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Developmental Light-Water Reactor Program:

Summary and Progress Report for FY 1989

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ABSTRACT

This report summarizes the progress of the Developmental Light-Water Reactor (DLWR) Program at Oak Ridge National Laboratory in FY 1989. It includes a brief description of the program, description of goals, achievements, and recommended future activities.

1. INTRODUCTION

This report summarizes the progress of the Developmental Light-Water Reactor (DLWR) Program at Oak Ridge National Laboratory in FY 1989. It also includes (1) a brief description of the program, (2) definition of goals, (3) earlier achievements, and (4) proposed future activities.

2. CHARACTERIZATION OF THE DEVELOPMENTAL LIGHT-WATER REACTOR PROGRAM

2.1 Program Objectives

The objectives of the DLWR program are to understand and develop the options for a DLWR with PRIME safety systems and significantly improved economics where:

- DLWRs are Light-Water Reactor (LWR) concepts with sufficiently innovative characteristics such that a prototype plant may be required.

- PRIME is the acronym for Passive, Resilient, Inherent, Malevolent Resistant, Extended Safety. Table 1 defines PRIME. (The safety goals are similar to a Modular High-Temperature Gas-Cooled Reactor [MHTGR], but the reactor is based on LWR technology – see Appendix A).

- Improved economics is a result of radical changes in safety system design and plant simplification (cost is driven by safety in current reactors).

Table 1. PRIME definition

Passive safety • Only passive safety systems; i.e., systems which do not need any external input to operate • No moving mechanical or electrical systems for plant safety (includes mechanical valves, pumps, etc.) **R**esilient safety • No operator incentive to bypass safety systems (Chernobyl and Bhopal problem) • Foolproof **I**nherent safety • Select materials and designs where possible to eliminate accidents (e.g., water inherently safe against fire) **M**alevolent resistance against: • Short-term terrorist plant takeover (encompasses all operator errors) • Off-the-shelf conventional military munitions **E**xtended safety after accident before core damage: • One week after major plant damage • Indefinitely after plant abandonment

Examples of DLWRs include but are not limited to Secure P® (PIUS) and PIUS/BWR. The program is defined by goals, not a specific reactor concept at this time.

2.2 Examples of DLWRs

There are many DLWR technical options including PIUS (Secure P®) being developed by Asea Brown Boveri (ABB) and PIUS/BWR, which was invented at Oak Ridge National Laboratory. Appendix C describes these concepts. DLWRs are relatively new concepts and other design options will likely be identified.

2.3 Relationship of DLWR Program to Other Reactor Programs in the United States

The DLWR program looks the furthest into the future of DOE's several Advanced Light-Water Reactor (ALWR) programs. A summary of some of these programs and their time scales follows. Table 2. provides a perspective on the differences between these programs.

Advanced Pressurized-Water Reactor (APWR)/Advanced Boiling-Water Reactor (ABWR)

The program objectives are licensable plants using the best of existing, proven technology and basic plant designs. The time frame is for Nuclear Regulatory Commission (NRC) Certification by 1991. The lead reactor vendors are General Electric and Combustion Engineering. The program includes heavy involvement by the Electric Power Research Institute (EPRI) and architect/engineers (A/Es).

Table 2. Relationship of DLWR to other LWR programs

	APWR/ABWR	SBWR/AP-600	DLWR
Goals	Refine/incorporate experience	Increased passivity of safety systems	Radical economic/ public acceptance/ safety improvements
Technical constraints	Proven technology	Near-term technology	None
First unit	Commercial	Commercial	Prototype may be required
Time frame	Today	1990s	Post 2000

Advanced Passive-600 (AP-600)/Simplified Boiling-Water Reactor (SBWR)

The program objectives are licensable plants with increased emphasis on the use of passive safety systems. Technologies used in these plants must be reasonably well developed and involve only small developmental programs before use. The time frame is for NRC certification by 1995. The lead reactor vendors are General Electric and Westinghouse, with major involvement by DOE and EPRI. There is limited national laboratory and university participation.

DLWR Program

The program objectives are to develop the technical options for licensable plants with radical improvements in safety and improved economics. There are no restrictions to the technologies which can be used, and new, innovative technologies are encouraged. The time frame for NRC licensing is beyond the year 2000. The technologies that are developed may (but not necessarily) require construction and operation of a demonstration reactor. The current program activities are primarily at national laboratories and universities.

In terms of safety systems, some perspective on the differences between various DOE programs is shown in Table 3. for various classes of reactors. The DLWR program can also be defined with respect to other reactor programs. Figure 1 shows the relationship of these programs.

Table 3. Summary of passive/inherent safety characteristics and limitations for various Light-Water Reactor (LWR) programs

LWR generation	Emergency power supply	Need pumps?[a]	Need valves?[a]	Need operator action?	Resists errors/ malice?	Certified when?
Present	Reliable AC	Yes	Yes	Yes (but not immediately)	No	Now
APWR/ABWR[b]	Reliable AC	Yes	Yes	Yes (but not immediately)	No	1991
AP-600/SBWR[c]	Batteries only	No	Few, small active valves	No	No	1995
DLWR[d]	NONE	No	NO	No	YES	Post 2000

[a]For safety functions.

[b]Advanced Pressurized-water reactor/advanced boiling-water reactor; several improvements made to enhance safety margins and reliability.

[c]Advanced passive-600/simplified boiling-water reactor.

[d]Developmental light-water reactor.

