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ORNL--6554

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Chemical Technology Division

PROPOSED AND EXISTING PASSIVE AND INHERENT SAFETY-RELATED STRUCTURES, SYSTEMS, AND COMPONENTS (BUILDING BLOCKS) FOR ADVANCED LIGHT-WATER REACTORS

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Date published – October 1989

Prepared for

U.S. Department of Energy Office of Nuclear Energy

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operated by
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FOR THE
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400

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Function: Other Structures, Systems and Components

PREFACE

This report is a compendium of safety-related passive and inherent structures, systems, and components (SSCs) for light-water reactors (LWRs). Systems, structures, and components are the "building blocks" that are combined to create a complete reactor design. The SSCs described herein range in development from new concepts to commercial technologies.

The SSCs for this report were identified using two approaches. First, detailed literature and patent searches were conducted of both domestic and foreign information sources. Second, an extensive review process was used. The draft report was sent to vendors, utilities, the Electric Power Research Institute, consultants, architects/engineers, national laboratories, and universities. The assistance of reviewers was requested in two areas: (1) evaluation of those technical areas of the report where the reviewer is knowledgeable and (2) identification of missing SSCs (building blocks) for inclusion in the final report. The authors of the report thank the reviewers for their comments and for identifying additional SSCs for inclusion in the final report.

An addendum to this report is planned for the future. This addendum would include new SSCs, newly identified SSCs, and SSCs currently being prepared for patent applications and, therefore, not included in this report. The authors request the assistance of readers in identifying additional SSCs. Comments may be sent to:

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ABSTRACT

A nuclear power plant is composed of many structures, systems, and components (SSCs). Examples include emergency core cooling systems, feedwater systems, and electrical systems. The design of a reactor consists of combining various SSCs (building blocks) into an integrated plant design. A new reactor design is the result of combining old SSCs in new ways or use of new SSCs.

This report identifies, describes, and characterizes SSCs with passive and inherent features that can be used to assure safety in light-water reactors. Existing, proposed, and speculative technologies are described. The following approaches were used to identify the technologies: world technical literature searches, world patent searches, and discussions with universities, national laboratories and industrial vendors.

1. INTRODUCTION

1.1 REPORT OBJECTIVES

The objectives of this report are to identify and characterize existing and proposed, passive and inherent, safety-related structures, systems, and components (SSCs) and design features for Advanced Light-Water Reactors (ALWRs). The first four chapters (~20 pages) define the report's contents and describe how the report was prepared. The last nine chapters provide descriptions of the technologies.

1.2 EXPLANATIONS OF TERMS IN STATED REPORT OBJECTIVE

Structures, systems, and components (SSCs) are the items of equipment that are used to accomplish safety and process functions and are the building blocks used to design a reactor. A reactor has many plant structures, systems, and components (building blocks). For example, there are safety-related and process-related core heat removal systems, steam supply systems, and electrical systems. Any new power reactor or other new industrial facility can be expected to consist of a mixture of SSCs (building blocks) that will be based on both existing and new technologies. New concepts for reactors may involve new arrangements of existing SSC technologies or combinations of old and new SSC technologies.

Safety-related refers to the functions and equipment performance requirements necessary to protect the health and safety of workers and the general public from radiological hazards. As used within this report, "safety-related" does not refer to other process functions or equipment required by law and regulation to provide nonradiological environmental protection or industrial safety against nonradiological hazards to workers.

Design features are those features of the reactor plant such as materials of construction and geometric arrangement that, if altered or modified, would have a significant impact on the capability to ensure plant radiological safety.

Existing and proposed SSCs and design features are included in this report. New proposed SSCs, in terms of development, may vary from a concept that has been proposed but not studied to a concept that has gone through extensive design and testing.

A passive safety system is a system composed of passive components and structures. A passive component is a component which does not need any external input to operate (1).

An inherent safety characteristic is a characteristic that results in the elimination of a specified hazard by means of the choice of material and design, through the laws of nature only (1).

A nuclear reactor can never be completely inherently safe because it contains large quantities of radioactive materials to generate usable heat-energy; but nuclear reactors can be made inherently safe against some types of events and have characteristics which limit the consequences of certain postulated accidents.

There are multiple types of safety characteristics: inherent, passive, and active. Examples of fire protection systems can clarify these terms. A concrete building full of glass bottles to be recycled is inherently safe against fire. 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.

1.3 REPORT RELATIONSHIP TO LIGHT-WATER REACTOR PROGRAMS

The United States has a large, complex program on light-water reactors including near-term, mid-term, and long-term objectives. An overview of various reactor options in terms of their development is shown in Fig. 1.1. This report provides support to various activities including long-term options. The contents of this report reflect this perspective by including existing, proposed, and speculative SSCs (building blocks) that could be applied to LWRs. An effort has been made to carefully define the status of the various technologies. It must, however, be understood that not all of these SSCs (building blocks) may work nor may all of these technologies be acceptable for reasons of economics, investment protection, or nonradiological safety considerations. Independent analysis and experiments by the authors of this report have not been done on the concepts described herein. The capabilities described of these SSCs (building blocks) are those identified in the literature. The process of research on advanced reactors is first to identify options and then to determine which options are feasible. This report is the first step and part of the second step of this process.

1.4 USES OF REPORT

This report is written for technical and management personnel involved in research and development (R&D) on existing and advanced LWRs. Particular SSCs or design features described herein may be applicable for retrofit on existing reactors or incorporated into various advanced reactor concepts.

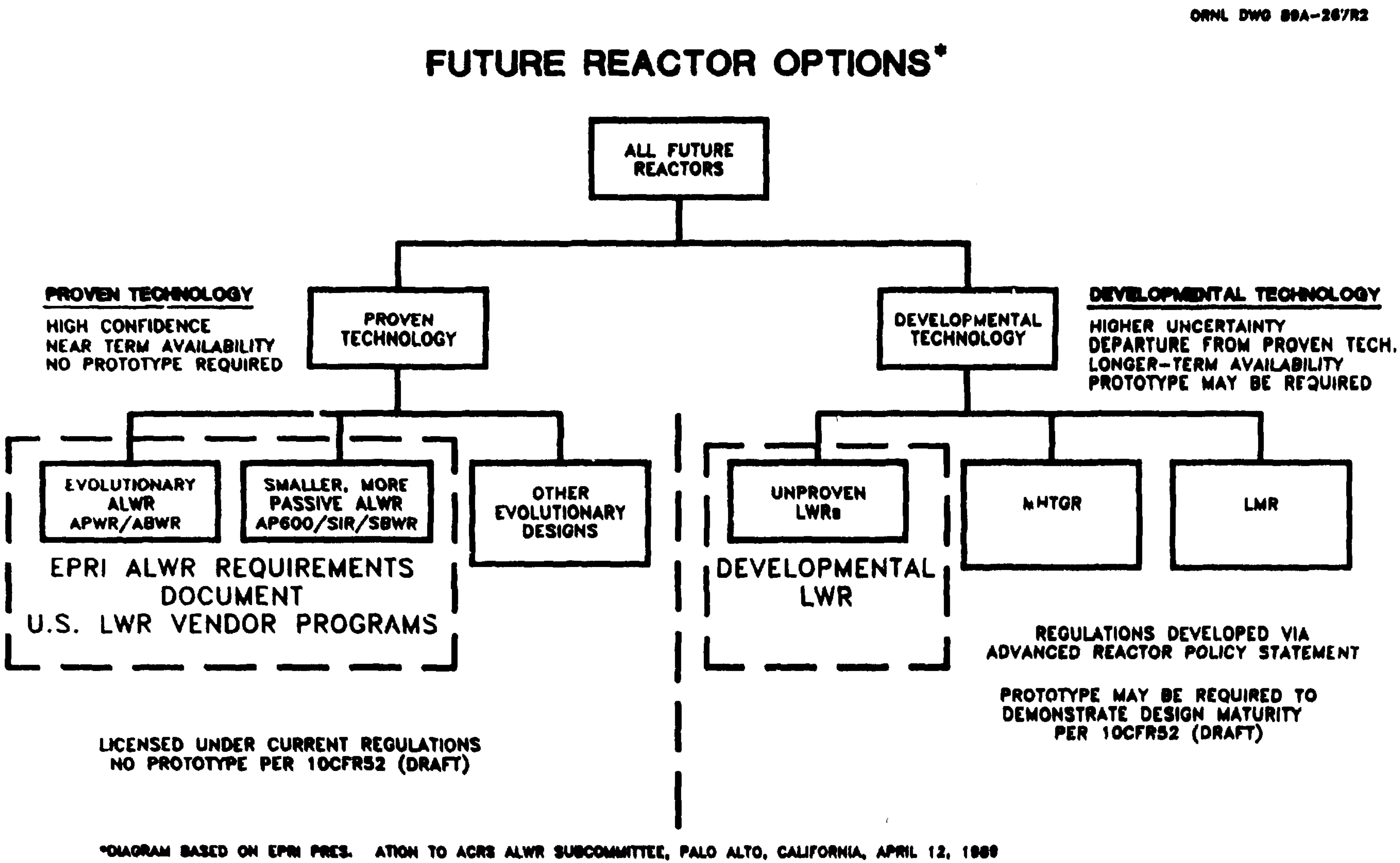

Fig. 1.1. Future reactor options.

The report has multiple uses:

- It defines current and proposed options for advanced reactors.
- It defines the current status of technology for various SSCs.
- It identifies functional requirements for an LWR in which there are no known passive safety system options or only options that would be difficult to implement. This assists in defining future directions of research.

1.5 DEFINITION OF REPORT CONTENTS

A power reactor has many SSC options. It is necessary to have a screening technique to define the building blocks (SSCs) or design features that should be included in this report and those that should be excluded. A two-part screening process is used to define the report content. This includes defining the characteristics desired of the safety systems and defining the safety-related functional requirements. The screening process is shown in Fig. 1.2.

1.5.1 Objectives and Goals

This report describes technologies for achievement of reactor safety with passive or inherent safety SSCs. This restriction on the characteristics of the safety systems was used as a screening mechanism to determine the contents of this report. Section 2 provides additional information on this constraint.

1.5.2 LWR Safety-Related Functional Requirements

The second screening step used to define the report contents was the requirement that each SSC included should meet a functional requirement for LWR safety. A functional analysis of LWR safety was completed to define these requirements in a systematic manner. As discussed in Chapter 3, the safety-related functions for LWRs are derived from a top-level assessment of the sources of potential challenge to fission product and radionuclide barriers in the reactor. These safety-related functions relate to (1) preventing and (2) mitigating the release of radiation through those barriers. A detailed functional analysis is provided in Fig. 1.3.

It should be noted that equipment that is not classified as "safety-related" may still execute safety functions under certain conditions. For example, there may be several sets of SSCs or design features that are dedicated to accomplish a specific functional requirement depending on the reactor operating mode or state of challenge. In current generation LWRs, heat removal from the reactor is accomplished in combination with the

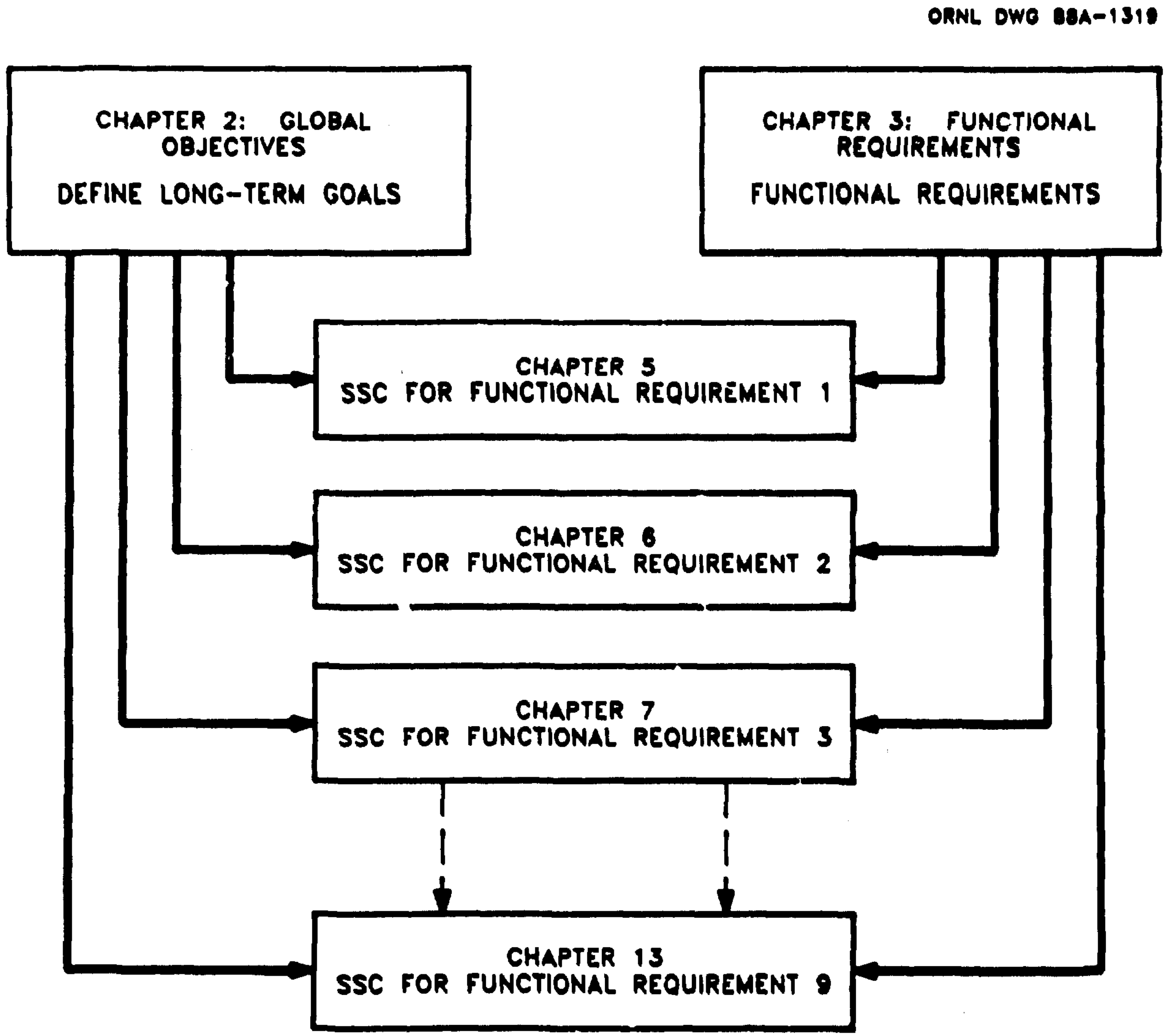

Fig. 1.2. Report logic and organization.

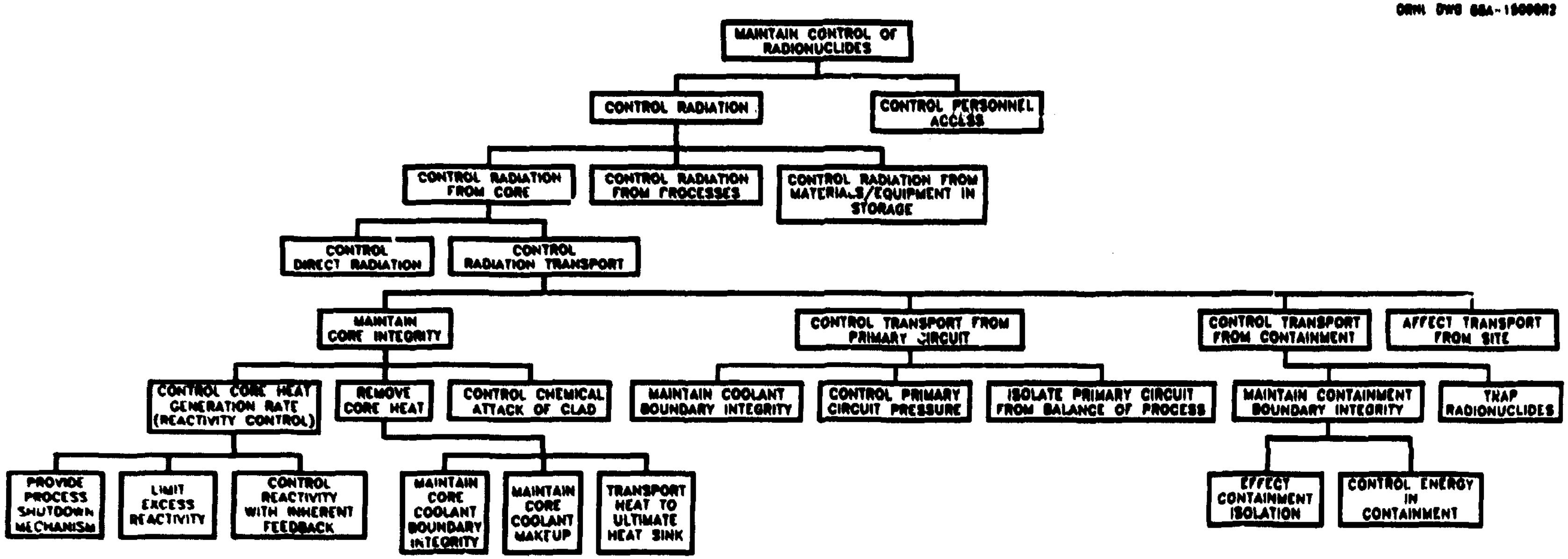

Fig. 1.3. Functional analysis requirements for light-water reactors.

primary coolant system by (1) the main steam and feedwater systems during normal powered operations, (2) the residual heat removal system during normal shutdown, (3) the emergency feedwater system during loss of normal heat sink, and (4) the emergency core cooling system during large break loss-of-coolant or total loss-of-heat-sink transients. Typically, process systems, such as main steam and feedwater systems, are not required by regulations to be seismically and environmentally qualified as "safety-related" equipment; but, the accomplishment of the safety function (core heat removal) by such process equipment is assured during normal operation by NRC technical specifications on normal process variables that are indicative of the continued operability of the equipment and the maintenance of normal conditions. Since the proper functioning of this type of equipment is indicative of the absence of a safety challenge as opposed to the mitigation of challenge, such equipment is not classified by regulations as being "safety-related." In general, the mitigation functions discussed in Section 3 reflect a consistent definition for safety-related functions as expressed in most current regulatory requirements.

1.6 ORGANIZATION OF SSC DESCRIPTIONS

This report organizes descriptions of SSCs by functional requirement. This organizational approach allows investigators to identify existing or proposed options for accomplishing LWR safety-related functional requirements. It also provides a way to identify where passive safety systems or inherent safety characteristics do not exist to meet a functional requirement. Sections 5 through 13 describe the SSCs meeting the respective functional requirements.

Within each functional requirement chapter, the top-level functional requirement may be broken into lower-level requirements. This reflects the fact that an SSC may meet either a top-level functional requirement or only a lower-level functional requirement that stems from the top-level requirement. For example, the functional requirement to remove core heat can be broken down into lower-level requirements such as maintain core coolant boundary integrity, maintain core coolant makeup, and transport heat to ultimate heat sink. A single SSC or combination of SSCs may meet a lower-level functional requirement such as maintain core coolant boundary integrity.

Some SSCs may meet multiple functional requirements. In such cases, the description of a particular SSC is included in only one section of the report, but is cross-referenced where appropriate.

The report does include some SSCs that may not fully qualify as passive safety systems or inherent safety characteristics. In many cases future R&D may be able to upgrade such building blocks to qualify; hence, such options are included.

1.7 REPORT ORGANIZATION

The first four sections define the organization and contents of the report. Chapter 2 summarizes objectives, chapter 3 summarizes functional requirements, and chapter 4 gives the standard format used to describe each SSC in the following chapters. Chapters 5 through 13 contain the actual SSC descriptions organized by functional requirement. Each SSC description is a complete, stand-alone description.

1.8 REFERENCES

1. International Atomic Energy Agency, "Technical Committee Meeting on Definition and Understanding of Engineered Safety, Passive Safety and Related Terms", Västeras, Sweden, May 30 - June 2, 1988. [Various draft descriptions of safety concepts for advanced reactors were prepared during the meeting by the meeting group as a whole and refined later by a smaller group. The useage herein is generally consistent with those descriptions.]

2. REPORT OBJECTIVES AND GOALS

For advanced light-water reactors, there are several approaches to improve economics and safety. One approach which has been recognized (1-3) as potentially offering major gains in power reactor economics and safety is the use of inherent safety characteristics and passive safety systems.

The development of inherent safety and passive safety systems has generally lagged behind that of other safety systems for a variety of reasons. An early philosophy of reactor operations emphasized the role of the operator. Operator selection, discipline, training, and knowledge were stressed as the basis for safe, efficient nuclear reactor operation. By the late 1970s, the increased complexity of nuclear power plants with size, the larger number of nuclear reactors with the need for more operators, the requirements placed on the operator for electric generation (high efficiency, operation within electrical grid constraints), and accidents where operator error was a factor changed the philosophical basis of depending primarily on the operator for safety. Simultaneously, major advances in passive safety systems were made. As a consequence, passive safety is receiving increased emphasis as an approach to improve safety while reducing costs. Passive systems tend to be simple to build, maintain, and operate, but difficult to invent. They are often the most advanced technologies.

In a complex plant such as a light-water reactor power plant, multiple structures, systems, and components are required to assure safety. If a power plant as a whole is to depend only on passive safety systems and inherent safety characteristics, each power plant system and subsystem can only use passive safety systems or inherent safety characteristic SSCs. In effect, the characteristics and performance of the entire reactor depend on the characteristics and performance of individual SSCs (building blocks).

If new reactor designs using passive and inherent safety or new SSCs with passive safety systems and inherent safety characteristics are to be developed, a prerequisite is to know what has been proposed or developed. This report catalogs SSCs releasable at the time of preparation.

REFERENCES

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

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

3. Taylor, J. J., "Improved and Safer Nuclear Power," Science, 318 (April 21, 1989).

3. SAFETY-RELATED FUNCTIONAL REQUIREMENTS

3.1 INTRODUCTION

As described in Fig. 1.3, the goals for LWRs are first to design and operate the facility to preclude challenges to fission product barriers, but, if such challenges are not precluded, to maintain control of the radionuclides that may be released through one or more fission product barriers. This approach to safety is that of defense in depth, both (1) by the provision of SSCs and design features that eliminate or substantially reduce the likelihood of challenges and (2) by the provision of SSCs and design features that either provide the redundant and diverse physical barriers to fission product release or otherwise ensure the integrity of those barriers under challenge. The basic functions are thus the (1) prevention and (2) mitigation of challenges.

For purposes of defining safety-related functional requirements, the fundamental safety-related function to be accomplished is the control of radionuclides to ensure meeting regulated dose guidelines for workers and the general public. Given that radiation and radionuclides are inherent in the fission process that generates the usable heat and energy in the reactor, the production of the source for radiation exposure cannot be prevented. The sources of radiation can only be controlled to prevent radionuclide leakage or with the defense in depth capability to mitigate its effects through attenuation of leakage. Thus, in defining safety-related functional requirements, the prevention of challenge is viewed as a functional subset to the mitigation of the challenge.

This premise has been incorporated into a functional analysis for LWRs. As illustrated in Fig. 1.3, the top-level safety-related functional requirement is to maintain control of radionuclides and the radiations that emanate from them. To be comprehensive, Fig. 1.3 acknowledges that radionuclides and radiation can exist in and emanate from other sources that may be located at a reactor site. However, during plant operation, the reactor core is the dominant source of radionuclides, and the development of the safety-related functional requirements for the LWR is thus focused on controlling radionuclides that are in the reactor core as the result of powered operation. As shown in Fig. 1.3, the key safety-related subfunctions that must be accomplished to control release of radionuclides from the core include, in the sequence of successive levels of challenge:

- maintain core integrity,
- control transport of radionuclides from the primary circuit,
- control transport of radionuclides from containment.

Table 3.1 Functional Safety Requirements for LWRs

Number	Functional requirements	SSC Chapter[a]
1.0	Maintain core integrity	
1.1	Control core heat generation rate (reactivity control)	Chapter 5
1.1.1	Provide process shutdown mechanism	
1.1.2	Limit excess reactivity	
1.1.3	Control reactivity with inherent feedback	
1.2	Remove core heat	Chapter 6
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	Chapter 7
2.0	Control transport from primary circuit	
2.1	Maintain coolant boundary integrity	Chapter 8
2.2	Control primary circuit pressure	Chapter 9
2.3	Isolate primary circuit from balance of process	Chapter 10
3.0	Control transport from containment	
3.1	Maintain containment boundary integrity	Chapter 11
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	Trap radionuclides	Chapter 12

[a]Chapter 13 contains SSCs which do not fit any of the above categories.

Each of these key subfunctions can be divided further into more detailed functional requirements that reflect the unique or inherent features of the water-cooled reactor (Table 3.1). The detailed functional requirements for LWR safety are discussed below.

3.2 MAINTAIN CORE INTEGRITY

Traditionally, LWR fuel pellets and cladding have been classified as the first barrier to the release of fission products. Maintaining the integrity of LWR fuel and cladding is based primarily on ensuring that the rate of core heat generation does not exceed the rate of core heat removal both during normal operation and under decay heat loads. A secondary function has been to ensure the integrity of the cladding to degrading effects such as normal corrosion, oxidation, hydriding, and other forms of chemical attack either from the hot irradiated fuel (pellet clad interaction) or from the ingress of undesirable chemicals such as chlorides into the primary coolant. As referred to herein, maintenance of core integrity excludes prevention of minor pinhole leaks in the fuel for which normal reactor water cleanup systems are designed to handle.

The control of core heat generation in turn depends upon, or at least is greatly facilitated by, meeting other functional requirements that relate to controlling the neutronic reactivity of the core configuration. At a minimum, these include:

- the provision of a process shutdown mechanism,
- the limiting of excess reactivity, both locally and globally in the core and in the control and protection system, and
- the control against positive reactivity insertions by control of reactivity with inherent feedback.

The need for a process shutdown mechanism is listed as a separate functional requirement but does not necessarily imply that a separate set of SSCs is required since the process shutdown mechanism may be effected through feedbacks that are initiated by an external manual or automatic feature. Limiting excess reactivity in the design of the core and of the control and protection systems limits the challenge that can be posed by equipment malfunction or externally induced core disruptive events. Limiting excess reactivity locally within the core achieves power tayloring so that local heat generation rates do not exceed the heat removal rate. The provision of reactivity feedback mechanisms (that is, negative reactivity coefficients) provides for the intrinsic stability of the core during challenges posed by positive reactivity insertions up to the design limits on available excess reactivity.

The required capacity to remove core heat stems from the need to balance heat generation against the capacity of the core to absorb heat without reaching temperatures that challenge the integrity of the core fission product barriers. In LWRs, the heat capacity of the core below fuel-clad damaging temperatures is relatively small compared to that of the water in the core. Because water/steam is a two-phase fluid that can change phases within the pressure-temperature conditions that occur in the conventional LWR core, water must be present to maintain core cooling. The need to keep the core supplied with water implies three additional subtier functional requirements:

- the maintenance of the core coolant boundary integrity so that water does not leak from the reactor vessel or primary circuit in excessive amounts,
- the maintenance of core coolant makeup to ensure adequate core inventory over all fuel, and
- the transport of heat to an ultimate sink from the core cooling water so that stable or decreasing temperatures can be achieved in the core and core coolant. (If the latter functional requirement is not met, the two functional requirements listed above may not be achievable in reasonable and stable fashion.)

3.3 CONTROL RADIONUCLIDE TRANSPORT FROM THE PRIMARY CIRCUIT

If core integrity is lost, fission product and/or irradiation product radionuclides will enter the primary circuit. If the primary coolant boundary integrity has been maintained, the release of radionuclides will be limited to within the reactor primary system, and the primary coolant cleanup systems can be used to remove fission products. If leak pathways exist from the primary circuit, further leakage can be controlled by controlling primary circuit pressure, which provides the driving force for leakage, and by isolating the primary circuit from the balance of the process plant. If a major breach has occurred in the primary circuit boundary, pathways will exist for leakage of radionuclides into the containment.

A distinction is made between "maintain coolant boundary integrity" associated with control of radionuclide transport from the primary circuit and "maintain core coolant boundary integrity" associated with maintaining core integrity. To prevent radionuclide transport, the boundary must be gas tight. To protect core integrity, liquid water must not leak out of the reactor core. A small opening at the highest point of the primary system might not affect core integrity, but it reduces control of radionuclide transport from the primary system.

3.4 CONTROL RADIONUCLIDE TRANSPORT FROM CONTAINMENT

The control of radionuclide transport from containment can be achieved by meeting two key functional requirements:

- the maintenance of containment boundary integrity so that radionuclides will not leak, and
- trapping the radionuclides within containment.

Maintaining containment boundary integrity requires the isolation of containment and the control of energy released in containment such that boundary integrity is neither lost nor degraded by dynamic effects.

The trapping of radionuclide releases into containment is the last functional barrier to external releases. Even if containment boundary integrity has been lost, the trapping of radionuclides inside containment can preclude releases to the environment.

3.5 AFFECT RADIONUCLIDE TRANSPORT FROM SITE

In the past, the assurance of elevated release points, the distance for attenuation provided by plant setting and exclusion areas, and the reliance on emergency procedures have been the mechanisms for mitigating the transport of radionuclides from the site to the surrounding population and environment. Advanced technologies with appropriate characteristics to meet the preceding sequence of functional requirements may obviate the need to rely on these latter mechanisms.

3.6 REPORT ORGANIZATION

The descriptions of SSCs in this report are organized using the preceding functional requirements analysis for LWRs. A summary of the chapter organization is shown in Table 3.1.

Within any chapter, the SSC descriptions are ordered in a hierarchical fashion from highest to lowest functional requirement. For example, Chapter 5 describes devices for the second level functional requirement of reactivity control. Descriptions of second-level SSC devices which provide complete reactor reactivity control (1.1) are first, followed by descriptions of devices which meet only third level functional requirements (1.1.1, 1.1.2, and 1.1.3) for reactivity control as shown in table 3.1.

4. FORMAT OF SSC DESCRIPTION

4.1 INTRODUCTION

This report describes and identifies structures, systems, and components (SSCs) that could be the building blocks for an advanced light-water reactor. These SSCs are described in a common format. Descriptions are not intended to provide detailed information, but rather, to describe the principles of operation of each SSC, current status of development, and references where detailed information can be obtained.

Many SSCs almost, but not quite, qualify as passive safety systems or inherent safety characteristics. Some of these are included here since additional work may upgrade such SSCs so they qualify.

4.2 SSC FORMAT

The format used to describe each SSC is shown in Table 4.1 with a description of each line. A simplified example of an SSC description is included in Table 4.2.

Table 4.1 Structures, Systems and Components Description Format

Format	Description
Title	Title of SSC
Functional requirements	List of the functional requirements identified in Chapter 3 of this report which may be met by this SSC
Safety type	Type of safety: inherent, passive, active, or other
Developmental status	The current status of this technology on a scale of 1 to 5 Status — Definition of status 1 Commercial — In commercial application in multiple light-water reactors 2 Demonstrated — Technology has been demonstrated in one or more reactors or sufficient design verification exists for first application in a power reactor 3 Evaluated — Prototype testing or equivalent is required before application 4 Analyzed — Technology has been analyzed for application 5 Concept — Conceptual ideal
Reactor type	Types of reactors to which the technology could be applied: LWR — light-water reactor (PWR and BWR) BWR — boiling-water reactor PWR — pressurized water reactor.
Examples of implementation	Examples of implementation. Examples may include non-LWR reactors
Organization (Name)	Type of organization (name of organization) which either developed or is using the particular building block

Table 4.1 (continued)

Format	Description
Description	Description of the technology. This will generally include a very simple nonmathematical description of the physics of the SSC. This may be followed by a summ.ary of an actual design using summary table or figure
Alternative versions	Alternative versions or use of the technology
Status of technology	Current status of technology
Advantages	Potential advantages of particular SSCs are described as identified from the literature. (No independent analysis of these advantages was done.) Advantages may include safety, cost, and operations.
Added requirements	Some safety systems may have additional functional requirements beyond those needed for LWR safety as defined in Chapter 4. Additional requirements raise questions about meeting safety goals
Comments	Any additional items of interest not included elsewhere.
References/contacts	References or contacts for additional information
Update date/compiler	Last update of description/Compiler of this information sheet

Table 4.2. Abbreviated/simplified example of SSC format and description

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TITLE: REACTOR PRESSURE VESSEL WITH NO BOTTOM PENETRATIONS

Functional Requirements: Level 1.2.1: Maintain Core Coolant Boundary Integrity

Safety Type: Inherent

Developmental Status 1: Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Vendors (Multiple)

Examples of Implementation: Yankee Rowe Nuclear Power Plant, Massachusetts, PWR

Description: Reactor pressure vessel is designed with no bottom penetrations. Lowest vessel penetrations are the cold and hot water nozzles near the top of the pressure vessel. The reactor core is placed at the bottom of the pressure vessel.

Alternative Versions: None

Status of Technology: Technology is standard practice for reactors built by several vendors.

Advantages:
1. No bottom vessel penetrations eliminate the possibility of pipe or instrument tube failure draining pressure vessel. Minimum emergency core cooling water quantity is ensured by the pressure vessel volume below the nozzles and above the reactor core.
2. Higher reliability, faster, simpler inspection of pressure vessel integrity

Additional Requirements: None

Comments:
1. See: "Reactor Pressure Vessel with Minimum Penetrations."
2. The largest existing pressure vessels with this feature are those for certain heavy water reactors (1). Heavy water reactors have lower power densities which require the use of larger pressure vessels than for LWRs.

References:

1. "Building a 745 MWe Pressure Vessel PHWR in the Argentine," Nuclear Engineering International, 30 (September 1982).

Update/Compiler: Nov. 1988; CWF

CHAPTER 5

Structures, System and Components to Control Core Heat Generation Rate (Reactivity Function)

Function 1.1

Table 5.1 Structures, Systems, and Components to Control Core Heat Generation Rate (Reactivity Control): Function 1.1

Description Title	Function	Location of Description	Page
Process Inherent Ultimate Safety (PIUS) Reactor Technology	1.1	Chapter 5	5-4
Passively Safe Pressurized-Water Reactor with Gas Bubble	1.1	Chapter 5	5-13
Passive Control of Power Levels by Variable BoronConcentration in Pressurized-Water Reactor (GEYSER)	1.1	Chapter 5	5-17
Use of Hydrides which Reversibly Adsorb and DesorbHydrogen to Control Reactivity in LWRs	1.1	Chapter 5	5-21
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	Chapter 5	5-29
Slow-Withdrawal Control Rods	1.1.1	Chapter 5	5-35
Fluidized Bed Control Rods	1.1.1	Chapter 5	5-38
Self-Actuating and Locking Shutoff Valve for Hydraulic/Fluidic Control Systems Initiated on Low Water Flow or High Temperature	1.1.1	Chapter 5	5-41
Neutron/Gamma/Coolant Thermal Fuse Control Rods/Devices	1.1.2	Chapter 5	5-44
In Reactor Core Power Fuses	1.1.2	Chapter 5	5-49
Use of Select Material Phase Changes to Enhance Doppler Reactivity Feedback in LWRs	1.1.3	Chapter 5	5-57
Large-Prompt Negative-Moderator Coefficient of Reactivity from Hydrogen Bond Structure	1.1.3	Chapter 5	5-59
High-Temperatures Ceramics, Clads, and Fuels for LWRs	1.1.3	Chapter 5	5-61
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Graphite-Disk UO_2 Fuel Elements for Enhanced Thermal Margin and Reduced Excess Reactivity	1.1.3	Chapter 5	5-65

TITLE : PROCESS INHERENT ULTIMATE SAFETY (PIUS) REACTOR TECHNOLOGY

Functional Requirements: Level 1.1 Control Core Heat Generation Rate (Reactivity Control)
Level 1.2 Remove Core Heat

Safety Type : Passive

Developmental Status: 2 Demonstrated

Reactor Type: Pressurized-Water Reactor (PWR)

Organization: Vendor (Asea Brown Boveri)

Examples of Implementation: None

Description: Process Inherent Ultimate Safety (PIUS) refers to a reactor concept and a particular technology. The PIUS concept[2-12] limits reactor power to safe levels and is an emergency core cooling system.

PIUS is a modified "swimming pool" PWR where the swimming pool is at full reactor pressure and contains high concentrations of cool borated water. The reactor is at the bottom of the swimming pool in a second volume of hot reactor water that contains a low concentration of boron. In the event of an accident, the cool borated (neutron poison) water enters the reactor core and shuts it down. The reactor core is cooled by boil-off of the borated water. The time period that the ECCS works in a passive mode depends upon the quantity of borated water available to be boiled off. Current proposed designs vary from two days to one week of passive heat removal.

The unique feature of PIUS is that the cold borated water zone is in direct contact with the hot, low-boron-concentration, reactor coolant water. The cool borated water does not enter the reactor core during normal operations due to a hydraulic balance maintained by the main recirculation pumps. The operating principles of PIUS are shown in Fig. 1.

Fig. 1a shows 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 quickly shut down by natural circulation of borated water into the reactor core from below (J) and out through the top of the riser (K).

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

In Figure 1c, 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 that depends upon 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 the 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 due to a careful hydraulic balance generated by the pump.

If the recirculation pump slows down below the natural water circulation rate (Fig. 1d) through the reactor core, 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. 1e). 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 ingress of cold borated water at only one pump speed for each set of operating conditions.

The hot reactor water is separated from cold borated water by interface zones (J, K). The large density differences between the two water zones make the interface very stable. Instruments sense if 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. 1f). Boiling causes major increases in natural circulation flows through the reactor core. Since the recirculation pump is sized so that it physically cannot handle these higher water flows through the reactor core, 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 by 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.

The most advanced reactor design[12] using this SSC is the proposed Secure P® reactor by Asea Brown Boveri (ABB) -Atom (Fig. 2). The design criteria for this reactor includes assurance of reactor safety after sabotage events or assault by conventional, off-the-shelf military weapons. The Secure P® is a 640 MW(e), 2000 Mw(t) reactor. A prestress concrete reactor vessel (PCRV) is used [see Prestress Concrete Reactor Vessel (PCRV) for Light-Water Reactors]. This allows the following:

1. The PCRV contains sufficient borated water to cool the reactor core for a period of one week after reactor shutdown.

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

3. The PCRV provides very high levels of protection against external threats.

Table 1 summarizes some of the design parameters.[12]

Alternative Versions: The basic concept of PIUS provides reactor core cooling until the volume of pool water has evaporated. For typical designs, a one-week supply of water is provided inside the pressure vessel. This cooling time can be extended indefinitely by use of any one of several long-term options to cool the borated water and, hence, the reactor core. The one-week grace period provided by the borated water evaporation implies that radioactive decay heat from the reactor core will have decreased significantly in this time period. Heat leaks through the insulation from the hot primary system to the cool borated water during normal operation; therefore, borated water coolers are needed for normal operation. The same cooling systems can handle long-term decay heat cooling loads.

There are multiple long-term options for cooling the borated water (see also SSC: Heat Pipe Cooling for Reactor Containment or Decay Heat Cooling Systems), including use of natural circulation air coolers (see Fig. 3). With this option,[7] the borated water coolers contain a fluid such as ammonia. The ammonia boils in the coolers, removing heat from the borated water. The vapor-state ammonia flows to an air cooler, is condensed, and the liquid flows back to the borated water coolers. The air coolers can operate either with fans or as natural air circulation units.

During normal operations, the borated water is kept near 50°C. In emergency conditions, the borated water will be at the boiling point of water (100°C) if the reactor is fully depressurized. Air cooler performance improves rapidly with higher temperature operation. In a number of proposed designs,[13] requirements for cooling borated water during normal operations controlled the design. For example, an air cooler requiring active fans for normal borated water cooling could handle higher temperature decay heat cooling requirements with the air cooler fans off.

The most recent[14-15] design of the ABB PIUS reactor includes long-term decay heat cooling with four natural circulation air cooling loops. With this design, after 14 hours, three of the four loops will handle the total decay heat with reactor water temperatures below 100°C. The system is used for both normal and emergency cooling of the borated water.

Status of Technology: There are several programs investigating the PIUS concept. The largest program is lead by ABB-Atom. Vendors in several countries are working with ABB. An alternative version of PIUS is being investigated in Japan at the University of Tokyo.

1. The ABB-Atom program[1-8] has as its goal commercial development of a passively safe PWR called Secure P®, which is based on this SSC. Current design studies are based on a 640 MW(e) power plant. The original concept of PIUS grew out of an earlier joint Swedish-Finnish program to develop an ultrasafe district heating reactor. The concrete reactor vessel technology came from an ABB program to develop concrete pressure vessels for 1000 MW(e) boiling-water reactors (BWRs).

   ABB-Atom has done extensive research on its Secure P® reactor and believes that the technology is now ready for application in a commercial-sized demonstration plant. Research on a Secure P® has included large scale test rigs,[8] with pressures up to 2.5 MPa, temperatures to 225°C, and fuel test bundles with heating capabilities to 2.5 MW. The test rig date has confirmed earlier computer models on the performance of the PIUS concept. Joint studies between various utilities (South Korea, China, and Italy) and ABB-Atom are currently under way.

2. A smaller research program[9-11] on the PIUS concept is located at the University of Tokyo. The studies emphasis a modular PIUS-type reactor called the Intrinsically Safe and Economical Reactor (ISER). The PIUS principles are used, but a steel (rather than concrete) pressure vessel is used. The long-term objectives are to develop a modular, barge-sited, nuclear power plant. The smaller pressure vessel limits cold borated water supplies in the vessel to one or two days of emergency cooling. The program is receiving some industrial support from Japanese reactor vendors.

Advantages: Totally passive safety systems for PWRs.

Additional Requirements: None

Comments: In the PIUS concept, all fluid lines into and out of the pressure vessel have siphon breakers which prevent siphoning of water from the vessel.

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Update/Compiler: Nov. 1988/CWF

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)

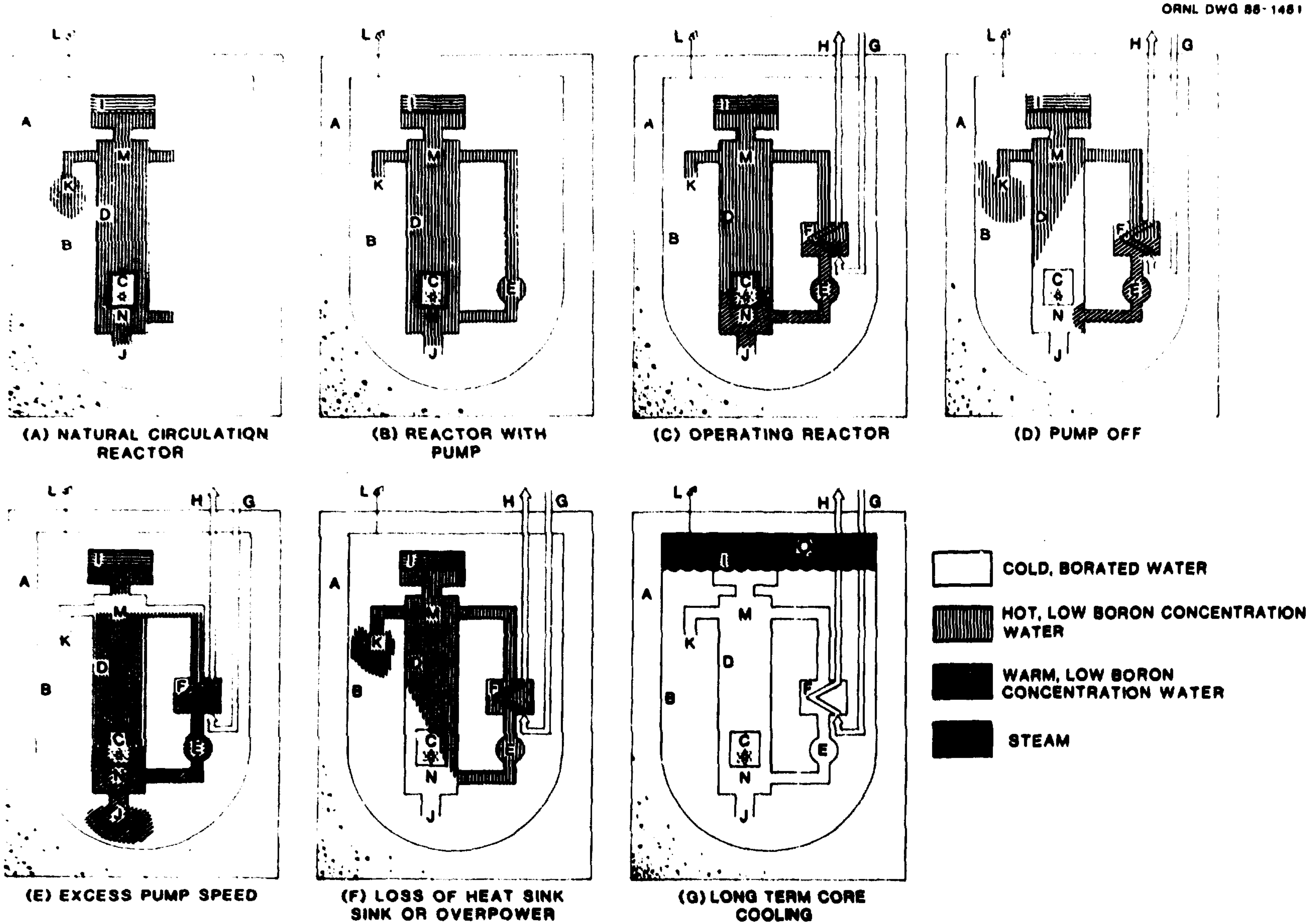

Fig. 1. Operating principles of PIUS.

