump is sized so that it physically cannot handle the water flow through the reactor core under these circumstances. Thus, the hydraulic balance breaks down, and cold, borated water enters the core from the bottom.

After the reactor shutdown, the cool, borated water heats up, absorbing radioactive decay heat. Eventually the borated water boils, and steam is released through pressure relief valves. The reactor will be cooled as long as water remains in the pressure vessel.

A recent design of the PIUS-PWR by ABB is shown in Fig. A.2; some design parameters are given in Table A.1. PIUS-PWR design options include steam generators on either the inside or the outside of the PCRV. Siphon breakers prevent siphoning of water from the PCRV if there is a pipe break. This design is for a 640-MW(e), 2000-MW(t) power reactor. The current design also includes four independent natural circulation cooling systems that transfer heat from the cool, borated water to the air during normal and emergency operations. During normal operations, heat leaks from hot water through the walls to the cold, borated water zone. During emergency operations, these cooling systems will remove all core decay heat from the high-boron-concentration water zone as water circulates between the two zones. The reactor core is protected essentially forever if the natural-circulation air coolers are operating or for at least 1 week in the event of air cooler failure. The air coolers can withstand normal expected events (storms, earthquakes, etc.) but, because they require good access to air, cannot be protected against some types of sabotage or external military assault.