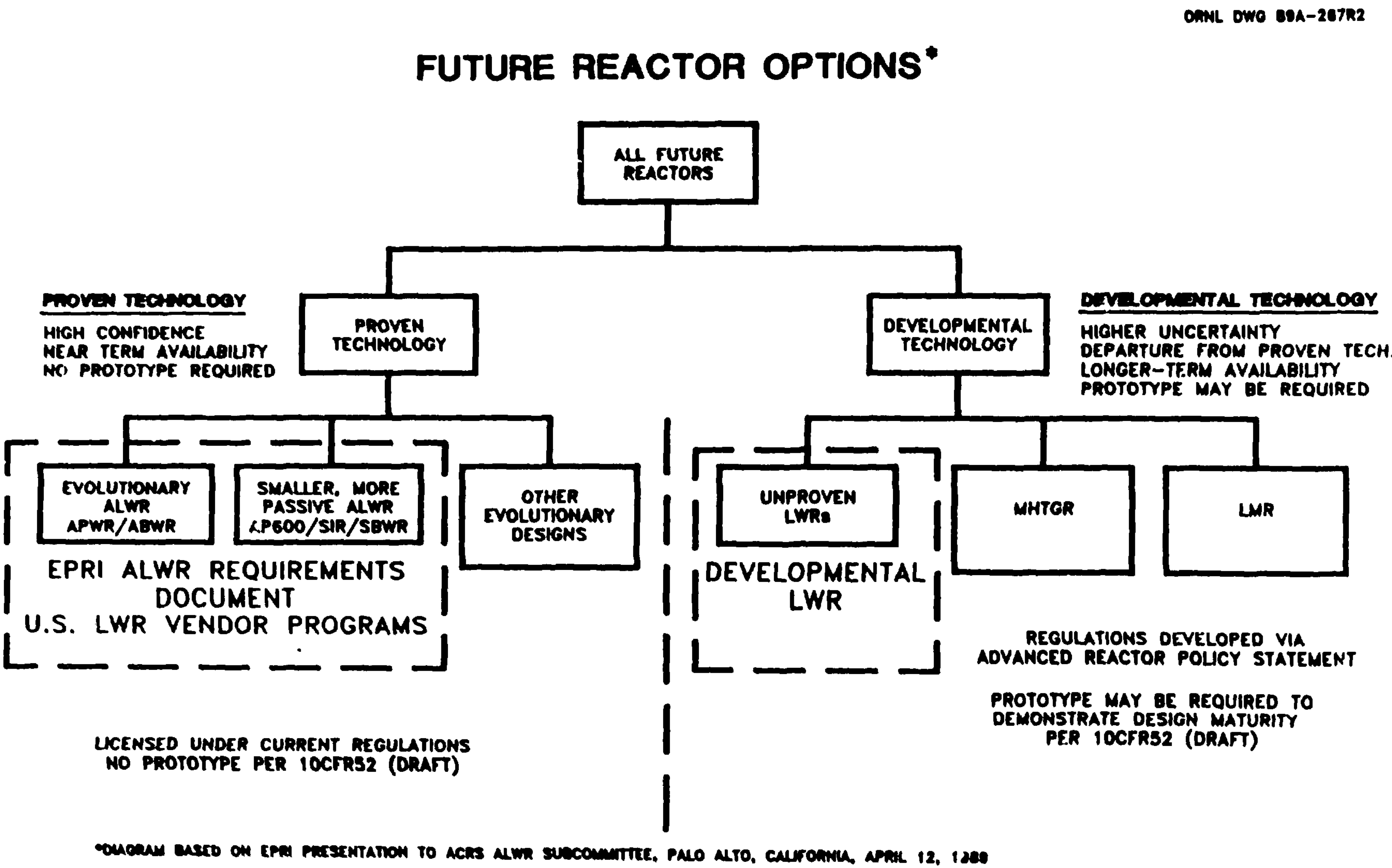

Fig. 1. Future reactor options

2.4 Basis for DLWR Program Objectives

The basis for a DLWR is a result of multiple near-term and long-term considerations. A brief description of these considerations is given.

Economic

The capital and operating costs of current designs of LWRs has increased rapidly in the last several years. Historically, capital costs of nuclear plants have exceeded those of coal fire plants but operating/fuel costs have been much lower. More recently, average operating/fuel costs of nuclear plants in the U.S. have exceeded equivalent costs at coal-fired power stations. This capital and operating cost growth, directly or indirectly, is a result of safety concerns and licensing. Since safety and licensing dominate costs, emphasis on major innovations in the approaches to safety is appropriate as a method to improve both safety and costs. Directions of research and development for safety and economics are the same.

Environment

Recent global environmental concerns, such as the carbon dioxide greenhouse effect, have created a renewed interest in use of nuclear power worldwide. In some parts of the world, political instabilities and/or low skill levels limit the use of nuclear energy. If reactors with PRIME characteristics can be developed, the option exists for wider use of nuclear power to assist in solving global environmental problems with lower accident risks than today.

Advanced Safety with Minimum Development Risk

Several advanced reactor concepts with potentially very high levels of safety have been proposed. An example is the Modular High-Temperature Gas-Cooled Reactor (MHTGR). The DLWR goals are to create the option of a LWR with these advanced safety characteristics but based on LWR technology and experience.

Using the existing LWR technology base can radically reduce the technical development risks and costs to create a new reactor option with such safety characteristics. The proposed DLWR's have several highly-innovative features which are responsible for their safety characteristics, but these are limited in number and well identified. It is the limited number of innovative features combined with an understood technical base which limits development risks.

International Competition in PRIME Reactors

Last, it is important to recognize that the United States is not alone in development of these advanced reactors. Many foreign countries have programs much larger than those in the United States. This includes the programs in Europe, Japan, and Canada. There is a general recognition overseas that if a DLWR with economics equal to or better than existing reactors can be developed, it will wipe out the competition. The resources being expended overseas by private and public sources also indicate the strong belief that an economic DLWR is possible. If the United States is to be competitive, it needs advanced reactor concepts which can compete in economics and safety. This program is the only effort in the United States in this area.

3. PREVIOUS ACCOMPLISHMENTS OF THE DLWR PROGRAM

3.1 Invention of the Process Inherent Ultimate Safety Boiling-Water Reactor (PIUS-BWR)

A major technical achievement for the program was the invention of the PIUS/BWR (see Appendix C). Evidence to date suggests that this reactor concept will meet PRIME safety goals. The PIUS/BWR is one of only two DLWR concepts currently in existence. The other DLWR option is the PIUS reactor – a modified pressurized-water reactor.