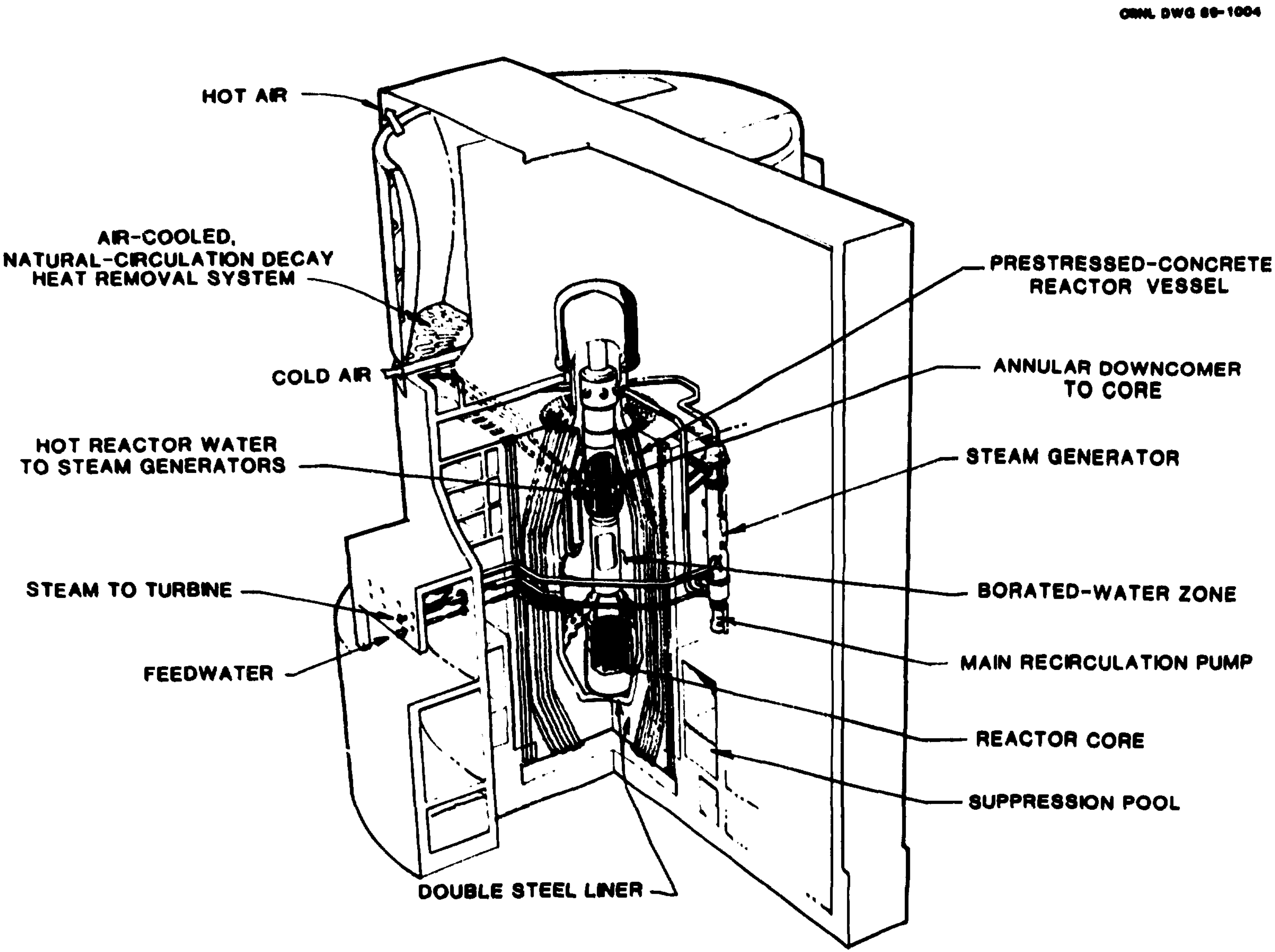

Fig. 2. Proposed PIUS design by ABB Atom.

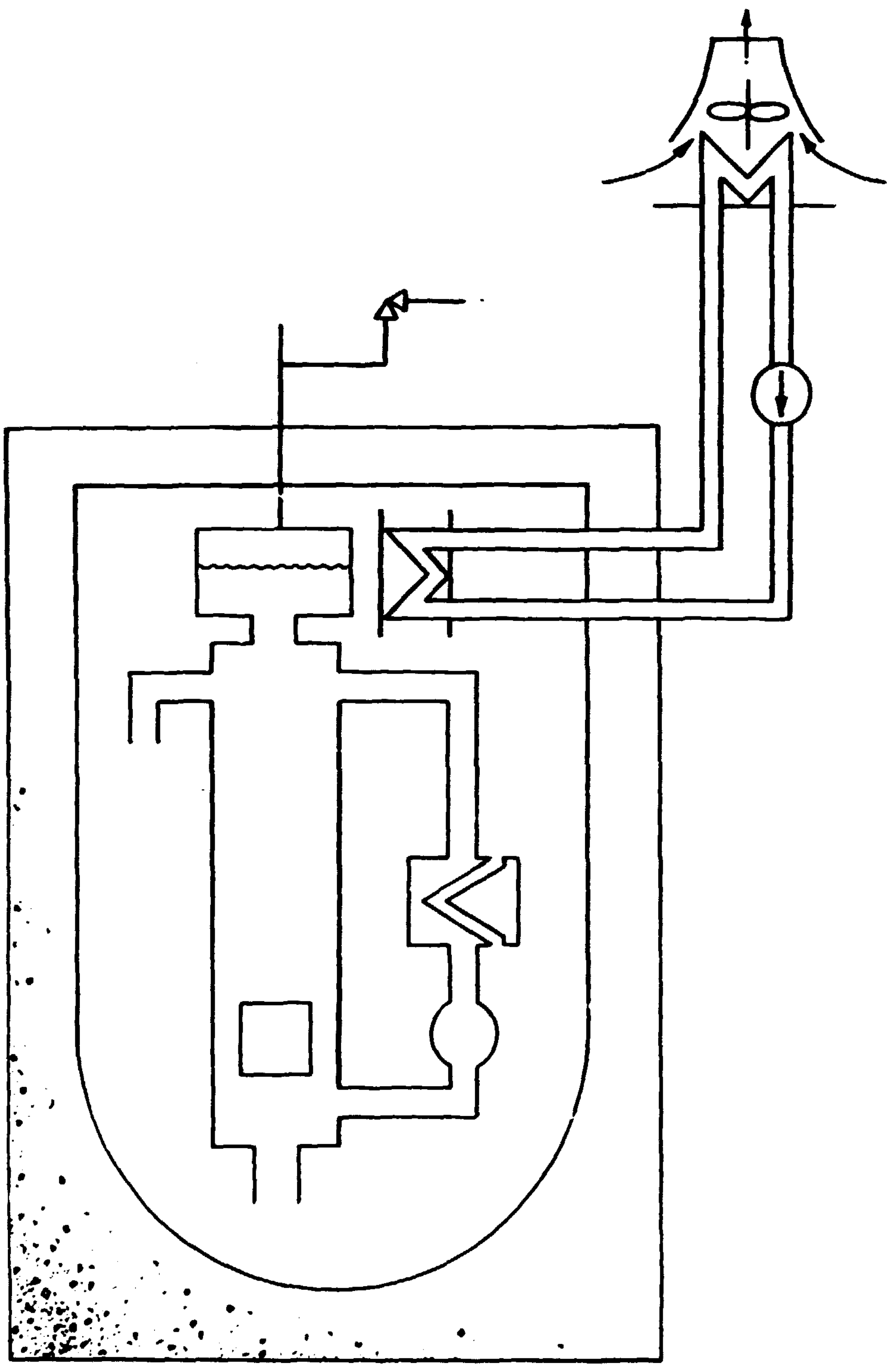

Fig. 3. Long-term cooling by forced or natural circulation air cooler.

TITLE: PASSIVELY SAFE PRESSURIZED-WATER REACTOR WITH GAS BUBBLE

Functional Requirements: 1.1 Control Core Heat Generation Rate (Reactivity Control)
1.2 Remove Core Heat

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Pressurized-water reactor (PWR)

Organization: Vendor (Asea Brown Boveri)

Examples of Implementation: None

Description: This reactor concept was originally developed as a low-temperature (< 200°C), low-pressure, district heating reactor, but can be used for higher temperature and pressure power reactors. The reactor vessel is contained within a large pressure vessel filled with cold, highly borated water (Fig. 1). The water in the primary circuit in the reactor vessel and in the heat exchanger has a relatively low boron concentration and, during normal operation, reactivity is controlled by the addition of either pure water or borated water to the primary circuit as needed. Normal reactor shutdown is accomplished by the injection of boric acid into the core (Fig. 2a). Under emergency conditions, the reactor is designed to shut down automatically through the inflow of cold, highly borated pool water into the core (Fig. 2b, 2c). Subsequently, natural circulation is established between the reactor vessel and the pressure vessel pool water to remove residual heat from the core (Fig. 2d).

The reactor can be cooled two ways:

1. Water can enter below the reactor core from the pressure vessel pool (A), flow through the reactor core, and exit back to the pool (B). Since the pressure vessel pool contains highly borated water, this cooling mode automatically results in reactor shutdown and decay heat removal.

2. Water from the recirculation pump can enter the reactor below the reactor core, go through the core, and be circulated back to the heat exchangers.

Both water flow paths are open with no shutoff valves. The hydraulic performance of the pump determines whether a hydraulic balance is maintained to keep cold, borated pool water out of the reactor core and allow operation of the reactor. The hydraulic balance requires proper operation of the recirculation pump and normal operating conditions in the reactor core. Abnormal conditions allow cold borated water into the reactor core.

The reactor uses a gas (e.g., nitrogen or steam) bubble above and below the reactor core to isolate the low-boron primary cooling water from the high-boron pressure vessel pool water during normal reactor operation. The upper part of the reactor vessel is a long pipe that is filled with gas, the volume of which is approximately equal to the volume of water normally in the reactor vessel. The pipe is open at the top and the upper end is enclosed by a hood so that a gas trap is formed, creating an upper gas-water boundary. During reactor startup, the main circulating pump is started at the same time gas is delivered to the upper bubble. The bubble pushes the water level in the pressure vessel down to a constriction above the core, where it remains during normal operation. The constriction contains the lower gas-water boundary. At the bottom of the reactor vessel there is also a gas trap, but the height of the bubble is negligible compared to that of the upper one.

Normal reactor operation occurs only when the flow of water pumped through the reactor core is such that the friction pressure drop across the reactor core equals the difference in the hydrostatic pressure head from A to B of cold, borated pressure vessel water minus that of the hot reactor water and gas column. If the flow of water from the pump stops, water inflow from the pool collapses the lower gas bubble, enters the core, and shuts down the reactor. It then enters the upper gas bubble and pushes that gas out of the reactor vessel into pool water. Even a relatively small reduction in the pumping rate (e.g., 20%) can produce shutdown if the boron concentration in the pool water is sufficiently high; lower boron concentrations merely reduce the reactor power. Since the reactor vessel has a relatively small cross section in the constriction where the lower gas-water boundary is maintained, small variations in the pumping rate can occur without causing a power reduction or shutdown.

The reactor also shuts down if the core temperature becomes too high. This is accomplished by the venturi in the exit line from the reactor core to the heat exchanger. In a venturi flow constriction, the moving fluid lowers the fluid pressure at the throat of the venturi. During normal operation, the reactor water is only in the liquid phase, despite the lower pressure in the venturi constriction. Pressure increase occurs in the recovery section of the venturi tube, so the resulting loss of pressure is insignificant. However, if the temperature of the reactor cooling water leaving the core becomes too high, steam is formed in the high-speed (throat) section of the venturi tube. The creation of a steam-water mixture in the venturi greatly increases the pressure drop of the fluids through the venturi. The pressure loss in the circuit produces cavitation and a reduction in flow from the pump, shutting down the reactor by allowing cold borated water to flow into the reactor core.

Alternative Versions: Many

Status of Technology: This technology is being considered for a district heating reactor.

Advantages: Passive reactor shutdown occurs in response to low reactor cooling water level or elevated core temperature.

Additional Requirements: None

Comments: This concept was originally developed for district heating but could be applied for power production. The concept was the predecessor of the PIUS reactor and has some of the features of the PIUS reactor.

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Update/Compiler: April 1989/EBL

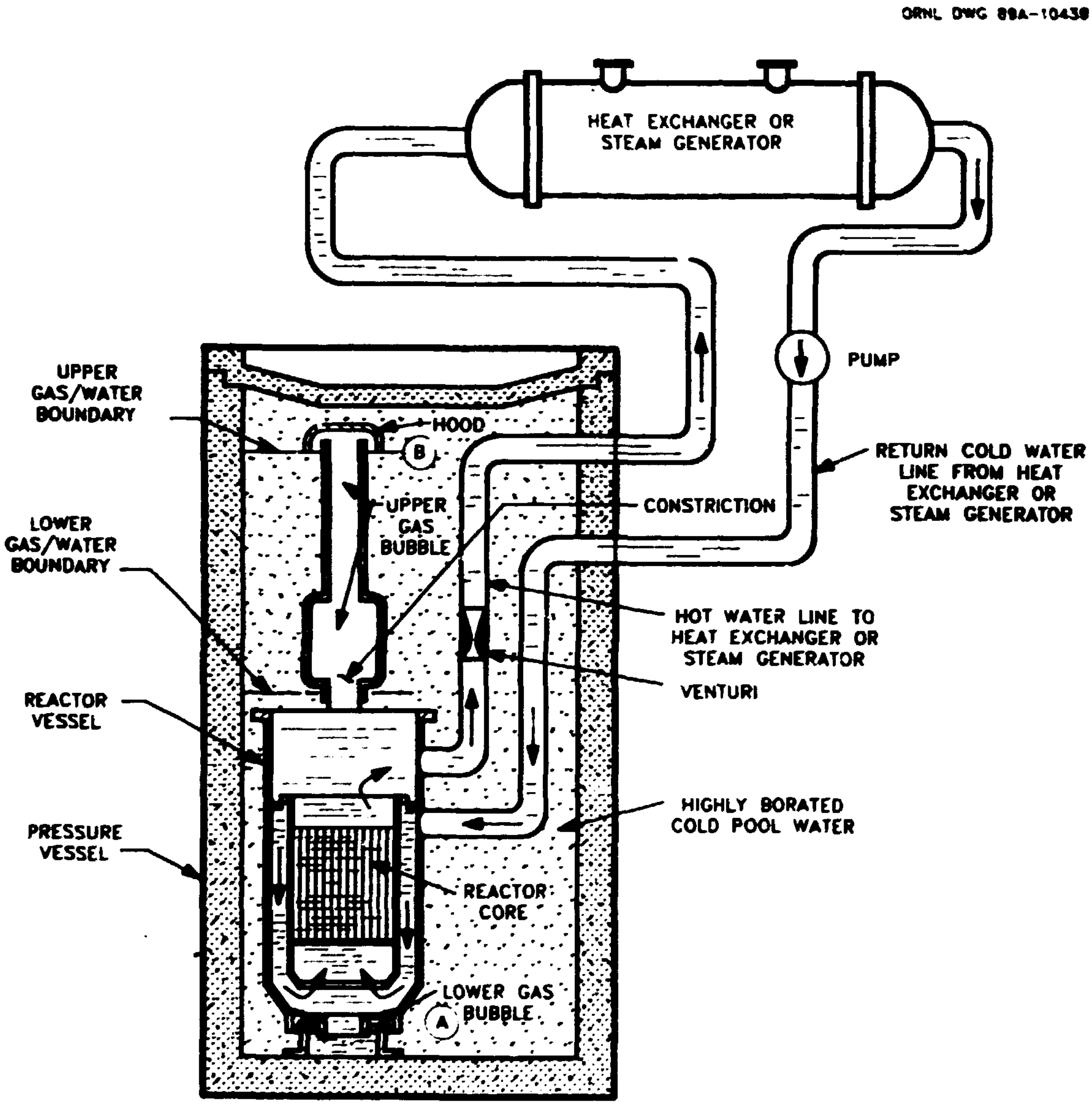

Fig. 1. Schematic of reactor system.

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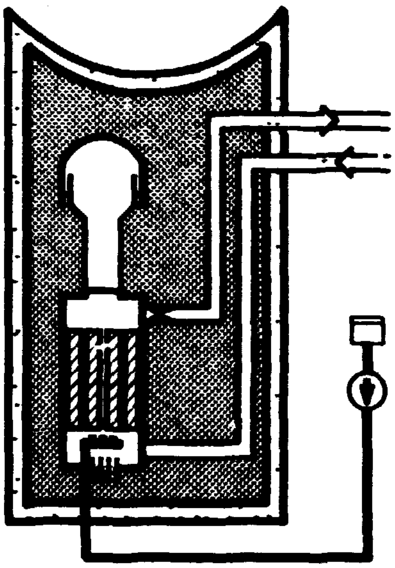
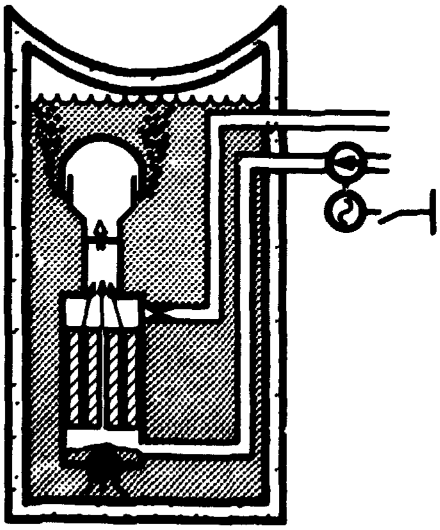
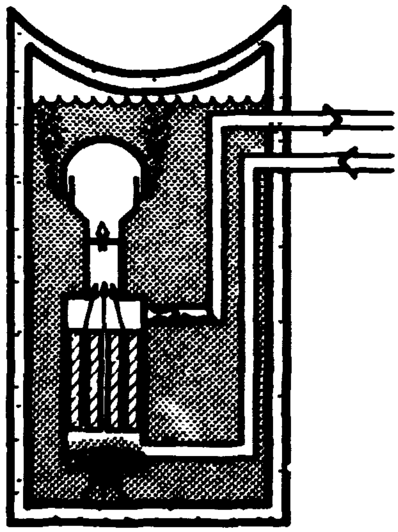


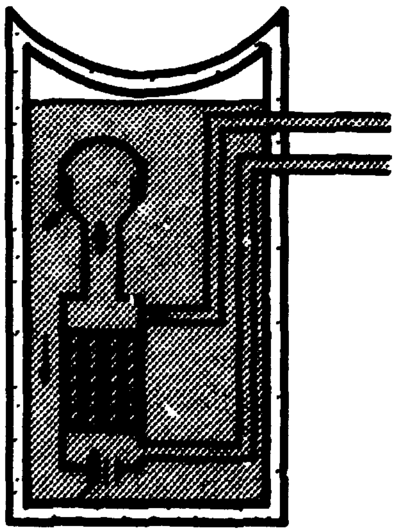
(a) NORMAL SHUTDOWN BY BORIC ACID INJECTION



(b) FAST SHUTDOWN BY MAIN CIRCULATION PUMP TRIP OR INSUFFICIENT COOLING WATER



(c) INHERENT SHUTDOWN BY VENTURI TUBE CAVITATION RESULTING FROM OVERLY HIGH CORE TEMPERATURE



(d) LONG-TERM SHUTDOWN COOLING BY BORATED POOL WATER

Fig. 2. Shutdown mechanisms for PWR with gas bubble.

TITLE: PASSIVE CONTROL OF POWER LEVELS BY VARIABLE BORON CONCENTRATION IN PRESSURIZED-WATER REACTORS (GEYSER)

Functional Requirements: 1.1 Control Core Heat Generation Rate (Reactivity Control)
1.2 Remove Core Heat

Safety Type: Passive

Developmental Status: 5 Concept (for power reactors)

Reactor Type: Pressurized-water reactor (PWR)

Organization: National Laboratory (Swiss Institute for Nuclear Research)

Examples of Implementation: None

Description: The GEYSER concept uses an entirely passive reactor system in which all safety-related and power control processes are governed by simple hydrostatic and thermohydraulic processes. A unique feature of this reactor concept is that the thermal hydraulics of the system automatically adjust the reactor power level to meet heat demand. GEYSER is being developed as a district heating reactor but may be possible to scale the concept in temperature and pressure to power reactor conditions.

The reactor vessel is contained in a large pool of cold, highly borated water (Fig. 1). Two reactor loops are used. The large primary loop moves heat from the reactor core, while the second loop is a small control loop. The primary cooling loop contains low-boron water that circulates up through the reactor core into an annular riser, where, as the water rises, the decrease in pressure allows some of the water to flash to steam. Steam from the conduit rises through the steam space to a condenser which, in turn, gives up useful heat to a heat exchanger outside the pressure vessel. The condenser is located in a bell that is open at the bottom. Under normal conditions, the pressure of the steam keeps the main steam bell free of pool water.

The water from the annular riser and the condensate from the condenser collect in a tray underneath the condenser. The tray is located above the normal pool water level and contains a pipe to return the coolant to the plenum below the reactor core. The difference in specific gravity of the water and steam mixture in the conduit and the water in the return pipe generates the drive for natural circulation of the coolant. If there is too much water in the primary cooling loop, the excess overflows the tray and returns to the pool.

The control loop, which is open to the pool water, has a riser pipe adjacent to the annular conduit of the primary cooling circuit (Fig. 1, Detail A). Above the reactor core, the primary cooling water from the core in the annular riser and the pool water in the control loop pass through a heat exchanger where some heat transfers from the primary coolant water in the riser to the highly borated water in the control loop. Steam formed in the control loop riser passes to the steam space to the condenser, while the water in the control loop riser returns to the pool.

Since the condensate of the steam generated from the pool water is boron-free, while the overflow from the tray contains boron at the mean concentration present in the primary cooling loop, the boron content of the primary cooling loop continually decreases. This results in a rise in reactivity and reactor power. Because steam delivery occurs continuously, the reactor power increases continuously until the quantity of steam produced exceeds the capacity of the condenser. There are two methods that limit power: the first is for normal operation, while the second is an emergency shutdown mechanism.

Method 1: Excess steam production lowers the pool water level in the main steam bell until a float drops enough to open a valve that allows highly borated pool water to flow through a pipe to the reactor core. Then, the increased boron content that results in the primary cooling loop lowers the reactivity of the core. In stationary operation, the supply of steam and the supply of pool water to the primary cooling loop remain in balance. Changes in power demand adjust the steam supplied to the condenser or the quantity of water supplied from the pool until the power that is produced has adapted to the demand.

Method 2: The reactor is designed to passively shut down when steam production greatly exceeds the capacity of the condenser to handle it or if there is a failure of the float mechanism. A steam siphon pipe connects the pool cooler with the steam bell containing the condenser. The siphon pipe contains a water-filled elbow that is submerged at pool water levels in the main steam bell maintained during normal reactor operation. A separate pipe connects the plenum around the reactor core with the pool water. This pipe contains a ribbed elbow located above the upper water level that is reached during normal reactor operation in the main steam bell. The heat of the steam space maintains a steam cushion in the elbow, preventing the pool water from entering the reactor vessel.

Gross overproduction of steam results in the pool water level falling below its normal lower limit, exposing the elbow of the pipe connecting the pool cooler and the main steam bell. The water in the elbow changes to steam, opening the way for steam to escape from the main steam bell and flow to the pool cooler. This creates a pressure drop in the main steam bell, allowing the pool water to rise above the collection tray and the borated pool water to enter the return pipe to the primary cooling loop. At the same time, the rising pool water submerges the elbow in the pipe connecting the reactor plenum to the pool, condensing the steam cushion in the elbow and allowing pool water to flow into the reactor core and shut it down. Steam produced by residual core heat condenses in the pool cooler until the water rising in the main steam bell submerges the inlet opening to the pool cooler. Long-term cooling for decay heat removal is provided by natural circulation of the large volume of pool water through the reactor.

During reactor startup, electric heaters next to the heat exchanger in the control loop provide the energy to heat the water in the primary cooling loop and start its circulation. The heaters maintain steam generation in the loop until the concentration of boron in the cooling water becomes low enough for the reactor to begin producing heat.

The primary cooling loop is thermally insulated from the cold pool water. To prevent any heat that does escape from excessively heating the pool water, a water cooler is located near the top of the pool. The cooler is composed of a cooling coil contained in a bell that is open at the bottom. During normal reactor operation the bell is filled with pool water. Warm water in the pool rises and is cooled by the coil, which carries the heat away to the environment.

Alternative Versions: This design concept is evolving with time.

Status of Technology: A pilot design study for district heating purposes was carried out at the Swiss Institute for Nuclear Research in 1986-87. Thermohydraulic calculations proved that stable conditions can be achieved in the entire operational range. A full-sized (32 m in height) model with a simulated power of 0.5 MW demonstrated stable operation, shutdown process, and power control feasibility in a power range of 15 to 100%.

Advantages: 1. No control rods are needed.

2. Simple, unmanned operation with walkaway safety.

3. Human influence is completely excluded from safety-related processes.

Additional Requirements: None

Comments: The GEYSER system was developed for district heating but may be applicable tu power reactors.

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2. G. Vecsey and P. G. Doroszlai, Method and Device fr r Passive Transfer of Heat from Nuclear Reactors to a Public Utility Network, w': Automatic Regulation of Reactor Power and Automatic Emergency Shutdown and Switchover to Emergency Cooling, U.S. Patent 4,783,306 (November 8, 1988).

Update/Compiler: May 1989/EBL

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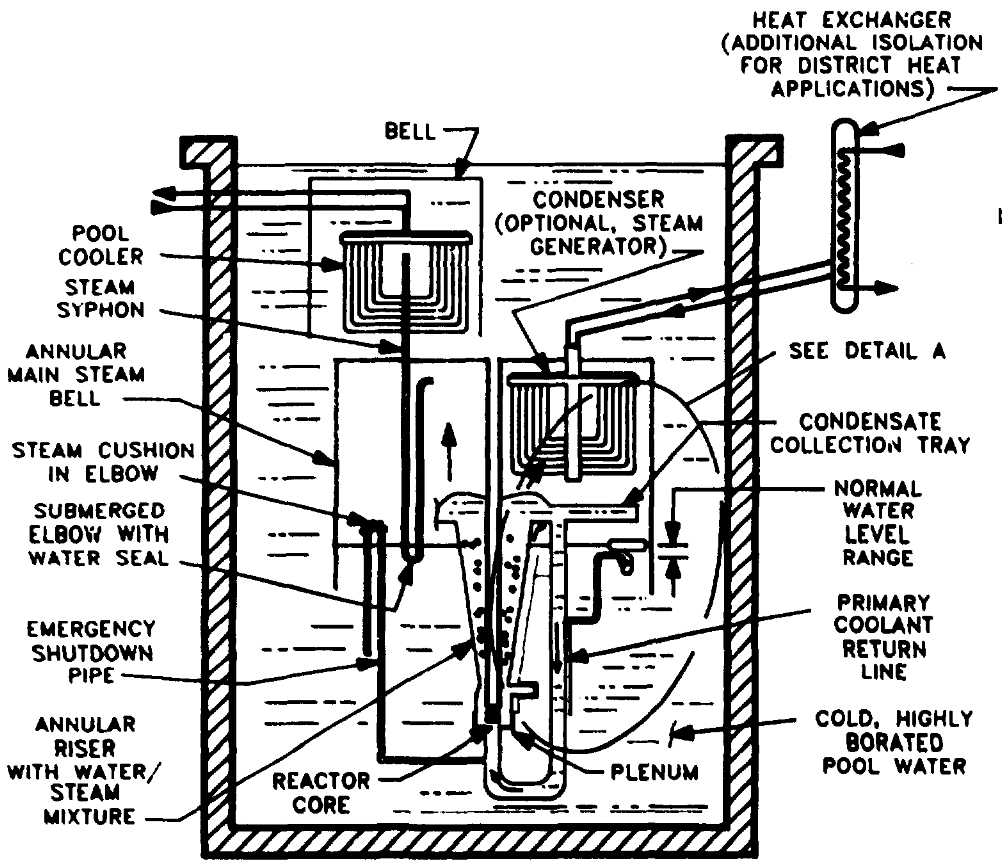

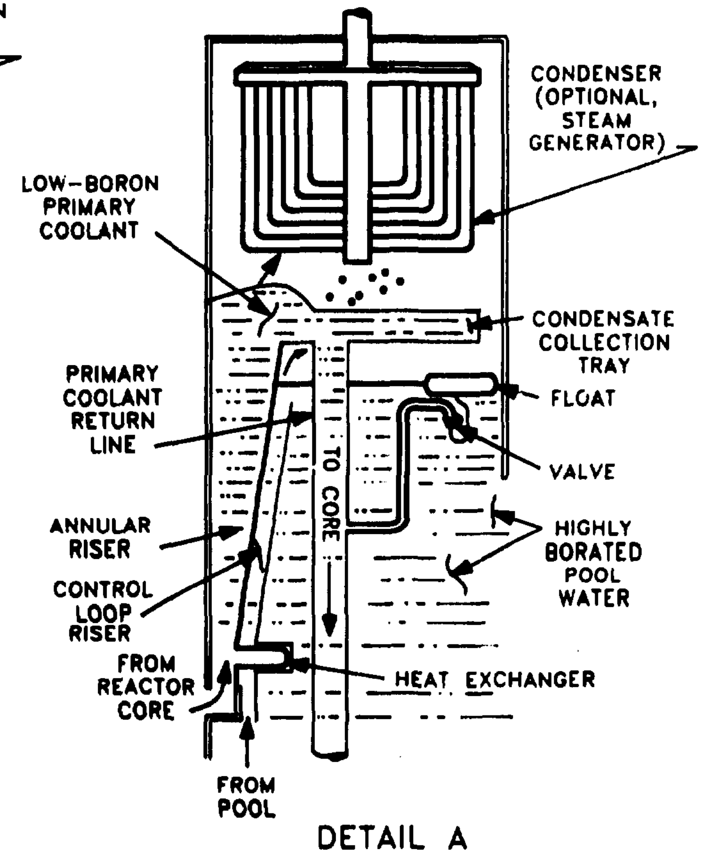

Fig. 1. Geyser reactor design (not to scale/simplified).

TITLE: USE OF HYDRIDES WHICH REVERSIBLY ABSORB AND DESORB HYDROGEN TO CONTROL REACTIVITY IN LWRs

Functional Requirements: Level 1.1: Control Core Heat Generation Rate (Reactivity Control)

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Light-Water Reactor (LWR)

Organization: Industrial (Multiple)

Examples of Implementation: None

Description: Several passive concepts using metal hydrides have been proposed to limit the reactivity, and hence, the power, of LWRs under various operating and accident conditions. These are based on the use of metal hydrides which reversibly absorb and desorb hydrogen depending upon reactor temperatures. The original concepts were developed for passive control of special purpose reactors.[1,2] The description herein is how the technology could be applied to LWRs.

These reactor reactivity control systems are based on the reactor physics of thermal neutron reactors, the chemistry of compounds which reversibly give off or take in hydrogen gas, and a special mechanical container in which the above compounds are located.

Reactor Physics

In a thermal neutron reactor, such as an LWR, a moderator is used to slow down neutrons so that they will react with fissile material (fuel). In water-cooled reactors, the moderator is hydrogen in the chemical form of water. If some of the "effective moderator" is removed from part of the core of a thermal neutron reactor during operation, the reactivity and, therefore, the power level of that part of the reactor core are reduced. In standard commercial LWRs, when hydrogen is removed, proportionally more neutrons are absorbed by ^{238}U and structural materials, and more neutrons escape the reactor core, resulting in fewer neutrons fissioning ^{235}U or other fissile materials.

Chemistry

Certain metals and their alloys (such as zirconium, magnesium, nickel, and titanium) will react with hydrogen to form hydrides.[3-6] The hydrogen concentration in these compounds is similar to water. When heated, these same materials will reversibly release hydrogen.

The absorption, depending upon the particular system, may vary from a surface adsorption effect to a strong chemical bond. With proper choice of chemical compound, the temperature of hydrogen release and absorption can be chosen. For example, magnesium and magnesium alloy hydride systems absorb or desorb near the operating temperatures of LWRs. A metal hydride system absorbs or releases hydrogen in a particular temperature range given a particular pressure.

Use of Hydrides

Two sets of applications can be identified for hydrides: (1) a passive technique to control reactor power levels and (2) a replacement for conventional control rods.

1. Passive Power Level Control

Metal hydrides can be used to limit power levels in LWRs. For this application, the external shape of the hydride device would be similar to an oversized fuel pin in a fuel assembly. The device would be somewhat longer than the fuel pins and have a greater diameter. An external perspective of such a device inside a boiling-water reactor (BWR) assembly and a pressurized-water reactor (PWR) assembly is shown in Fig. 1. The BWR assembly is a Kraftwork Union design (U.S. subsidiary: Advanced Nuclear Fuels) where the fuel assembly normally contains a "square hole" in the middle. This square hole in the fuel assembly during reactor operation is filled with water to moderate neutrons. The device would be placed into this hole. The PWR assembly is a Combustion Engineering (C. E.) Design which normally has five holes in it for control rods. When the reactor is operating, the control rods are normally out of the reactor core and the holes are full of water, which moderates the neutrons. In a typical C. E. design, only a few assemblies have holes for control rods. With this device, fuel assembly locations without control rods would contain hydride rods.

Each hydride rod (as shown in Fig. 2) is filled with a metal hydride mixture with a significant fraction of the metal in hydride form. The metal hydride fills the tube from top to bottom, but has space to allow hydrogen to flow up or down the length of the tube. The tube may have an insulation barrier so that most of the temperature drop from outside to inside the tube is across the insulation. In cold conditions, the hydride will form a uniform composition along the full length of the rod. This occurs because the chemistry (Gibbs free energy) of hydride formation prefers the same concentration of hydrogen throughout the metal hydride.

In a reactor at power, the neutron/gamma flux will heat the hydride tube. Heat will be conducted out of the tube to the coolant. In the reactor core, the highest power levels from fissioning create the highest neutron and gamma ray fluxes. The sections of hydride rod in these locations will absorb the most heat and will be the hottest. At higher temperatures, the hot metal hydride will release hydrogen, increasing hydrogen pressure in the tube. The hydrogen will flow to the cooler sections of the tube and form hydrides. The loss of hydrogen in hot spot locations is also a loss of moderator. The loss of moderator implies loss of reactivity and loss of power in the hottest reactor core. If the entire reactor is too hot, much of the hydrogen will leave the core and be just outside the reactor core in the parts of the hydride tube extending beyond the fuel pins. This limits total power levels.

An alternative design of hydride tube is shown in Fig. 3. Many water-cooled reactors have an integral grid structure into which individual or groups of fuel assemblies are placed. With this option, the grid structure is composed of hydride rods (or squares or rectangles) perpendicular to the fuel assemblies and parallel to the horizon (Fig. 3). With this orientation of hydride rods across multiple fuel assemblies, the hydride rods would act to level power across the reactor core. This levels the power output between fuel assemblies, and the leveling process would be expected to occur down to relatively low power levels.

2. Hydride Control Rods

Hydrides in tube form may also be used as control rods by connecting to external control devices. To increase power, the hydrogen pressure in the tube would be increased, thereby increasing hydrogen absorption. To decrease power, hydrogen would be removed from the tube via vacuum. Theoretical and experimental work to evaluate this option has been done for special-purpose, nonwater-cooled reactors.(7,8) Two types of external control systems may be used:

1 Tubes from the hydride control rods may be attached to external hydrogen and vacuum tanks with appropriated valving.

2. Tubes from the hydride control rods may be attached to cylinders filled with hydrides and incorporating heaters. To add hydrogen to control rods, the external hydride tanks are heated releasing hydrogen.

Such rods would have two unique characteristics compared to conventional control rods.

a. If reactor power is excessive, the hydride will decompose and release the hydrogen gas at very high pressure forcing hydrogen out of the hydride control rods. The control rod activates (shuts down reactor) when the core overheats, regardless of control room conditions or commands.

b. Hydride control rods self adjust power levels in the reactor. Control rods will increase neutron moderation in sections of the reactor core at low power and decrease neutron moderation in sections of the reactor core at high power.

<u>Alternative Versions</u>: For stronger coupling of hydride temperatures to reactor temperature, there is the option to incorporate nuclear fuel into the hydride tubes or place an annular hydride tube around a fuel pin. In either case, hydride temperatures are directly coupled to fuel pin temperatures.

<u>Status of Technology</u>: Only limited work has been done on the use of hydrides to limit or control reactor power levels. Most of this work was done in the 1960s for special purpose reactors. The base technology for such devices has expanded greatly in the last 20 years due to two other national programs.

1. In the 1970's, a major R&D effort was initiated to develop methods to store hydrogen.[4-6] Development of hydrides for hydrogen storage was a major component of this program. The knowledge of thermodynamics, kinetics, and manufacturing techniques for hydride devices has expanded greatly.

2. For very high temperature gas-cooled reactors and fusion reactors, there has been a concern about diffusion of tritium out of the reactors to the environment. Major R&D efforts[9] have been initiated to develop technologies to contain hydrogen gas in metal equipment in high temperature radiation fields. This technology is identical to that required for use in LWRs to prevent hydrogen from diffusing from hydride to coolant through tube walls.

Advantages: This use of hydrides is a passive technique to control LWR reactivity. Multiple potential applications may exist.

1. Hydride devices could be used to eliminate various nuclear-thermal-hydraulic instabilities in power reactors (such as the xenon oscillation effect).

2. The technology may be applicable to development of more economic high burnup fuels. In recent years, there has been a major emphasis to increase fuel burnup to (1) reduce fuel costs, (2) minimize reactor refueling time, and (3) minimize the cost of spent fuel storage and disposal by minimizing the tons of spent fuel produced. If a fuel assembly is to provide more energy, the fissile enrichment must be made higher so there is fissile fuel to produce energy over an extended period of time. Simultaneously, the higher enrichment of fuel increases the possibility of excess power output from the fuel assembly. This problem is currently solved using control rods and burnable absorbers to minimize power peaking. Hydride devices may partially eliminate the use of burnable absorbers in order to save on fuel costs or to create the potential of further extension of fuel burnup.

3. There is one particularly unique characteristic of hydrides as control rods. The rods automatically shut down the reactor if the reactor temperature is excessive because the control rod is sensitive to core conditions.

Additional Requirements: None

Comments: The base technologies required for hydride technologies have progressed rapidly in the last 20 years. There have been no power reactor studies of hydrides for several decades; hence, their potential is not well understood.

References/Contacts:

1. R. Magladry, *Hydrogen Diffusion Reactor Control*, U.S. Patent 3,351,534 (November 7, 1967).

2. J. L. Anderson, W. Mayo, and E. Lantz, *Reactivity Control of Fast-Spectrum Reactors by Reversible Hydriding of Yttrium Zones*, NASA TN D-4615, (June, 1968).

3. J. J. Reilly and R. H. Wiswall, "The Reaction of Hydrogen with Alloys of Magnesium and Copper," *Inorganic Chemistry*, 6, 2220 (1967).

4. J. J. Reilly, "Metal Hydride Technology," *Zeitschrift fur Physikalische Chemic Neue Folge*, 117, 655 (1979).

5. J. J. Reilly, R. H. Wiswall, and C. H. Waide, *Motor Vehicle Storage of Hydrogen Using Metal Hydrides*, TEC-75/001, (October, 1974).

6. J. O'M. Bockris, *Energy Options: Real Economics and the Solar-Hydrogen System*, J. Wiley & Sons, New York.

7. G. F. Zindler, *Feasibility Study of Terrestrial Thermoelectric Nuclear Power Plant*, AFWL-TR-66-16, June, 1966.

8. G. F. Zindler, *Terrestrial Direct Conversion Nuclear Power Plant Conceptual Design*, WL-TR-64-168, (March, 1965).

9. *Nuclear Technology* (entire issue), 66, No. 3 (Sept. 1984).

Update/Compiler: Dec. 1988/CWF

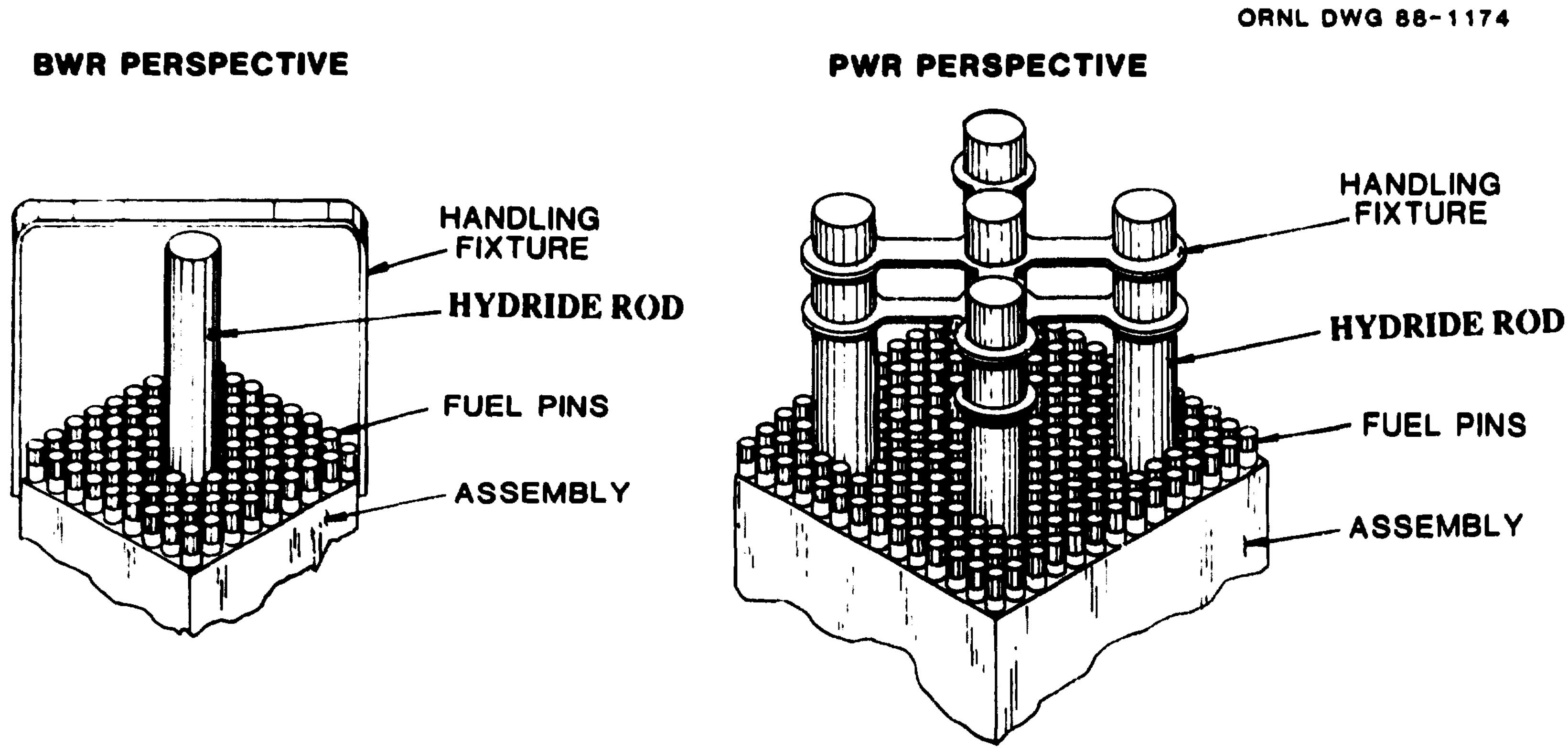

Fig. 1. Options to incorporate hydride tubes into LWR fuel assemblies (top of fuel assembly perspective).