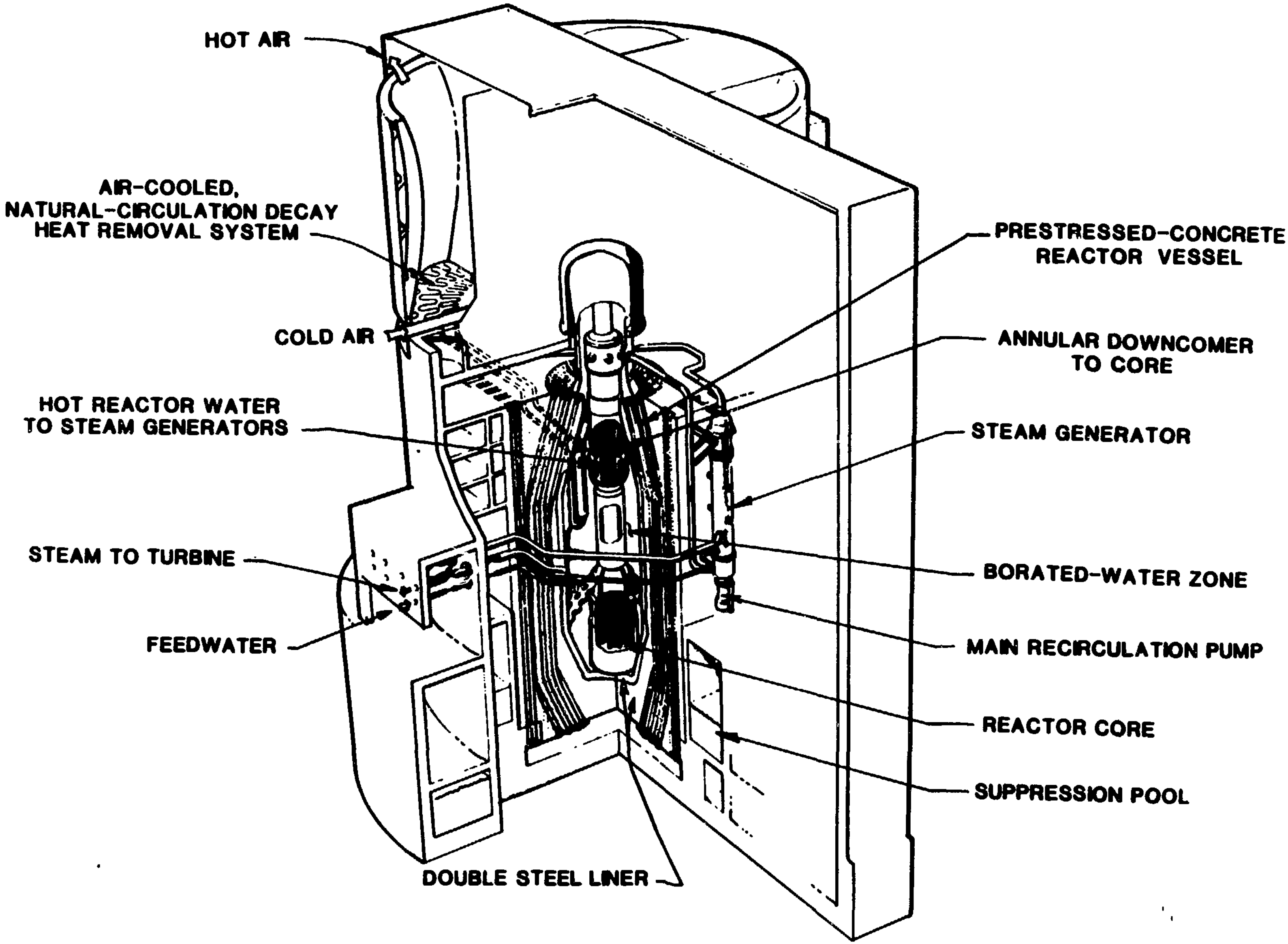


Fig. A.2. Proposed PIUS design by ABB Atom.

Table A.1. Some key design data for the Secure-P Reactor

| | | |
|---|---|---|
| Thermal power | MW | 2000 |
| Electric power (net) | MW | 640 |
| Core exit temperature (full power) | °C | 290 |
| Core inlet temperature (full power) | °C | 260 |
| Core coolant flow | kg/s | 13000 |
| Primary system pressure (pressurizer) | MPa | 9.0 |
| Number of fuel assemblies | | 213 |
| Number of fuel rod/assembly | | 316[a] |
| Fuel enrichment, reload fuel | % | 3.5 |
| Average burnup | mwd/t | 45,500 |
| Core height (active) | m | 2.50 |
| Core diameter (equivalent) | m | 3.76 |
| Core pressure drop (dynamic) | MPa | 0.039 |
| Number of steam generators | | 4 |
| Steam pressure (steam generator exit) | MPa | 4.0 |
| Steam temperature | °C | 270 |
| Number of reactor coolant pumps | | 4 |
| Pool temperature (normal operation) | °C | 50 |
| Concrete vessel cavity diameter | m | 12.2 |
| Concrete vessel cavity total height | m | 44 |
| Concrete vessel cavity volume | $m^3$ | 3300 |
| Concrete vessel thickness (minimum) | m | 7 |

[a]Up to 32 fuel rods containing burnable absorber ($Gd_2O_3$).

## A.3. PIUS-BWR

The PIUS-BWR was invented by C. Forsberg of ORNL. The basic rationale for a BWR type rather than a PWR type is that a BWR has lower operating pressures. Because PIUS reactors use very large pressure vessels, there is a strong technical and economic incentive for lowering reactor operating pressures. For technical reasons, the safety systems of the PIUS-PWR will not work for a PIUS-BWR. The invention of the PIUS-BWR followed the PIUS-PWR by several years; hence, technical uncertainties with the PIUS-BWR are larger.

In the PIUS-BWR, the conventional BWR system is placed within a very large PCRV along with a 1-week supply of borated, emergency core cooling water (Fig. A.3). A Fluidic In-Vessel Emergency Core Cooling System (FIVES) protects the reactor core against accidents. The FIVES has three major components: (1) a large volume of cool, borated water at reactor pressure; (2) a fluidic valve assembly that separates cool, borated, emergency water supplies from hot reactor water; and (3) the FIVES water pump system that provides power to the fluidic valve and detects water shortages in the reactor core.

The pressure vessel is divided into two water zones: (1) a reactor coolant zone with core, riser, downcomer, and steam separators; and (2) a supply of cool, borated emergency core-cooling water. The two water zones are separated by an insulated wall that is not a pressure boundary. The zones are in direct contact with each other near the top of the pressure vessel through a hot/cold interface where hot, low-density, clean reactor water lies on top of cold, high-density, borated water.