For the United States, the PIUS/BWR creates the option of development of a U.S. concept (patents owned by the U.S. government) or the option of a partnership with Asea Brown Boveri (a European company), the owner of the PIUS patents. Many of the new technologies being developed are applicable to both concepts. Perhaps of equal importance, the existence of two technical options suggests that additional unknown options may exist.

Comparisons of the PIUS and PIUS/BWR indicate advantages and disadvantages for each concept. The PIUS/BWR is a somewhat more thermally efficient machine and should have lower capital costs because its system operating pressures are lower. Both concepts use large prestressed concrete reactor vessels which are a significant factor in total plant capital costs. Lower system pressures and higher plant efficiencies of the PIUS/BWR reduce pressure vessel costs. Higher efficiency lowers fuel cycle costs. Because of the extensive development work, including large-scale hydraulic tests, the technical and cost uncertainties associated with PIUS are much less than those of PIUS/BWR.

3.2 Program Plan

A multi-year, multi-phase program plan has been prepared for the Developmental Light-Water Reactor program. Because of the long-term exploratory characteristics of some of the work, the program plan includes heavy involvement of universities, national laboratories and vendors.

3.3 International Atomic Energy Agency (IAEA) Activities

In support of the U.S.Department of Energy, the DLWR program has worked with the IAEA International Working Group on Advanced Technologies for Water-Cooled Reactors (IWGATWR). This included support of consultant meetings and preparation/

presentation of papers at the IAEA Technical Committee Meeting on "Definitions and Understanding of Engineered Safety, Passive Safety and Related Terms," Västerås, Sweden, May 30 - June 2, 1988. The goals of this IAEA meeting were to better understand the characteristics and benefits of passive and inherent safety applied to water-cooled reactors. This included developing a common understanding of safety terms.

4. FY 1989 ACCOMPLISHMENTS OF THE DLWR PROGRAM

4.1 Report: Proposed and Existing Passive and Inherent Safety-Related Structures, Systems and Components (Building Blocks) for Advanced Light-Water Reactors (ORNL-6554)

This report was the major program activity in FY1989. It is a compendium of safety-related passive and inherent structures, systems, and components (SSCs) for light-water reactors. Systems, structures, and components are the "building blocks" that are combined to create a complete reactor design. The concepts described in the report range from current commercial technologies to speculative ideas. Seventy major categories of technologies were identified.

The report provides a basis to define future directions of research. It also documents the large base of technology available to create DLWRs with PRIME safety characteristics.

4.2 Invented/Identified Possible Technologies to Eliminate Size Restrictions for all DLWRs

Some designs of DLWRs are limited in size to ~700 Mw(e) due to nuclear core power instabilities at larger core sizes. In current nuclear reactors, such instabilities are not a concern due to different core designs and because design criteria allow the use of control rods to limit such instabilities. Because of the PRIME design criteria, control rods in DLWRs cannot be used for safety although they are allowed for operations. Control rods as active components have potential failure modes and create possibilities for misoperation or sabotage. For this reason, it is desirable not to include them as a safety system in very advanced reactors.

As a part of the DLWR program, patent reports of possible subject invention have been filed on passive devices to limit nuclear reactor core instability problems and maximum power levels. If these devices are practical, there are no theoretical reactor core size limits for DLWRs. This potentially may have a major impact on economics of DLWRs and creates the option of building such reactors in any desired size to match utility interests. The devices may also have applications to other reactor types.

4.3 Invented/Identified Passive Decay Heat Cooling System Applicable to Current and Future LWRs

Patent reports of possible subject invention have been filed on a passive decay heat removal system applicable to all water-cooled reactors. This passive safety system would prevent core damage after a loss-of-heat-sink accident or after most small- and medium-size loss-of-coolant accidents. Such a system would prevent core damage in an accident such as occured at Three Mile Island. The system appears applicable (based on limited work) to both near-term and more advanced reactor designs. The system can be applied to pressurized-water reactors and boiling-water reactors.

4.4 International Atomic Energy Agency (IAEA) Activities

In support of the U.S. Department of Energy, the DLWR program has worked with the IAEA International Working Group on Advanced Technologies for Water-Cooled

Reactors (IWGATWR). This included in FY1989 preparation and presentation of papers at the IAEA Technical Committee Meeting on Passive Safety Features in Current and Future Water-Cooled Reactors, Moscow, USSR, March 21-24, 1989. This meeting also provided a useful mechanism to obtain information for the compendium report on safety building blocks described in Section 4.1.

4.5 Other

Appendix B includes a complete list of DLWR inventions, disclosures, articles, and reports.

5.0 FUTURE ACTIVITIES

The program plan lays out the long-term program. This section briefly describes recommended near-term activities (Table 4.). These activities can be broken into two categories: (1) collection of baseline information to direct, formulate, and aid the program and (2) research activities.

5.1 Baseline Information

The starting point for future development activities is an understanding of what has been done, what was learned, and what is important. It is very expensive in time and resources to "reinvent the wheel." Two activities to understand where we are now and help define where we should go are planned. These activities are part of the DLWR program but are expected to provide major assistance to other reactor programs.

Designs of Advanced Plants

The last several years have seen an explosive growth in proposed reactor concepts from vendors, national laboratories, and universities. This activity is to develop a single, consistent source of information on the characteristics of nuclear power plants which are

Table 4. Planned future Developmental Light-Water Reactor Program activities

Activity	Description	Why important
1. Designs of advanced plants	Description and Analysis of Advanced Reactor Concepts (domestic and foreign) including goals, criteria, and designs	Single source update on proposed reactor designs. Input to future work: "not reinvent wheel."
2. Economics of nuclear power	Analysis of nuclear operating cost by functional plant requirement (reactivity control, core cooling, security, etc.)	Input to focus research on areas with high cost (Note: operating and fuel cost of current nuclear plants now exceed coal plant costs)
3. New concepts for passive safety	Analysis of New Concepts for Passive Safety	Identify/create new technical options to eliminate economic, operational, and safety limits for DLWRs
(a) Passive LWR core power control devices	(a) Theoretical analysis of proposed passive devices to control power levels	(a) Most DLWR concepts are limited in size due to core power instability problems. If it can be shown that solutions exist to this problem, no size limit for DLWR. This has major economic implications. The technology may also be applicable to other reactors.