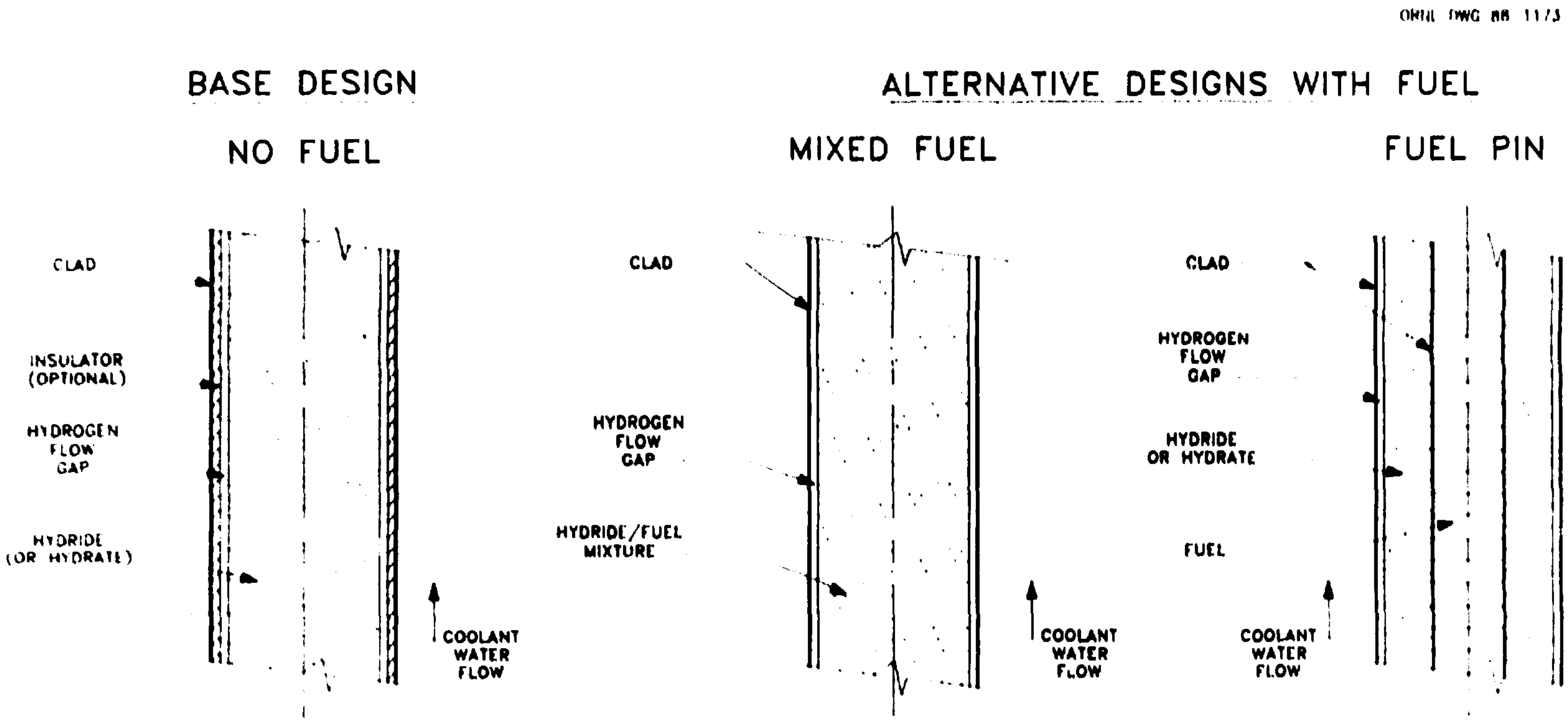

Fig. 2. Cross section of various hydride design options.

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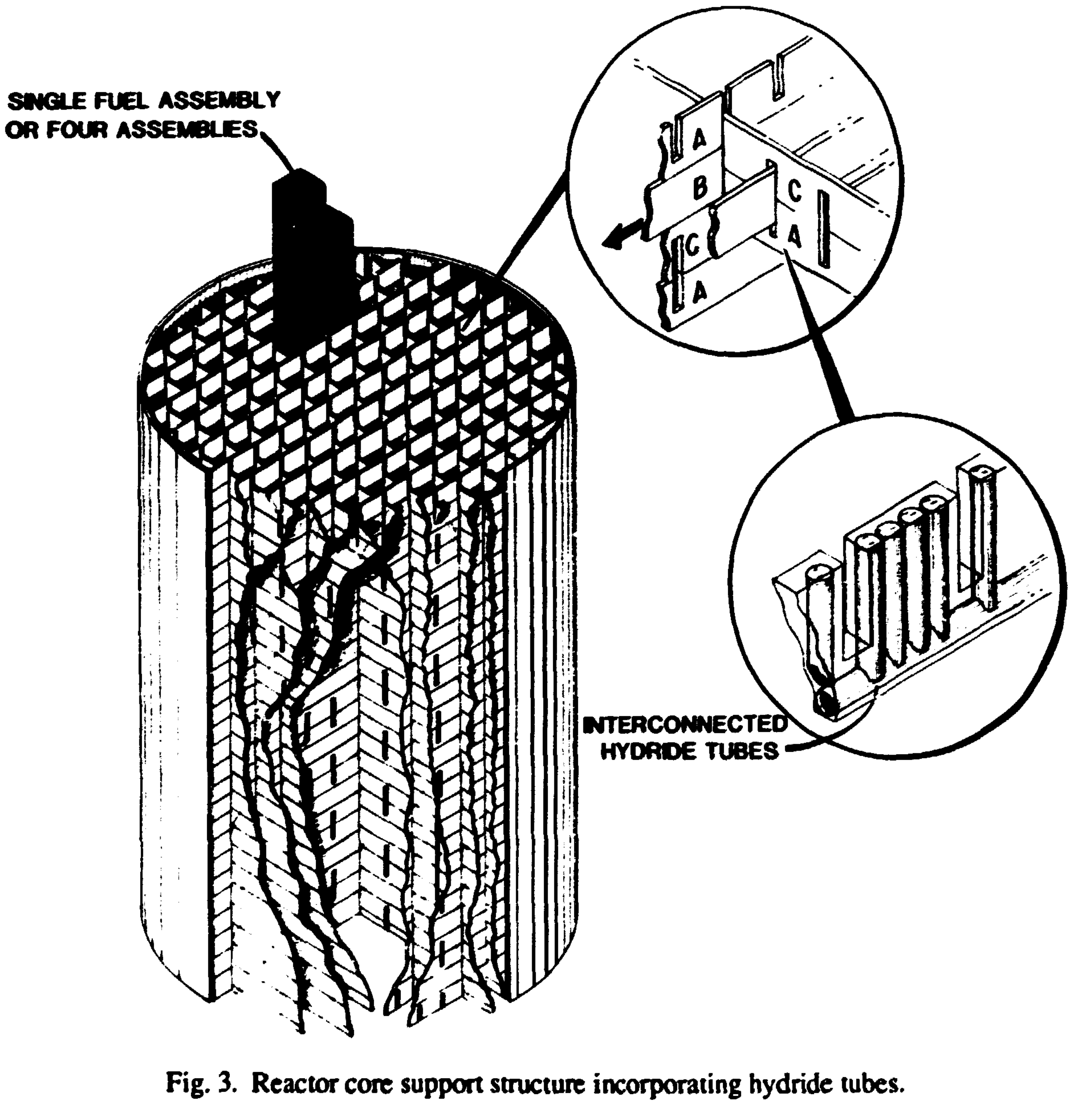


Fig. 3. Reactor core support structure incorporating hydride tubes.

TITLE: HYDRAULIC CONTROL ROD SYSTEM WITH PASSIVE SHUTDOWN MECHANISMS FOR WHOLE CORE DISTURBANCES (LOW WATER LEVEL, RAPID PRESSURE CHANGES) IN A BOILING-WATER REACTOR

Functional Requirements: Level 1.1.1 Provide Process Shutdown Mechanism

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (Asea Brown Boveri)

Examples of Implementation: None

Description: This is a hydraulically controlled system to shut down the reactor by dropping control rods into the reactor core if certain accident conditions occur. The control rods are contained in vertical chambers that run through and above the reactor core (Fig. 1). Each control rod moves freely in its chamber and has only two stable positions: completely in the core or completely out of and above the core. The hydraulic force to hold the control rods above the core is supplied by a safety control system pump that takes cooling water from the pressure vessel and circulates it through a distribution box that runs beneath the reactor core and connects to the control rod chambers. The chambers are not completely closed at the top, allowing the water from the pump to return to the cooling volume. When the safety control pump is not supplied with enough water to keep the control rods suspended over the reactor core, they fall by gravity into the core and shut down the reactor. The safety control system automatically operates to shut down the reactor in response to a (1) manual shutdown, (2) a low cooling-water level, (3) a rapid increase in vessel pressure, or (4) a rapid decrease in vessel pressure.

1. Manual Shutdown -- In a manual shutdown, the safety control distribution pump is turned off, removing the hydraulic force holding up the control rods, which then fall into the reactor core.

2. Low Water Level -- The safety control pump intake is located at a level in the pressure vessel below which not enough water would be present for safe reactor operation. If the cooling water level in the pressure vessel falls below the safety control pump intake, the safety control pump fails, allowing the control rods to fall into the reactor core.

3. Rapid Pressure Variations -- To provide for reactor shutdown in case rapid pressurization or depressurization of the pressure vessel, the safety control system uses differential pressure valves to control the amount of water reaching the distribution box. This is accomplished through the addition of a short water circuit connected to the pump circuit. The short circuit contains a membrane valve that is controlled by either of two differential pressure valves (Fig. 2). During normal operation, the safety control pump supplies pressure to both sides of the membrane valve, keeping it closed. If one of the differential pressure valves (A or B) opens, the pressure on that side of the membrane is decreased and the valve opens, shorting

the safety control pump circuit so that not enough water reaches the distribution box to keep the control rods out of the core.

One of the short circuit differential pressure valves (Valve B) is connected to a venturi located in the pressure vessel steam outlet; the other one (Valve A) is connected to a steam container located within the reactor steam space. The steam container communicates with the steam space by a flap valve or fluidic diode, and is connected to two water-filled tanks located outside the pressure vessel. One tank contains cold water. Its upper end is connected to the reactor steam space by an open pipe and its lower end is connected to the steam container by a pipe containing a differential pressure valve (Valve D). The other tank contains a heating coil that keeps the water in the upper part of the tank near the boiling point. The water in the lower part of the tank is cold. The upper end of this hot/cold tank is connected to the reactor steam space by a pipe containing a throttle regulator, while the lower end is connected to the steam container by a pipe containing a pressure differential valve (Valve C).

A rapid pressure increase in the pressure vessel, which would result if the steam turbine suddenly stopped, can produce a reactor shutdown in two ways. First, steam is forced from the reactor steam space into the steam container until the flow velocity closes the flap valve (or fluidic valve) that separates them. A difference in pressure then occurs between the steam container and Valve A; Valve A opens with water flowing to the steam container, leading to a short circuit of the pump. Second, the pressure surge also forces steam from the steam space into the cold water tank, increasing the pressure there. If the pressure becomes great enough, Valve D opens and cold water flows to the steam container, condensing the steam and decreasing the pressure. This creates a rapid pressure decrease in the pipe connecting the steam container to Valve A, causing it to open and short circuit the safety control pump.

A rapid pressure decrease in the pressure vessel, which would result from a steam pipe break, also triggers a double shutdown system. First, the increased steam flow through the venturi in the pressure vessel steam outlet causes a pressure decrease in the connecting pipe and opens Valve B, leading to a short circuit of the pump. Second, the quick loss of pressure from the pressure vessel and the steam container creates a difference in pressure between the two of them and the hot/cold water tank. However, the difference in pressure between the steam space and the water tank is not as great due to the throttle regulator and the boiling of the water in the upper half of the tank. The greater difference in pressure between the steam container and the hot/cold water tank opens Valve D, allowing cold water to flow into the steam container, condensing the steam and producing a brief pressure drop. The pressure drop closes the flap valve in the steam container and opens Valve A; this leads a short circuit of the pump.

The above safety control system assures that all control rods will drop into the reactor core under any accident conditions. The system also allows individual operation of each control rod within the constraints of the described system. Raising a single control rod requires the proper operation of this system <u>and</u> a second system. Each control rod chamber is connected not only to the safety control distribution box, but also to a separate operating control pipe that runs through the safety control distribution box and out of the pressure vessel to a separate operating control pump and valve system. The control rod chamber is designed so that when a control rod is in the lowered position, not enough water can enter the chamber from the safety control distribution box to lift the control rod (Fig. 3). Water from the safety control distribution box enters the control rod chamber through the lower openings in the chamber wall, flows down the gap between the chamber wall and the control rod assembly, and up to the control rod through the center openings in the control rod assembly. The narrowness of the gap between the chamber wall and the control rod assembly limits the amount of water that can enter the lower openings and exert force on the control rod.

To lift the control rod, high-pressure water is supplied through the operating control pipe. Once the control rod assembly is raised above the lower openings in the chamber wall, enough water can enter from the distribution box to hold the control rod assembly out of the core, and the flow from the operating control pipe is shut off. A throttle regulator that is between the bottom of the control rod chamber and the pipe leading to the safety control distribution box limits the return of water from the control rod chamber to the safety control distribution box. To lower an individual control rod, a valve on the operating control pipe is opened, allowing water to flow out and reduce the lifting force supplied to the control rod, which drops the control rod into the core. A pressure gauge attached to the operating control pipe can be used to determine whether a control rod is in or out of the core. If the pressure in the operating control pipe is almost equal to the reactor pressure, the control rod is in the lowered position; if the pressure is almost equal to that in the distribution chamber, the control rod is in its upper position.

Alternative Versions: None

Status of Technology: Preliminary studies of the system were conducted by Asea Brown Boveri of Sweden for the Model 85 Boiling-Water Reactor.

Advantages: 1. Automatically shuts down the reactor in response to abnormalities in either water level or pressure level.

2. The shutdown system can be placed in a room completely inaccessible to operating personnel or outsiders during normal reactor operation, increasing security of the facility.

Additional Requirements: None

Comments: None

References/Contacts:

1. K. Hannerz, R. Rytterne, and T. Holm, Kernreaktor des Siedewassertype (Nuclear Reactor of the Boiling Water Type), European Patent No. 31,541 A2, (December 16, 1980): ORNL/TR-89/8 (1989).

Update/Compiler: March 1989/EBL

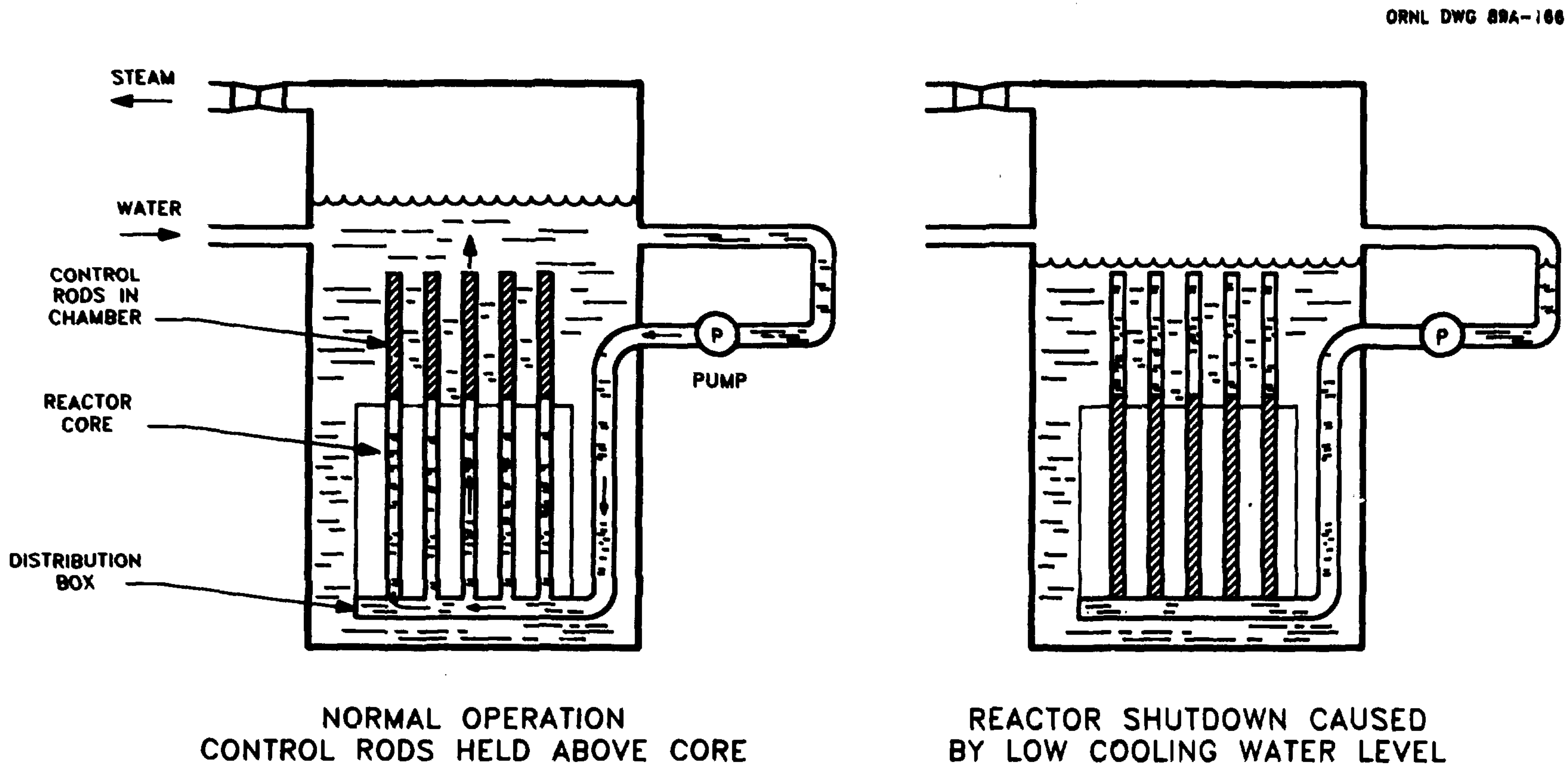


Fig. 1. Hydraulic control rod system for boiling water reactor.

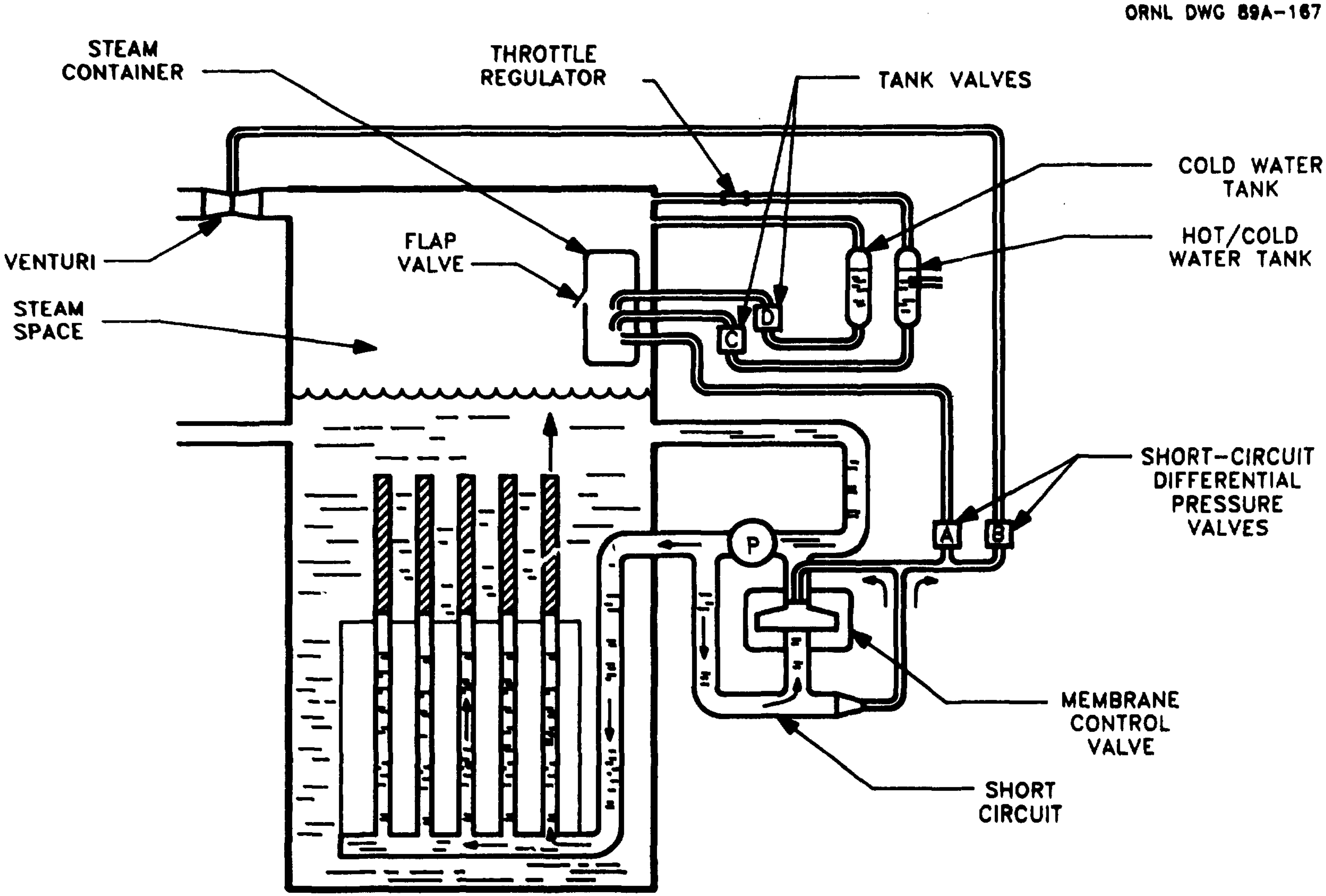


Fig. 2. Passive system for hydraulic operation of control rods.

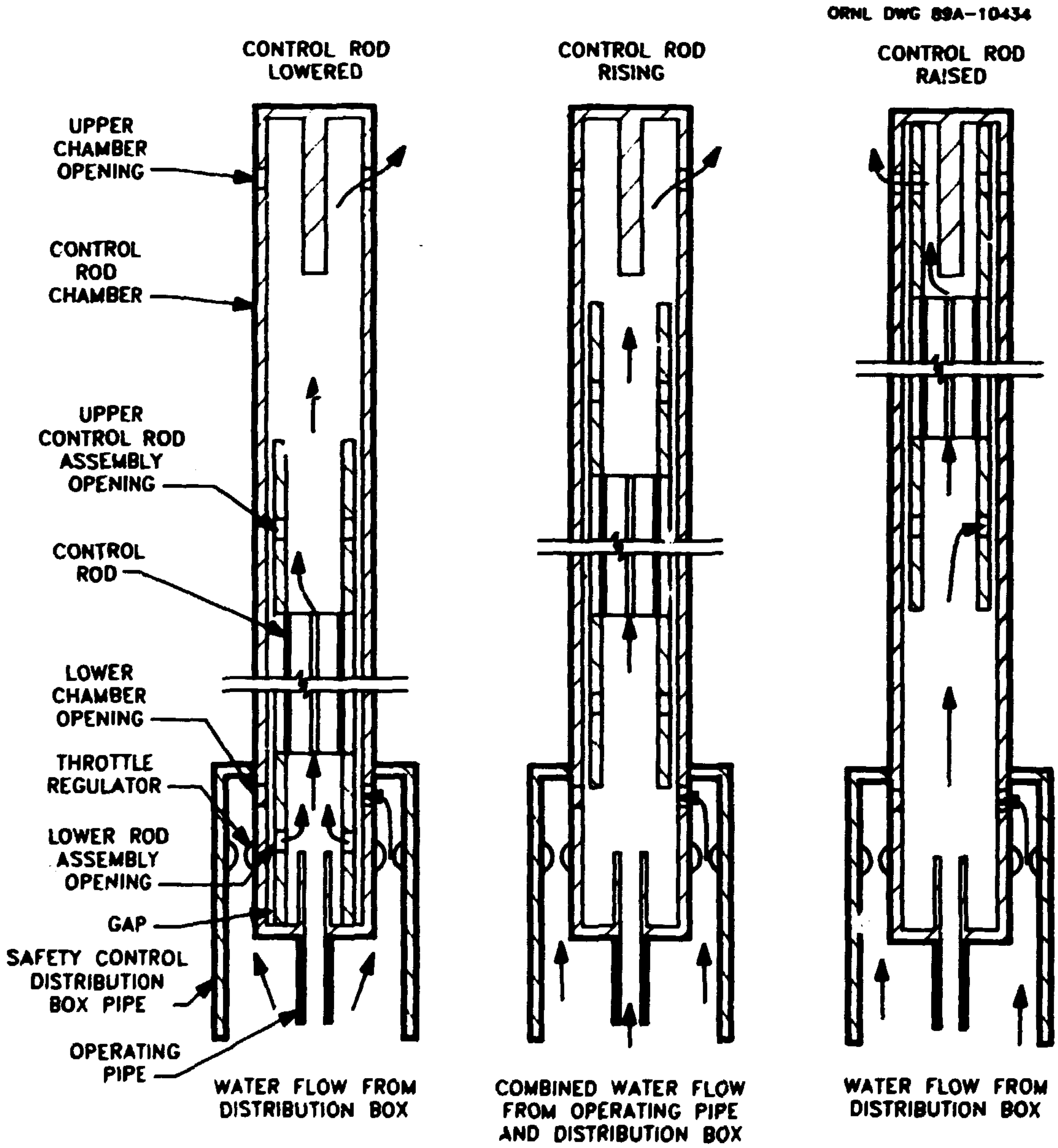


Fig. 3. System to manipulate individual control rods.

TITLE: SLOW-WITHDRAWAL CONTROL RODS

Functional Requirements: 1.1.1 Provide Process Shutdown Mechanism

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (General Electric)

Examples of Implementation: Multiple

Description: This device is one of a family of devices which passively limits the rate at which a control rod may be removed from the reactor core, but allows very rapid movement of the control rod into the core to shut it down. By limiting the maximum rate of control rod withdrawal, it provides time for slower inherent reactivity control mechanisms (such as void reactivity coefficients) to operate. The device described is for a bottom entry control rod for a boiling water reactor (BWR), but is applicable to control rods mounted in any direction.

The velocity control device is composed of a toroidally-shaped control member with a pair of fluid guide vanes positioned underneath (Fig. 1). The control member is coaxially positioned around the control rod drive hub and secured to the hub by radially extending webs. The webs are fitted with rollers at their outer ends to center and guide the hub within the control rod guide tube. The control member is constructed to allow for fluid movement between the member and the inner guide tube surface and between the member and the drive hub. The upper side of the control member has a smooth conical surface that minimizes fluid resistance when the rod moves upward into the reactor core. The underside of the control member contains adjacent inner and outer grooves to increase water resistance when the rod moves downward out of the core.

Positioned below the control member is a pair of inner and outer annular fluid guide vanes, also positioned radially around the control rod drive hub and secured to it by webs. The vanes are angled upward toward the common edge of the two grooves in the underside of the control member. When the control rod moves downward, the vanes guide a stream of fluid at increased velocity against the common edge of the two grooves. This greatly increases turbulence in the flow of the fluid upward through the inner and outer fluid passages, thus increasing resistance to movement of the control rod out of the reactor core.

Alternative Versions: Many related devices have been proposed.

Status of Technology: This is an advanced version of existing devices.

Advantages: Passively decreases drop velocity of control rods being withdrawn from the reactor core.

Added Requirements: None

Comments: None

References/Contacts:
1. J. E. Cearley, J. C. Carruth, R. C. Dixon, S. S. Spencer, and J. A. Zuloaga, Jr., Control Rod Velocity Limiter, U.S. Patent No. 4,624,826 (November 25, 1986).

Update/Compiler: April 1989/EBL

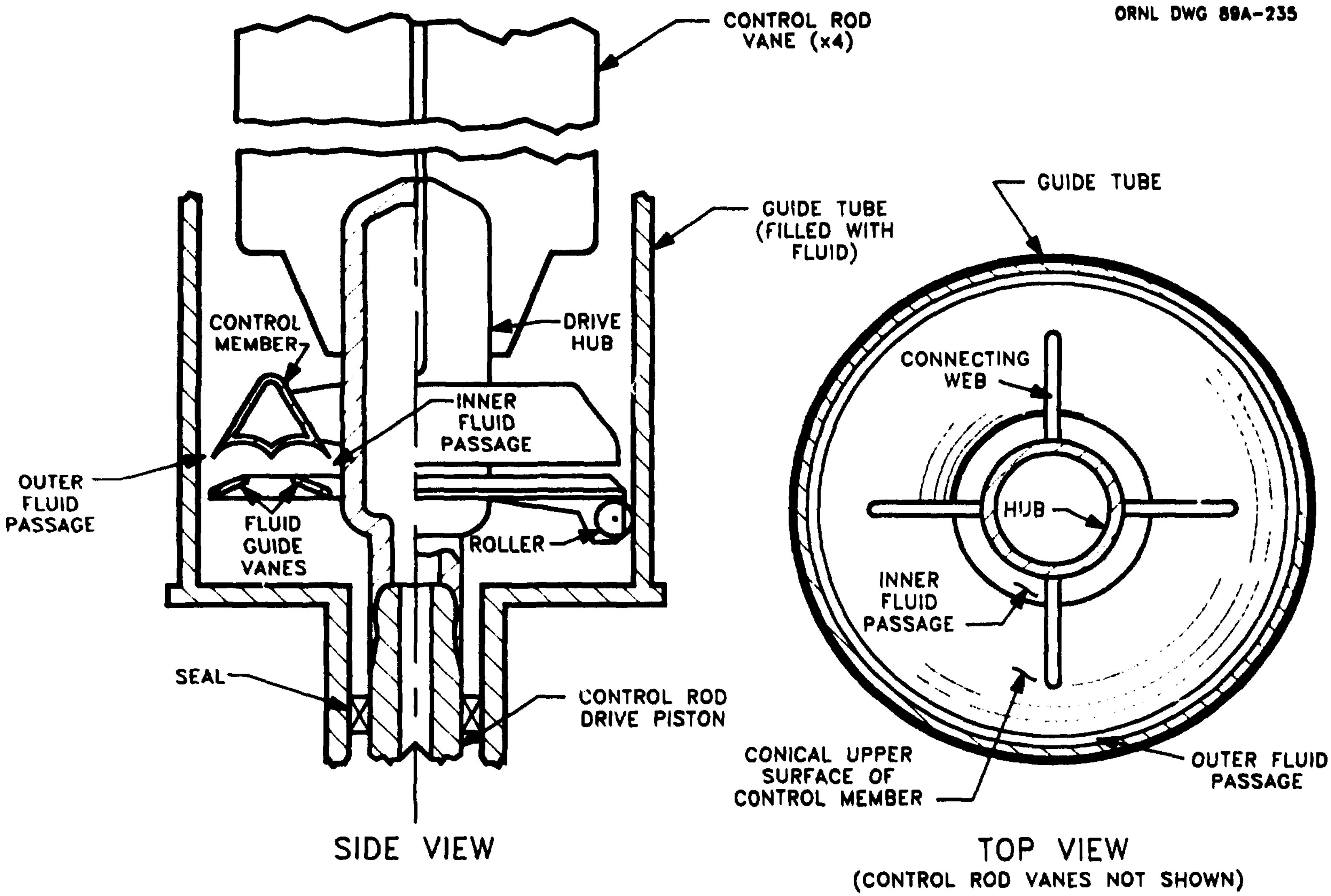


Fig. 1. Device to limit rate of control rod withdrawal.

TITLE: FLUIDIZED-BED CONTROL RODS

Functional Requirements: 1.1.1 Provide Process Shutdown Mechanism

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Pressurized-water reactor (PWR)

Organization: University (University of Florida) and Industrial (General Nuclear Engineering Corporation)

Examples of Implementation: None

Description: Fluidized-bed control rods (FBCRs) are tubes through the reactor core filled with particles of a neutron absorber. A FBCR uses the reactor cooling water as the fluidizing medium (Fig. 1). In a FBCR, when there is no water flow, the solid neutron absorbing particles form beds at the bottom of the control rod tubes in the reactor core. When water begins to flow, the absorber particles are fluidized. As water flow increases, the particle bed expands and absorber particles spread out over the length of the control rod tubes that extends above the reactor core. In effect, the control rod becomes more transparent to neutrons as the absorber particle density decreases.

With the FBCR concept, the reactor control system becomes a flow-regulating system using either variable-speed pumps (Option A) or motor-driven control valves (Option B) to control water flow to the FBCR tubes and, hence, control rod worth. This system can be coupled to a variable-speed water pump and trip the reactor if the reactor water level drops below the pump inlet. For this application, the water pump inlet is located at the lowest allowable reactor water level for normal operations.

Alternative Versions: A similar control device developed by General Nuclear Engineering Corporation is called HY-BALL (hydraulic-ball) control. In the HY-BALL system, the absorber is distributed in relatively large spheres that are suspended hydraulically above the reactor core during normal operation. The spheres are either completely inserted or withdrawn from the reactor core; this offers less control flexibility than a true FBCR.

Status of Technology: Construction of prototype models and testing of the hydraulic and nuclear characteristics have been performed for transmission- (thickness < 2 mean-free paths) and reflection- (thickness > 4 mean-free paths) type rods. Acceptable characteristics are possible with both types. The feasibility of controlling low-power reactors by transmission- or reflection-type FBCRs has been established, but the technology has not been developed for commercial reactors.

Advantages:
1. Reduces pressure vessel head penetrations.
2. Loss-of-flow accident automatically shuts down reactor.
3. Axial power distribution can be shaped by using contoured channels or variable-sized particles.

4. Water gap flux peaking can be reduced for a partially withdrawn control rod.

5. Distortion or bending of control tubes will not prevent their operation.

Additional Requirements: None

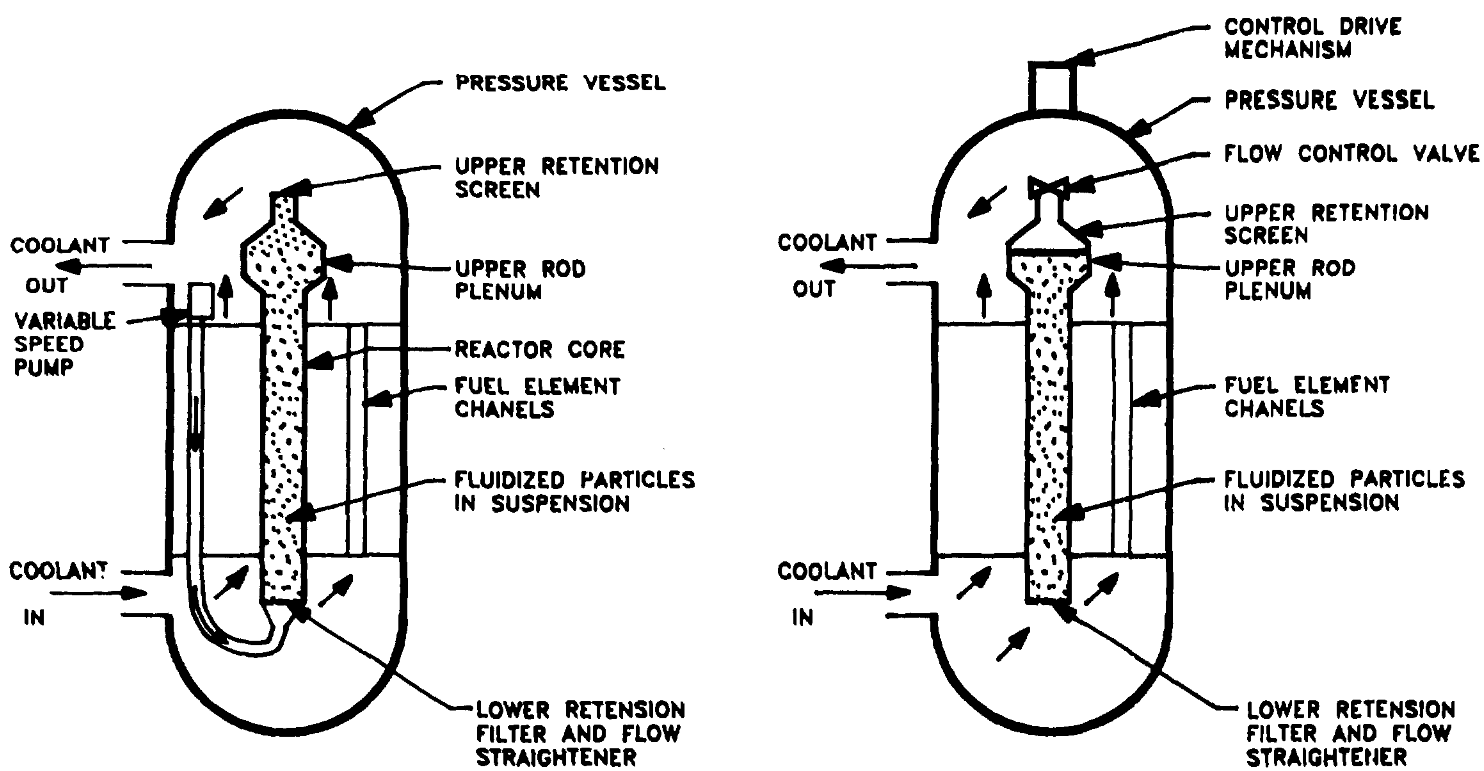
Comments: None

References/Contacts:

1. D. J. Blair, M. J. Driscoll, G. R. Dalton, and T. F. Parkinson, "A Study of Fluidized-Bed Control Rods," Physics and Material Problems of Reactor Control Rods, IAEA, Vienna (1964).

2. S. J. Weems, S. E. Turner, R. L. Lyerly, and R. W. Taylor, Jr., Development of Hydraulic Ball (HY-BALL) Control System: Second Quarterly Progress Report, GNEC264, EURAEC-533 (Jan. 2, 1963).

Update/Compiler: Feb. 1989/EBL



OPTION A: VARIABLE SPEED PUMP CONTROL

OPTION B: VALVE CONTROL (PRESSURE DROP ACROSS CORE EXCEEDS PRESSURE DROP ACROSS FBCR)

Fig. 1. Reactor with fluidized bed control rods.

TITLE: SELF-ACTUATING AND LOCKING SHUTOFF VALVE FOR HYDRAULIC/FLUIDIC CONTROL SYSTEMS INITIATED ON LOW WATER FLOW OR HIGH TEMPERATURE

Functional Requirements: Level 1.1.1 Provide Process Shutdown Mechanism

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Light-water reactor (LWR)

Organization: Industrial (Rockwell International Corporation)

Examples of Implementation: None

Description: Many types of hydraulically supported control rods, balls, and particles have been proposed for nuclear reactors. In each case the neutron absorber is held out of the reactor core by fluid friction forces during power operations. If the water flow slows or stops, the neutron absorber falls, by gravity, to the core and shuts down the reactor. This provides automatic reactor shutdown upon power failure, loss-of-coolant accident, or low reactor water levels, if the inlets for the pumps that push fluid by the control rods, balls, and particles are located above the reactor core but below normal reactor water levels. A drawback to gravity-based systems with absorbers held out of the core by fluid flow is that any residual fluid flow, even an amount below the minimum needed to safely operate the reactor, retards the fall of the neutron absorber into the reactor core.

This drawback can be overcome by using a self-actuating flow cutoff valve in the fluid stream above the neutron absorbing element. If the fluid flow drops below a set value, the valve automatically closes to cut off the fluid flow and allow the neutron absorber to fall into the core. The valve also has a self-locking mechanism enabling it to remain closed in the event of subsequent short-term pulses of fluid. This device could be used for any type of hydraulically suspended neutron absorber. When the valve closes, gravity would cause neutron-absorbing rods, balls, and particles to fall into the core. In the case of control rods, the valve could be used in conjunction with a hydraulic cylinder, which would insert a control rod into the core when the valve closes.

When the valve is open, fluid passes through it and out the holes in the top of the fluid flow cylinder (Fig. 1a). The valve balance member is pushed against the aperture plate, covering some of the holes so that a pressure drop is created across the balance member to hold the valve open. The balance member can be sized so that a predetermined minimum flow rate is required to hold the valve open. When the flow rate falls below the minimum, it is no longer sufficient to support the valve, which then closes by gravity, shutting off the water flow (Fig. 1b). This removes the fluid support for the neutron-absorbing elements, allowing them to fall into the core and shut down the reactor.

The valve is designed so that when it is closed, the amount of area for fluid pressure to exert downward force (α) is greater than the area for it to exert upward force (β). Thus, any subsequent fluid pressure, such as that from pressure surges, only acts to maintain the closure.

The valve could also be actuated by a weighted member suspended over the valve and controlled by a Curie point alloy magnet or an electromagnet. In the case of the Curie point alloy magnet, when the fluid flowing through the valve exceeds a designed temperature, the magnet would release the weight to force the valve closed. The electromagnet could be used as an active control and deenergized to release the weighted member to close the valve.

Alternative Versions: None

Status of Technology: As part of the U. S. Liquid Metal Reactor (LMR) Program, the described valve has been installed in a test neutron absorber column and tested at various rates of fluid flow through the assembly. The time for all the neutron-absorbing elements to fall into the core zone was substantially constant and faster than in the assembly without the valve.

Advantages:
1. Reduces time required by current hydraulic or fluidic control systems to achieve reactor shutdown.
2. The balance member can be designed so that the valve will close at any desired minimum flow rate.
3. Reduces the risk of mechanical failure of reactor shutdown system.

Additional Requirements: None

Comments: This technology was developed for LMRs. Since the liquid properties of sodium and water are similar, the device is expected to work in water-cooled reactors.

References/Contacts:

1. D. K. Chung, Self-Actuating and Locking Control for Nuclear Reactor, U.S. Patent 4,313,794 (February 2,1982).

Update/Compiler: February 1989/EBL

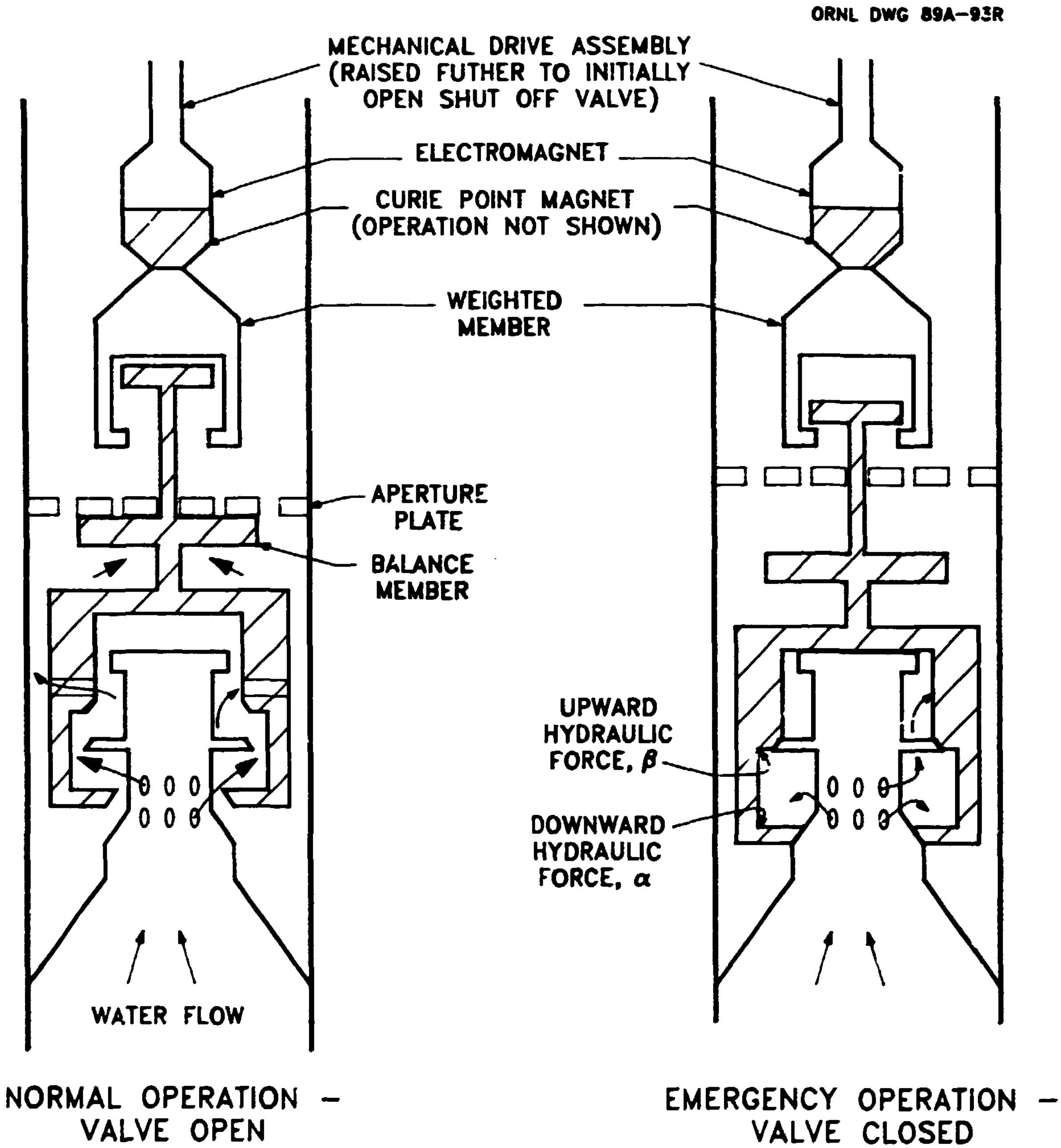


Fig. 1. Self-actuating and locking shutoff valve operation.