Near the bottom of the pressure vessel, the two water zones are connected by a fluidic valve. The pressure of the cool, borated water is somewhat higher at this location than that

ORNL DWG 88A-640

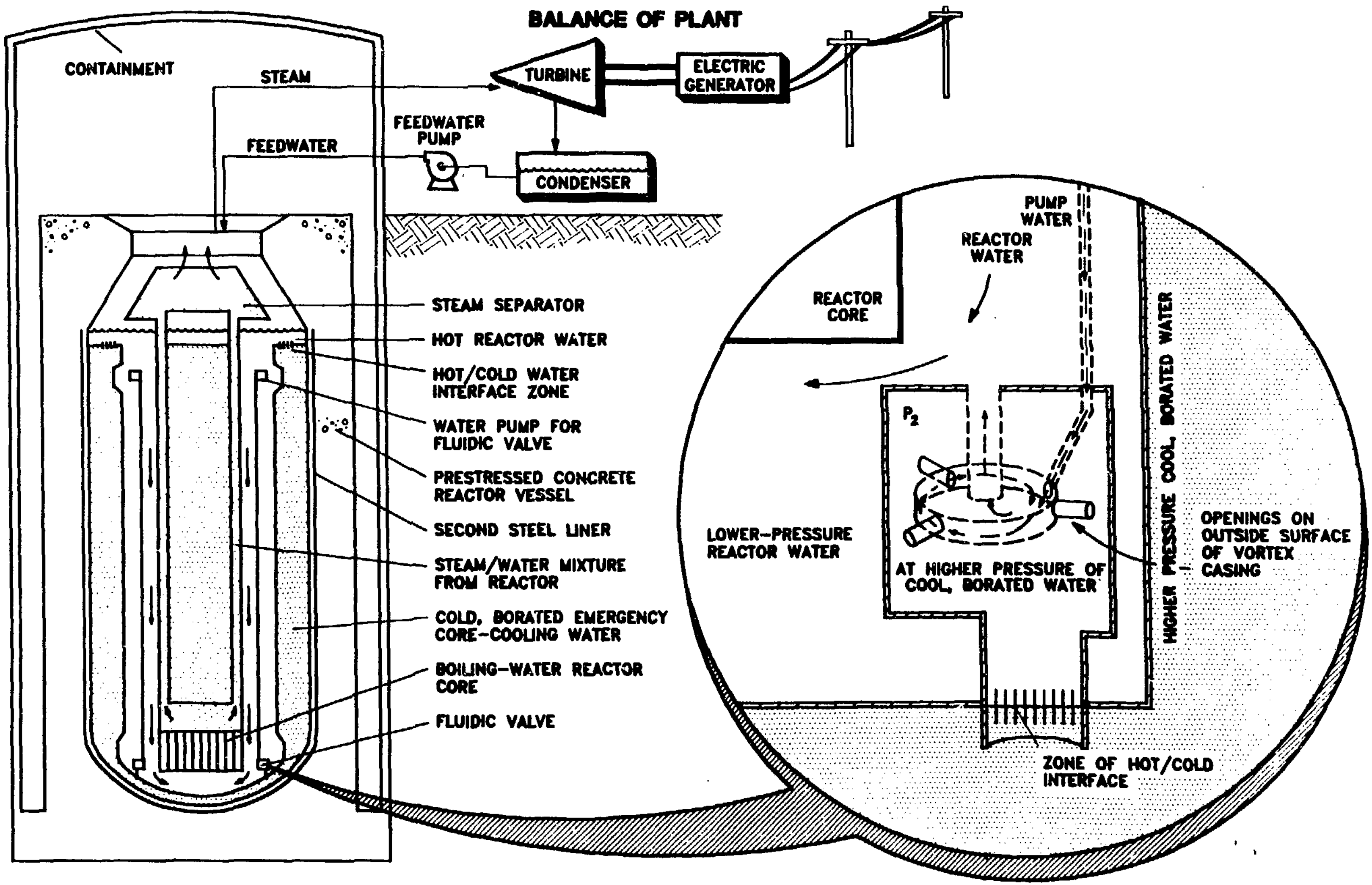


Fig. A.3. Process Inherent Ultimate Safety Boiling-Water Reactor.

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