Table 4. Planned future Developmental Light-Water Reactor activities (continued)

Activity	Description	Why important
(b) Passive decay heat cooling system	(b) Analyze concept performance	(b) Concept applicable to BWR/PWR and current/ future reactors. The concept could prevent core damage in the type of accident that occured at TMI.
(c) Performance limits of fluidic valves	(c) Analysis of the limits of performance of fluidic valves	(c) Many DLWR concepts use fluidic valves to assure passive safety. Analysis to date shows the devices will work for the various proposed concepts. If high performance devices can be built, there are many additional DLWR options.
(d) New ideas RFP	(d) RFP	(d) RFP to universities and industry for new concepts

expected to be marketed in the near-term, under development, or proposed. Important characteristics include: (1) design goals and requirements, (2) technical descriptions of the concept, (3) current status of development, and (4) organizations responsible for development. This information is required to:

1. Provide a basis to understand requirements and goals of different proposed reactors to assure the DLWR program has appropriate requirements and goals.
2. Provide a basis to understand different ways safety and reactor systems can be combined to meet particular goals. Particular approaches used to solve a particular requirement in one reactor may be useable in other reactor designs.
3. Provide an understanding of the commercial aspects of reactor development and design with respect to trends and international competition.

This activity is a parallel activity to the FY1989 effort to produce a compendium of passive and inherent safety-related building blocks for reactors. The FY 1989 activity looked at the pieces (building blocks) used for advanced reactor design. This activity looks at the integrated designs.

Economics of Nuclear Power

In the United States, nuclear power plants have historically had higher capital costs than coal power plants but lower operating/fuel costs. In recent years, the operating/fuel costs of the average nuclear power plant have increased rapidly until today these costs are 10% higher than the average operating/fuel costs of coal power plants. This is a serious economic concern. If new reactors are to be developed, it is essential to understand why the costs have increased rapidly so appropriate requirements and goals are defined for new reactors. For example, if the cost of security to protect plant safety systems is a major cost driver with current reactors, then a requirement for future reactors must be safety systems with high resistance to sabotage/assault to minimize the need for security forces.

Existing economic studies of nuclear power costs have broken costs by categories such as equipment, labor, management, and quality assurance. For direction of research and development, costs must be broken down by functions such as (1) power production, (2) security, (3) core cooling, (4) accident mitigation, and so forth. Research must be targeted to real problems.

5.2 Research Activities

The proposed research activities are identified in Table 4 and are an outgrowth of earlier research. The criteria for selection of research options are as follows:

1. The technology must be a new innovative light-water reactor technology. The DOE has a broad LWR research program; and the DLWR program charter is in new technologies, particularly those associated with passive and inherent safety for LWRs.
2. The technology, if successful, should result in major improvements to LWRs.
3. The technology should be broadly applicable to LWRs – not limited to use in a single design of LWR. This broad applicability can exist in two forms:
   (a) The technology is applicable to both boiling-water reactors (BWR) and pressurized-water reactors (PWR).
   (b) The technology is applicable to current, near-term [Advanced Boiling-Water Reactor (ABWR) and Advanced Pressurized-Water Reactor (APWR)], mid-term Simplified Boiling-Water Reactor (SBWR) and AP-600] and long-term [PIUS (Secure P®), PIUS-BWR, other] light-water reactors.

Appendix A: PRIME Safety Characteristics

The DLWR goals with respect to radiological safety characteristics are summarized by the acronym PRIME (Passive, Resilient, Inherent, Malevolent-resistant, and Extended safety).

PRIME refers to the level of safety desired for a DLWR and how that level is to be achieved. For example, from a purely technical perspective, approaches other than the use of passive safety systems exist. Employing the PRIME safety characteristics as an approach to meet safety goals was chosen because it appears to have a reasonable probability of technical success and addresses the various social and economic issues.

The definition and explanation of the acronym PRIME follows.

A.1 Passive Safety

The DLWR should be designed with passive (rather than active) safety systems. This excludes the use of moving parts such as motors, pumps, and mechanical valves for the operation of safety systems.

A passive safety system is "a system composed of passive components and structures" (IAEA, 1988). A passive component is defined as "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 self-acting means" (IAEA, 1988).

There are multiple types of safety systems: inherent, passive, and active. Examples of fire protection systems can clarify the definition of these terms. A concrete building full of glass bottles to be recycled is inherently safe against fire, since a fire cannot occur. A tank-fed water sprinkler system is a type of passive safety system. An active system is the use of fire detectors, pump units, and firemen.

Passive safety, by itself, does not ensure a high level of safety. A poorly designed, built, and maintained passive safety system may not offer as much protection as a well-designed, -built, and -maintained active safety system. Passive safety systems have the potential to achieve high levels of safety if properly designed and built. Furthermore, they are a prerequisite for other desired safety characteristics such as malevolence resistance.

The development of passive safety systems generally lags behind that of other types of safety systems because of the difficulty of design. A historical example of this is the development of the jet engine for the commercial aircraft industry. The jet engine is not a passive device, but it contains fewer moving mechanical parts than a piston engine. Because it is simpler, it is much more reliable. The jet engine followed the piston engine by 40 years primarily because it was so difficult to design. Passive systems tend to be simple to build, maintain, and operate, but very difficult to invent/design. They are often the most advanced technologies.

A.2 Resilient Safety

The DLWR safety and operating systems must be resilient; that is, they must activate when required, but not interfere with the normal operation of the plant. Given sufficient time and effort, safety systems can be bypassed. For very high levels of safety, it is important to eliminate incentives to bypass safety systems and to make it difficult or impossible to bypass them (no off switch).

A.3 Inherent Safety

The DLWR safety system should have inherent safety whenever possible.

An inherent safety characteristic is "a characteristic [that] refers to the elimination of a specified hazard by means of the ch[illegible] of material and design, through the laws of nature only" (IAEA, 1988).