TITLE: NEUTRON/GAMMA/COOLANT THERMAL FUSE CONTROL RODS/DEVICES

Functional Requirement: 1.1.2 Limit Excess Reactivity

Safety Type: Passive

Developmental Status: 3 Evaluated (See Comments)

Reactor Type: Light-Water Reactor (LWR)

Organization University (Multiple)

Example of Implementation: None

Description: Thermal fuses have been proposed as a mechanism to initiate reactor shutdown for several types of reactor overpower conditions. The thermal fuses would hold control rods and/or absorber spheres out of the reactor during normal operations. If reactor overheating occurs, the fuses would fail allowing gravity drop of control rods and/or absorber spheres into the reactor core.

There are multiple types of conventional thermal fuses. With each type of fuse, the fuse components are held together at low temperatures, but separate if the temperature exceeds some design temperature. There are several mechanisms that can be used to hold the fuse components together at lower temperatures:

Solder – Many thermal fuses are held together by a low melting point solder. The solder melts at a defined temperature which allows the fuse components to separate.

Curie Point Magnets – Thermal fuses may contain magnets to hold the components together. All magnetic materials posses a curie point temperature at which the magnetic field fails. When a magnetic material in a thermal fuse exceeds the curie temperature (which can be selected by choice of material), the magnetic field collapses and, the fuse components separate. Curie point fuses may be reusable.

Bimetallic Strips – Bimetallic strips are strips of dissimilar metals fused together with different coefficients of thermal expansion. As the temperature rises, bimetallic strips bend due to the different rates of thermal expansion of the two metals. Strips can hold components together at low temperatures, but separate at high temperatures. Many thermostats operate on this principle where an electric switch is opened or closed. Bimetallic thermal fuses can be reusable.

Bimetallic Cylinders - Bimetallic cylinders are concentric cylinders of dissimilar metals with different coefficients of thermal expansion connected together at one end. As the temperature rises, the cylinders elongate at different rates. This can initiate control rod action (see below).

Thermal fuses are common industrial devices, particularly in fire protection systems. Spring loaded fire doors and ventilation dampers are often held open during normal operations by wires containing thermal fuses. If a fire occurs, the fuses fail on high temperature and doors or dampers close.

For nuclear applications, thermal fuses in or just above the reactor core would hold control rods or absorber spheres out of the reactor core. Three types of thermal fuses are possible based on the heating mechanisms. In practice, practical devices are likely to be a combination of these.

Gamma Ray Thermal Fuse: In a gamma ray thermal fuse, the thermal fuse contains materials which absorb gamma rays. When gamma rays are absorbed, heat is generated. The device is designed to overheat and cause thermal fuse failure under three circumstances:

- If power levels are excessive, the gamma flux will be high which overheats the fuse and causes failure.
- If water levels are low, the gamma flux will be higher than normal due to the lack of shielding from water coolant between fuse and reactor core. This causes the fuse to overheat and fail.
- If there is a mismatch between power level and coolant flow, the thermal fuse will be undercooled and fail.

Neutron Thermal Fuses: Neutron thermal fuses are similar to gamma ray fuses except the fuse body contains a fissile material such as ^{235}U. Neutrons from the reactor core cause fission which, in turn, generates heat. The device fails due to overheating in the same circumstances as causes failure of gamma ray thermal fuses except neutron flux determines heat generation rate rather than gamma flux.

Coolant Thermal Fuses: These are fuses which are heated by a rise of coolant temperature.

A wide variety of thermal fuse control rods have been investigated for use in Liquid Metal Reactors (2,3). Some of these devices, such as the Self-Actuated Shutdown System (SASS) which is a currie point control rod, have undergone significant development and testing. Thermal fuses have not been used in LWRs.

Some examples of thermal fuses are shown in Fig. 1 and Fig. 2. Figure 1 is a schematic of one type of curie point control rod where the control rod is held out of core by magnets. If the magnets overheat, they fail and drop the control rod into the reactor core. Figure 2 shows a bimetallic cylinder control rod fuse (4,5) where heating of the bimetallic cylinder leads to differential expansion of the two metals with the drop of the control rods.

Alternative Versions: There are multiple design options.

Status of Technology: For normal industrial applications, many types of thermal fuses are in use. Neutron and gamma ray thermal fuses have been proposed but not implemented (see Comments).

Advantages: Potential advantages of thermal fuses include:

1. Passive safety device

2. Base of industrial technology exists from which to develop nuclear thermal fuses

Additional Requirements: None

Comments: The technology is developed for LMRs but significant development would be required for use in LWRs due to the different environments (radiation level, neutron spectrum, coolant temperatures).

References/Contacts:

1. B. I. Spinrad, personal correspondence to C. W. Forsberg (Aug 12, 1989).

2. R. B. Tupper, M. H. Cooper, R. K. Sievers, "A Self-Actuated Shutdown System for Liquid Metal Reactors," Third International Conference on Liquid Metal Engineering and Technology in Energy Production, Oxford, United Kingdom (April 9-13, 1984).

3. R. B. Tupper, W. A. Brummond, R. K. Paschall, and S. S. Borys, "Development of a Self Actuated Shutdown System, ASME81-JPGC-NE-8 (October 1981).

4. Department of Energetics of La Sapienza University of Rome (Italy), Draft: Multipurpose Advanced Reactor Inherently Safe (MARS), (July 1989).

5. M. Caira, M. Cumo, and N. Naviglio, "MARS Reactor: A Proven PWR Technology Combined with Advanced, Safety Requirements," Energia Nuclear 23 (May/September 1987).

Update/Compiler: August 1989/CWF

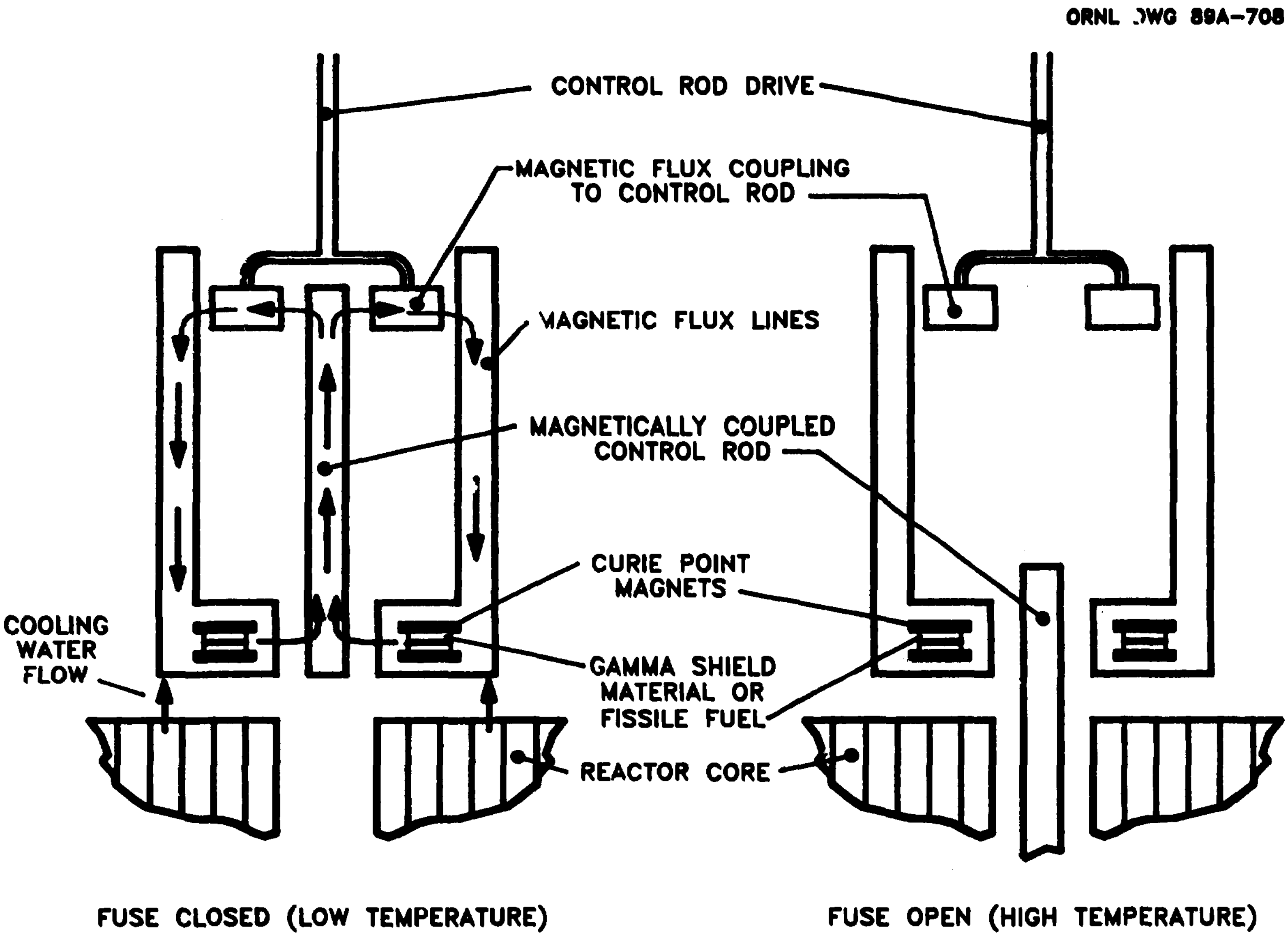


Fig. 1. Schematic of curie point thermal fuse for LWR shutdown rods.

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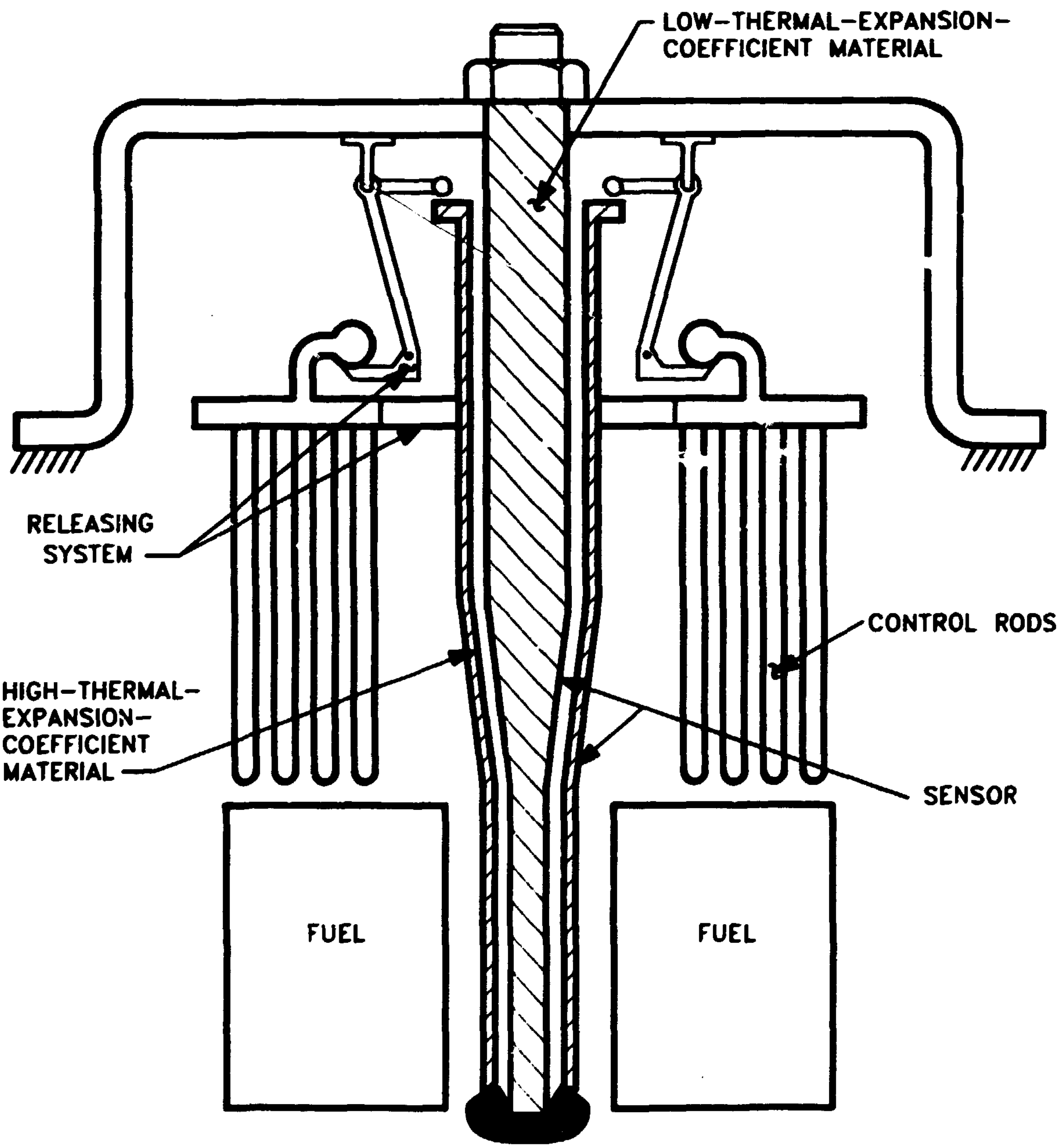

Fig. 2. Schematic of thermal metal expansion fuse for LWR shutdown rods.

TITLE: IN-REACTOR-CORE POWER FUSES

Functional Requirements: 1.1.2 Limit Excess Reactivity
1.2.2 Maintain Core Coolant Inventory
2.2 Control Primary Circuit Pressure

Safety Type: Passive

Developmental Status: 5 Concept

Reactor Type: Light-water reactor (LWR)

Organization: National Laboratory (Oak Ridge National Laboratory) and Regulatory Agency (Direzione Sicurezza Nucleare E. Protezione Sanitaria)

Examples of Implementation: None

Description: In-core power fuses have been proposed for use with a variety of safety systems. Both electrical and fluid fuses have been proposed (1, 2).

Electrical Fuses – In all electrical fuse systems, an electric power line goes through the reactor core. The power line contains an in-core power fuse that cuts off electric power if nuclear power levels are excessive or if there is insufficient core cooling. The electric power operates one or more "fail-safe" safety systems, which are designed so that the loss of electric power causes their activation. Two examples show the use of in-core power fuses:

- In-core power fuses can be located in the power lines that supply power to pressurized-water reactor (PWR) control rods. In most PWRs, the control rods are held out of the reactor core by electromagnets. If the electromagnets lose power, the control rods drop back into the reactor core by gravity and shut down the reactor.

- In-core power fuses can be located in the power lines that supply power to certain types of emergency core cooling systems [see: Fluidic In-Vessel Emergency Core Cooling System], where the operation of pumps is required to hold emergency supplies of cold, borated water out of the reactor core. Failure of the electric power causes the pump to stop, borated water to enter the reactor core, and the reactor core to be shut down and cooled.

In most cases, the in-core power fuse accomplishes the same functions as is achieved in some current reactors with in-core sensors, control systems, and power relays, as shown in Fig. 1. The benefit of the in-core fuse is that, with appropriate design, many possible safety-system failure modes are eliminated (e.g., such as from operator error, maintenance error, or sabotage).

An example of an in-core power fuse is the Critical Heat Flux Scram System (CHEFSS), which can be used with control-rod systems or emergency-core cooling systems. CHEFSS is an electrical power fuse that "blows" upon experiencing an excess power/coolant flow rate mismatch sufficient to cause excessive core heating. CHEFSS consists of one or more annular sensor fuel pins with a power line extending through each pin. If more than one pin is used, the power line is run in series through the pins so that burnout of the power line in any one of the pins will disrupt the power supply.

A CHEFSS fuel pin has two components (Fig. 2). The first is an annular fuel pin. The second is the power line that runs through the middle of the annular pin. There is an annular cooling channel between the power line and the annular fuel pin. The power line must be cooled or it will burn out due to high temperatures caused by neutron/gamma heating and electric resistance heating from voltage drops across the power line. The annular fuel pin from inside to outside has layers of clad, insulation, cermet fuel, and clad. With this structure, most of the heat generated by the fuel goes to the outside surface of the annular fuel pin during normal operations. The coolant channel between the inside of the annular pin and the power line primarily cools the power line.

The annular fuel pin is designed to reach the critical heat flux before any conventional fuel pin in the reactor does so or before any pin reaches any other operating limit. The critical heat flux (Fig. 3) is a heat-transfer phenomenon where, as power levels increase, a layer of steam forms on the heat-transfer surface of the annular pin. When this occurs, the rate of heat transfer between pin and cooling water drops rapidly due to the insulating properties of steam and the temperature of the annular fuel pin rises rapidly. This occurs suddenly at a well-defined temperature.

If the critical heat flux is exceeded, the temperature of the annular pin rises rapidly and, with the higher temperatures, heat is transferred to the inner cooling channel through the insulated layer in the annular pin. This causes steam to form in the coolant channel between the annular pin and power the line, which rapidly reduces coolant flow and heat transfer. This, in turn, causes the power line to heat up and fail. In effect, heat-transfer conditions inside the reactor core determine when the in-core fuse will fail.

There are alternative versions of CHEFSS, including systems for fluid power systems rather than electric power. In these systems, a high-pressure water line replaces the electric power line. When CHEFSS is activated, water flow rates rapidly decrease due to the formation of steam inside the annular pin and liquid water is converted to steam. A reduced water flow rate or the presence of steam is used to trigger safety systems.

Thermal Fluid Fuses – In-core thermal fluid fuses have been proposed for several applications. In particular, they have been proposed to ensure reactor depressurization in the event of reactor core overheating or reactor core melt. There are two reasons for depressurization under these circumstances:

- If the reactor is depressurized, it is possible to add emergency core-cooling water by gravity drain of tanks above the reactor core or with the use of low-pressure pumps.

- If a core melt occurs when the reactor vessel fails, molten core will be injected rapidly into the containment upon vessel failure. High-pressure injection of molten core into containment increases the potential for containment failure due to very rapid heating in containment, increases the production of fine aerosols from sprayed molten core material, and creates the possibility of rapid chemical reaction between core material and containment atmosphere due to the fine aerosols and high temperature.

An example in-core thermal fluid fuse system is shown in Fig. 4 as part of the larger pressure-relief valve system. In a PWR, the reactor is depressurized by a pressure-relief valve attached to the pressurizer. This valve is opened by applying pressure to a large piston in the valve which, in turn, opens the valve.

During normal operations, the reactor can be depressurized by opening a small motor-operated valve that allows fluid from the pressurizer to push the large piston in the relief valve down, thus opening the valve. Since the surface area of the piston is larger than that of the valve, equal pressure on both sides of the piston/valve results in the valve going to the open position.

If the reactor overpressurizes, a small spring-loaded pilot valve opens, allowing pressurizer fluid to push the large piston down and open the valve.

If the reactor overheats, the valve is opened by the following sequence of events:

- High temperatures in the reactor core cause fusible plugs in the reactor in line to the relief valve to melt and fail. These plugs are similar in design to those used in water sprinkler systems. (See: Temperature-Activated Water Sprinklers.)
- Water, high-pressure steam, and other gases pressurize the lines.
- High-pressure fluid pushes the large piston down in the pressure relief valve, which depressurizes the reactor.

Alternative Versions: Many design options exist.

Status of Technology: Limited theoretical calculations of system performance have been made.

Advantages:
1. In-core conditions directly initiate operation of the safety system. Conventional systems measure various properties and, from these properties, infer in-core reactor conditions. The potential exists to misunderstand actual in-core conditions such as occurred during the TMI-2 accident (3).
2. In-core fuses are more resilient against many types of operator error, maintenance error, or sabotage than are conventional sensor and control systems.

Additional Requirements: None

Comments: In-core fuses are an unexplored technology where both advantages and disadvantages are not well understood.

References/Contacts:

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

2. G. Petrangeli, More Intrinsically Safe and Simplified Light-Water Reactors, Direzione Sicurezza Nucleare E Protezione Sanitaria (Rome), RTI-DISP (85) (September 1985).

3. Nucl. Technol. 87, No. 1 (entire issue) (August 1989).

Update/Compiler: September 1989/CWF

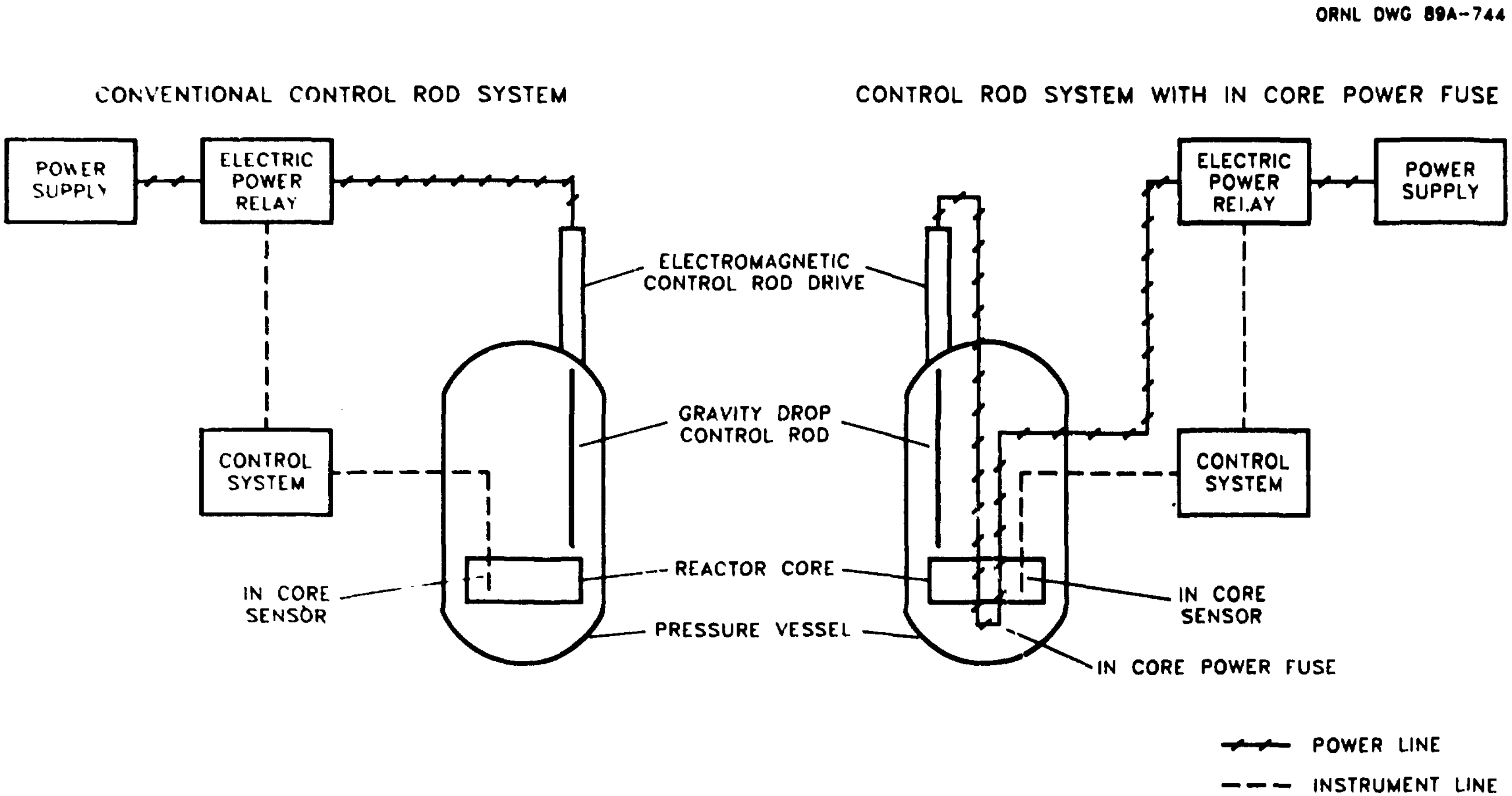

Fig. 1. Alternative control rod system.

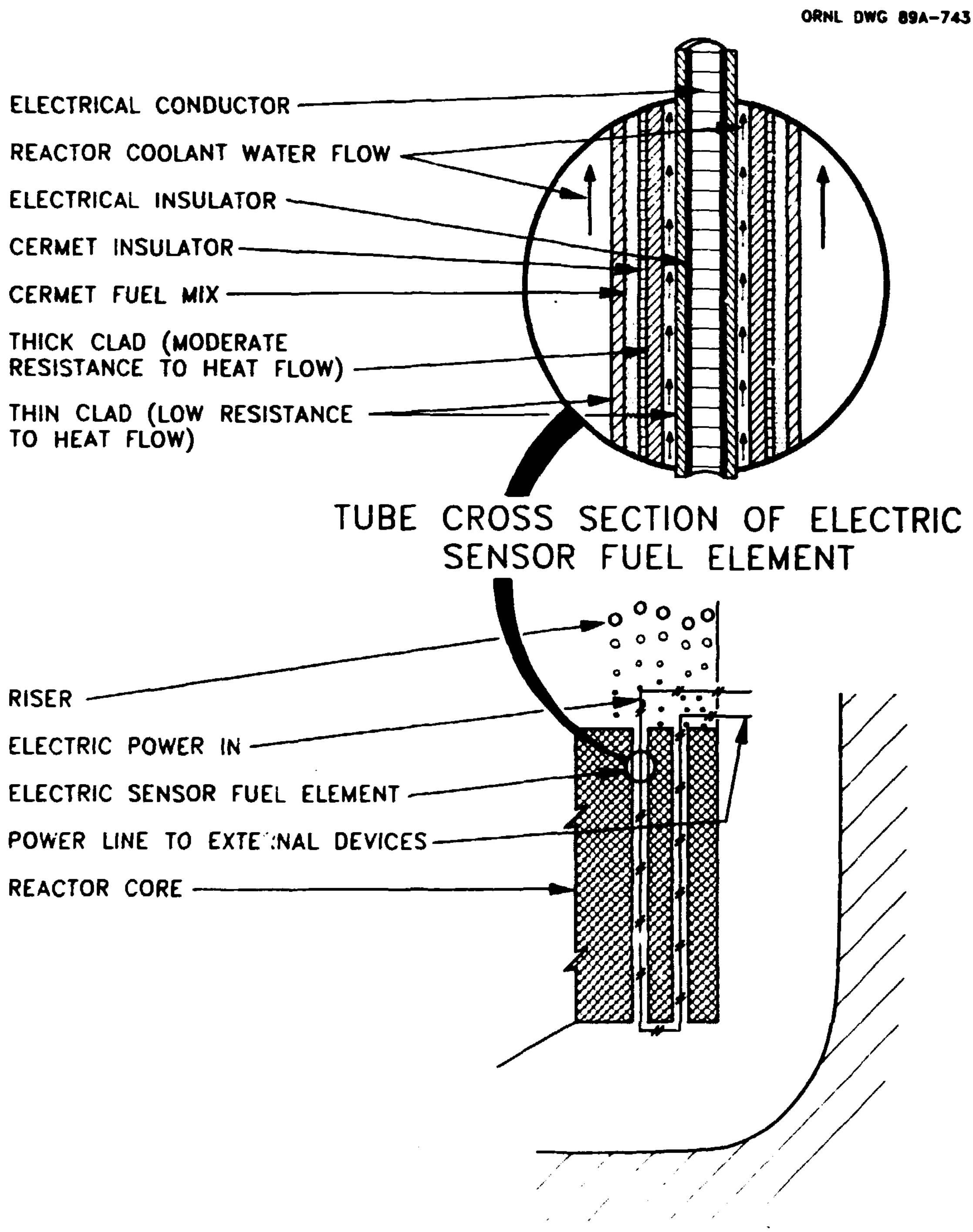

Fig. 2. CHEFSS with electrical sensor fuel element.

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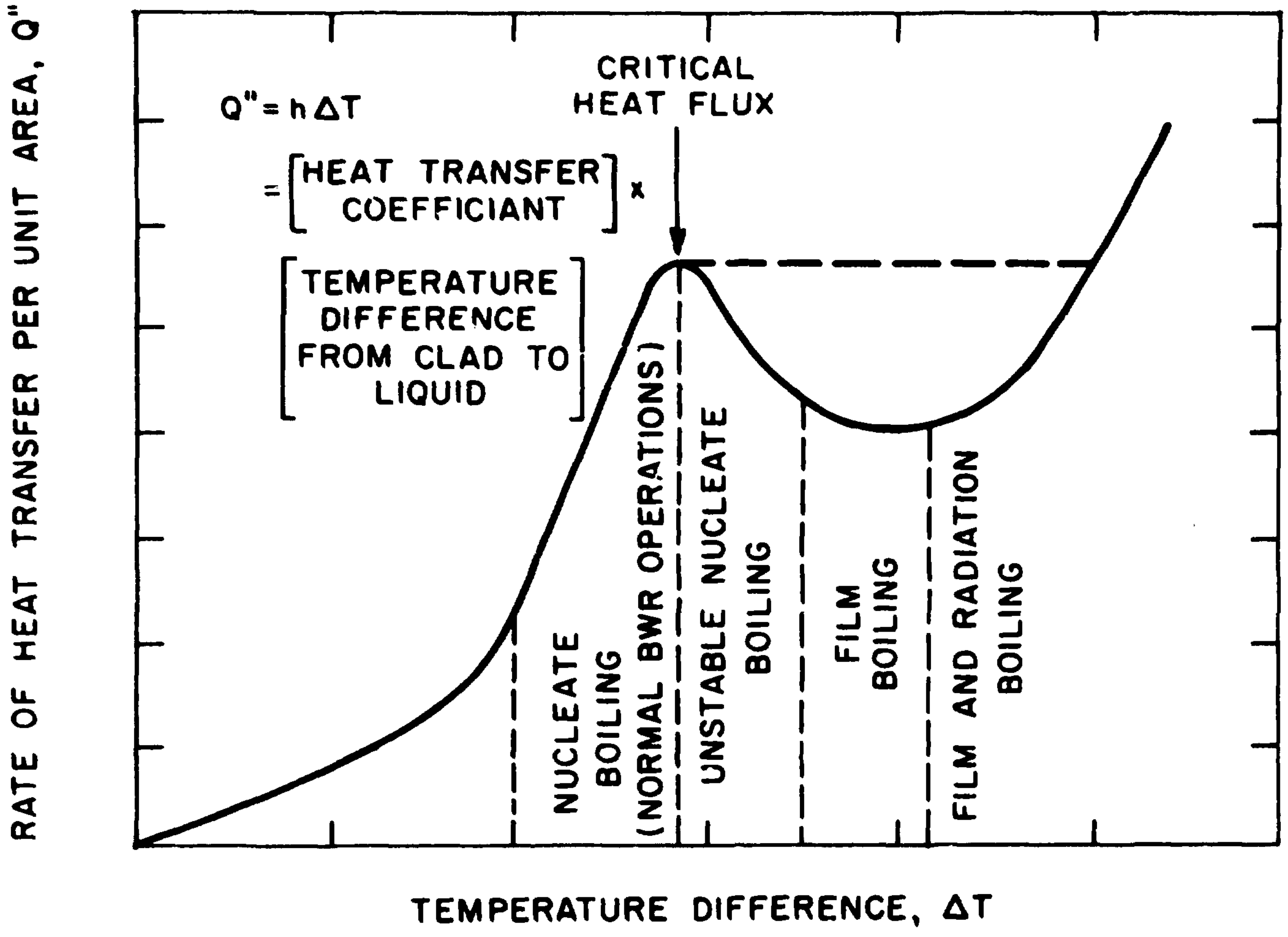

Fig. 3. Generalized boiling heat transfer performance.

ORNL DWG 88A-745

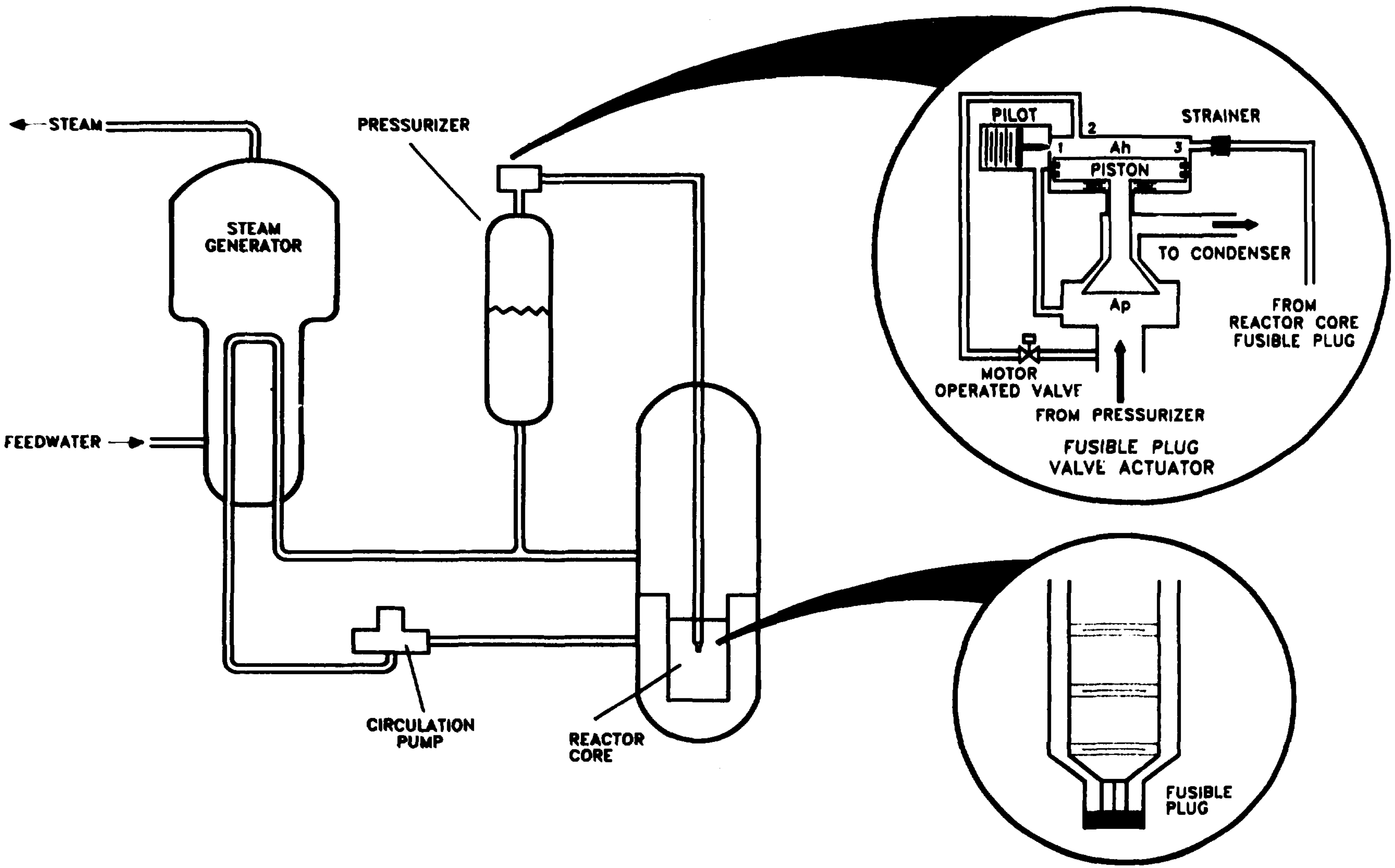

Fig. 4. Fusible plug depressurization system.

TITLE: USE OF SELECT MATERIAL PHASE CHANGES TO ENHANCE DOPPLER REACTIVITY FEEDBACK IN LWRS

Functional Requirements: Level 1.1.3: Control Reactivity with Inherent Feedback

Safety Type: Inherent

Developmental Status: 5 Concept

Reactor Type: Light-water reactor (LWR)

Organization: University (University of Missouri)

Examples of Implementation: None

Description: The effect of temperature-induced, density-corrected phase changes on the average total nuclear cross section of various materials has been demonstrated experimentally in the laboratory[1,2] and is shown in Fig. 1 for tin. As used in this discussion, temperature-induced phase changes include all realignments of the material crystalline lattice, including melting. With an appropriate selection of fuel and absorber materials, it may be feasible to enhance the Doppler effect at phase changes for fertile fuel material and resonance-absorbing absorber materials so as to increase the reactivity feedback during reactor temperature excursions. The materials selection must meet operational requirements of burnup, radiation exposure, temperature, and chemical environment of the reactor while providing the desired feedback effects at temperatures above normal operating values. As indicated, this effect is independent of changes in material densities caused by changes in phase.

Alternative Versions: None

Status of Technology: This is an observed physical and neutronic phenomenon that has not been investigated for practical applications.

Advantages: Inherent reactivity control mechanism

Added Requirements: Maintenance of structural integrity in the reactor environment

Comments: None

References/Contacts:

1. F. Y. Tsang and R. M. Brugger, "Doppler Effect Measurements of Tin by the Filtered Neutron Beam Technique," Nucl. Sci. and Eng., 74, 34 (1980).

2. R. M. Brugger and H. Amintar, "Doppler Measurements of ^{238}U," Joint IAEA/NEA Consultants Meeting on Uranium and Plutonium Resonance Parameters, Vienna, September 28 - October 2, 1981.

Update/Compiler: Apr. 1989/DLM

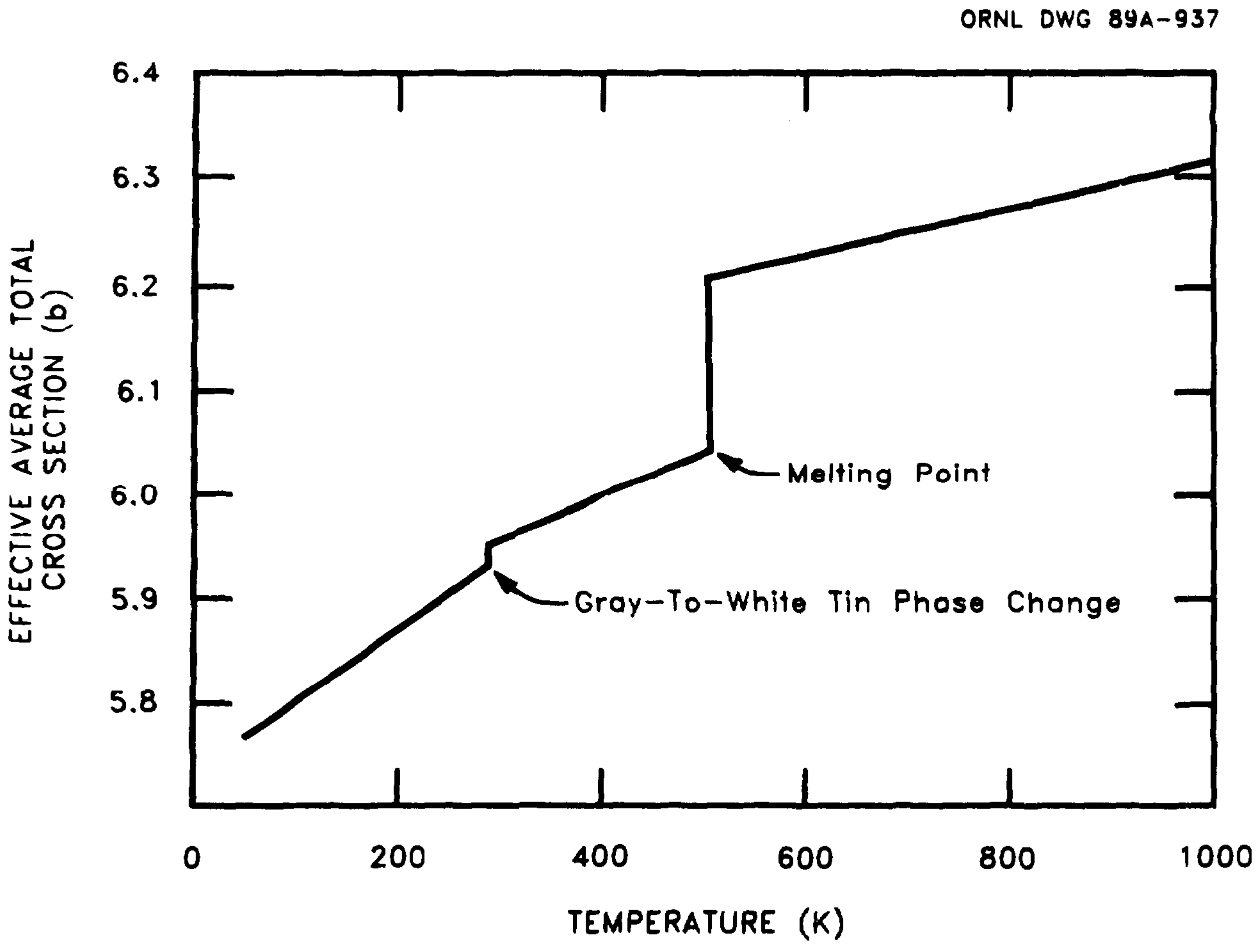

Fig. 1. Effective average total neutron cross section of tin for neutrons in the 24 KeV band.

TITLE: LARGE-PROMPT, NEGATIVE-MODERATOR COEFFICIENT OF REACTIVITY FROM HYDROGEN BOND STRUCTURE

Functional Requirements: Level 1.1.3: Control Reactivity with Inherent Feedback

Safety Type: Inherent

Development Status: 5 Concept

Reactor Type: Light-water reactor (LWR)

Organization: None

Description: In thermal neutron reactors such as LWRs, hydrogen is the moderator. Hydrogen can provide a negative moderator coefficient through two effects.

1. In LWRs, reactivity feedback of the moderator temperature coefficient is primarily due to density changes in the water affecting neutron slowing down, neutron leakage, and the density of chemical shim. Change in the thermal neutron scattering characteristics of the hydrogen bound in the water molecule has only a small temperature reactivity effect compared to the change in water density.

2. In the specialized water-cooled reactors such as the TRIGA research reactors, the neutron moderation is primarily effected by the hydrogen bound in the hydride material that is intimately mixed with the fuel in the fuel element matrix. This effect is described herein.

In TRIGA and some other specialized reactors, heat up of the fuel element containing a hydride material introduces little immediate density change in the hydrogen, but does immediately change the "thermal scattering" of neutrons off of the hydrogen bound in the hydrides. This effect is due to the chemical bond structure of the hydride as a function of temperature. As the hydride heats up, the hydrogen atoms enter chemically excited states. This increases the probability that a thermal neutron will gain energy from the oscillatory hydrogen atoms in the hydride lattice. The effect is to "harden" the thermal neutron energy spectrum in the fuel element and thereby increase the leakage of thermal neutrons from the fuel element into the cooler water-cooling channels where rethermalization and preferential parasitic capture of the neutrons in the hydrogen atoms that are bound in water take place. Increases in Uranium-235 loading or high burnup introducing Plutonium-239 (with its large thermal fission resonance) tend to make the moderator coefficient of reactivity of TRIGA-type fuel less negative, but the use of a thermal resonance absorber such as Erbium-167 will compensate for such positive reactivity effects.

Status of Technology: TRIGA fuel does not meet the power density and temperature stability requirements of LWRs; however, similar inherent neutronic properties of TRIGA-type fuels with large-prompt, negative-temperature coefficients may be achievable in advanced LWR fuels with appropriate research. The technology is undeveloped for LWRs.

Advantages: Inherent safety

Additional Requirements: Materials properties compatible with the high temperature , high burnup, and high power density environment of current LWRs.

Comments: None

References/Contacts:

1. M. T. Simnad, F. C. Foushee, and G. B. West, "Fuel Elements for Pulsed TRIGA Research Reactors," Nucl. Technol., 28, 31 (1976).

Update/Compiler: Apr. 1989/DLM

TITLE: HIGH-TEMPERATURE CERAMICS, CLADS, AND FUELS FOR LWR

Functional Requirement: Level 1.1.3: Control Reactivity with Inherent Feedback
Level 1.2: Remove Core Heat

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Light-Water Reactor (LWR)

Organization: National Laboratory (Kernforschungsanlage Jülich)

Examples of Implementation: None

Description: The current fuel form for LWRs consists of metallic tubes filled with UO_2 pellets. The preferred clad, zircaloy, begins to weaken if temperatures exceed 600°C. Normal fuel clad operating temperatures are near 300°C. Many of the safety characteristics of LWRs are dependent on the maximum allowable fuel or clad temperature.

If fuel clad and fuel temperature could be raised 500 to 1000°C without major fuel failure, there would be major benefits in safety with major simplifications in reactor design. In particular:

1. The consequences of most, and perhaps, all types of reactivity accidents could be reduced. LWRs have negative temperature reactivity coefficients (i.e., raising fuel temperature lowers reactor power levels). This is a safety benefit, but with current fuels, not all the consequences of various types of reactivity accidents can be eliminated. Fuel damage occurs before the fuel temperature is high enough to shut down the reactor with a negative temperature reactivity coefficient. With higher temperature fuels, reactor shutdown could be assured before fuel damage.

2. The consequences of loss-of-coolant accidents could be eliminated for small LWRs and reduced for larger reactors. A high-temperature fuel allows decay heat to escape the core by thermal radiation. For small reactor cores, all decay heat can be removed from the reactor core by thermal radiation and heat conduction. For larger cores, the requirements for emergency core cooling are reduced.

In the 30 years since the initial use of zircaloy, major advances in high-temperature ceramic fuels have been made. For example, the high-temperature gas-cooled reactor uses a fuel where fuel damage is insignificant below 1600°C[1]. With current technology, a high-temperature totally ceramic fuel could be developed for LWRs. There are two basic options.

The first option is to use a modified form of high-temperature gas-cooled reactor (HTGR) fuel for LWRs. Preliminary theoretical and experimental studies have been conducted for using these fuels in pressurized water reactors (PWRs)[2] and water-cooled research reactors[3]. The designs proposed for PWRs would have reactor core pressure drops similar to existing reactors so as not to change most of the characteristics of existing PWRs.

The particular fuels investigated were coated particles of 100 to 800 μm in diameter. They contain oxide or carbide fuel kernels enclosed in several coatings deposited by pyrolytic means. Typically, the inner coating is made of a porous pyrocarbon that contains (1) gaseous fission products, (2) kernel swelling due to burnup, and (3) recoil atoms from the kernel surface during fissioning. The inner coating is covered with one or more layers of high-density pyrocarbon that act as pressure vessels for the fission products and as a diffusion barrier to fission products. The outer coat would be silcon carbide due to its good corrosion and abrasion resistance.

Figure 1 shows a proposed fuel element, the particle bed, and an individual coated particle. Unlike existing fuels, a PWR with coated-particle fuel would have only a very short water flow path across the particle beds. The particles have a very large surface to volume ratio and very good heat transfer coefficients.

The second option is to develop a specific high-temperature ceramic fuel for LWRs. The characteristics of ceramics are such that they require a spherical fuel form. Thermal hydraulic characteristics of LWRs would probably make it desirable to have a spherical fuel form ranging in size from 1 to 10 cm.

Alternative Versions: Many possibie options

Status of Technology: Limited experimental work has been conducted in Europe on using HTGR microspheres for LWRs[2]. Major work would be required to identify the preferred high-temperature ceramic fuel elements for LWRs and understand the economics of such fuels.

Advantages:
1. Potential to eliminate consequences of all types of reactivity accidents in LWRs.
2. Reduction of emergency core-cooling system requirements.
3. With current fuels, in a postulated reactor core meltdown accident, hydrogen is generated by reaction of metal clad with steam. Ceramic clad fuels would eliminate hydrogen generation in an accident and the problems of controlling hydrogen in a post-accident environment.

Additional Requirements: None

Comments: Neither the benefits nor the costs of developing this technical option are well understood. There have been major advances in ceramics in the past several decades, but the total impact is unknown.

References/Contacts:

1. H. Nabielek, W. Schenk, W. Heit, A. Mehner, and D. T. Goodin, "The Performance of High-Temperature Reactor Fuel Particles at Extreme Temperatures," Nucl. Technol. 84, 62 (Jan. 1989).

2. W. Katscher, "Coated Particle Fuel Element for Pressurized Waste Reactors," Nucl. Technol. 35, 557 (Sept. 1977).

3. J. R. Powell, H. Takahashi, and F. L. Horn, "High-Flux Research Reactors Based on Particulate Fuel," Nuclear Instruments and Methods in Physics Research, A249, 66 (Aug. 15, 1986).

Update/Compiler: Jan. 1989/CWF

ORNL DWG 88-14972

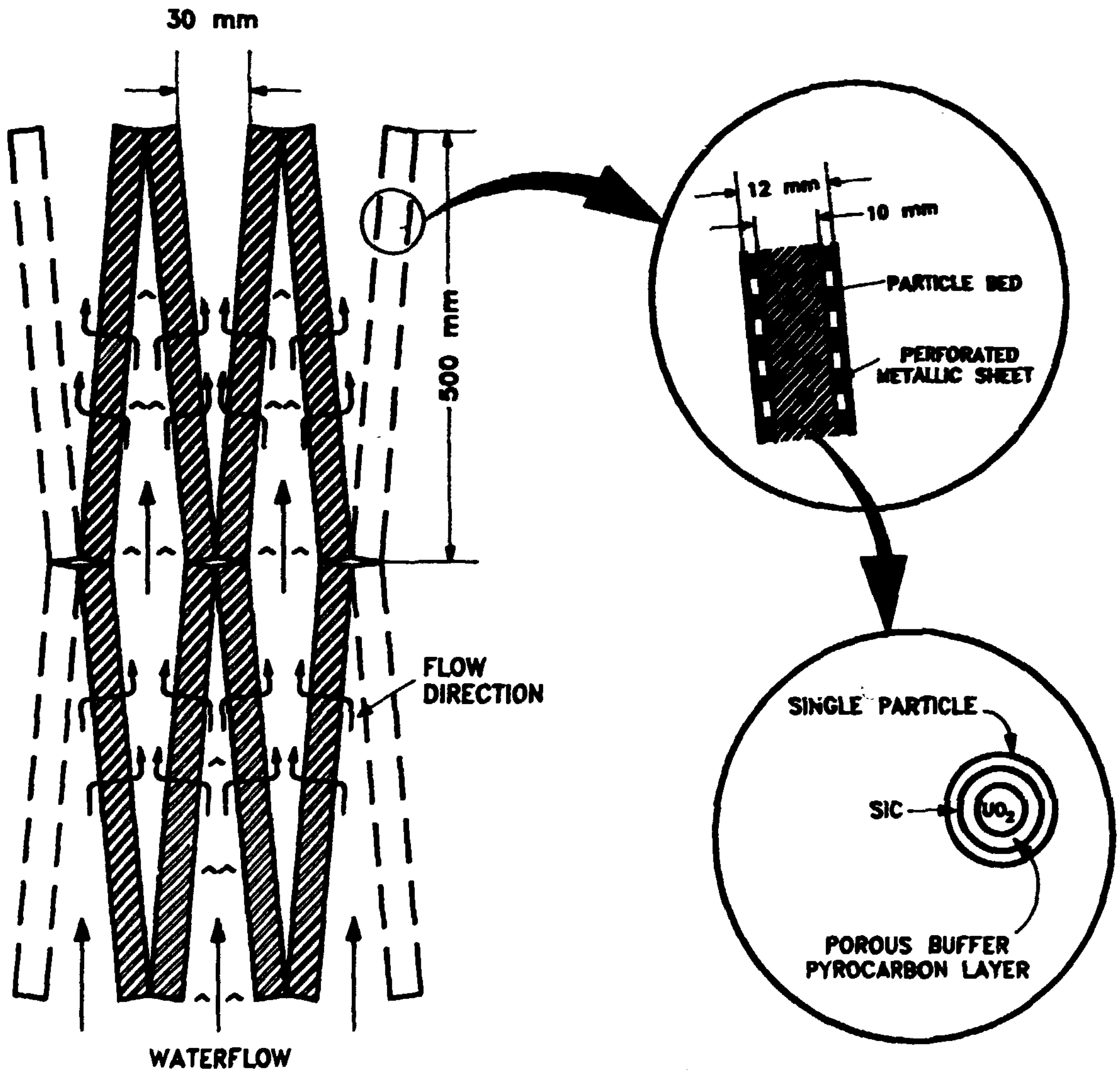

Fig. 1. Schematic of PWR coated-particle fuel element.

TITLE: GRAPHITE-DISK UO_2 FUEL ELEMENTS FOR ENHANCED THERMAL MARGIN AND REDUCED EXCESS REACTIVITY

Functional Requirements: Level 1.2: Remove Core Heat
Level 1.1.2: Limit Excess Reactivity
Level 1.1.3 Control Reactivity with Inherent Feedback

Safety Type: Inherent

Development Status: 3 Evaluated

Reactor Type: Light-water reactor (LWR)

Organization: National Laboratory (Chalk River National Laboratories)

Examples of Implementation: None

Description: The use of dished and/or annular UO_2 fuel pellets in fuel rods with axially interspaced thin graphite disks has been tested extensively for applications to CANDU fuel systems. The pellet and disk configuration is used to decrease average operating fuel pellet temperatures and change the temperature distribution within the fuel pellets. The graphite has a high thermal conductivity; therefore, it provides a heat conduction path to remove heat from the center sections of adjacent uranium oxide pellets (Fig. 1). Uranium oxide has a very low thermal conductivity, which implies that normally high fuel pellet temperatures exist in the center of the fuel pellets. Reduced pellet temperatures and redistribution of temperatures throughout the pellet have multiple benefits:

- The average and center line fuel temperatures are lower, thus the reactor core can absorb more heat under any postulated set of accident conditions before fuel failure or exceeding center line melting of the fuel.
- The thermal margin between normal operation and fuel damage is larger. Because reactors have negative temperature reactivity coefficients, the fuel can withstand larger reactivity accidents before fuel failure due to excess temperatures.
- The temperature distribution of the fuel pellets is changed which allows temperature control of the fuel to maximize negative reactivity effects at different power levels.
- Reduced power reactivity requirements of fuel allow for extended burnup.

Status of Technology: Experimental data and calculational analyses have been performed for CANDU reactors; no analysis has been performed for LWRs. For CANDU reactors, this fuel design is providing the basis for high-burnup, high-power, low-enriched fuel cycles.

Advantages: Inherent safety

Additional Requirements: None

Comments: In principle, a wide variety of materials with different physical and nuclear properties could be used for the disks. Benefits and costs for this technology in LWRs are not well known.

References/Contacts:

1. R. D. MacDonald and I. J. Hastings, "Graphite Disk UO_2 Fuel Elements Designed for Extended Burnups at High Powers," Nucl. Technol., 71, 430 (1985).

Update/Compiler: April 1989/DLM

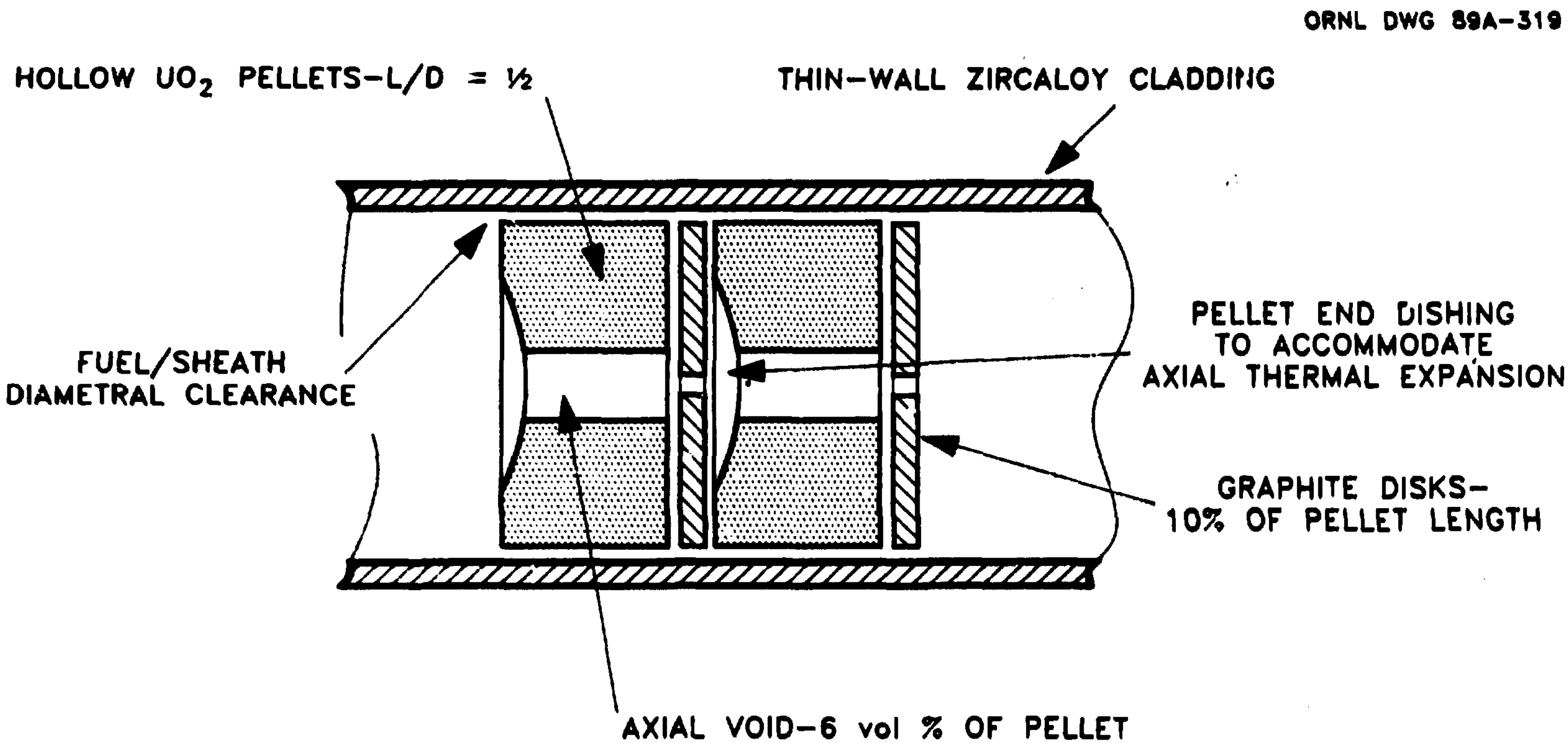

Fig. 1. Schematic of Candu reactor fuel design with annular fuel and graphite disk.

TITLE: ABSORBER (POISON) PILL FOR REACTOR SHUTDOWN ON HIGH TEMPERATURES

Functional Requirements: Level 1.1.3 Control Reactivity with Inherent Feedback

Safety Type: Passive

Developmental Status: 5 Concept

Reactor Type: Light-water reactor (LWR)

Organization: Unknown

Examples of Implementation: None

Description: The absorber (poison) pill is a use-once, emergency shutdown mechanism initiated on high fuel temperatures. An example design (Fig. 1) consists of metal-encapsulated pills of a very strong neutron adsorber (e.g., cadmium, enriched boron, or enriched gadolinium) inside selected annular fuel pellets which are inside LWR fuel pins.

During normal operations, relatively few neutrons are adsorbed by the absorber (poison) pills . The effective neutron flux does not see the pills because of self-shielding effects of the fuel around the pill and due to the small dimensions of each pill (few neutrons go through the volume of any pill).

If the fuel overheats, the poison pill container fails and releases the neutron absorber inside the fuel pin. The neutron absorber, by gas diffusion, moves up and down the inside of the fuel pin. Some neutron absorber may flow by gravity down the annular holes in the fuel pellets. As the neutron absorber spreads out, it is seen by the neutron flux, adsorbs neutrons, and shuts down the reactor.

There are several important requirements and characteristics for absorber (poison) pills. To minimize parasitic neutron adsorption during normal operations, small pills with very high neutron cross-section materials are desirable. Absorber (poison) pill failure could be initiated by melting of the coating material, high thermal stresses at high temperatures, or expansion of the absorber with temperature (such as cadmium melting to a liquid) and bursting of the pill.

Alternative Versions: None

Status of Technology: No studies found of concept.

Advantages: Passive technology that cannot be bypassed or stopped by an operator.

Additional Requirements: None

Comments: No analysis of this concept has been identified. The closest related work are studies on burnable absorbers. The concentrations of absorbers in fuel pellets in these studies[1] have been several orders of magnitude less than would exist in an absorber (poison) pill.

References/Contacts:

1. R. P Harris, Use of Gadolinium in PWR Extended Burnup Fuel Cycles, CEND-397 (May 1982).

Update/Compiler: June 1989/CWF

ORNL DWG 89A-449

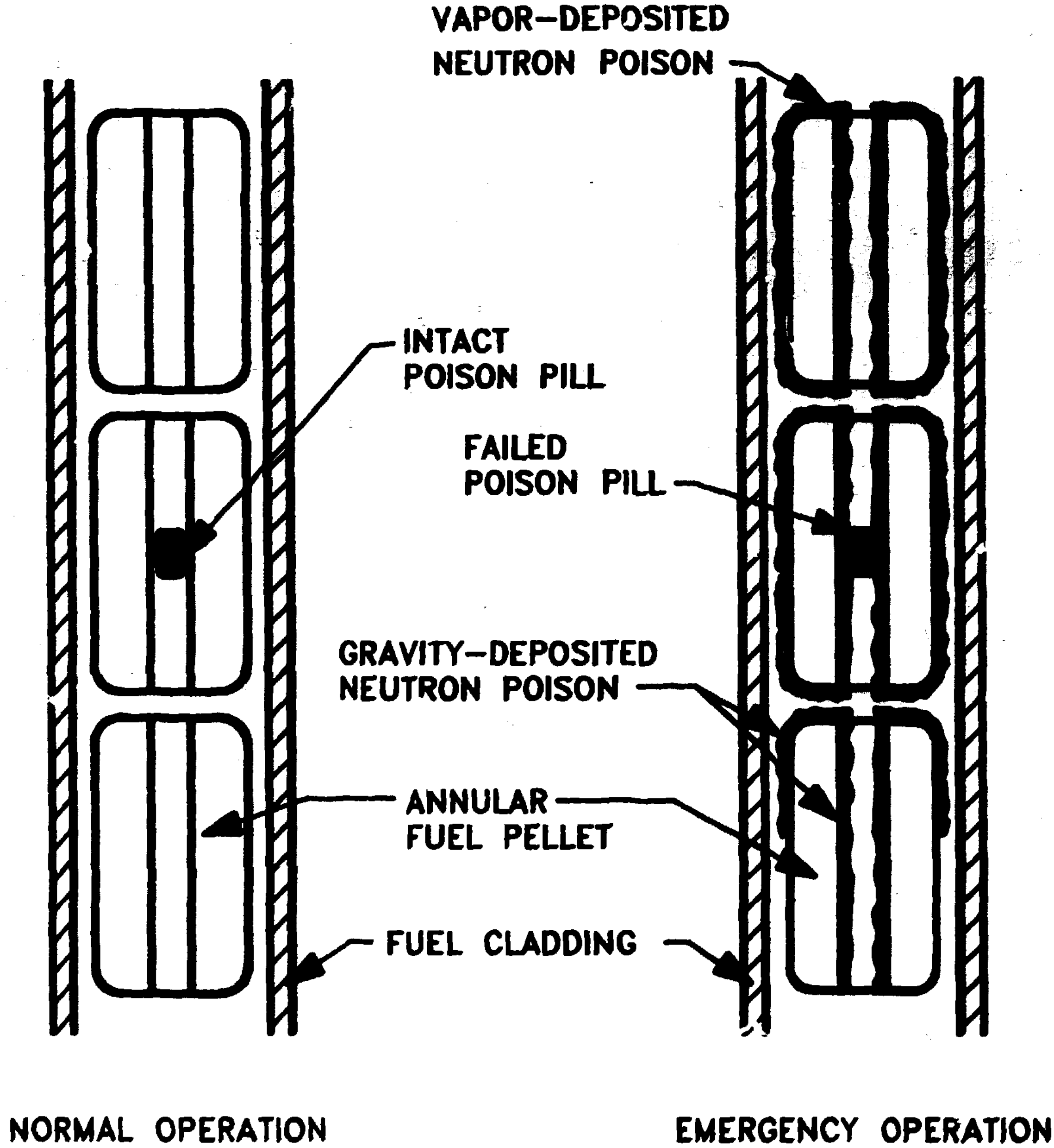

Fig. 1. Poison-pill fuel pin.

CHAPTER 6

Structures, Systems and Components to Remove Core Heat

Function 1.2

6. STRUCTURES, SYSTEMS, AND COMPONENTS TO REMOVE CORE HEAT

6.1 INTRODUCTION

This chapter describes structures, systems, and components (SSCs) for light-water reactors (LWRs) that meet the functional requirement to remove core heat (Table 3.1; Function 1.2). This requirement is met by ensuring the presence of an adequate amount of water in the LWR core. There are three subtier functional requirements: (1) maintain core coolant boundary integrity, (2) maintain core coolant makeup, and (3) transport heat to ultimate heat sink.

The decay heat that must be removed from the core is a function of the decay power after shutdown. The decay power and integral decay heat per MW(t) of reactor output are shown in Fig. 1 for a typical pressurized-water reactor (PWR). The assumed fuel cycle for Fig. 1 consists of three irradiation cycles, with one-third of the fuel irradiated to a burnup of 11 gigawatt-days per metric ton of initial heavy metal [GW(d)/MTIHM], another one-third to 22 GW(d)/MTIHM, and the final one-third to 33 GW(d)/MTIHM. The decay heat values were obtained using the ORIGEN2 computer code.[1] The amount of cooling water required to remove the core decay heat per MW(t) of reactor output is shown in Fig. 2. The assumptions were that heat removal was achieved by water boiloff at a pressure of 1 atm. The rate of heat removal and the cumulative heat removal requirements per MW(t) of reactor output are given in Table 6.2 for various conditions. Table 6.1 lists the SSCs found in Sect. 6.

REFERENCES

1. A. G. Croff, ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1980.

Table 6.1 Structures, Systems, and Components to Remove Core Heat: Function 1.2

Description Title	Function	Location of Description	Page
Process Inherent Ultimate Safety (PIUS) Reactor Technology	1.2	Chapter 5	5-4
Passively Safe Pressurized-Water Reactor with Gas Bubble	1.2	Chapter 5	5-13
Passive Control of Power Levels by Variable Boron Concentration in Pressurized-Water Reactors (GEYSER)	1.2	Chapter 5	5-17
In-Reactor-Core-Power Fuses	1.2.2	Chapter 5	5-49
High Temperature Ceramics, Clads, and Fuels for LWR	1.2	Chapter 5	5-61
Graphite-Disk UO_2 Fuel Elements for Enhanced Thermal Margin and Reduced Excess Reactivity	1.2	Chapter 5	5-65
Fluidic In-Vessel Emergency Core-Cooling System	1.2	Chapter 6	6-9
Low-Water-Level Initiated, Hydraulic-Valve Operated,Emergency Core Cooling and Shutdown Systems	1.2	Chapter 6	6-15
Main Recirculation Pump Failure Initiated, Core Coolingand Shutdown System	1.2	Chapter 6	6-19
Passive Safety and Shutdown System	1.2	Chapter 6	6-26
Fluidized Bed Pressurized-Water Reactor	1.2	Chapter 6	6-29
Prestress Concrete Reactor Vessel (PCRV) for Light-Water Reactors	1.2.1	Chapter 6	6-32
Metal Pressure Vessel in Pool	1.2	Chapter 6	6-38
Metal Pressure Vessel in Pool	1.2.1	Chapter 6	6-38
Passive Reactor Core-Cooling System Which Can Operate with Primary Pressure Vessel Leak	1.2.1	Chapter 6	6-42
Reactor Pressure Vessel with no Bottom Penetrations	1.2.1	Chapter 6	6-45
Integral PWR with no Primary System Piping	1.2.1	Chapter 6	6-46
Fluidic Diodes	1.2.1	Chapter 6	6-50

Table 6.2. Decay Heat Removal Requirements Per 1 MWth Output of a Reactor

Time after Shutdown		Rate of Heat Removal Requirement			Cumulative Heat Removal Requirement				
Time (seconds)	Time (days)	Decay Power (MWth)	Boil Water (p=1 atm) (m3/hour)	Heat Water 10 C (p= 1 atm) (m3/hour)	Integral Decay Power (MWD)	Boil Water (p=1 atm) (m3)	Boil Water (p=1000 psi) (m3)	Boil Water (p=2000 psi) (m3)	Heat Water 10 C (p= 1 atm) (m3)
0	0	5.775E-02	9.616E-02	5.158E+00	0	0	0	0	0
60	6.94E-04	3.295E-02	5.487E-02	2.943E+00	3.150E-05	1.259E-03	2.429E-03	4.051E-03	6.751E-02
300	3.47E-03	2.399E-02	3.995E-02	2.143E+00	1.106E-04	4.419E-03	8.530E-03	1.422E-02	2.370E-01
600	6.94E-03	2.084E-02	3.469E-02	1.861E+00	1.884E-04	7.529E-03	1.453E-02	2.423E-02	4.039E-01
1200	1.39E-02	1.765E-02	2.940E-02	1.577E+00	3.221E-04	1.287E-02	2.484E-02	4.142E-02	6.903E-01
2400	2.78E-02	1.441E-02	2.399E-02	[illegible]	5.447E-04	2.177E-02	4.202E-02	7.006E-02	1.168E+00
3600	4.17E-02	1.265E-02	2.106E-02	1.130E+00	7.326E-04	2.928E-02	5.651E-02	9.423E-02	1.570E+00
14400	1.67E-01	8.347E-03	1.390E-02	7.454E-01	2.045E-03	8.171E-02	1.577E-01	2.630E-01	4.383E+00
28800	3.33E-01	6.948E-03	1.157E-02	6.205E-01	3.319E-03	1.326E-01	2.560E-01	4.269E-01	7.115E+00
43200	5.00E-01	6.233E-03	1.038E-02	5.567E-01	4.418E-03	1.765E-01	3.408E-01	5.682E-01	9.469E+00
86400	1	5.110E-03	8.508E-03	4.564E-01	7.253E-03	2.898E-01	5.595E-01	9.329E-01	1.555E+01
172800	2	4.110E-03	6.844E-03	3.671E-01	1.186E-02	4.741E-01	9.151E-01	1.526E+00	2.543E+01
259200	3	3.562E-03	5.931E-03	3.181E-01	1.570E-02	6.274E-01	1.211E+00	2.019E+00	3.365E+01
345600	4	3.190E-03	5.311E-03	2.849E-01	1.908E-02	7.623E-01	1.471E+00	2.453E+00	4.089E+01
432000	5	2.909E-03	4.844E-03	2.598E-01	2.212E-02	8.841E-01	1.707E+00	2.846E+00	4.742E+01
518400	6	2.689E-03	4.477E-03	2.402E-01	2.492E-02	9.960E-01	1.922E+00	3.208E+00	5.342E+01
604800	7	2.510E-03	4.180E-03	2.242E-01	2.752E-02	1.100E+00	2.123E+00	3.540E+00	5.900E+01
691200	8	2.363E-03	3.934E-03	2.110E-01	2.996E-02	1.197E+00	2.311E+00	3.853E+00	6.422E+01
777600	9	2.238E-03	3.726E-03	1.999E-01	3.226E-02	1.289E+00	2.488E+00	4.149E+00	6.915E+01
864000	10	2.132E-03	3.549E-03	1.904E-01	3.445E-02	1.376E+00	2.657E+00	4.430E+00	7.383E+01
950400	11	2.040E-03	3.396E-03	1.822E-01	3.653E-02	1.460E+00	2.818E+00	4.698E+00	7.830E+01
1036800	12	1.959E-03	3.262E-03	1.750E-01	3.853E-02	1.540E+00	2.972E+00	4.956E+00	8.259E+01
1123200	13	1.888E-03	3.143E-03	1.686E-01	4.045E-02	1.617E+00	3.120E+00	5.203E+00	8.671E+01
1209600	14	1.824E-03	3.037E-03	1.629E-01	4.231E-02	1.691E+00	3.264E+00	5.442E+00	9.069E+01
1814400	21	1.506E-03	2.508E-03	1.345E-01	5.396E-02	2.156E+00	4.163E+00	6.941E+00	1.157E+02
2419200	28	1.305E-03	2.173E-03	1.166E-01	6.380E-02	2.550E+00	4.922E+00	8.206E+00	1.368E+02
3024000	35	1.181E-03	1.933E-03	1.037E-01	7.244E-02	2.895E+00	5.567E+00	9.316E+00	1.553E+02
3628800	42	1.050E-03	1.749E-03	9.382E-02	8.017E-02	3.204E+00	6.184E+00	1.031E+01	1.719E+02
4233600	49	9.631E-04	1.604E-03	8.601E-02	8.722E-02	3.485E+00	6.728E+00	1.122E+01	1.870E+02
4838400	56	8.921E-04	1.485E-03	7.968E-02	9.371E-02	3.745E+00	7.229E+00	1.205E+01	2.009E+02
5443200	63	8.326E-04	1.386E-03	7.436E-02	9.975E-02	3.986E+00	7.394E+00	1.283E+01	2.138E+02
6048000	70	7.816E-04	1.301E-03	6.981E-02	1.054E-01	4.212E+00	8.130E+00	1.356E+01	2.259E+02
6652800	77	7.370E-04	1.227E-03	6.583E-02	1.107E-01	4.424E+00	8.540E+00	1.424E+01	2.373E+02
7257600	84	6.977E-04	1.162E-03	6.232E-02	1.157E-01	4.625E+00	8.927E+00	1.489E+01	2.481E+02
15779000	182.627315	3.861E-04	6.429E-04	3.448E-02	1.692E-01	6.761E+00	1.305E+01	2.176E+01	3.626E+02
31558000	365.25463	2.007E-04	3.342E-04	1.793E-02	2.228E-01	8.902E+00	1.718E+01	2.865E+01	4.775E+02

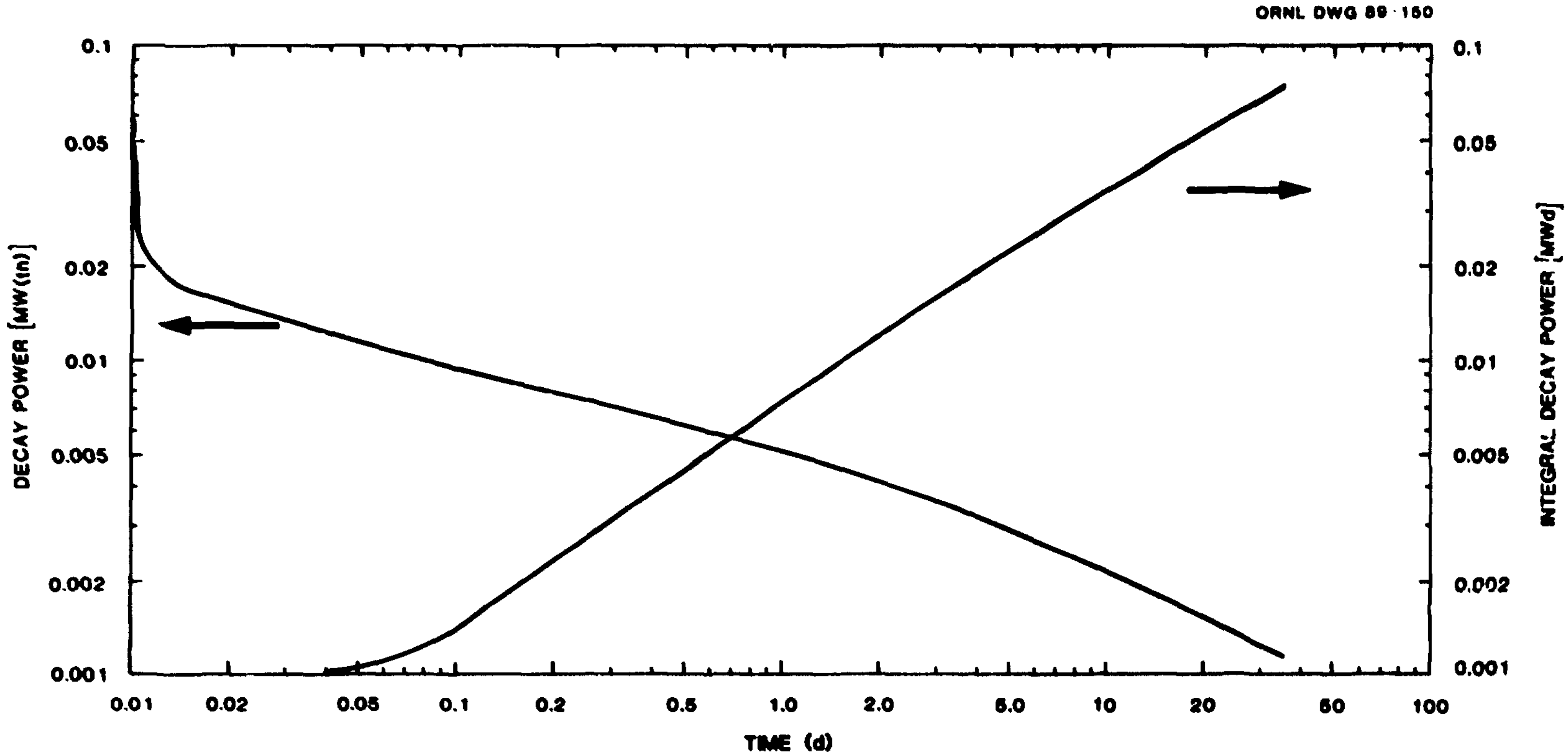

Fig. 1. Decay power and integral decay power as a function of time.

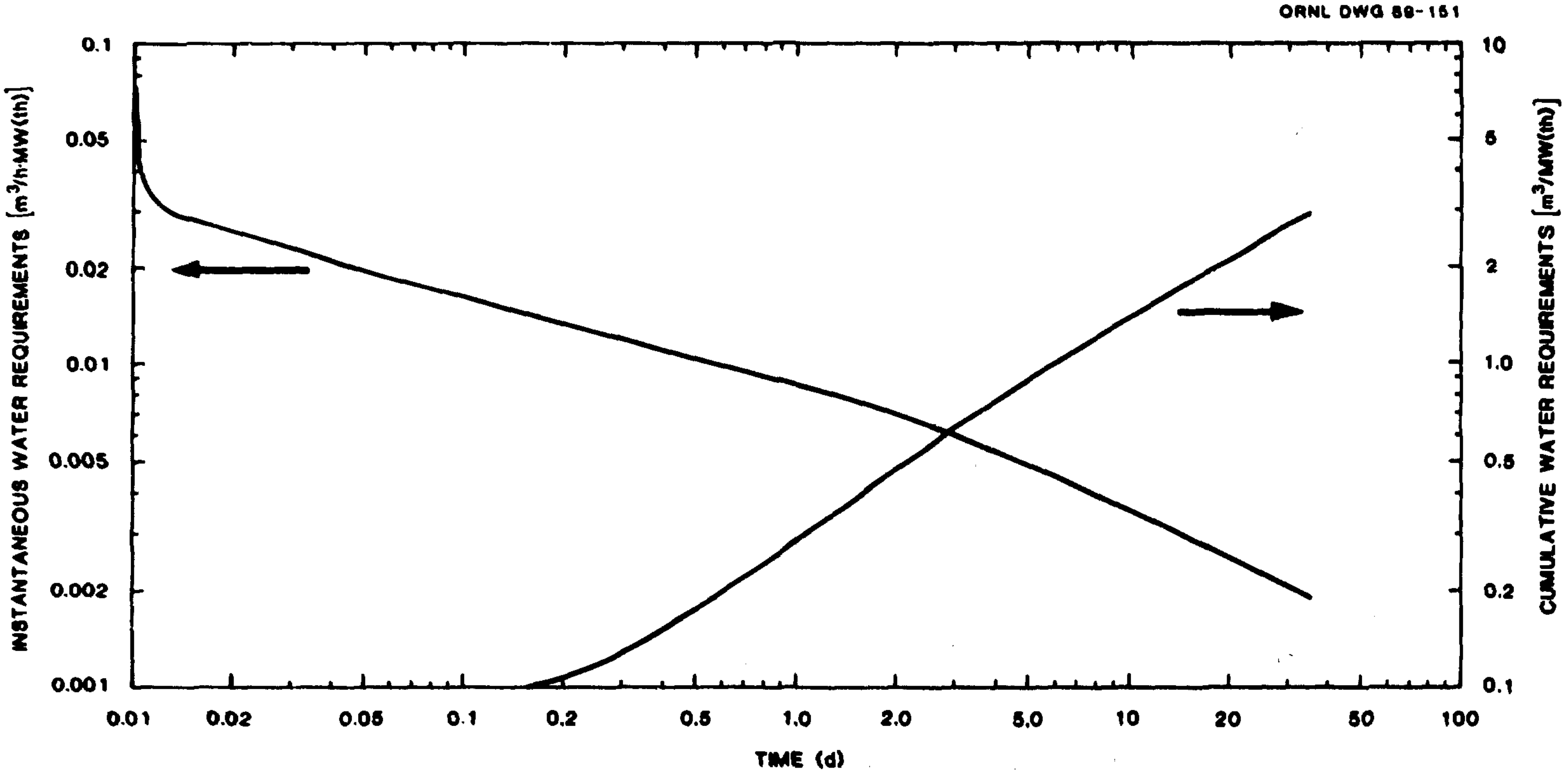

Fig. 2. Cumulative cooling water requirements for decay heat removal as a function of time.

TITLE: FLUIDIC IN-VESSEL EMERGENCY CORE-COOLING SYSTEM

Functional Requirement: Level 1.2: Remove Core Heat
Level 1.1.1: Provide Process Shutdown Mechanism

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Light-Water Reactor (LWR)

Organization: National Laboratory (Oak Ridge National Laboratory)

Examples of Implementation: None

Description: The Fluidic In-Vessel Emergency Core Cooling System (FIVES) is a key component [1-3] of a reactor concept called the Process Inherent Ultimate Safety boiling water reactor (PIUS BWR). FIVES can, however, be used as an emergency core-cooling system for either a BWR or a pressurized-water reactor (PWR).

In the PIUS/BWR, the conventional BWR system is placed within a very large prestress concrete reactor vessel (PCRV) along with a supply of borated emergency core-cooling water (see Fig. 1). The reactor system and borated water supply are at the same pressure. FIVES includes the mechanisms to allow the borated water to enter the reactor core and shut it down and cool it in an emergency. The in-vessel borated water supply is sufficiently large to cool the reactor core by boil-off of water for a period of one week after an accident.

As shown in Fig. 1 the FIVES has three major components (1) a large volume of cool borated water at reactor pressure (2) a fluidic valve assembly (see Fig. 2) 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 (steel or prestress concrete reactor 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 in this location than the pressure of the reactor coolant because it is more dense. If the fluidic valve opens, the core is flooded with borated water, which shuts down the reactor and cools it. The fluidic valve assembly contains no moving parts. It is designed to remain closed only as long as it receives a continuous flow of water from the FIVES water pump system.

Protection against a low water level in the reactor vessel is provided by positioning the FIVES water pump system high above the reactor core (Fig. 1). 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 the 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; in other words, a fail-safe sensor.

The central component of FIVES is the vortex fluidic valve assembly (see Fig. 2), which is a modified vortex fluidic amplifier operated as a valve. This 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 the output of the FIVES water pump, these pressures can be made to match the pressures of the two water zones. In effect, a valve exists that uses the dynamic forces of water, rather than pieces of metal, to prevent flow through the valve.

Below the fluidic valve, hot/cold water interface zone 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 interface, so the exits to the reactor coolant cleanup systems (not shown) are located within the fluidic valve.

Alternative Versions: The basic concept of FIVES provides reactor core cooling until the volume of pool water has evaporated. For typical designs, a one-week supply of water is provided inside the pressure vessel. This cooling time can be extended indefinitely by use of any one of several long-term options to cool the borated water and, hence, the reactor core. The one-week grace period provided by the borated water evaporation implies that radioactive decay heat from the reactor core will have decreased significantly in this time period. In practice, heat leaks through the insulation from the hot primary system to the cool borated water during normal operation; therefore, borated water coolers are needed for normal operation and can be used in an emergency.

There are multiple long-term operations for cooling the borated water (see also SSC: Heat Pipe Cooling for Reactor Containment or Decay-Heat Cooling Systems), including use of natural circulation air coolers (see Fig. 3). With this option[2,3], the borated water coolers, contain a fluid such as ammonia. The ammonia boils in the coolers removing heat from the borated water. The vapor state ammonia flows to an air cooler, is condensed, and the liquid flows back to the borated water coolers. The air coolers can operate either with fans or as natural air circulation units.

During normal operations, the borated water is kept near 50°C. In emergency conditions, the borated water will be at the boiling point of water (100°C) if the reactor is fully depressurized. Air cooler performance improves rapidly with higher-temperature operation. In a number of proposed designs[2], requirements for cooling borated water during normal operations controlled the designs. For example, an air cooler requiring active fans for normal borated water cooling could handle higher-temperature decay-heat cooling requirements with air cooler fans off.

These auxiliary cooling methods provide protection essentially forever; however, such cooling techniques are not as reliable against natural or man-made assault as is the basic FIVES system.

Status of Technology: Preliminary engineering calculations have been done for FIVES. The fluidic valve technology is not totally new. Some types of fluidic valves have been developed for nuclear fuel reprocessing plants.

Advantages: Totally passive technology

Additional Requirements: None

Comments: None

References/Contacts:

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

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

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

Update/Compiler: Jan. 1989/CWF

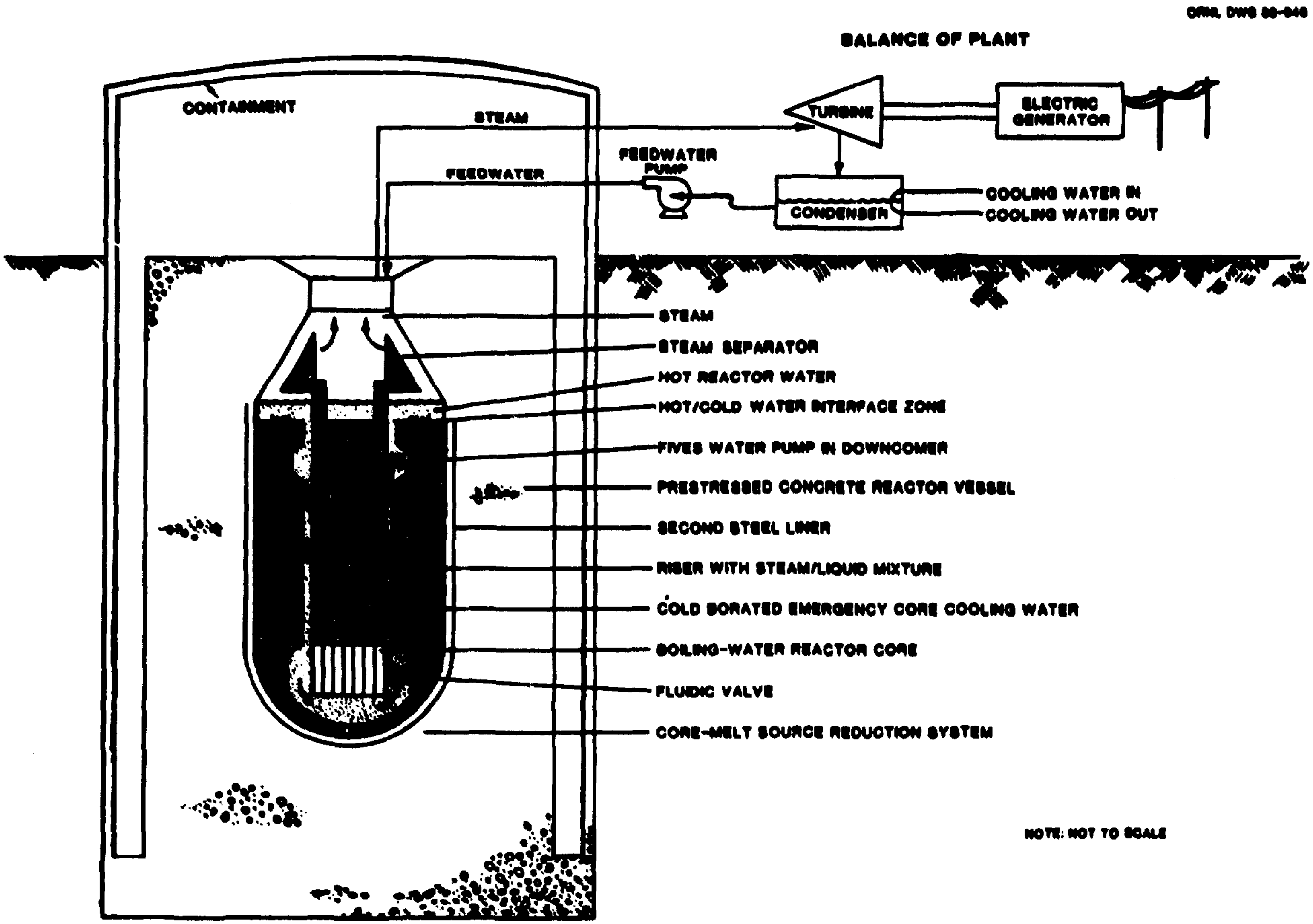

Fig. 1. Schematic of PIUS/BWR.