A nuclear reactor can never be completely inherently safe because it contains large quantities of radioactive materials as the result of its inherent mechanism for generating useable heat and energy. However, nuclear reactors can be made inherently safe against many types of accidents that can challenge the integrity of barriers that retain radioactive materials.

Inherent safety may also limit the maximum possible consequence of a reactor accident. Similar to chemical plant accidents involving potential release of toxic materials (Kletz, 1985; CCPS, 1985), the maximum potential radionuclide release from a reactor accident depends on the energy generated during the accident and the chemical forms of the radionuclides. If energy release during an accident is slow and the number and type of volatiles or aerosols carrying released radionuclides is limited, the maximum possible release of radioactivity after an accident (including core meltdown) will be limited. There are only four significant sources of energy in a reactor accident: (1) nuclear power excursion, (2) thermal reactions (steam explosions), (3) chemical reactions (zirconium/water and core/concrete), and (4) radioactive decay heat. The first three can be limited or controlled by proper selection of materials – a form of inherent safety. The fourth energy source, decay heat, is a slow and inherently restricted form of energy release.

A.4 Malevolence Resistance

The DLWR should be capable of passively withstanding malevolent acts of man without a significant release of radionuclides to the environment.

Plant security for ensuring public health and safety should depend primarily on passive (thermal/neutronic sluggishness, physical barriers, etc.) rather than active (guards, security checks, etc.) techniques. In practice, this design characteristic also covers operator errors and inaction. The reduction of dependence on active security methods (guards) can result in major savings in operating costs.

A.5 Extended Safety

DLWR safety should be assured for extended time periods after an accident without human intervention.

The time period for continued assurance of reactor safety is defined as essentially indefinite for cases of plant abandonment and as a period of one week after military assault or internal sabotage.

A.6 REFERENCES

Center for Chemical Process Safety, 1985. Guidelines for Hazard Evaluation Procedures, American Institute of Chemical Engineers, New York.

International Atomic Energy Agency International Working Group on Advanced Technologies for Water-Cooled Reactors, Oct. 3-5, 1988. "Working Paper: Description of Safety-Related Terms," Vienna, Austria [The descriptions herein are consensus draft descriptions which will be modified with time].

Kletz, T. A., 1985. Cheaper, Safer Plants or Wealth and Safety at Work: Notes on Inherently Safer and Simpler Plants, Institution of Chemical Engineers, Warwickshire, England.

APPENDIX B: PATENTS, NOTICES OF INVENTION, REPORTS, AND ARTICLES

This appendix summarizes patents, notices of invention, reports and articles on Developmental Light-Water Reactors from ORNL.

B.1 PATENTS

1. C. W. Forsberg, Boiling Water Neutronic Reactor Incorporating A Process Inherent Design, U.S. Patent 4,666,654 (May 19, 1987)

B.2 NOTICES OF INVENTION

The general areas of patent notices of invention are identified below. Specific titles in some cases are not provided because disclosures have not yet been made to the U.S. patent office or no decision on whether to file for a patent has been made.

Methods to Passively Control Power Levels in LWRs

Five notices of invention were filed in FY 1989.

Methods to Cool Reactor Core

One notice of invention was filed in FY 1989.

Mitigate Accidents

1. Passive decay heat sink for Nuclear Reactors and Radioactive Waste (1988)

2. Passive Short-Term Cooling for Nuclear Reactor Containments (1988)

B.3 GENERAL TECHNICAL REPORTS

1. C. W. Forsberg, D. L. Moses, E. B. Lewis, R. Gibson, R. Pearson, W. J. Reich, G. A. Murphy, R. H. Staunton, and W. E. Kohn, Proposed and Existing Passive and Inherent Safety-Related Structures, Systems and Components (Building Blocks) for Advanced Light-Water Reactors, ORNL-6554 (to be published).

B.4 ARTICLES ON A PROCESS INHERENT ULTIMATE SAFETY BOILING-WATER REACTOR

1. C. W. Forsberg, "Passive Emergency Cooling Systems for Boiling-Water Reactors (PECOS-BWR), Nucl. Tech., 76, 185 (January 1987).

2. C. W. Forsberg, "A Passive Inherent Ultimate Safety Boiling-Water Reactor," Nucl. Tech., 72, 121 (1986).

3. C. W. Forsberg, "A Passive Inherent, Ultimately Safe BWR," Trans. Am. Nucl. Soc., 50, 428 (November 1985).

4. C. W. Forsberg, "A Process Inherent Ultimate Safe Boiling-Water Reactor," Nuclear Safety, 26, 5 (September - October 1985).

5. C. W. Forsberg, "Summary of a Process Inherent, Ultimate-Safe (PIUS) Boiling-Water Reactor", Proceedings of the Workshop on Intrinsically Safe and Economical Reactors, Institute for Energy Analysis (Oak Ridge) and the Nuclear Engineering Research Laboratory of the University of Tokyo (Japan), (August 14-15, 1985).

B.5 INTERNATIONAL ATOMIC ENERGY AGENCY PAPERS/REPORTS

1. C. W. Forsberg, "Report of Foreign Travel; IAEA Technical Committee meeting on Passive Safety Features in Current and Future Water-Cooled Reactors," ORNL/FTR-3199 (April 5, 1989).

2. C. W. Forsberg, "Identification and Characterization of Passive Safety System and Inherent Safety Feature Building Blocks for Advanced Light-Water Reactors," IAEA Technical Committee Meeting on Passive Safety Features in Current and Future Water-Cooled Reactors, Moscow, USSR (March 21-24, 1989).

3. International Atomic Energy Agency, Status of Advanced Technology and Design for Water-Cooled Reactors: Light-Water Reactors, IAEA-TECDOC-479 (1988).

4. C. W. Forsberg, "Implications of Passive Safety Based on Historical Industrial Experience," IAEA Technical Committee Meeting on Definitions and Understanding of Engineered Safety, Passive Safety and Related Terms, Västerås, Sweden (May 30 - June 2, 1988).