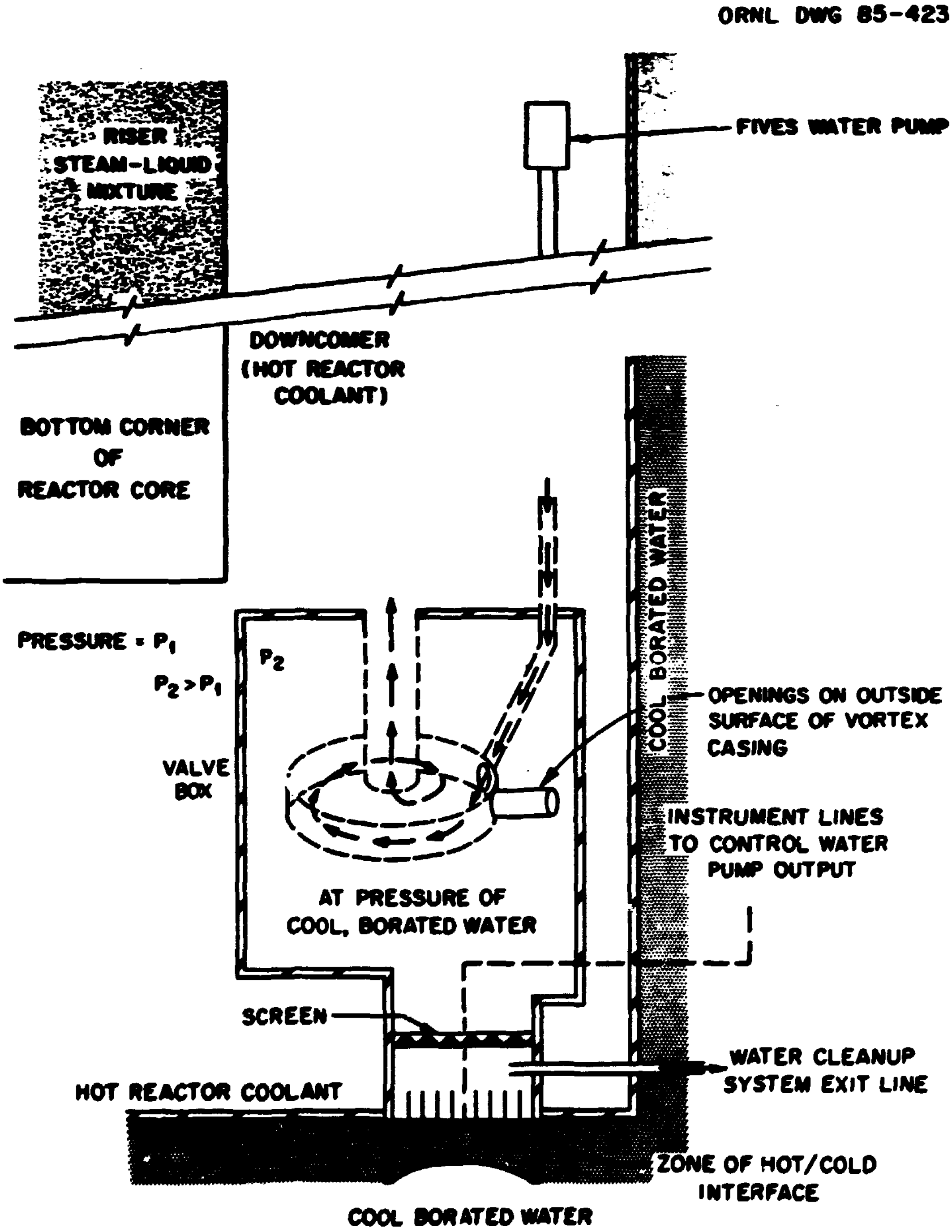

Fig. 2. Vortex fluidic valve assembly.

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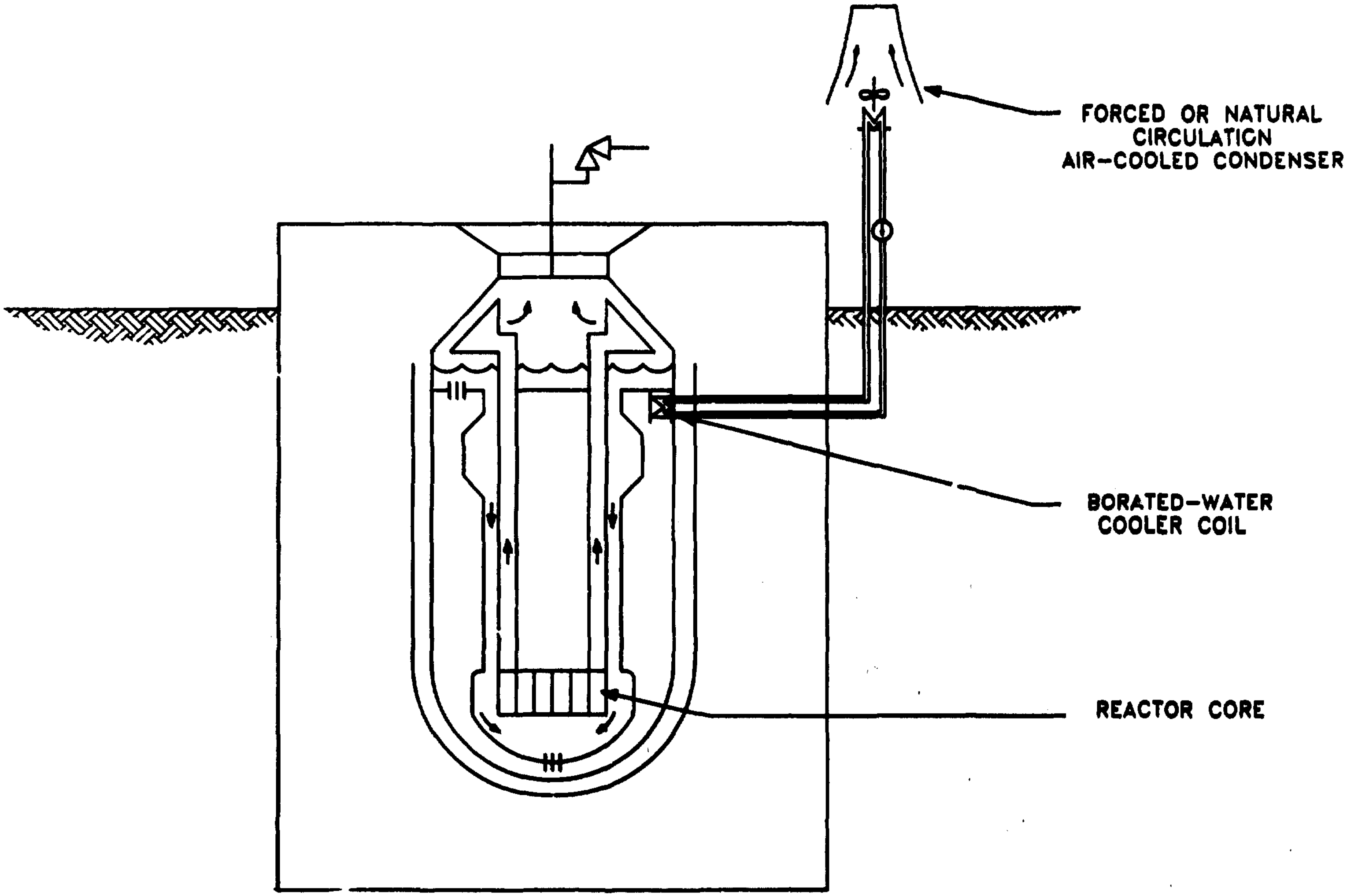

Fig. 3. Air cooling system for PIUS/BWR.

TITLE: LOW-WATER-LEVEL INITIATED, HYDRAULIC-VALVE OPERATED, EMERGENCY CORE COOLING AND SHUTDOWN SYSTEMS

Functional Requirements: Level 1.1.1 Provide Process Shutdown Mechanism
Level 1.2 Remove Core Heat

Safety Type: Passive/Active

Developmental Status: 4 Analyzed

Reactor Type: Boiling-water reactor (BWR)

Organization: Vendor (Asea Brown Boveri)

Examples of Implementation: None

Description: The emergency core-cooling system (ECCS) consists of a volume of ECCS water above the reactor core and pressure vessel that is connected to the reactor pressure vessel through special hydraulic valves (Fig. 1a). In an emergency, the valves open, the reactor is depressurized, and the core is flooded with water. The key components in these systems are the hydraulic valves. The valves are designed to remain closed only if they receive a continuous flow of high-pressure water. The pump providing the high-pressure water is above the reactor core. When the water level in the reactor vessel falls below a minimum level, the water-level sensing pump fails and delivers no water to the ECCS valves. The valves in the emergency cooling pipes then open, allowing steam to flow from the reactor vessel and condense in the pool (Fig. 1b). This depressurizes the reactor. When the difference in static pressure of the pool between the upper and lower emergency cooling pipes is greater than the pressure drop of the steam flux passing through the upper emergency cooling pipe, water flows from the pool into the reactor vessel. Emergency cooling is then achieved by natural circulation between the reactor core and the pool.

The valve used in the ECCS pipes is connected to a piston normally held closed by hydraulic pressure from a pump supplied with water from the reactor vessel (Fig. 2a). When the cooling water level falls below the water inlet to the pump, the pump fails and shuts down, allowing the hydraulic piston to retract and open the valve (Fig. 2b). The inlet of the pipe transporting water from the reactor pressure vessel to the pump can be located near the valve or at a different vessel elevation than the valve.

Alternative Versions: None

Status of Technology: This system was studied in the mid-1980s by Asea Brown Boveri (ABB). With the development of the Process Inherent Ultimate Safety (PIUS) reactor concept, ABB concentrated its development program on PIUS.

Advantages:
1. Emergency cooling is achieved without the use of delivery pumps, motors, or diesel generator systems.
2. Although mechanical operation of equipment is necessary for reactor operation, the safety performance of the system operates passively.
3. If the ECCS water is borated, its entry into the reactor will assure long-term reactor shutdown.

Additional Requirements: None

Comments: In the event of a coolant leak in the pressure vessel and subsequent reduction of the pressure vessel water level, this mechanism will depressurize the reactor. As depressurization progresses, less coolant will be forced out the leak and such a leak may eventually stop.

References/Contacts:

1. K. Hannerz, Boiling Reactor, U.S. Patent Number 4,363,780 (Dec. 14, 1982).

Update/Compiler: Feb. 1989/EBL

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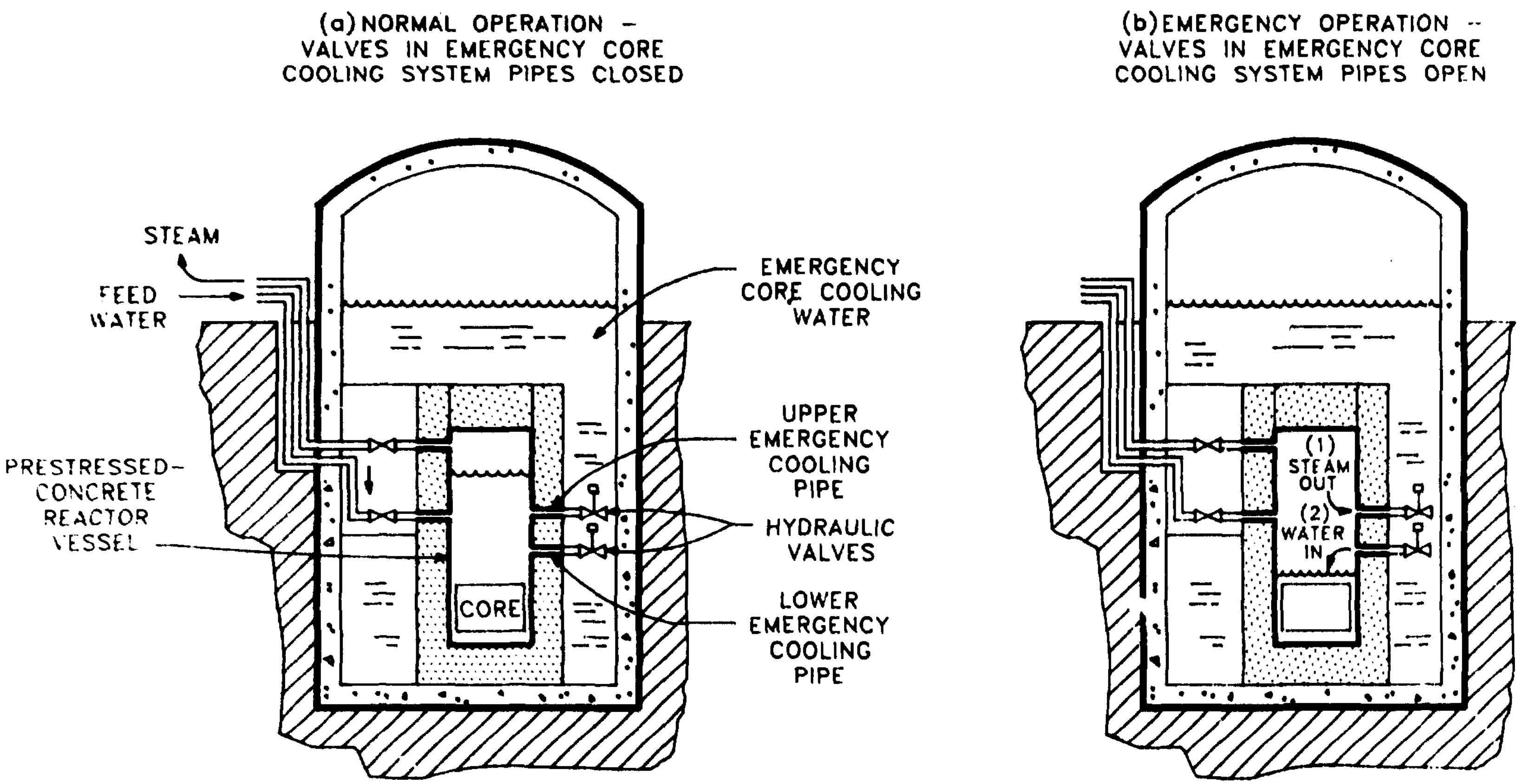

Fig. 1. Hannerz emergency core cooling system applied to boiling-water reactor.

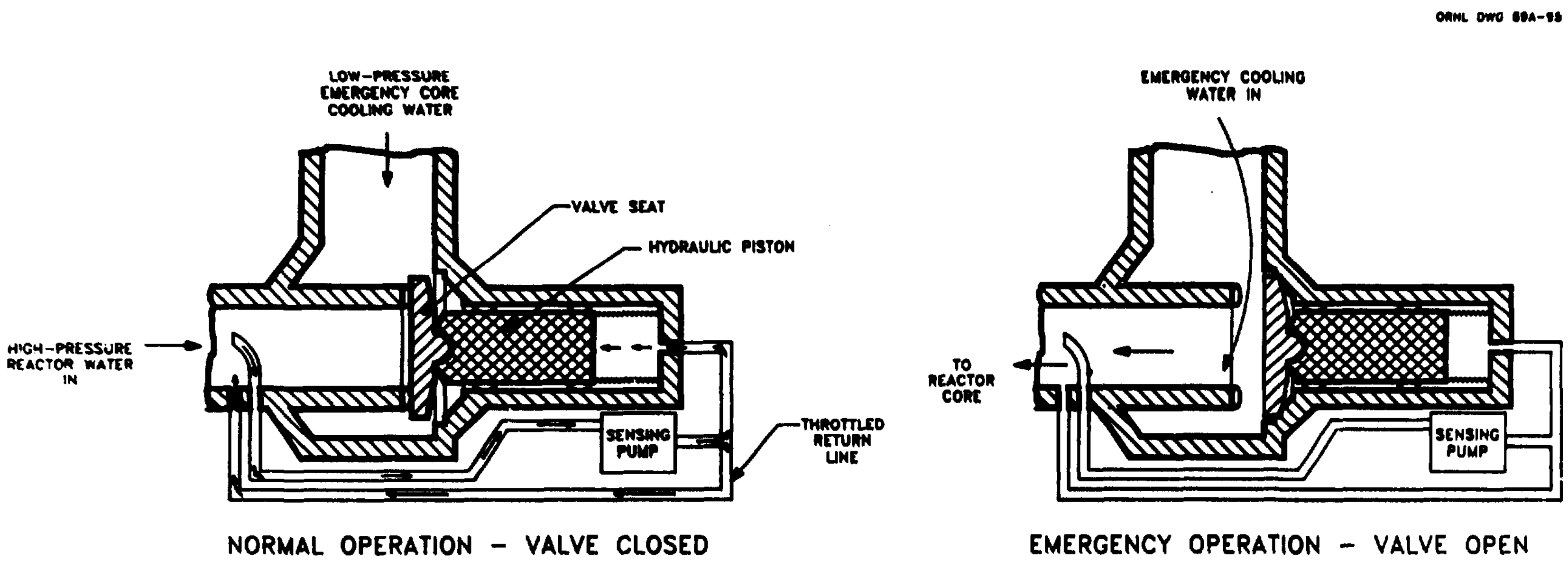

Fig. 2. Hannerz high-pressure ECCS valve system.

TITLE: MAIN RECIRCULATION PUMP FAILURE INITIATED, CORE COOLING AND SHUTDOWN SYSTEMS

Functional Requirements: Level 1.1.1 Provide Process Shutdown Mechanism
Level 1.2 Remove Core Heat

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Pressurized-water reactor (PWR)

Organization: National Laboratory (Japan Atomic Energy Research Institute) and University (University of Rome [Italy]).

Examples of Implementation: None

Description: The shutdown of the primary recirculation pumps in a PWR can be used as a mechanism to trigger into operation various passive emergency core cooling systems, passive decay heat removal systems, or shutdown systems. Primary recirculation pump shutdown can be initiated by control systems or failure such as caused by low reactor water levels and steam in the primary circuit. The safety mechanisms herein are initiated by the change in relative pressure throughout the primary circuit on pump shutdown.

1. SPWR System - The Japanese System-Integrated Pressurized-Water Reactor (SPWR) is similar in some respects to the Process Inherent Ultimate Safety (PIUS) concept (see Process Inherent Ultimate Safety (PIUS) Technology). The reactor core (Fig. 1) is immersed in water of a low boron concentration and is connected through a hot/cold interface zone underneath the core to a poison tank of cool, highly borated water. The cool, highly borated water serves two functions: (1) reactor shutdown and (2) emergency reactor core cooling. The cooler high-boron water does not mix with the hot, lower-density, low-boron reactor coolant at the hot/cold water interface zone. The top of the neutron absorber tank is connected to the reactor coolant volume above the reactor core by passive hydraulic pressure valves (Fig. 2). When the valves open, water or steam enters the top of the neutron absorber tank and cool, borated water exits the bottom of the poison tank and enters the reactor core and it shuts down. Cooling is achieved by boiling off the borated water. If necessary, a separate set of actively controlled valves can also be used to deliver the poison to the reactor core via pumps.

   The valve to the emergency borated water supply is controlled by a hydraulic piston normally held closed by the force of water from the main reactor recirculation pumps (Figs. 2a and 2b). The large clearance between the piston and the cylinder wall allows some water leakage around the piston. This prevents the possibility of the piston binding against the wall. The piston is attached by a connecting rod to a weight that is sized to overcome a minimum delivery pressure from the main reactor recirculation pump. If a low water level to the pump causes the pump delivery pressure to fall below the minimum required to overcome the weight, the weight pulls open the valve, allowing regular reactor coolant or steam into the top of the poison tank and cool, borated water out of the bottom of the poison tank to the reactor core (Fig. 4b).

2. MARS System - The proposed Multipurpose Advanced Reactor inherently Safe (MARS) pressurized-water reactor has a passive decay heat removal system (2,3) which is initiated upon pump failure. The system has three components:

   - The primary system has a natural circulation decay heat piping system from reactor vessel to a heat exchanger above the reactor core. The piping loop contains a one way valve and is connected to the inlet (higher pressure) and outlet (lower pressure) of the reactor pressure vessel. When the main circulation pump operates, there is no flow through this loop because flow is stopped by the one way valve.

   - A natural circulation heat transfer loop transfers heat from the heat exchanger with primary fluid flow to an external heat sink.

   - The natural circulation heat transfer loop dumps the heat to an external heat sink. There are multiple options for external heat sinks. The MARS concept uses a boiling-water bath where heat is rejected by boiling water (see: Condensation of Pressurized Steam or Cooling Hot Water with Boiling Water Bath). Some of the water is condensed in an air cooler and returned to the water bath. Exiting steam handles high decay heat loads immediately after reactor shutdown while the air cooled steam condenser handles long-term decay heat cooling loads.

   In the event of pump shutdown, the pressure in the primary circuit equalizes and a natural circulation flow of primary water or steam begins between the reactor core and lower heat exchanger in the natural circulation heat transfer loop. The primary system fluid flow is in the opposite direction expected during normal operation with the main pumps. The system passively dumps decay heat to the environment.

   There are multiple design options including:

   - The decay heat piping system may be loaded with borated water. In this case, pump failure results in borated water injection into the reactor core and reactor shutdown.

   - The one way valve can be a no-moving parts fluidic valve (see: Fluidic Diodes). This results in some movement of water through the primary system natural circulation decay heat piping system with some loss of heat during normal operation but allows for a totally passive system with a no-moving-part one way valve.

<u>Alternative Versions</u>: None

<u>Status of Technology</u>: 1. Japanese System - A preliminary design study has been carried out and laboratory-scale development is underway.

2. Italian System - Preliminary studies have been initiated.

<u>Advantages</u>: 1. Emergency cooling is achieved without the use of delivery pumps, motors, or diesel generator systems.

2. Although mechanical operation of equipment is necessary for reactor operation, the safety performance of the system operates passively.

Additional Requirements: None

Comments: None

References/Contacts:

1. K. Sako, "Conceptual Design of SPWR," American Nuclear Society Topical Meeting on Safety of Next Generation Power Reactors, Seattle, Washington (May 1988).

2. Department of Energetics of La Sa enza University of Rome (Italy), Draft: Multipurpose Advanced Reactor Inherently Safe (July 1989).

3. M. Caira, M. Cumo, and A. Naviglio, "MARS Reactor: A Proven PWR Technology Combined with Advanced, Safety Requirements," Energia Nucleare 23 (May/Sept. 1987)

Update/Compiler: Oct. 1989/CWF

Table 1 Major Design Parameters for the Japanese System-Integrated Pressurized-Water Reactor (SPWR)

Reactor	
Thermal power	1,100 MW(t)
Coolant inlet/outlet temperature	280/310° C
Coolant flow rate	24,000 t/h
Core outlet pressure	13 MPa
Core pressure drop	0.035 MPa
Steam generator pressure drop	0.18 MPa
Reactor pressure vessel inside diameter	6.6 m
Core	
Equivalent core diameter/height	2.89/2.0 m
Core average power density	84 $MW(t)/m^3$
Main circulating pump (one unit)	
Flow rate	26,000 t/h
Delivery pressure	0.23 MPa
Rotating speed	600 rpm
Diameter of impeller	1.5 m
Steam generator	
Steam temperature/pressure	285°C/5 MPa
Feed water temperature	210°C
Steam flow rate	2,000 t/h
Steam generator inner/outer diameter	3.2/6.1 m
Steam generator height	7.6 m
Poison injection system	
Natural boron content	>4,000 ppm
Poison temperature	150 ± 10°C
Initial driving force	0.031 MPa
Number of hydraulic pressure valves	3
Number of rapid-opening valves	3
Hydraulic pressure valves	
Valve Port	200 mm
Cylinder diameter	300 mm
Piston diameter	290 mm
Annular space gap diameter	10 mm
Estimated leakage flow	50 l/s
Percent of rated main circulating pump flow required for closure	40%
Attached weight	170 kg
Lifting force supplied to piston at full power operation	1.26 t

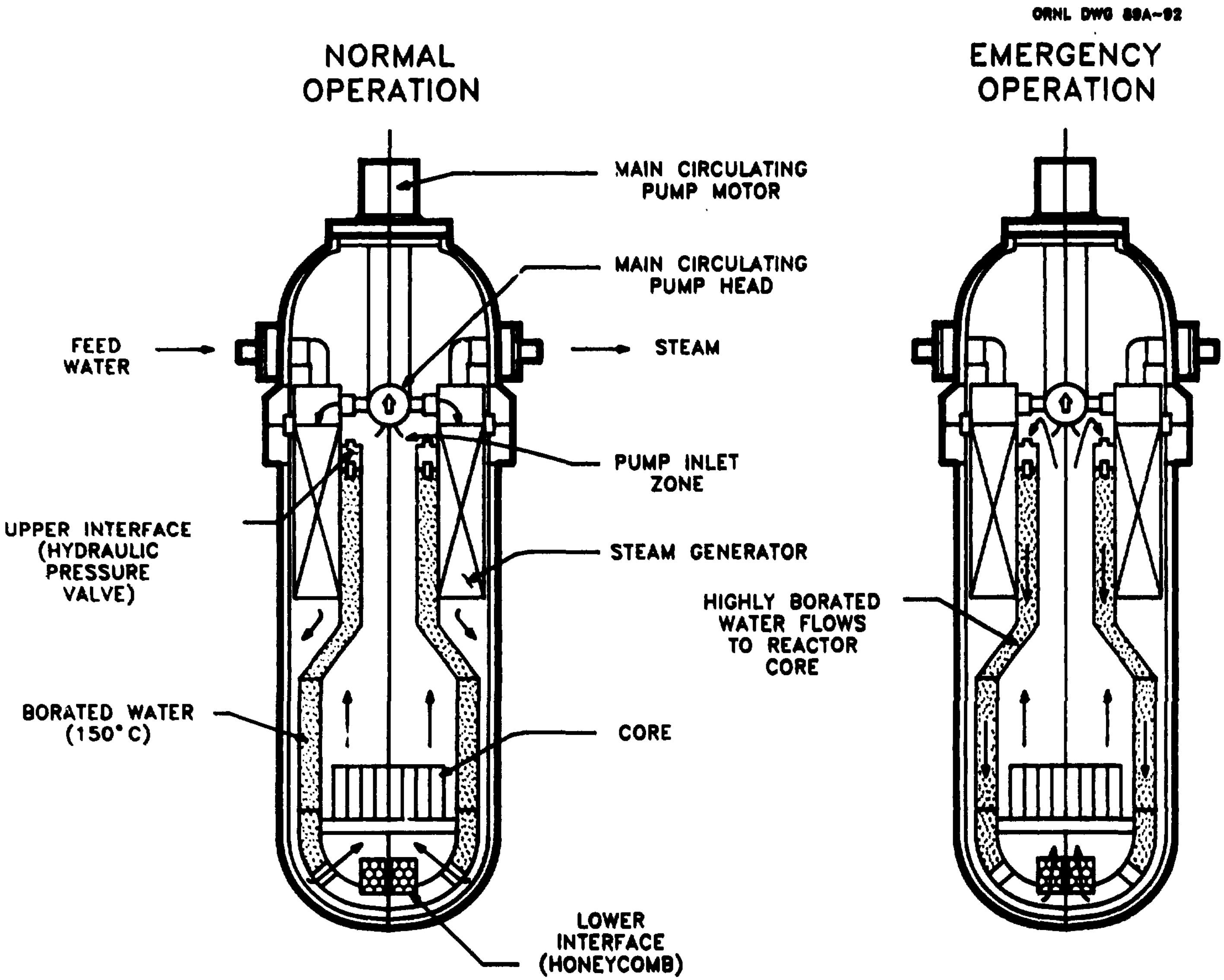

Fig. 1. Japanese system-integrated pressurized-water reactor (SPWR).

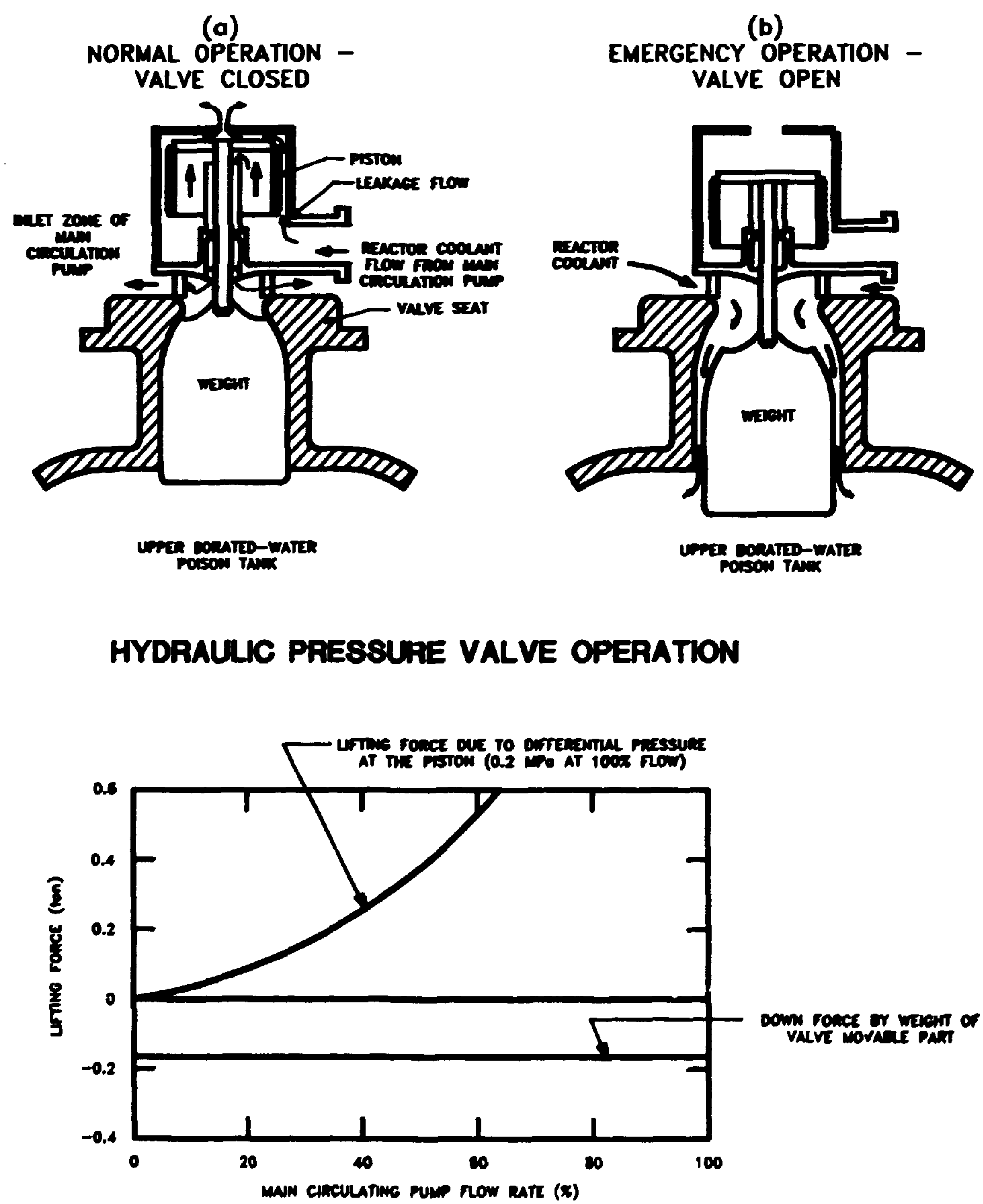

Fig. 2. Characteristics of hydraulic pressure valve.

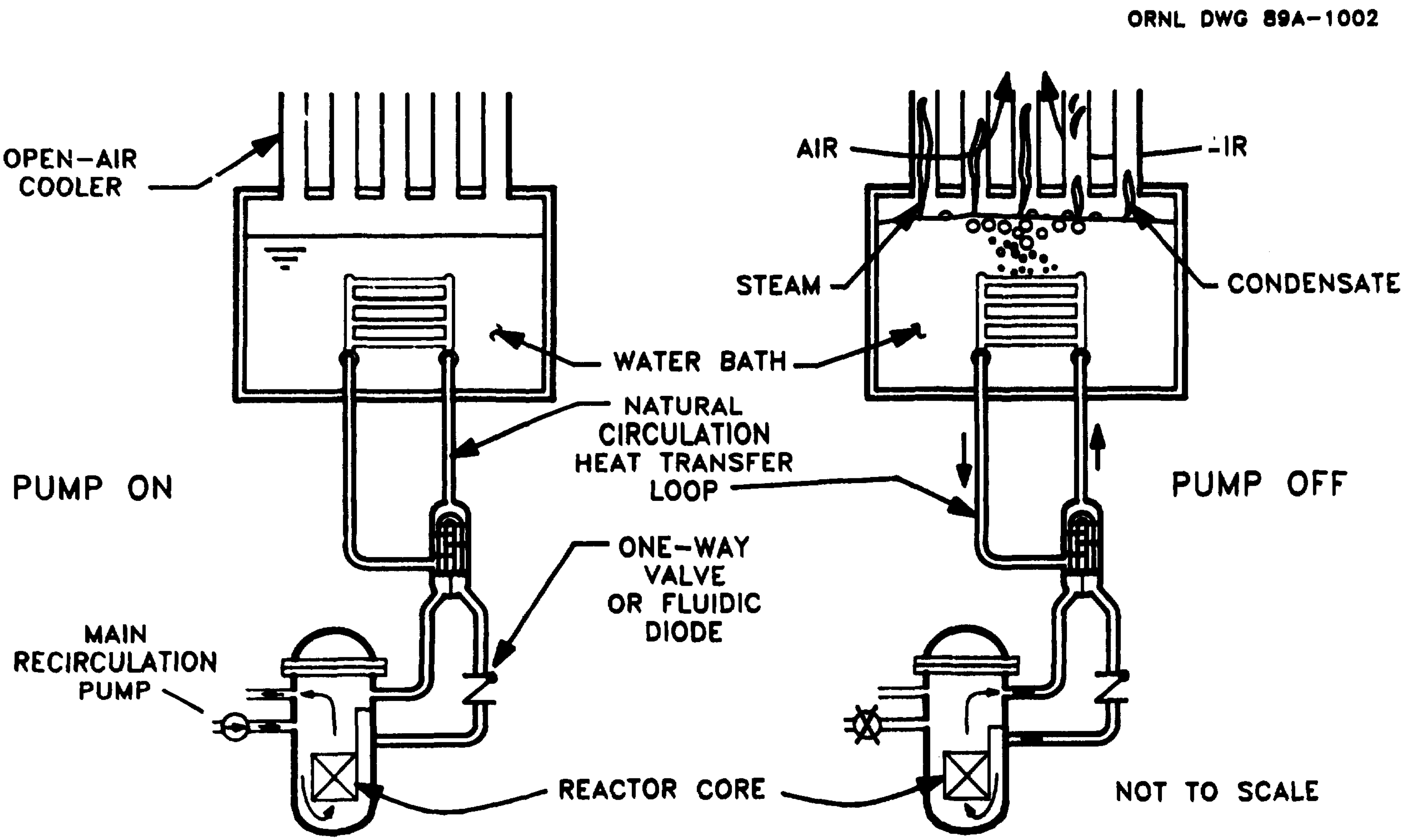

Fig. 3. Schematic of pump-failure-initiated decay heat removal system with air cooler for pressurized water reactor.

TITLE: PASSIVE SAFETY AND SHUTDOWN SYSTEM

Functional Requirements: Level 1.2: Remove Core Heat
Level 1.1.1: Provide Process Shutdown Mechanism

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Pressurized-water reactor (PWR)

Organization: National Laboratory (Japan Atomic Energy Research Institute)

Examples of Implementation: None

Description: A passive safety and shutdown system (PSSS) can act as a passive emergency core-cooling system (ECCS) and a reactor shutdown mechanism during postulated severe transients or reactor accidents. The primary function of a PSSS is to introduce additional coolant into the reactor coolant system (RCS) after a severe reactor transient.

The PSSS consists of a tank of cold water that is connected between the hot and cold legs of the primary coolant loop(s) of a PWR as shown in Fig. 1. During normal reactor operation the coolant inside the PSSS is hydraulically isolated from the rest of the RCS. Only during a severe reactor transient will the stagnant PSSS coolant start to flow from the hot leg to the cold leg. The hydraulic isolation of the PSSS is achieved by controlling the head and loss coefficient of regulating pumps A and B in the cold leg of the primary coolant loop (as shown in Fig. 1). By carefully controlling these centrifugal pumps, the pressure drop across the PSSS (from point 1 to point 2 in Fig. 1) can be eliminated, resulting in a stagnant flow condition. The coolant may contain neutron absorbers that act to passively shut down the reactor.

Application of the PSSS to enhance PWR safety during a postulated accident has been studied[1] using the THYDE-W Analysis Code[2, 3]. The analysis modeled a typical nuclear steam supply system (NSSS) of a four-loop 1100-MW(e) PWR. The PSSS in this model contained as much coolant as is found in a typical NSSS accumulator. The neutron absorber in the PSSS was boron at a concentration of 5000 ppm.

Calculations with and without a PSSS in the Nuclear Steam Supply System (NSSS) showed that the PSSS enhanced passive reactor safety during an anticipated transient without scram (ATWS) under loss of normal ac power. The conservative assumptions were complete loss of feedwater and main steam flow in 5 seconds, complete coast down of all pumps in 15 seconds, and insertion failure of the scram rods. The results indicate that the PSSS adequately compensated for the ATWS without operator intervention or the use of other ECCS components. Small-and large-break loss-of-coolant accidents (LOCA) were also analyzed, with the results showing an increase in the walkaway period with the PSSS in the system

It has been suggested[1] that the PSSS could possibly simplify existing reactor designs by allowing certain components to be removed from the RCS. Such components may include the scram rods. the high-pressure injection system, or the accumulators. None of these components was required to halt the postulated transients when a PSSS was in the modeled systems.

Alternative Versions: None

Status of Technology: Preliminary modeling of this concept has shown how the PSSS could act as a passive ECCS in a PWR during an ATWS under loss of normal ac power.

Advantages:
1. Maintains core cooling during an ATWS accident.
2. Increases the walkaway period of a PWR during both a small- and large-break LOCA.
3. Provides a passive reactor shutdown mechanism during anticipated transients.
4. The device is passive in operation.

Additional Requirements: Obtaining hydraulic isolation of the PSSS during all normal operating procedures may require the use of other passive systems. The hot and cold leg interfaces must be vertical to help stabilize and prevent horizontal coolant flow. The interfaces will be disturbed by thermal and molecular diffusion. Baffles also may be used to prevent horizontal flow.

Comments: The technology has some characteristics of the PIUS reactor concept (see: Process Inherent Ultimate Safety Reactor Technology).

References:

1. Yoshiro Asahi, Hiroaki Wakabayashi, Tadashi Watanabe, "Improvement of Passive Safety of Reactors," Nucl. Sci. and Eng., 96, 73-84 (May 1987).

2. Y. Asahi and Y. Suzuki, Journal of Nuclear Science and Technology, 21 (9), 657 (1984).

3. Y. Asahi et al., THYDE-P2:RCS (Reactor-Coolant System) Analysis Code, JAERI-1300, Japan Atomic Energy Research Institute (1986).

Update/Compiler: May 1989/WJR

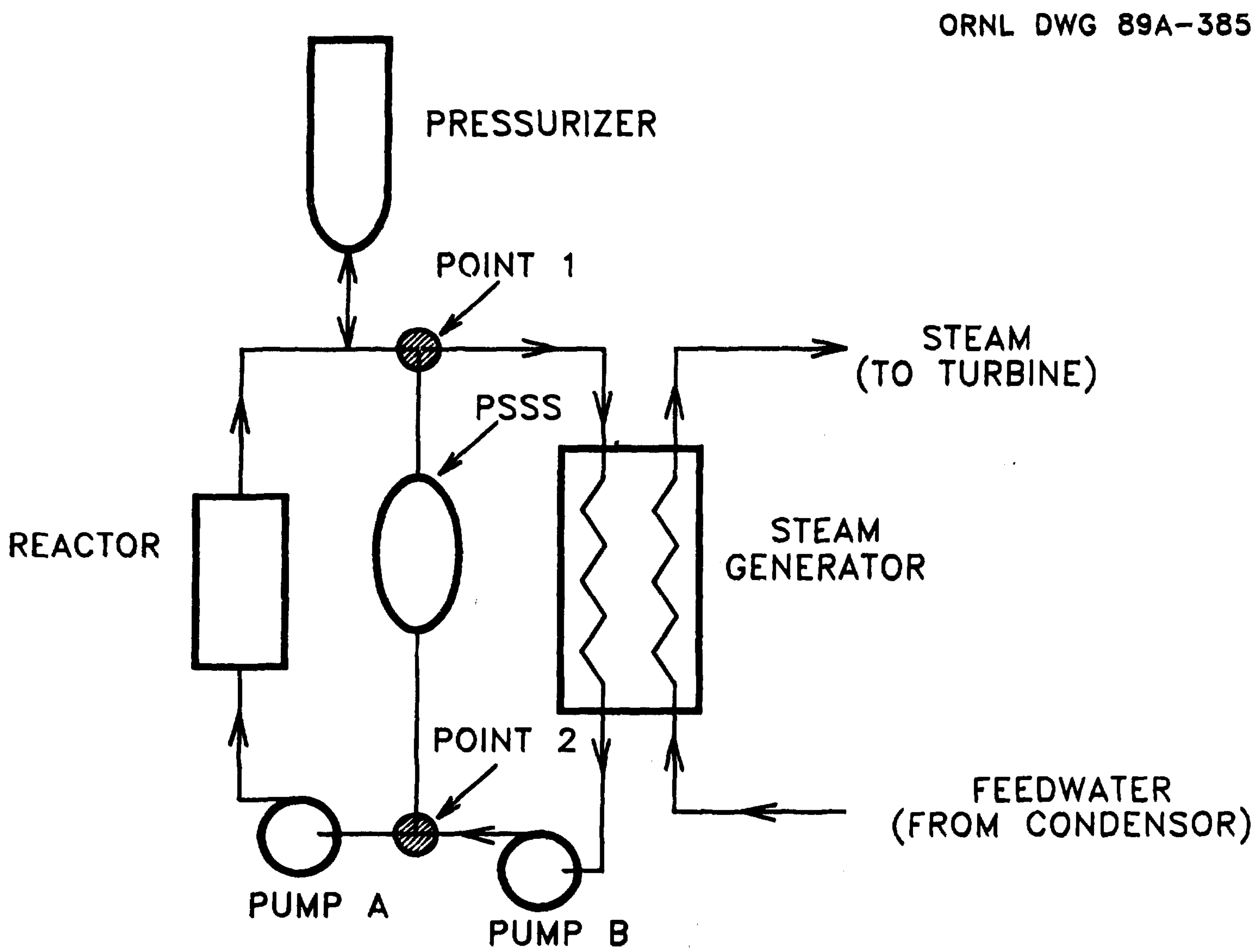

Fig. 1. Passive safety and shutdown system.

TITLE: FLUIDIZED-BED PRESSURIZED-WATER REACTOR

Functional Requirements: Level 1.1: Control Core Heat Generation Rate
Level 1.2: Remove Core Heat

Safety Type: Passive

Developmental Status: 5 Concept

Reactor Type: Pressurized-water reactor (PWR)

Organization: Multiple

Examples of Implementation: None

Description: Several fluidized-bed PWR concepts have been proposed with several passive and inherent safety features that may help maintain core integrity during various postulated reactor accidents. A fluidized-bed reactor is composed of a layer or bed of solid fuel pellets that are "fluidized" or suspended in an upward-flowing liquid such as water. Unlike a gas-cooled pebble bed reactor, the fuel pellets are actually suspended in the liquid moderator during reactor operation.

Control of the fluidized-bed reactor is achieved by controlling the coolant flow rate. Increased coolant flow rate will increase the moderator-to-fuel ratio as the fuel pellets rise from the bed. At first the reactor reactivity will increase. This reactivity increase with increased coolant flow rate eventually reaches a maximum and then the reactivity decreases with increasing flow as the fuel is further diluted with water. This provides an inherent limit on additional reactivity. Physical barriers that limit the active core height can also be employed to control the maximum reactivity. These reactivity control techniques may eliminate the need for burnable absorber and control rods and provide safety against reactivity accidents.

Other unique characteristics of a fluidized-bed reactor include the high heat-transfer rates due to the large surface area of the fuel. The thorough mixing of fuel and moderator results in an even temperature distribution and a large, homogeneous fuel pellet burnup. Fuel loading can occur on-line through access ports in the reactor vessel.

Various reactor designs based on fluidized beds have been proposed including a single fluidized bed and a more modular type of design. Spherical fuel composed of Zircaloy-clad, slightly enriched UO_2 with a pressurized light-water coolant has been proposed, as well as the use of thorium fuel pellets with organic coolants or natural uranium with a heavy-water coolant.

An example of the concept of a fluidized bed reactor is shown in Fig. 1. This proposed reactor uses Zircaloy-clad, slightly enriched UO_2 with light-water coolant. Figure 1 is of a single reactor module. An actual fluidized-bed reactor would be composed of a number of these modules. Each module would produce approximately 5 MW thermal power. Each module has a reactor core, a steam generator, and a coolant pump. The fuel chamber located below the core provides passive reactor safety during an LOCA or pump failure. After a loss of coolant flow, the fuel pellets would settle into the fuel chamber, which is geometrically designed to keep the fuel in a subcritical configuration. A spent fuel pool is located under the reactor modules and is capable of absorbing the decay heat after reactor shutdown. The fuel can also be discharged from the bottom of the modules into the spent fuel pool during refueling or after a loss of coolant flow. Criticality control and enhanced cooling could be achieved by placing fuel chutes or tubes under the fuel discharge valves to direct the fuel pellets into a wider, less critical geometry in the spent fuel pool.

Other fluidized-bed reactor concepts have been proposed that are not modular in design. Physical barriers to limit the active core height, such as the level limiter shown in Fig. 1, are not always used and, instead, the reactor pressure vessel is designed with a flared top so that the resulting decrease in the coolant flow velocity at the top of the core will contain the fuel pellets. Reactivity control is then achieved by controlling the coolant flow rate and by limiting the amount of fuel available so as to achieve a minimal critical mass. Fuel pellets can be added on-line to correct for fuel depletion and fission product accumulation.

Alternative Versions: There are many alternative proposed designs.

Status of Technology: Preliminary studies of fluidized-bed concepts have been performed and have involved studies of fuel and cladding abrasion, experimental verification of heat-transfer and fluid-flow characteristics, reactor control and transient phenomena, and other reactor physics calculations.

Advantages:
1. Passive control of reactivity eliminates need for control rods and burnable absorber. Fluid bed characteristics provide inherent limit to reactivity.
2. Some fluidized bed designs provide for passive emergency-core cooling capabilities.

Additional Requirements: None

Comments: A significant research and development effort would be required to properly assess the viability of these concepts.

References:

1. F. Sefidvash, "A Fluidized-Bed Nuclear Reactor Concept," Nucl. Technol., 71, 527 (Dec. 1985).

2. F. Sefidvash and M. R. Harron, "Preliminary Reactor Physics Calculations of a Fluidized-Bed Nuclear Reactor Concept," Atomkernenergie/Kerntechnik, 35 (3), 191 (1980).

3. F. Sefidvash, "Preliminary Thermal Design Calculations of the Fluidized-Bed Nuclear Power Reactor," Atomkernenergie/Kerntechnik, 41, 45 (1982).

4. F. Sefidvash, "Loss-of-Coolant Accident in the Fluidized-Bed Nuclear Power Reactor," Atomkernenergie/Kerntechnik, 42 (2), 125 (1983).

5. M. R. Scheve, "Liquid Fluidized-Bed Reactor Program," Proc. American Power Conf., Chicago, Illinois, March 1960, Vol. 22, p. 138, Illinois Institute of Technology (1960).

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Update/Compiler: May 1989/WJR

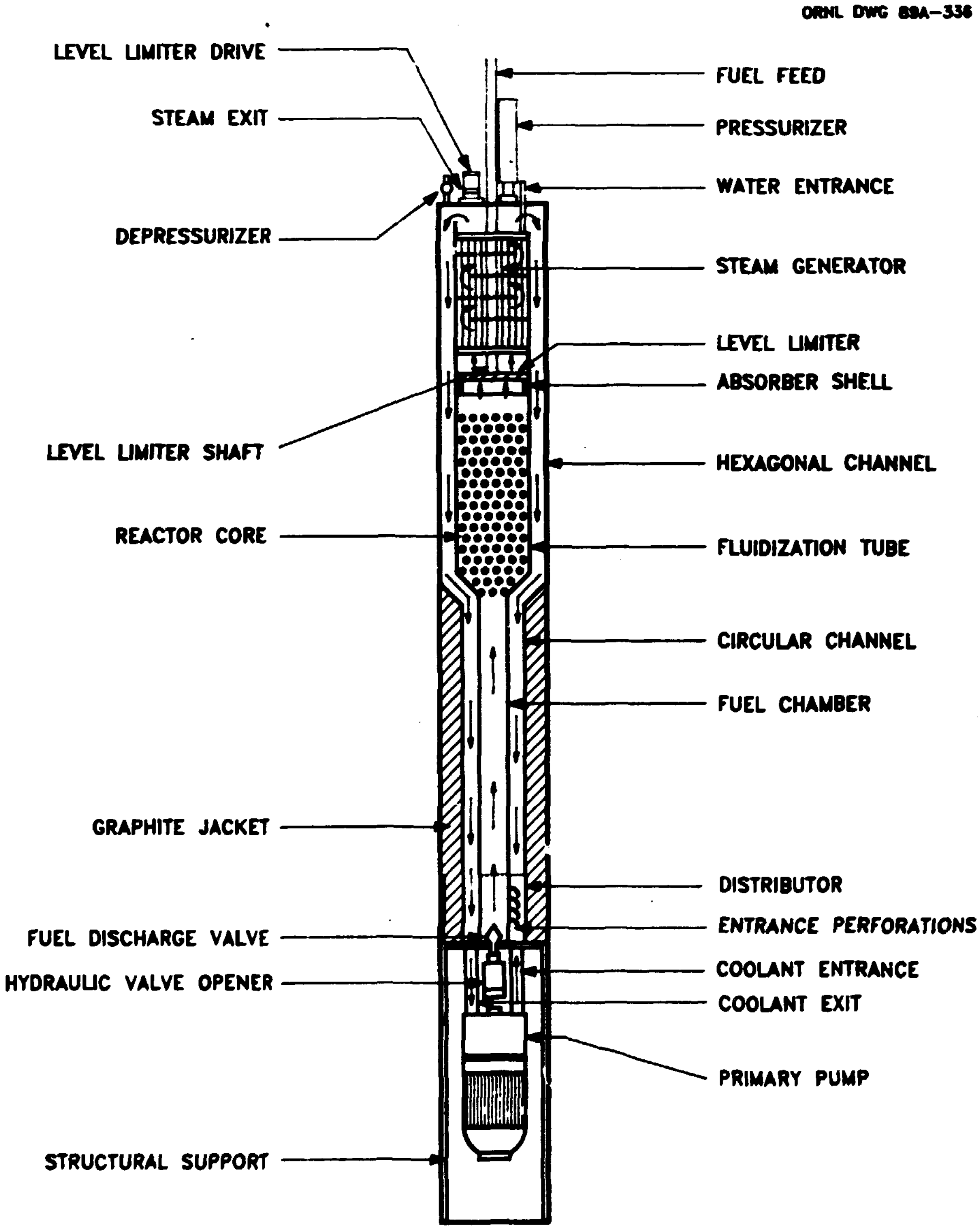


Fig. 1. Fluidized-bed pressurized-water reactor.

TITLE: PRESTRESSED CONCRETE REACTOR VESSEL (PCRV) FOR LIGHT-WATER REACTORS

Functional Requirements: 1.2.1 Maintain Core Coolant Boundary Integrity

Safety Type: Passive

Developmental Status: 2

Reactor Type: Light-Water Reactor (LWR)

Organization: Vendor (General Atomics, Great Britain)

Examples of Implementation: None for LWRs
All Advanced Gas-Cooled Reactors

Description: Prestressed concrete reactor vessels (PCRV) are being considered for use with advanced light-water reactors. The potential advantages include: (1) PCRVs can be built in sizes much larger than steel vessels, (2) the experience with PCRVs in gas-cooled reactors has been excellent, and (3) PCRVs may be designed with relatively benign failure modes. Figure 1 shows the design of a current British Advanced Gas-Cooled Reactor vessel and a proposed vessel for PIUS-type reactors. (See Process Inherent Ultimate Safety Emergency Core Cooling System and Fluidic In-Vessel Emergency Core-Cooling System.)

A PCRV is a composite concrete-and-steel pressure vessel where the concrete structure contains both steel cables and steel rebar. The steel cables "wrapping" the pressure vessel are prestressed to hold the vessel together in compression at all operating pressures (Fig. 2). During normal operations, the structural effect of the steel rebar is to ensure local distribution of stresses within the vessel and thus reduce or prevent crack formation. The concrete is the filler and provides the mass of the vessel. This mass results in a vessel that can withstand, without failure, impact (accident) or shock (explosive) loadings that are much larger than can be tolerated with other types of pressure vessels.

PCRVs are designed for only benign failure modes. A PCRV has thousands of prestressed cables. With this redundant structural characteristic, the loss of multiple cables does not threaten vessel integrity. With nuclear-grade vessels, sufficient steel rebar is included to ensure vessel integrity (but with permanent damage) in the event that all prestressed cables are destroyed. If a vessel is overpressurized, small cracks will open in the concrete and, thus, will vent the vessel fluid. With the prestressed cables, the vessel will reseal after sufficient fluid exists to lower the pressure.

Current developments of PCRVs for LWR applications have included one additional feature to ensure vessel integrity against leaks. This is a double-liner system where the vessel cross section from the inside to the outside the vessel consists of stainless steel liner, 1 m of concrete, embedded steel liner, and 6 to 8 m of concrete. The embedded liner is protected against extreme events by the 1 m of concrete. The secondary liner protects the bottom 90% of the vessel against leaks. The secondary liner is not extended to the top of the vessel. In the event of a very severe overpressure accident, the vessel is designed to leak near the top of the pressure vessel. This feature allows steam to escape from the reactor vessel but prevents release of the water around the reactor core. The technology for the embedded steel liners was developed and has been used for several decades in Swedish reactor containment structures to ensure leak-tight containments.

Alternative Versions: There are many design variations of PCRVs.

Status of Technology: PCRVs are the standard, commercial technology of choice for large gas-cooled reactors. Large-scale, commercial experience with these systems is available. The technology has been developed but has not been deployed for LWRs as part of the Scandinavian PCRV Development Program. This program included large-scale model tests.

Advantages: PCRVs have multiple advantages compared with other types of pressure vessels such as:

1. two independent mechanisms (cables and rebar) to ensure vessel integrity,
2. two independent liners to ensure leak-tightness,
3. slow failure with benign vessel failure mechanism,
4. very high resistance to impact (major accident) and shock (explosive) loadings, and
5. feasibility of fabrication in very large sizes.

Additional Requirements: None

Comments: None

References/Contacts:

1. Z. P. Bazant, "Safety Advantages of Prestress Concrete Reactor Vessels," Nucl. Technol., 30, 256 (September 1976).
2. C. Elter and G. Becker, "Design Criteria for Prestress Concrete Reactor Vessels for High-Temperature Reactors," Nucl. Technol., 59, 228 (November 1982).
3. H. Norman, "Enclosing Means for Housing a Nuclear Reactor Core," U.S. Patent No. 4,533,513 (Aug. 6, 1985).
4. American Concrete Institute Committee 361, Composite Concrete and Steel High-Pressure Vessels for General Industrial Use, ACI 361R-86, American Concrete Institute (April 1986).
5. S. Menon, I. Rasmussen, T. Taranai, and W. Kraemer, "Scandinavian PCRV Development: Present Status and Planned Work, 1973-75," Proceedings of the 2nd International Conference on Structural Mechanics in Reactor Technology (SMIRT), Vol. 3, H3/9, Berlin (1973).

6. K. Hannerz et al., "PIUS – The Next Generation Water Reactor," Proceedings of Safety of the Next Generation Power Reactors, American Nuclear Society, Seattle, Washington (May 1988).

7. K. Hannerz, Towards Intrinsically Safe Light-Water Reactors, ORAU/IEA-83-2 (M) - Rev. (July 1983).

Update/Compiler: August 1989/CWF

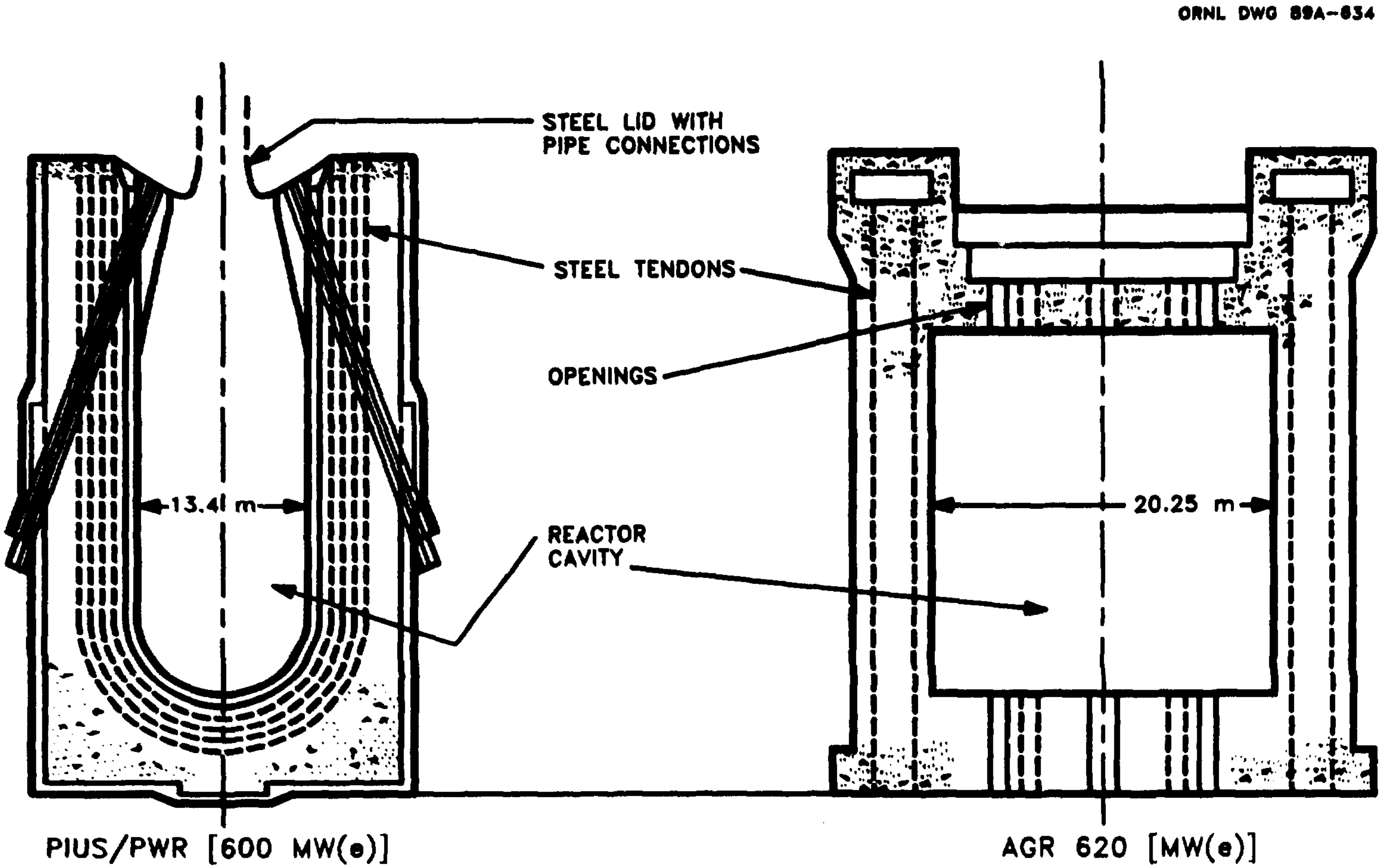
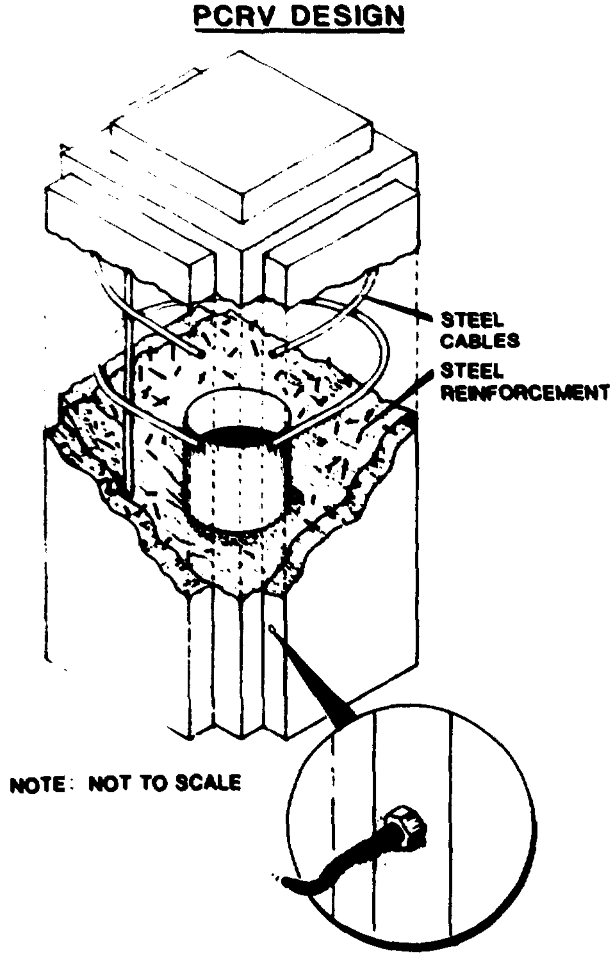


Fig. 1. Process inherent ultimate safety pressurized water reactor (PIUS/PWR) and British advanced gas-cooled reactor (AGR) prestress concrete reactor vessel designs.

ORNL DWG- 88-534



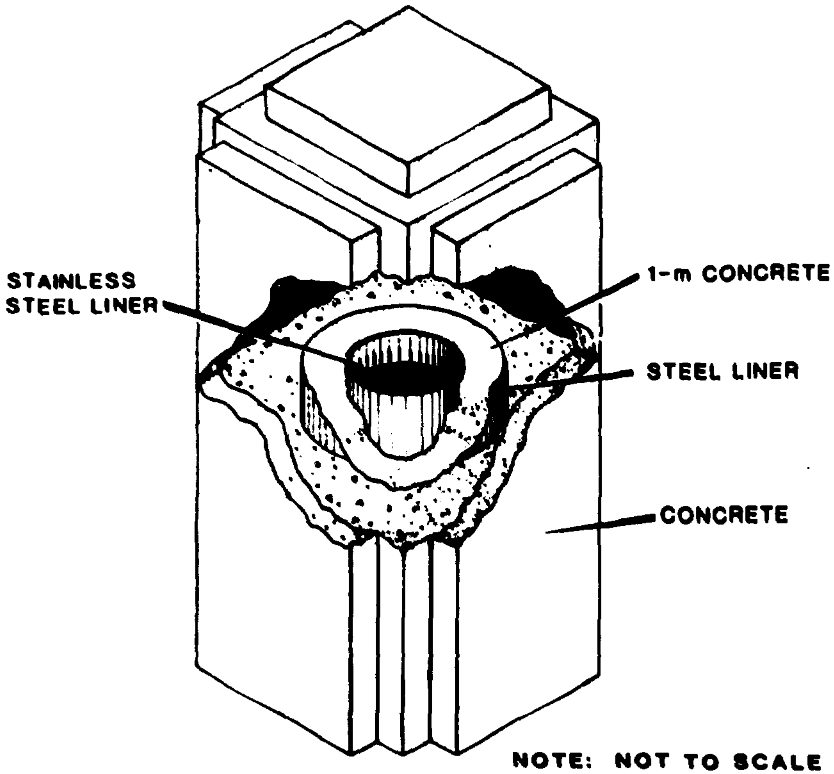
CHARACTERISTICS

- PCRV INTEGRITY INDEPENDENTLY ENSURED BY
  - -PRESTRESS STEEL CABLES, AND
  - -STEEL REINFORCEMENT
- LOSS OF EITHER CAN CAUSE DEPRESSURIZATION BUT NOT FAILURE
- SMALL-SCALE TEST PCRV BUILT IN SWEDEN FOR LWR

Fig. 2. Critical design features of PCRV to ensure integrity-independent structural integrity.

ORNL-DWG 88-536R

PCRV DESIGN



CHARACTERISTICS

- INTEGRITY OF EACH LINER CAN BE TESTED
- ONE-METER PROTECTION AGAINST VIOLENT INTERNAL EVENTS
- BASIS OF SWEDISH REACTOR CONTAINMENT DESIGN

Fig. 3. Critical design features of PCRV to ensure integrity-double liner on inside of PCRV.

TITLE: METAL PRESSURE VESSEL IN POOL

Functional Requirements: 1.2.1 Maintain Coolant Boundary Integrity
1.2 Remove Core Heat
3.2 Immobilize Releases

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Light-water reactor

Organization: University (University of Tokyo), Industry (Sumitomo Heavy Industries, Inc.), and National Laboratory (Japan Atomic Energy Research Institute)

Examples of Implementation: None

Description: A safety concern is the integrity of the reactor pressure vessel and its potential for failure due to severe accidents, sabotage, or other causes. One option is to locate the metal pressure vessel in a large pool so that if the metal pressure vessel fails, the pool will keep the reactor core submerged under water. This concept is currently being investigated in Japan for the Intrinsically Safe and Economic Reactor (ISER). The ISER system is shown in Fig. 1; technical details are listed in Table 1.

In this particular application, the pool surrounding the reactor is also used as part of the emergency core cooling system, spent fuel storage system, and pressure-relief valve blowdown system, as well as for several other functions.

ISER is a PIUS-type pressurized water reactor (PWR) (see Process Inherent Ultimate Safety [PIUS] Technology) in a metal pressure vessel. Using this type of design, the pressure vessel operates in a thermally cold condition. This is in contrast to conventional LWRs, where the pressure vessel is thermally hot. With a cold pressure vessel, the heat loss from the pressure vessel to the pool during normal operations is very low.

ISER can operate with a cold pressure vessel because the internals of the vessel are divided into two zones: (1) a low-boron-concentration, hot-water zone with reactor core, steam generators, and recirculation pumps; and (2) a high-boron-concentration, cold-water zone that is located next to the pressure vessel wall. The two water zones are in hydraulic contact with each other near the top and the bottom of the pressure vessel. A hydraulic balance mechanism (the PIUS concept) prevents circulation of water between the two water zones during normal operation.

The emergency core cooling system for ISER uses the pool as a heat sink. The system (see Process Inherent Ultimate Safety [PIUS] Technology) is designed to allow the two water zones inside the pressure vessel to become one zone if the reactor core overheats for any reason. When this occurs, the pressure vessel goes from a thermally cold state to a thermally hot state and core decay heat is conducted through the pressure vessel wall to the pool water. Pool inventory is sufficient to provide core cooling through the vessel walls for ~47 d.

The pool, which has a design pressure of 20 kg/cm^3, is also designed to withstand the pressure associated with vessel failure. In such a case, the pool becomes the "effective" pressure vessel but operates at a much lower absolute pressure than does the metal pressure vessel.

In the event of any extreme accident with significant core damage, the pool maintains the damaged core underwater. The pool water would scrub any gaseous releases from the core.

Alternative Versions: The ISER reactor is relatively small; however, the basic approach should be applicable to reactors of all sizes. The major constraint for scale-up involves the pressure vessel surface area limitations that limit the amount of heat which can be transferred from inside the pressure vessel to the pool water after an accident. These limitations can be eliminated by the addition of heat pipes or other devices through the pressure vessel wall from the in-vessel cold water zone to the pool water.

Status of Technology: Multiple studies are under way in Japan at Sumitomo Heavy Industries, the Japan Atomic Energy Research Institute, and the University of Tokyo on the ISER concept. Similar concepts have been proposed for various low-temperature district heating reactors.

Advantages:

1. Provides backup in case of failure of the metal pressure vessel.
2. Serves as possible emergency core cooling system for some reactor designs.
3. Limits the maximum accident since the pool scrubs fission products from a core damage accident.

Additional Requirements: The selection of the materials to be used for fabrication of the pressure vessel in the swimming pool requires special consideration due to the low vessel temperatures encountered.

Comments: The use of cold reactor pressure vessels and their placement in a pool as a backup safety mechanism are relatively new concepts. Additional studies of thermal shock and similar issues must be addressed.

References/Contacts:

1. T. Arita, "Various Technical Problems Relating to the ISER Reactor," Collected Papers of the Safety Reactor System Design Research Group, UTNL-R 0229, Nuclear Engineering Research Laboratory, University of Tokyo (December 1988) (Oak Ridge National Laboratory translations ORNL/OLS-89/21)
2. T. Yoshida, H. Wakabayashi, and Y. Asahi, "System Design for ISER," Collected Papers of the Safety Reactor System Design Research Group, UTNL-R0229, Nuclear Engineering Research Laboratory, University of Tokyo (December 1988), (Oak Ridge National Laboratory translation ORNL/OLS-89/21)

Update/Compiler: July 1989/CWF

Table 1: D~sign parameters of ISER reactor with metal pressure vessel in swimming pool

General ISER Parameters	
Type	PWR
Power	645 MWth
System pressure	15.9 MPa
Metal Pressure Vessel Parameters	
Inside diameter	6 m
Height	26.5 m
Wall thickness	0.3 m
Average vessel temperature	75°C
Swimming Pool Parameters	
Design pressure	20 kg/cm^2
Diameter	12 m
Height	33.5 m
Material of construction	Concrete
Normal operating temperature	50°C
Reactor Parameters	
Reactor coolant outlet temperature	323°C
Reactor coolant inlet temperature	288°C
Power density	62.2 kWh
Cold Water Zone Temperature	97°C

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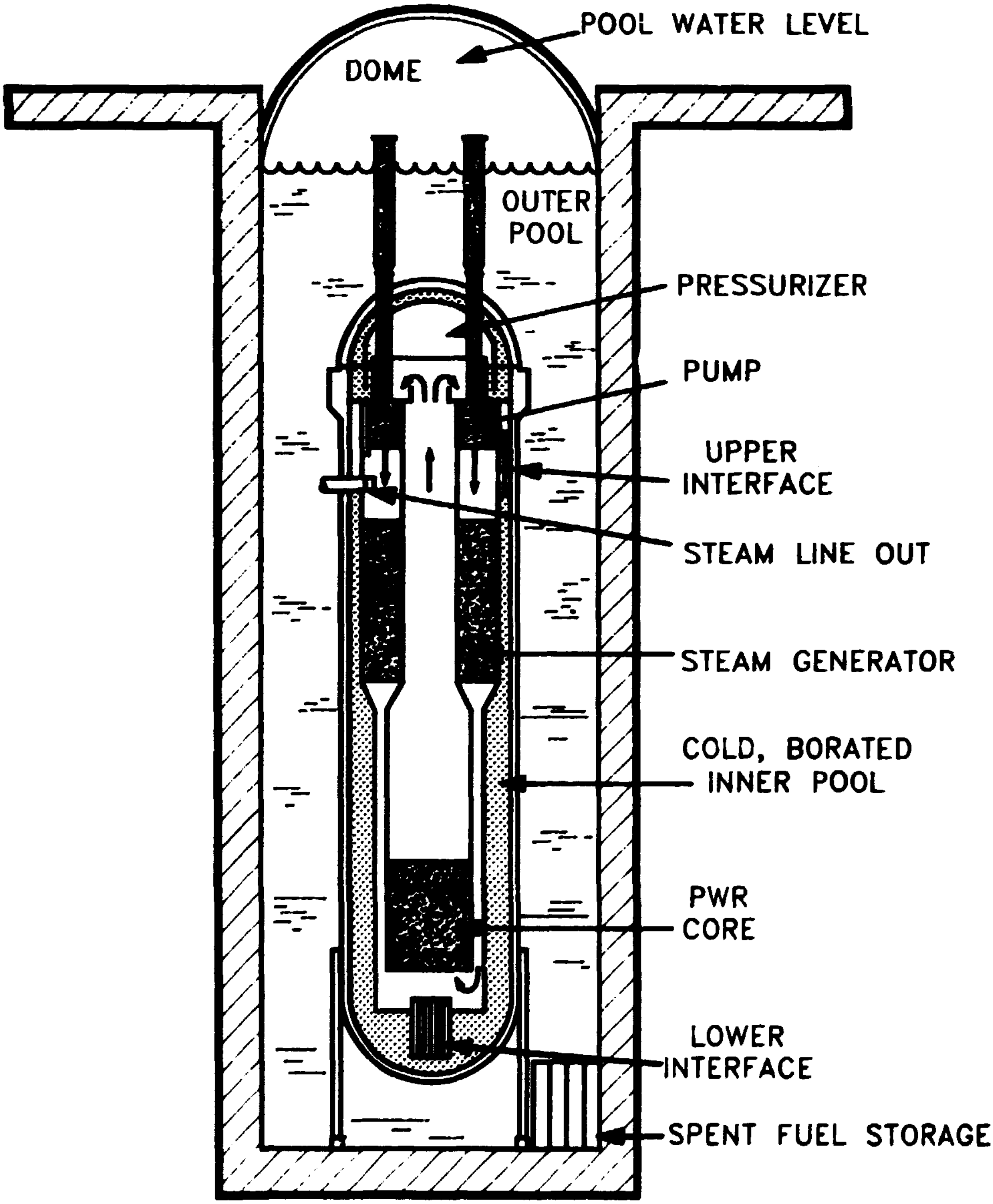

Fig. 1. ISER with pools.

TITLE: **PASSIVE REACTOR CORE-COOLING SYSTEM THAT CAN OPERATE WITH PRIMARY PRESSURE VESSEL LEAK**

Functional Requirements: Level 1.2.1 Maintain Core Coolant Boundary Integrity

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Light-Water Reactor (LWR)

Organization: Vendor (Asea Brown Boveri)

Examples of Implementation: None

Description: Emergency cooling of a reactor core enclosed in a large pressure vessel can be secured for a relatively long period by filling the pressure vessel with water. The heat from the reactor core is absorbed when the water is evaporated, with the resulting steam being vented from the vessel. However, the cooling ability can be lost if a rupture occurs in the lower part of the pressure vessel and the water leaks out as a liquid before it can be vaporized for cooling purposes.

Loss of cooling ability due to leakage could be prevented by a system composed of (1) an outer vessel enclosing the pressure vessel, (2) an evaporation pool above the pressure vessel, and (3) a cooling coil in the upper part of the pressure vessel with both ends connected to the evaporation pool (Fig. 1a). Addition of the outer vessel creates an auxiliary space between the two vessels, connected to the evaporation pool by one or more hydraulic connections. Because of the difference in pressure between the pressure vessel and the auxiliary space, a leak in the lower part of the pressure vessel would produce a flow of water from the auxiliary space to the evaporation pool and the cooling coils (Fig. 1b). During normal operation, the evaporation pool may be dry. After a leak, the pool would begin to fill with water and flood the cooling coils. High-pressure steam in the pressure vessel would condense on the cooling coil, generating steam in the coil and emitting it to the evaporation pool. The liquid in the evaporation pool would flow by self-circulation through the cooling coil and then boil at a point near the pool surface. Thus, a leak in the lower portion of the pressure vessel would not result in a loss of the heat sink.

This system is self-balancing. If a leak occurs in the pressure vessel, the high pressure in the vessel forces the water to the evaporation pool and coils. At high pressures and corresponding high water and steam temperatures, the resulting temperatures in the pressure vessel allow excellent heat transfer to the cooler water in the coils. Over time, the steam condensation on the cooling coils decreases the pressure in the vessel and the amount of water forced out of the leak is reduced or stopped, leaving more in the vessel for core cooling. As a result, less heat transfer to the coils occurs. Note that the evaporation pool temperature never exceeds 100° C- the atmospheric boiling point of water. Since the evaporation pool is above the pressure vessel, the hydraulic head ensures that the pressure of water in the pressure vessel exceeds one atmosphere with the boiling point of water in the vessel significantly above 100° C. This ensures that the higher temperature needed to transfer heat across the cooling coils will be produced and that the evaporation pool will evaporate to dryness before the reactor core and assure full utilization of all cooling water, assuming proper sizing of the cooling coils.

Alternative Versions: Several coils could be used in a variety of arrangements.

Status of Technology: Preliminary work on this concept has been done by Asea Brown Boveri (ABB) as part of that organization's PIUS reactor development program.

Advantages:
1. Maintains core cooling in the event of a leak in the lower portion of the pressure vessel.
2. No moving parts.

Additional Requirements: None

Comments: This technology was originally developed for large prestressed concrete reactor vessels with sufficient water in the pressure vessel to cool the reactor core for approximately one week.

References/Contacts:

1. J. Fredell and K. Hannerz, Means for Cooling a Heat-Generating Device, U.S. Patent Number 4,666,661 (May 19, 1987).

Update/Compiler: Feb. 1989/EBL

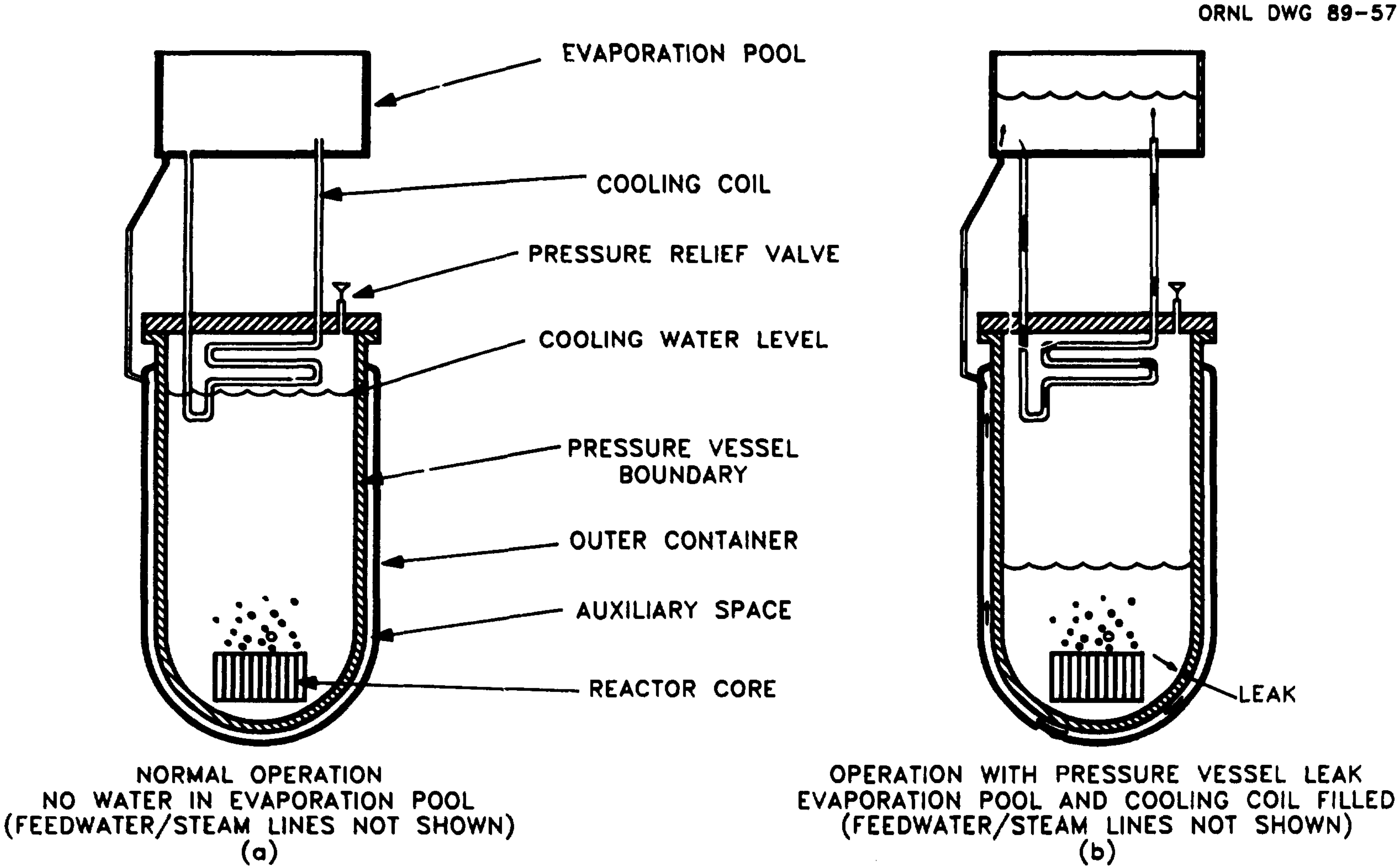

Fig. 1. Passive core-cooling system for leaky pressure vessel.

TITLE: REACTOR PRESSURE VESSEL WITH NO BOTTOM PENETRATIONS

Functional Requirements: Level 1.2.1: Maintain Core Coolant Boundary Integrity

Safety Type: Inherent

Developmental Status 1: Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Vendors (Multiple)

Examples of Implementation: Yankee Rowe Nuclear Power Plant, Massachusetts, PWR

Description: Reactor pressure vessel is designed with no bottom penetrations. Lowest vessel penetrations are the cold and hot water nozzles near the top of the pressure vessel. The reactor core is placed at the bottom of the pressure vessel.

Alternative Versions: None

Status of Technology: Technology is standard practice for reactors built by several vendors.

Advantages:
1. No bottom vessel penetrations eliminate the possibility of pipe or instrument tube failure draining pressure vessel. Minimum emergency core cooling water quantity is ensured by the pressure vessel volume below the nozzles and above the reactor core.
2. Higher reliability, faster, simpler inspection of pressure vessel integrity

Additional Requirements: None

Comments:
1. See: "Reactor Pressure Vessel with Minimum Penetrations."
2. The largest existing pressure vessels with this feature are those for certain heavy water reactors (1). Heavy water reactors have lower power densities which require the use of larger pressure vessels than for LWRs.

References:

1. "Building a 745 MWe Pressure Vessel PHWR in the Argentine," Nuclear Engineering International, 30 (September 1982).

Update/Compiler: Nov. 1988; CWF

TITLE: INTEGRAL PWR WITH NO PRIMARY SYSTEM PIPING

Functional Requirement: Level 1.2.1 Maintain Core Coolant Boundary Integrity

Safety Type: Inherent

Developmental Status: 2 Demonstrated

Reactor Type: Pressurized Water Reactor (PWR)

Organization: Vendor (Combustion)

Examples of Implementation: Otto Hahn (Merchant Ship)

Description: Current PWRs have separate reactor vessels, steam generators, pressurizers, and pumps that are interconnected by pipes. Pipe failures, pump seal failures, and various other failures create the potential for loss of coolant from the primary system. Many of these failure modes can be eliminated by incorporating all primary system components into a single pressure vessel.

An example of such a design is the Safe Integral Reactor (SIR) proposed by Combustion Engineering, Rolls-Royce, and other partners (1-4). Specific design features of SIR are shown in Fig. 1 and listed in Table 1. The following design features should be noted:

- The 12 steam generators are inside the main pressure vessel with small-bore tubing for feedwater and steam. The largest external pipe connected to the pressure vessel is <7 cm (<2.8 in.). This limits the maximum rate of loss of coolant after an accident.
- Sealed primary circulation pumps are used with no high-pressure seals.
- The primary system has a large water inventory for short-term emergency core cooling (see: High Water Inventory).
- All vessel penetrations are near the top of the pressure vessel (see: Reactor Pressure Vessel with no Bottom Penetrations)

Alternative Versions: Multiple

Status of Technology: The technology has been used for various marine reactors and considered for power plants by different vendors (5,6).

Advantages:

1. Integral designs eliminate primary pipes and, hence, primary pipe failures.
2. Many potential safety-related equipment failures become either nonsafety-related or have reduced importance because the primary pressure barrier remains intact. For example, failure of the casing for the main circulation pump is not a loss of coolant accident for the SIR.

3. Many possible seismic-related failures caused by differential movement of heavy-pressure vessel/steam generator equipment are eliminated.

4. The thick-wall primary pressure boundary area is significantly reduced with simpler geometry, which improves manufacture and inspectability of the pressure boundary. This reduces probability of failure. (Note: Design does not change the steam-generator tube area; however, steam generator tubes are small and the failure of a steam generator tube allows only a slow loss of coolant. Failure of the thick-wall primary pressure boundary may imply rapid loss of coolant.)

5. The equipment arrangement allows feedwater/steam generation inside the steam generator tubes with primary circuit water outside the tubes. This places the steam generator tubes in compression which reduces the probability of tube rupture.

6. For typical designs such as SIR, the distance between reactor core and pressure vessel increases with resultant lower radiation damage to pressure vessel (radiation doses ~10^4 less for SIR pressure vessel compared to conventional PWR).

Additional Requirements: None

Comments: None

References:

1. "SIR - An Imaginative Way Ahead," Nucl. Eng. Intern. (June 1989).

2. "The SIR Project," Atom (June 1989).

3. R. Bradbury, J. Longo, R. Strong, and M. Hayns, "The Design Goals and Significant Features of the Safe Integral Reactor," Trans. Am. Nucl. Soc., Annual Meeting, Atlanta, Georgia (June 4-8, 1989).

4. R. A. Matzie, J. Longo, R. B. Bradbury, K. R. Teare, and M. R. Hayns, Design of the Safe Integral Reactor, Combustion Engineering, TIS-8471 (1989).

5. Babcock and Wilcox, 400 MWe Consolidated Nuclear Steam System (CNSS): 1255 MWt CNSS Design/Cost Update, ORNL/Sub/82-17456/1 [BAW-1754] (July 1984).

6. United Engineers and Constructors, Inc., The CNSS Plant Concept, Capital Cost, and Multi-Unit Station Economics, ORNL/Sub/82-17455/4 [UE&C-DOE-ORNL-830615] (July 1984).

Update/Compiler: September 1989/CWF

Table 1. Basic parameters for the SIR system

Plant data

Power output (design)	320 MWe
Reactor power	1000 MWth
Reactor type	Pressurized-water reactor (PWR)
Plant style	Integral primary circuit

Primary circuit

Design pressure	19.4 MPa (194 bar)
Operating pressure	15.5 MPa (155 bar)
Coolant flow	7500kg/s (7.38t/s)
Core inlet temperature	295°C (563°F)
Core outlet	Height
temperature	318°C (604°F)

Reactor core

Moderator	Light water
Fuel	Low-enriched UO_2
Fuel enrichment	3.3 - 4.0%
Reactivity control	Fuel loading, burnable poison, control element assemblies, no soluble boron
Clad material	Zircaloy-4
Power density	55 kW/L
Refuel cycle	24 months

Steam generators (SGS)

Number	12
Type	Modular once-through
Steam temperature	298°C (568°F)
Steam pressure	5.5 MPa (55 bar)
Superheat	28°C (82.4°F)
Feedwater temperature	224°C (435°F)
Feedwater flow	516 kg/s (1138 lb/s)
Tube bundle length	8.5 m (27 ft 10 in.)
Heat transfer area	11,140 m^2 (13,323 yd^2)
Material	Inconel 690

Pressurizer

Type	Integral with reactor vessel (in head)
Volume	80 m^3 (2825 ft^3)

Reactor coolant pumps

Number	6
Type	Glandless wet winding
Power (design)	1100kW
Operating power	700 kW

Pressure vessel

Height	19.2 m (63 ft)
Diameter	5.8 m (19 ft)
Wall thickness	28 cm (11 in)
Weight	~1000 tons
Free volume	2400 m^3

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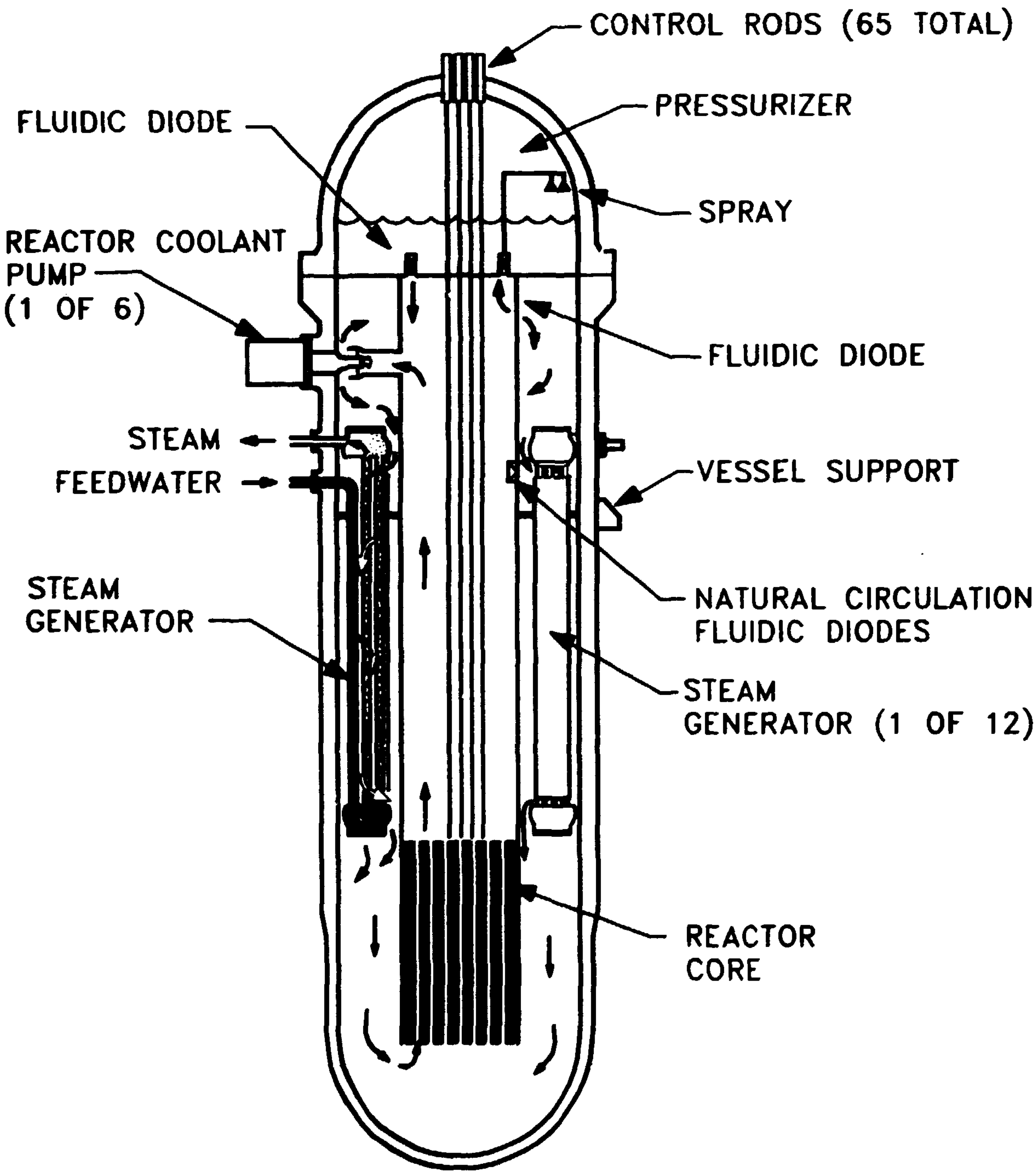

Fig. 1. Primary circuit flow diagram for safe integral reactor.