B.6 OTHER

1. Chemical Technology Division Progress Report for the Period January 1, 1987 to June 30, 1988, ORNL-6490 (February 1989).

2. C. W. Forsberg, "New Developments in Reactor Design," Wattec '89, Knoxville, Tennessee (February 14-17, 1989).

APPENDIX C: TECHNICAL CHARACTERISTICS OF PIUS LWRs

C.1 INTRODUCTION

This appendix provides a brief description of the two PIUS concepts that have proposed: the PIUS-PWR and the PIUS-BWR. The PIUS-PWR was invented earlier, thus is further along in development. Each 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. A more complete description is provided of the PIUS-PWR which reflects its advanced development. Common features of both reactors are not described a second time in the description of the PIUS-BWR, but instead are referenced back to the earlier description.

It is noteworthy that two different technical approaches to creating a DLWR have been found. The area is a new area of research where major innovations can be expected.

C.2 PIUS-PWR

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

The PIUS-PWR is a modified "swimming pool" pressurized water reactor; 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 and shuts it down. The reactor core is cooled by boil-off of the borated water. The time period this Emergency Core Cooling System (ECCS) works in a passive mode depends upon the quantity of borated water available to be boiled off. Current proposed designs provide 1 week of passive heat removal.

This reactor has two unique features. The first feature is a very large pressure vessel in which are included the reactor core and all key safety systems. The second feature is 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 exceeds 13 m.

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 exceeds 7 m.

The PCRV has several unique design features:

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

Figure C.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 reactor 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 reactor core from below (J) and out through the top of the riser (K).

In Fig. C1(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. C1(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 upon differences in water densities of the high-temperature, low-boron concentration water in the riser and the low-temperature, high-boron 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 reactor core, it will be shut down. This does not happen in operation because of a careful hydraulic balance generated by the pump.

If the recirculation pump slows down below the natural water circulation rate [Fig.C.1(d)] through the reactor core, then cold, borated water will enter the reactor core from point J and shut the reactor down. If the pump operates too fast, pump suction will draw cold, borated water into the system near point M and through the steam generator and pump [Fig. C.1(e)]. The pump discharge will push some highly borated water into the reactor core near point N and push 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 water zones make the interface very stable. Instruments sense whether the hot/cold interface zone is moving up or down and adjust the pump speed accordingly.

Power levels in the reactor 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. C.1(f)]. Boiling causes major increases in natural circulation flows through the reactor core. The recirculation pump is sized so that it physically cannot handle the water flow through the reactor core under these circumstances. The hydraulic balance breaks down, and cold, borated water enters the reactor core from the bottom.

After the reactor shutdown, the cool, borated water heats up, adsorbing 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. C.2 with some design parameters given in Table 1. PIUS-PWR design options include steam generators on either the inside or 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-MWe, 2000-MWth power reactor. The current design also includes four independent natural circulation cooling systems which 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 borated 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 at least one week after 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.

C.3. PIUS-BWR

The PIUS-BWR was invented by C. Forsberg of ORNL. A basic rationale for consideration of a PIUS type-BWR rather than a PIUS-PWR 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. C.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,

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

Thermal power	MW	2000
Electric power (net)	MW	640
Core exit temperature	°C	289.8
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*
Fuel enrichment, reload fuel	%	3.5
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	13.4
Concrete vessel cavity total height	m	34
Concrete vessel cavity volume	m^3	3820

* Up to 32 fuel rods containing BA (Gd_2O_3)

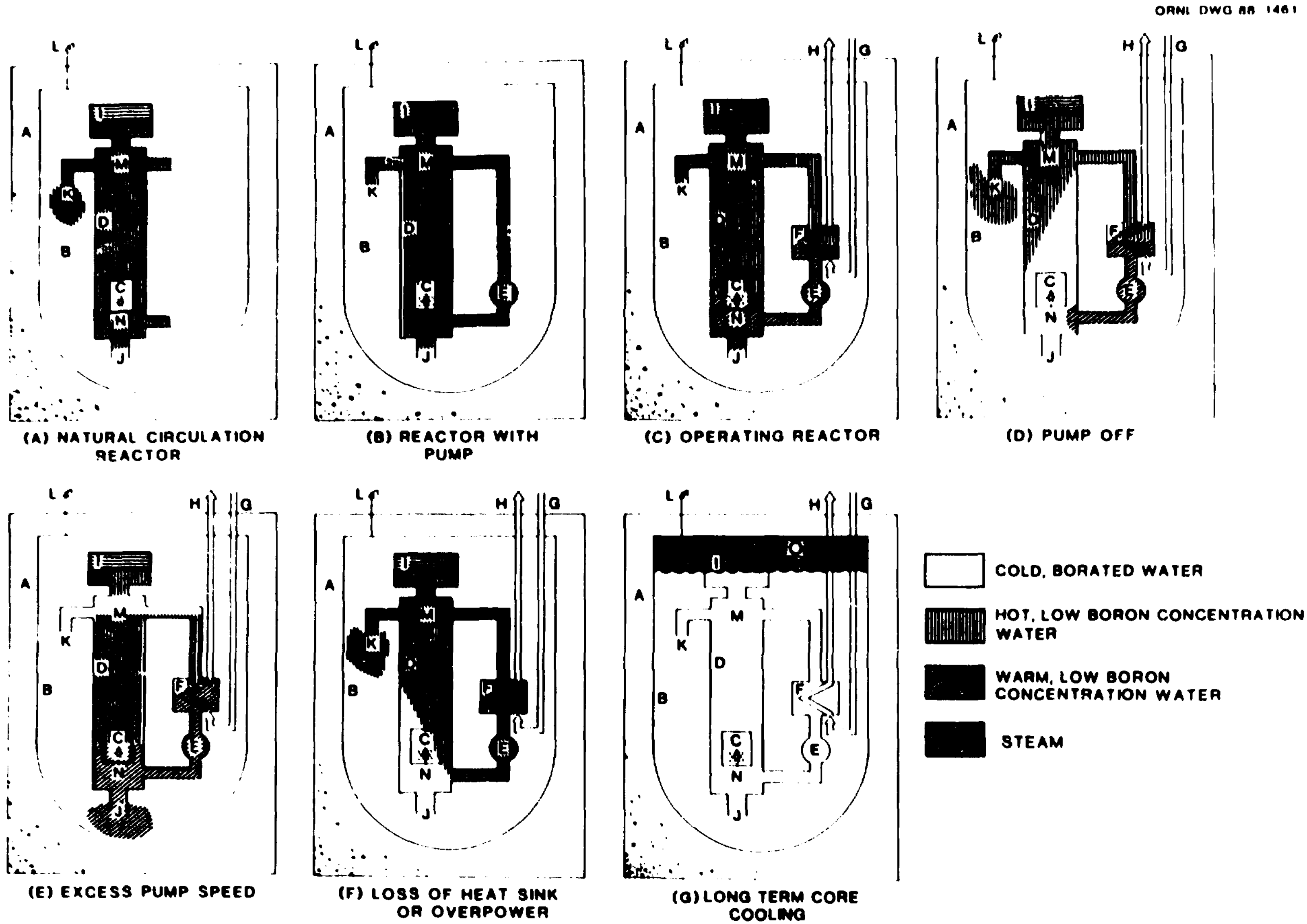

Fig. C.1. Operating principles of PIUS

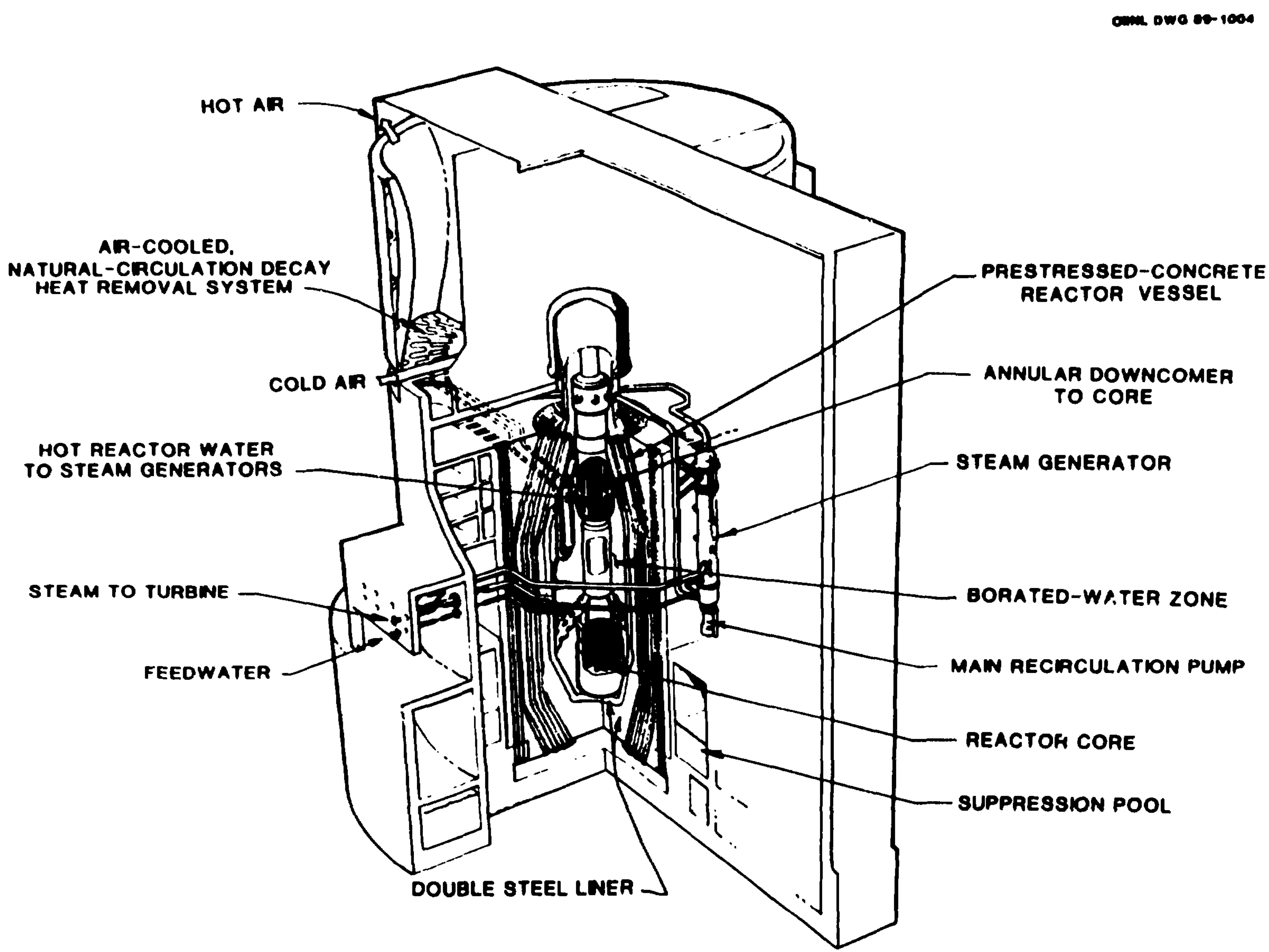

Fig. C.2. Proposed PIUS design by ABB Atom

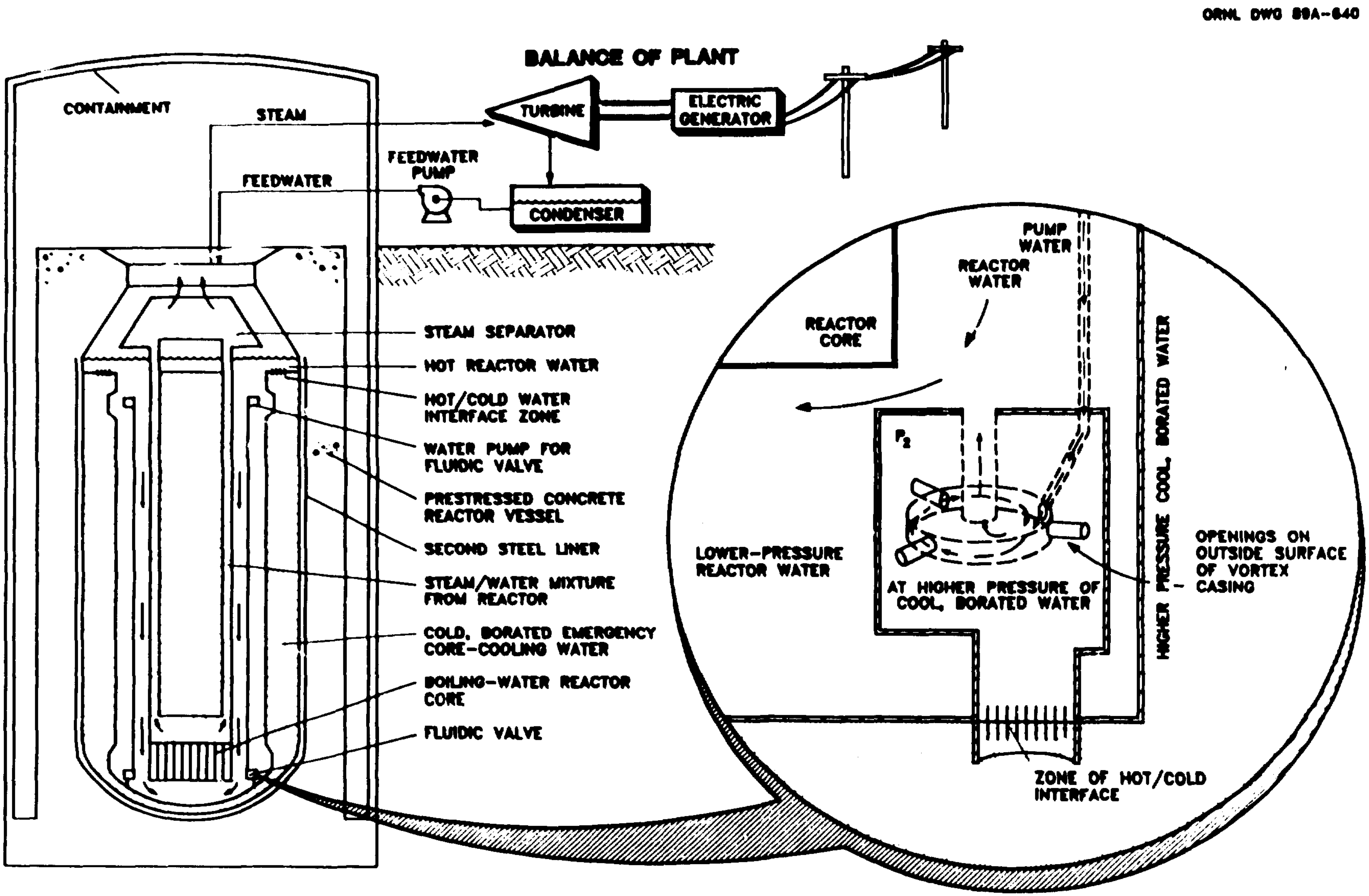

Fig. C.3. Process Inherent Ultimate Safety Boiling-Water Reactor borated water at reactor pressure; (2) a fluidic valve assembly that separates cool, borated, emergency water supplies from hot reactor water; and (3) the FIVES water pump system that provides power to the fluidic valve and detects water 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 two water zones are in direct contact with each other near the top of the pressure vessel through a hot/cold interface zone 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 of the reactor coolant because it is denser. If the fluidic valve opens, the core is flooded with borated water, which shuts down the reactor and cools it. The volume of borated water is sufficient to cool the reactor core by boil-off for a period of 1-week.

The fluidic valve assembly contains no moving parts. It is designed to remain closed only if it receives a continuous flow of water from the FIVES water-pump system.

Protection against a low water level in the reactor core is provided by positioning the FIVES water-pump system high above the reactor core (Fig. C.3). If a loss of feedwater occurs, the pumps will go dry before the reactor core is uncovered, and no water will be sent to the fluidic valve. This lack of water triggers the valve to open and floods the reactor with cool, borated water within seconds. The volume of reactor water in the downcomer between the elevation of the water in the steam separator and that in the water pump intake is sized so that normal plant transients will not activate the FIVES. The FIVES will self-activate only if there is a major threat to core integrity. The FIVES water pump system is, in effect, both the power supply system for the fluidic valve and a sensor of water level in the reactor – a fail-safe sensor.

The central component of FIVES is the vortex fluidic valve assembly , which is a modified vortex fluidic amplifier operated as a valve. This component is similar to a conventional centrifugal pump with a blocked exit line. The incoming FIVES water is injected tangentially at high velocities into the vortex casing, causing the water to move in a circle. The centrifugal forces create higher pressures near the outside surface of the vortex valve casing and lower pressures near the inside. The outside surface has holes (short lengths of tubing) that connect it to a zone of clean, higher pressure reactor water, which, in turn, is in contact with the borated water zone through a hot/cold water interface zone. The center of the vortex casing is connected to the downcomer and exhausts FIVES water to the downcomer. By adjusting water-pump output of FIVES, these pressures can be made to match the pressures of the two water zones. In effect, a no-moving-parts valve exists that uses the dynamic forces of water rather than pieces of metal to prevent flow through the valve.

Below the fluidic valve is the hot/cold water interface zone, where sensors determine the interface location. These sensors are used to control the speed of the FIVES water-pump system during normal operation. If the borated water interface rises relative to the reactor coolant interface, the FIVES water pump speeds up, increasing the pressure of the reactor coolant in the vortex valve box and pushing the interface back to its correct position. The reverse operation occurs if the borated water interface drops. Boron may leak through the hot/cold interface zone, so the exits to the reactor coolant cleanup systems (not shown) are located within the fluidic valve.

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