TITLE: FLUIDIC DIODES

Functional Requirements: 1.2.1 Maintain Core Coolant Boundary Integrity
1.2.3 Transport Heat to Ultimate Heat Sink

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (Rolls Royce and Associates, Inc.)

Description: Fluidic diodes are one-way valves with no moving parts. Fluidic diodes differ in performance compared with mechanical one-way valves in the following ways:

- Fluidic diodes are very reliable because they contain no moving parts.
- Fluidic diodes can operate under conditions where check valves will fail. This is partly due to the large physical clearances in such devices, which can be made larger than the connecting pipes. Fluidic diodes will normally handle high debris levels, varying pressure levels, and pressure pulses.
- Fluidic diodes do not totally shut off fluid flow. The pressure drop in one direction for a particular flow rate may be 50 to 100 times that in the opposite direction in high-performance devices.

Three designs of fluidic diodes (1-4) are shown in Fig. 1.

The vortex diode is a high-efficiency diode with flow resistance 50 times higher in one direction than the other for typical designs. In the low-flow-resistance direction, flow resistance is roughly equal to two 90° pipe bends. In the high-flow-resistance direction, the fluid flow sets up high centrifugal forces that resist fluid flow. The scroll diode operates on similar principles. For typical designs, the scroll diode will have 30 times more resistance to fluid flow in one direction versus the other. The scroll diode has the desirable characteristic that its volume can be very small, which is important for certain applications.

The fluid flow rectifier is a lower-performance device. It is an example of a class of diodes where flow resistance and flow turbulence create the diode effect.

Fluid diodes are being used, or are being considered, for several applications, as follows:

1. Fluidic diodes can reduce the rate of fluid loss in a loss-of-coolant accident by 50%. If a pipe breaks in half, fluid will pour out both ends of the pipe break. With a fluidic diode in the line, the fluid cannot reverse direction; it can only leak out one side of the break. Fluidic diodes are currently used in British Advanced Gas-Cooled Reactors (AGRs) for such an application. Figure 2 illustrates this concept for AGRs and shows where fluidic diodes could be used in a PWR.
2. Fluidic diodes can prevent equipment drainage by fluid flow reversals in accident scenarios. In the above example, the diodes also protect compressors and other

equipment. In some cases, fluidic diodes can prevent mechanical equipment failures, which, in turn, will prevent secondary accident initiation events caused by equipment failure.

3. Fluidic diodes can create alternative water flow paths to assist emergency core cooling after initiation of an accident. An example of this application (5-6) is the proposed Safe Integral Reactor (SIR) - a PWR being developed by Combustion Engineering and Rolls Royce and Associates, Inc. (Fig. 3). During normal operation, water flow is from core to recirculation pump, to steam generator, and back to the reactor core. If there is a pump failure or low water level, vortex fluidic valves allow water flow from the reactor core directly to the steam generator (or decay heat cooler) and back to the reactor core. This avoids the high flow resistance of the dead pump and ensures water circulation even with lower water levels in the pressure vessel.

Alternative Versions: There are many types of fluidic valves.

Status of Technology: Fluidic diodes are used in gas-cooled nuclear reactors and nuclear fuel reprocessing plants; they are also used in the oil and chemical industries.

Advantages: Fluidic diodes are a passive, very reliable technology. They are primarily used in very critical applications where mechanical one-way valves are not considered to be sufficiently reliable.

Additional Requirements: None

Comments: None

References/Contacts:

1. C. F. King, "Power Fluidics for Nuclear and Process Plant Safety and Control," C94/83, Heat and Fluid Flow in Nuclear and Process Plant Safety, Institute for Mechanical Engineers (1983).

2. C. Etherington, "Power Fluidics Technology and Its Application in the Nuclear Industry," Nucl. Energy 23, No. 4, 227 (1984).

3. P. J. Baker, "A Comparison of Fluid Diodes," Paper D6 presented at the Second Cranfield Fluidics Conference, Cambridge, England, January 1967.

4. J. Grant, "Power Fluidics and the Environment," Critical Reviews in Environmental Control, CRC Press, Inc., Boca Raton, Fla., 1977.

5. R. Bradbury, J. Longo, R. Strong, and M. Hayns, "The Design Goals and Significant Features of the Safe Integral Reactor," Trans. Am. Nucl. Soc. (June 1989).

6. R. A. Matzie, J. Longo, R. B. Bradbury, K. R. Teare, and M. R. Hayns, Design of the Safe Integral Reactor, Combustion Engineering, TIS-8471 (1989).

Upgrade/Compiler: July 1989/CWF

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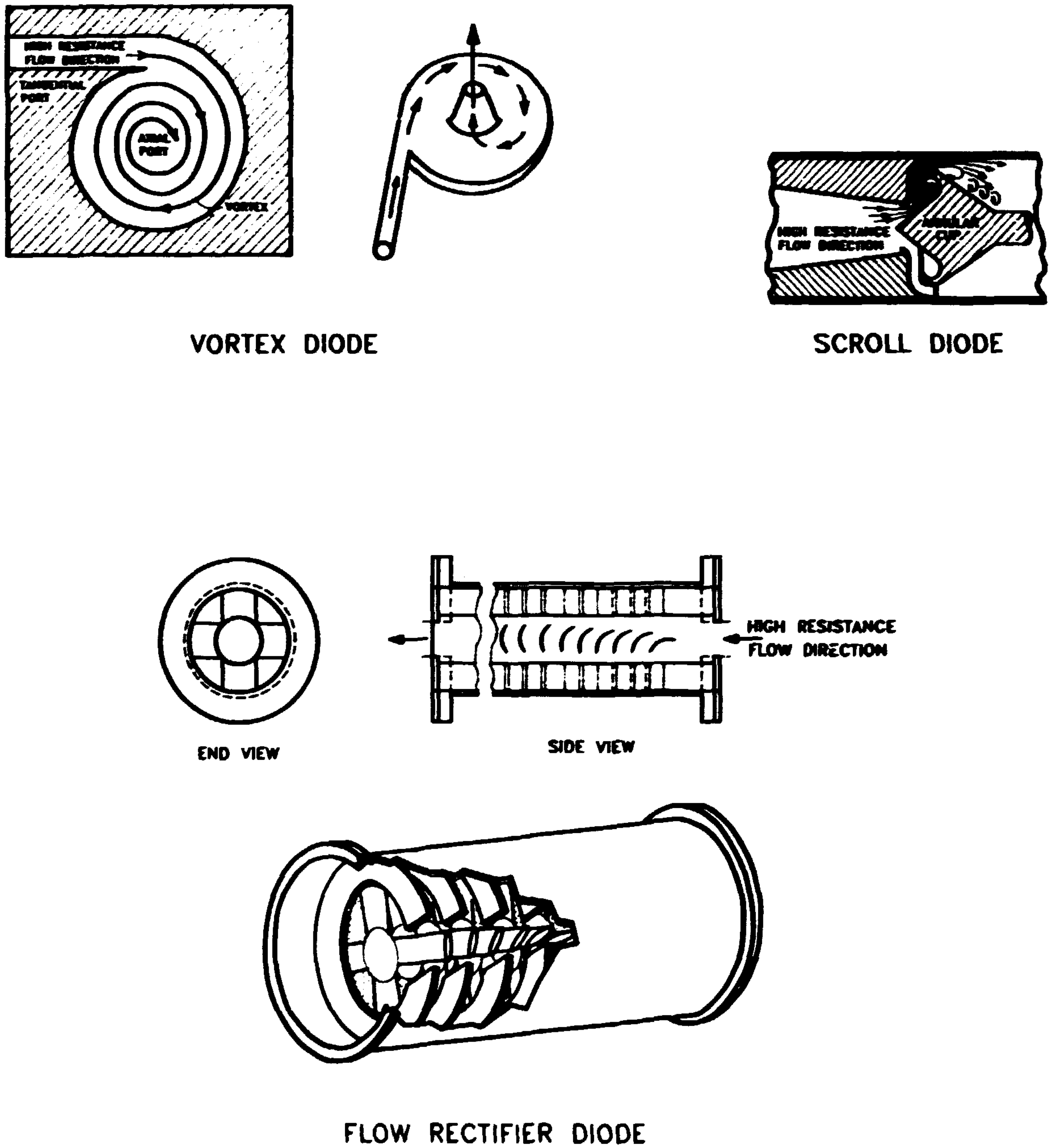

Fig. 1. Fluidic diodes.

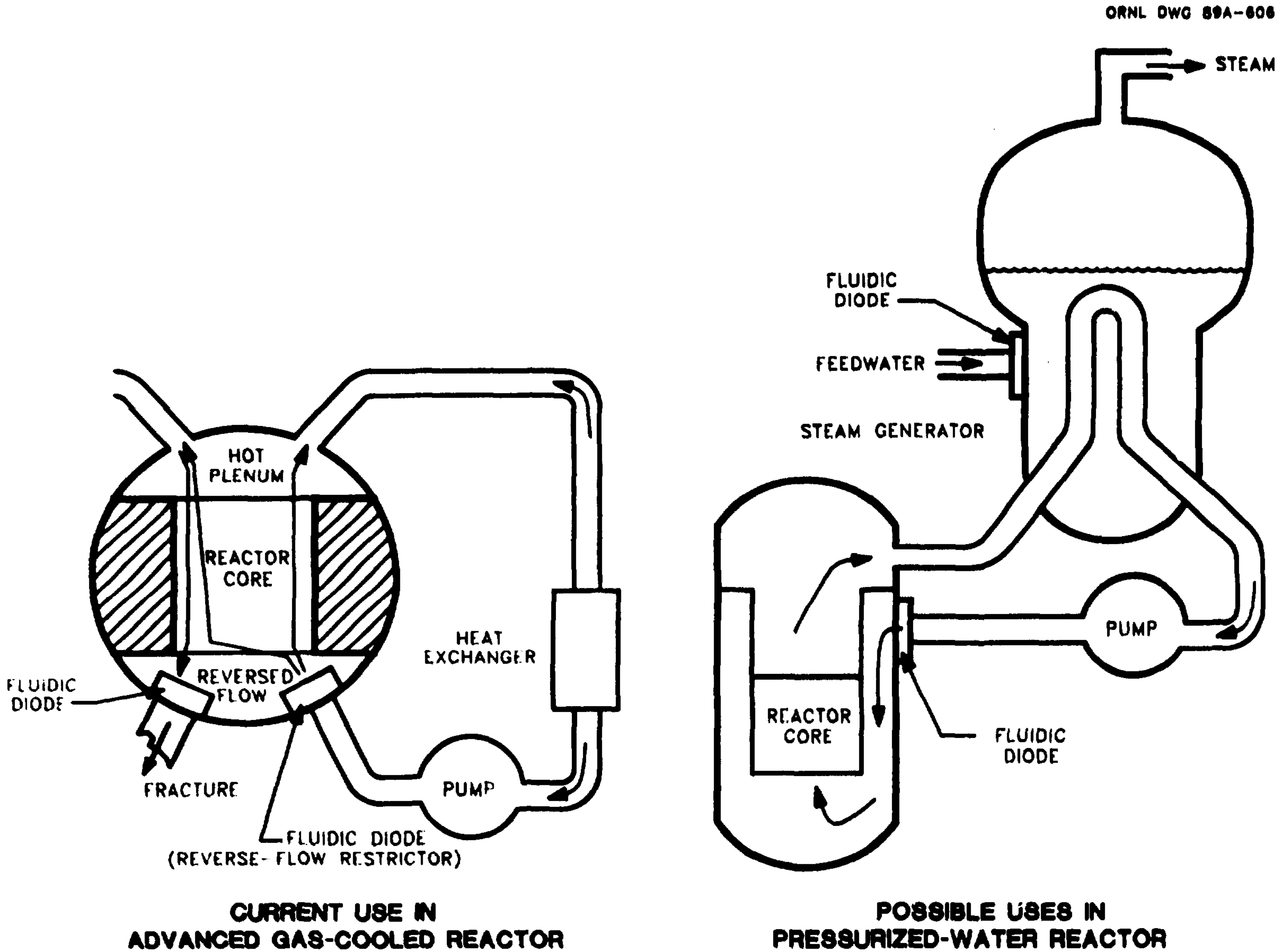

Fig. 2. Applications for fluidic diodes in power reactors.

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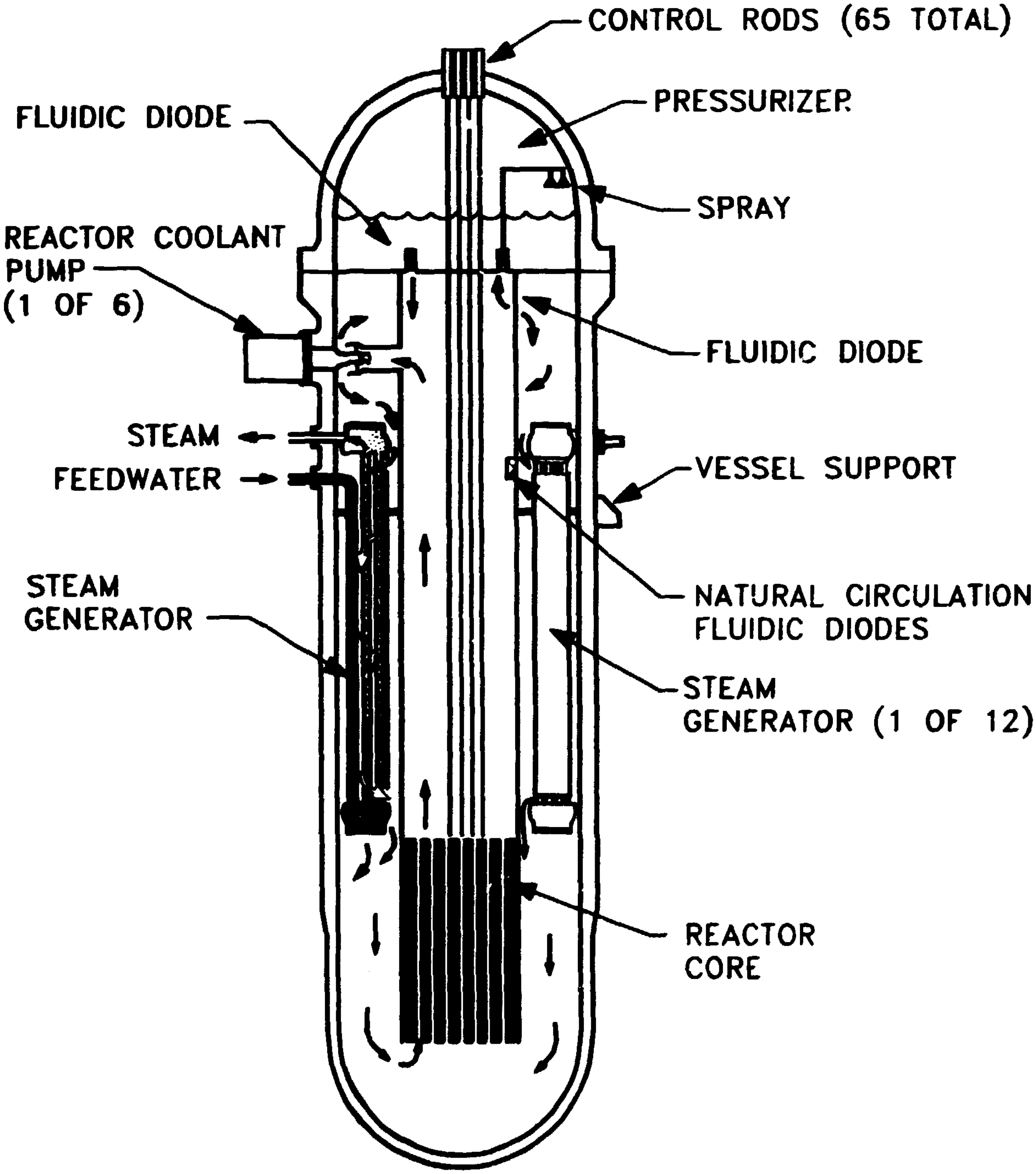

Fig. 3. Primary circuit flow diagram for safe integral reactor.

TITLE: HIGH WATER INVENTORY

Functional Requirements: 1.2.2 Maintain Core Coolant Makeup

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Light-water reactor (LWR)

Organization: Vendors (Asea Brown Boveri, Combustion)

Examples of Implementation: All LWRs

Description: A major safety concern for LWRs is maintenance of reactor core cooling after normal or post-accident shutdown. If the reactor primary system is intact, the reactor core can be cooled by the inventory of water that is in the primary system (pressure vessel, steam generator, etc.) by conversion of water to steam and release of steam through pressure-relief valves. This cooling mechanism works regardless of the operation of other emergency systems, including various heat sinks. Because the water is in and next to the core, it is the most reliable source of reactor core cooling.

This safety feature can be enhanced by increasing the primary system water inventory as measured in volume of water per unit of thermal power output. Table 1 shows per unit water inventories for various LWR designs and the core-cooling time they provide after reactor shutdown.

Alternative Versions: Many options exist to increase water inventories in the primary system.

Status of Technology: Historically, LWRs have had relatively small inventories of water in the primary system. Advanced reactor concepts have placed greater emphasis on increasing primary water system inventories as a safety feature.

Advantages:

1. Increased water inventories increase the time between initiation of an accident and the possibility of core damage. This reduces the need for and speed of response required of emergency core-cooling systems.

2. Large water inventories usually imply slower response of the reactor system to normal operational transients. This slower response makes the reactor easier to control, reduces wear and tear from operational transients, and allows the reactor to "ride out" many transients without reactor shutdown.

Additional Requirements: None

Comments: In addition to the primary circuit water inventories, pressurized water reactors (PWRs) have a secondary water inventory in the steam generator. This water inventory assists in long-term cooling *if* the primary system is not depressurized. If the reactor is depressurized, temperatures are low in the primary system and there is no temperature driving force to move heat to the steam generator and boil steam generator water. Steam generator water inventory will typically add 1 to 3 hours of cooling time for the reactor core.

References/Contacts:

1. United Kingdom Atomic Energy Authority, "The SIR Project," ATOM (June 1989).

2. "SIR-An Imaginative Way Ahead," Nuclear Engineering International (June 1989).

3. Asea Brown Boveri, PIUS: Technical Information, Vasterås, Sweden (January 1989).

4. I. Haarala and T. Kukkola, "Passive Safety Features in VVER-440 Type Plant Designed for Finland," International Atomic Energy Agency Technical Committee Meeting on Passive Safety Features in Current and Future Water-Cooled Reactors, Moscow, USSR (March 21-24, 1989).

Update/Compiler: July 1989/CWF

Table 1. Primary reactor system water inventories and reactor core cooling times provided by inventory

Reactor	Reactor type	Power output MW(t)[b]	Power output MW(e)[c]	Water inventory m^3/MW(t)	Cooling time after reactor shutdown (H)	References
<u>Existing reactors</u>						
Oconee (B&W)	PWR	2568	846	0.133	~1 to 2	1
Calvert Cliffs (Combustion Eng.)	PWR	2700	825	0.116	~1 to 2	1
H. B. Robinson (Westinghouse)	PWR	2300	665	0.112	~1 to 2	1
Loviisa[a] (Finland)	VVER	1500	440	0.140	~1 to 2	4
<u>Proposed reactors</u>						
Safe Integral Reactor (Combustion)	PWR	1000	320	0.402	~6 to 8	1
Secure P® (PIUS) (Asea Brown Boveri)	PWR	2000	640	1.750	>170	3

[a]A Soviet-designed PWR. This design also has a very large secondary side water inventory which provides ~6 h of cooling before the steam generator dries out.
[b]MW(t) = megawatt (thermal).
[c]MW(e) = megawatt (electrical).

TITLE: INTEGRAL SAFETY INJECTION SYSTEM

Functional Requirements: 1.1.1 Provide Process Shutdown Mechanism
1.2.2 Maintain Core Coolant Makeup

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Pressurized-water reactor (PWR)

Organization: Vendor (Westinghouse)

Examples of Implementation: None

Description: The Integral Safety Injection System is a modified water accumulator integrated with reactor pressure vessel (Fig. 1). The accumulator contains borated water that would be injected into the reactor core during a loss-of-coolant-accident (LOCA) to cool the reactor core and assure shutdown of the reactor.

The accumulator filled with cold water surrounds the reactor core (except on top). The accumulator is at the same pressure as the reactor core, but is separated from the reactor core by a water-tight insulated wall. Open tubes from the accumulator extend into the bottom of the reactor core through empty control rod or instrument thimble locations. The outside wall of the accumulator is the outside wall of the primary system. This outside, primary-system, pressure-vessel wall is cooled by flooding the bottom of the vessel with water. At the top of the accumulator, the accumulator tank is partially insulated. Heat conducts through the inner accumulator walls from the reactor to the accumulator water. Pool cooling of the pressure-vessel wall maintains most of the accumulator water at cold conditions, except for water near the top of the accumulator that becomes hot due to a lack of local vessel cooling.

In an emergency such as an LOCA, the reactor will depressurize. The depressurization will cause the hot water in the accumulator to flash to steam and push the colder accumulator water into the reactor core.

Alternative Versions: None

Status of Technology: Initial studies of this concept were conducted by Westinghouse Electric Company in 1986.

Advantages: Passive accumulator that does not inject inert gases into the primary system.

Additional Requirements: None

Comments: System only operates when the reactor is initially at full pressure and temperature.

References/Contacts:

1. Burns and Roe, Inc., "Preliminary Conceptual Design Study for a Small LWR Plant: Phase I Report", Westinghouse Electric Company, March 1986.

Update/Compiler: May 1989/CWF

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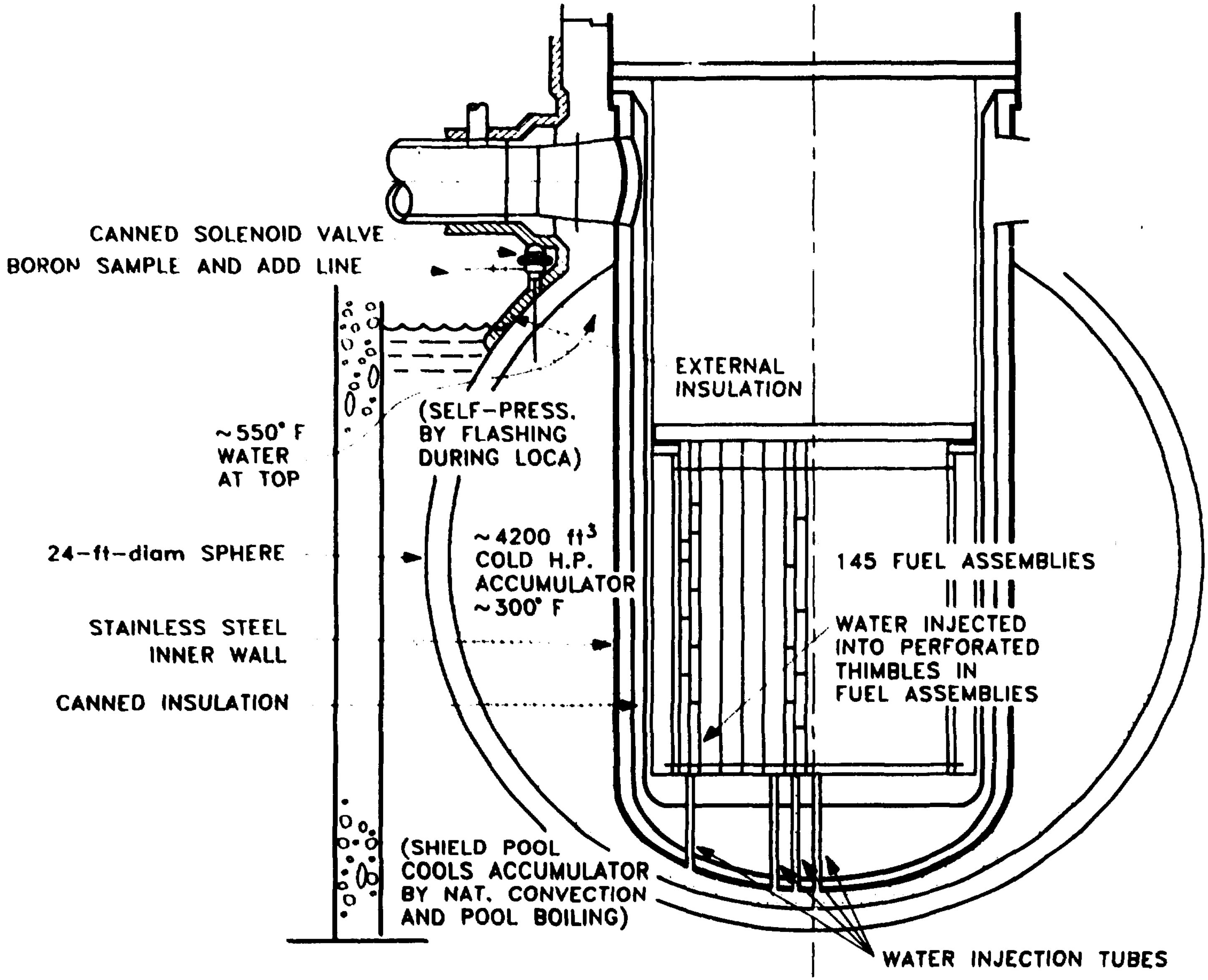

Fig. 1. Integral safety injection accumulator for 1800-MWTH PWR.

TITLE: JET INJECTOR DECAY-HEAT CORE-COOLING SYSTEM

Functional Requirements: 1.2.2 Maintain Core Coolant Makeup

Safety Type: Passive

Developmental Status: 2,3 Demonstrated/Evaluated (See Status of Technology)

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (Canada, General Electric, Combustion Engineering)

Examples of Implementation: None

Description: One method to cool a reactor core during emergency operation is to remove the heat by boiling water. Water lost from the pressure vessel or steam generator in steam production is replaced through a suitable makeup (PWR) or feedwater (BWR) system. If the makeup (PWR) or feedwater (BWR) system becomes unavailable, a replacement water supply system is needed to provide makeup (PWR) or feedwater (BWR) for passive removal of decay heat from the core. The replacement system can use water from the makeup (PWR) or feedwater (BWR) supply system condensate tank, fuel storage pools, or suppression pools. One passive method to pump this water into the pressure vessel or steam generator is to use a jet injector system operated with decay heat steam from the reactor core or steam generator. The key technology for these applications is the jet injector system, which takes medium-pressure steam and low-pressure cold water and produces a small high-pressure stream of warm water and a second low-pressure stream, also of warm water. This device, which uses two steam injectors and a jet pump, contains no moving parts.

Jet injector systems have been proposed for several reactor concepts:

- Boiling-Water Reactors (BWRs) – In a system proposed for use in the General Electric Small/Simplified Boiling-Water Reactor (1) and the Ohio State University Inherently Safe Reactor (2), the steam generated in the pressure vessel is proposed to power a jet injector system that pumps water into the pressure vessel at ~1000 psi if a failure of the feedwater system occurs (Fig. 1).

- Pressurized-Water Reactor (PWR) Steam Generator Feed – A similar approach can be used with a pressurized-water reactor to add emergency feed water to the steam generators (Fig. 1).

- Pressurized-Water Reactor Primary Circuit – The Combustion Engineering (C-E) Safe Integral Reactor (SIR) also proposes to use jet injectors to deliver makeup water to the reactor vessel when normal makeup is lost. The SIR Emergency Coolant Injection System (ECIS) consists of a steam injector receiving water from the containment pressure suppression tanks and injecting it into the pressure vessel downcomer. Two independent ECIS trains are provided. The steam comes from the upper head of the reactor vessel via a branch line in the reactor coolant vent system (3).

An example of a jet injector system(4) is shown in Fig. 2. Through the use of steam injectors, the enthalpy of steam is exchanged for increased water pressure and velocity.

Two steam injectors are arranged in series and are separated by a heat exchanger to cool the water flowing between them. Following loss of the normal feedwater supply system, steam from the pressure vessel is diverted from the turbines to the steam injectors. Steam enters the injectors and initially flows out check valves to an overflow. As steam passes through an injector, the expansion nozzle of the steam injector accelerates the steam to supersonic velocity at subatmospheric pressure. The high-velocity steam mixes with a mist of cold replacement water. The high-velocity mixture then enters the condensation region of each injector where the steam-water mixture condenses, creating a high-velocity fluid stream that pulls shut the check valve on the overflow line, keeping the output water in the system lines. The high-velocity, low-pressure water stream is converted to a higher-pressure low-velocity exit stream from each steam injector.

A jet pump is used to bring replacement water to steam injector A. A portion of the output of moderate-pressure water from steam injector A is fed back through a branch line containing the jet pump, where the moderate-pressure water from the steam injector is driven through the jet pump nozzle. The jet pump nozzle produces a high-velocity stream of water to establish low pressure within the pump, creating the suction needed to draw water from the replacement water supply into the system. On the output side of the jet pump there is a diffuser that is shaped to reduce flow velocity and gradually boost the water pressure to steam injector A. The efficiency of injector A is maintained by cooling the steam-heated water output in a heat exchanger before it enters the jet pump. The coolant for the heat exchanger can be circulated by a pump or by natural convection.

Most of the output water from steam injector A does not enter the jet pump circuit; instead, it proceeds through heat exchanger B to steam injector B. Cooling the output water via the heat exchanger allows injector B to operate at an acceptably high efficiency level. The steam-driven replacement water from injector B is supplied to the replenishment water line of the reactor system at an output pressure exceeding 1000 psi, which is sufficient to allow it to enter the pressure vessel and remove decay heat from the reactor core.

Alternative Versions: Many designs of jet injectors exist.

Status of Technology: Steam-driven injectors are not new. They have a long heritage in the chemical industry and in steam-driven merchant ships after World War II. In the merchant marine, they were used as feedwater pumps. Steam injectors are used in emergency cooling systems of CANDU reactors(5) with output pressures of 4.14 MPa (600 psi). Recent proposed applications are to generate feedwater pressures in excess of 1000 psi. These high-performance devices have not been used in industrial applications nor are fully tested.

Advantages:
1. Reactor core cooling is maintained without the use of electrically operated pumps.
2. The steam injection system is inexpensive, uncomplicated, and simple to maintain.

Additional Requirements: This technology requires maintenance of the primary reactor system pressure boundary integrity. This requirement is not necessarily needed for reactor safety. Small leaks in the primary reactor system can be handled by jet injectors.

Comments: High-performance, high-pressure steam injectors imply a significant advance over currently available proven technology and may require a significant development effort.

References/Contacts:

1. General Electric Company, Preliminary Conceptual Design Study for a Small LWR. EPRI NP-5150M (June 1987).

2. S. Jayanti and R. N. Christensen, "A Passive Steam-Driven Injector as ECCS for an Inherently Safe BWR," Transactions of the American Nuclear Society 53:310-12 (1986).

3. R. A. Matzie, J. Longo, R. B. Bradbury, K. R. Teare, and M. R. Hayns, Design of the Safe Integral Reactor, Combustion Engineering, TIS-8471 (1989).

4. R. W. Howard, Steam-Driven Water Injection, U.S. Patent No. 4,440,719 (April 3, 1984).

5. S. Suurman, "Steam-Driven Injectors Act as Emergency Feedwater Reactor Supply," Power (March 1986).

Update/Compiler: April 1989/EBL

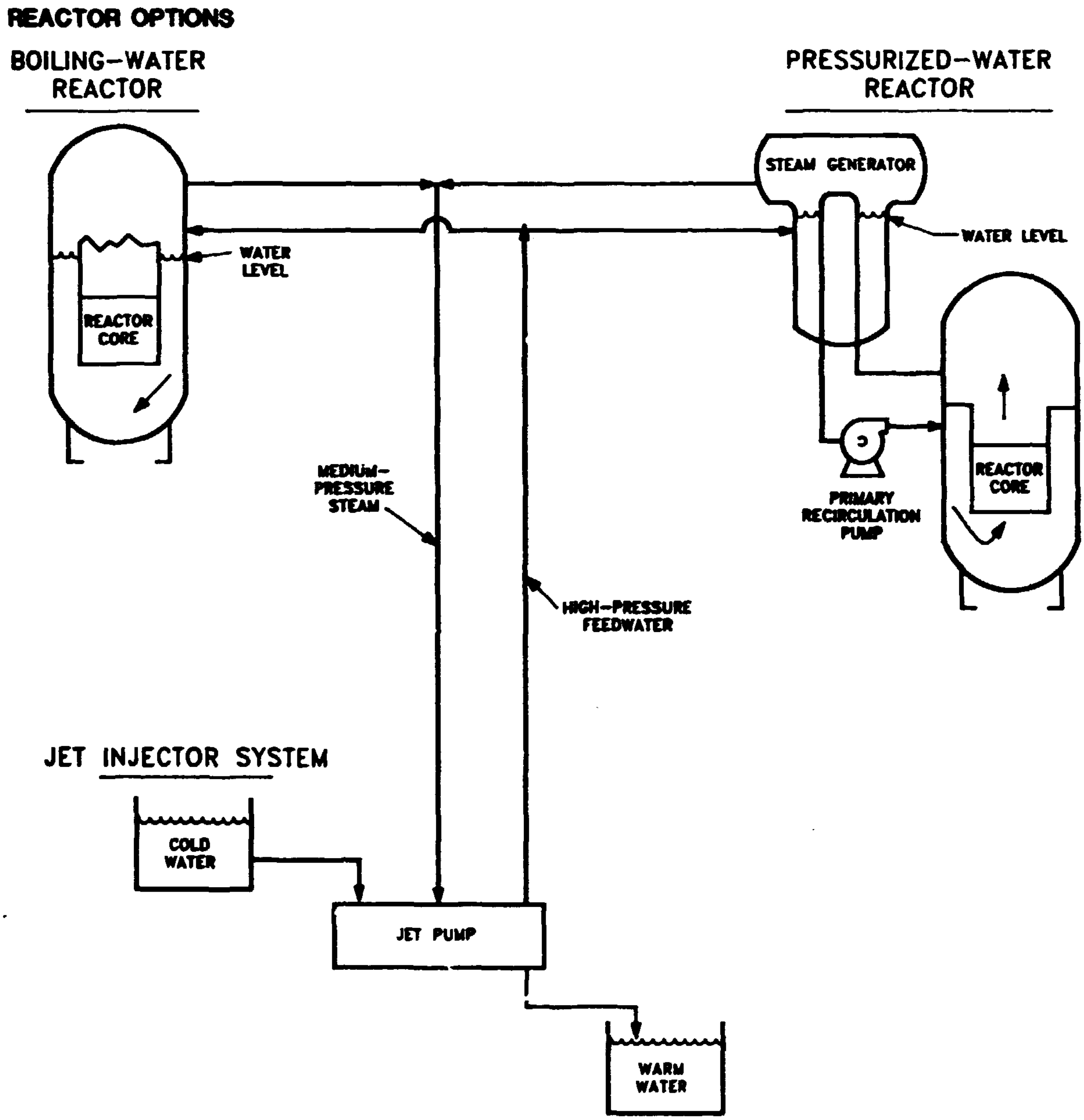

Fig. 1. Use of jet injector system for decay heat removal or small loss-of-coolant accidents.

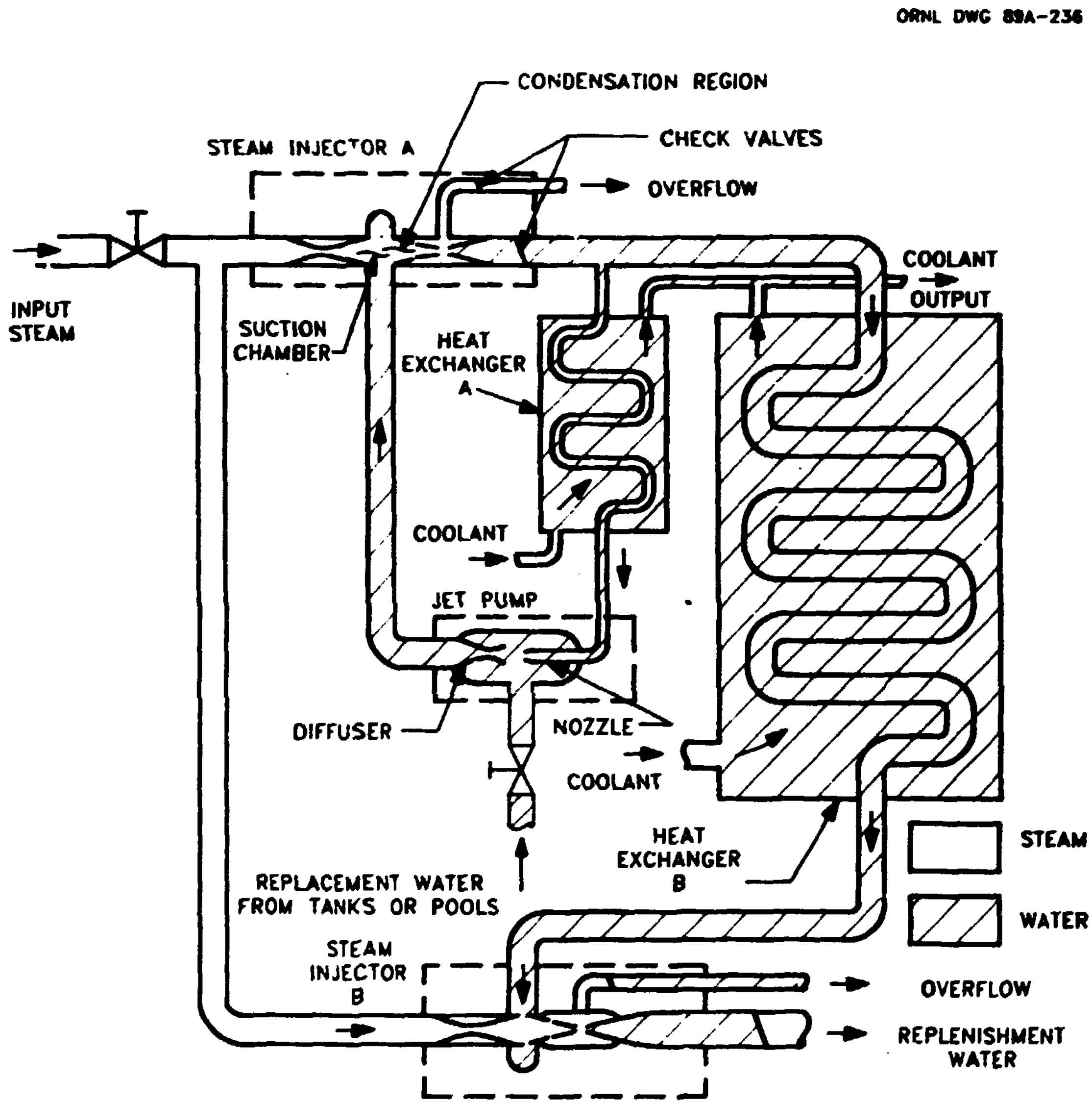

Fig. 2. Jet injector system to provide high-pressure water for decay heat removal.

TITLE: NATURAL CIRCULATION OF WATER

Functional Requirements: 1.2.3 Transport Heat to Ultimate Heat Sink

Safety Type: Passive

Developmental Status: 1 Commercial
2 Demonstrated

Reactor Type: Light Water Reactor (LWR)

Organizations: Multiple

Examples of Implementation: All LWRs

Description: A major passive safety characteristic of Light Water Reactors (LWRs) is natural circulation of water to transfer heat from the reactor core to the remainder of the primary system and the heat sink. The power density in an LWR may vary from 30 to 100 kW/L, depending on the design. Decay heat removal levels after reactor shutdown are initially several percent of full power levels and remain above 1% of full power levels for several hours. At these power densities, heat removal by conduction of heat out the reactor core is insufficient to cool the reactor core. The reactor core must have water circulation through it to remove the reactor core heat during normal operations and the decay heat after normal or accident reactor shutdown.

Water can be circulated through the reactor core to the rest of the primary system with the heat dumped from the system by forced circulation of water using pumps and/or natural circulation of water. Natural circulation of water is based on one of two phenomena: (1) hot water has a lower density than does cold water; and/or (2) water is converted to steam, which has a much lower density than liquid water. For natural circulation to occur, it is required that the heat source (the reactor core) be below the heat sink. This allows the lower-density, hotter fluid (or low-density vapor) to rise to the top of the primary reactor system and the colder, higher-density fluid to sink to the bottom of the primary system and to the reactor core.

The capabilities of natural circulation for removing heat from the reactor core depend on the design of the primary system (1). Characteristics that encourage natural circulation in heat transfer are:

- Large differences in the density of the hot water (or high steam content in water/steam mixtures) versus that of the cold return water.
- Large elevation differences with heat sink far above reactor core.
- Low pressure drops from flow obstructions between reactor core and heat sink.

A brief description of natural circulation in several pressurized water reactors (PWRs) and boiling-water reactors (BWRs) shows how improved designs can increase natural circulation flow in a reactor under all conditions.

Pressurized Water Reactor — In a PWR, water transports heat from the reactor core to the steam generator. During normal operations, hot water flows from the reactor core to the steam generator, the steam generator generates steam while cooling the primary system water, and the cool primary water flows to the pump and then back to the reactor core. Several layouts of primary PWR systems are presented in Fig. 1, which shows the evolution of designs that encourage natural circulation flow.

Figure 1(a) shows the primary system layout for the reactor involved in the Three Mile Island (2, 3) accident. This particular layout resulted in a low natural circulation of water, which was a contributor to the accident. Two characteristics limited natural circulation:

- The center of the reactor core is only slightly below the center of the steam generator (heat sink), providing little driving force for natural circulation.
- The plant has siphon loops (high points in piping between reactor and steam generator). In the accident, these siphon loops filled with gas, first steam and then steam/hydrogen mixtures. These gas pockets prevented the flow of water around the loop during much of the accident. The accident sequence was stopped only when these siphon loops were filled with water – a situation which reestablished circulation between the reactor core and the steam generator.

Figure 1(b) shows schematics of more recent plant layouts (4-6) that assist natural circulation flow. In each design, the steam generators are located entirely above the reactor core and siphon loops are reduced or eliminated. The layout on the left is similar to many current operating reactors, while the design on the right is that proposed for the AP-600(5) and several other next-generation reactors. The best of these designs results in reactors where natural circulation heat removal can be as high as 25% of full reactor power. Such designs ensure natural circulation sufficient to handle all accident conditions.

Figure 1(c) shows schematics of advanced reactors (7-10) with high natural-circulation capabilities. Some proposed versions of these reactors are natural-circulation reactors with no pumps, where all reactor heat during normal operation is removed by natural circulation. The design on the left is the Combustion Engineering Safe Integral Reactor (SIR), while that on the right is that of the advanced Soviet VVER. The SIR includes very low-pressure drops across the reactor core and through the primary circuit. The unique characteristic of the Soviet VVER is the horizontal steam generator. This feature has two potential advantages: a lower pressure drop and location of the entire heat sink (steam generator) at the high point in the circuit. [Note: A vertical steam generator removes heat over the entire elevation of the steam generator with cold, high-density water existing only at the bottom of the steam generator.]

Boiling Water Reactor — In a BWR, the reactor core generates a mixture of low-density steam and high density water. The mixture flows up a riser, and separates, the steam is sent to the turbine, and the water flows through the downcomer to the bottom of the reactor core. Because of the very large density differences between a steam/water mixture and pure steam, all BWRs have high natural circulation rates compared with a PWR. The design evolution of large BWRs has been toward natural circulation. Figure 2 shows early designs (11), current designs (12, 13), and proposed future designs (14, 15). Many current BWRs have natural circulation rates from 30 to 50% of full power levels.

Improved natural circulation in BWRs is achieved by reduction of the pressure drops. Early large reactors had external pump loops and jet pumps that offered significant pressure drops for natural circulation when pumps were shut off. Some current reactors have internal recirculation pumps with low flow resistance when the pump is shut off. Many advanced designs have no pumps and use natural circulation water flow for normal power operations.

Early small BWRs used natural circulation flow of water. Uncertainties about hydraulic instabilities and various economic factors resulted in early large BWRs with recirculation pumps. As a better understanding of hydraulic instabilities has been developed, natural circulation flow has been applied to larger BWRs.

Alternative Versions: Many Options to Encourage Natural Circulation Water Flow

Status of Technology: The technology is used in all LWRs to assist safety. Improvements in design are increasing the capability of natural water circulation to remove heat from a reactor core at higher power levels and under more extreme accident conditions.

Advantages: Passive technology

Additional Requirements: None

Comments: None

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Update/Compiler: July 1989/CWF

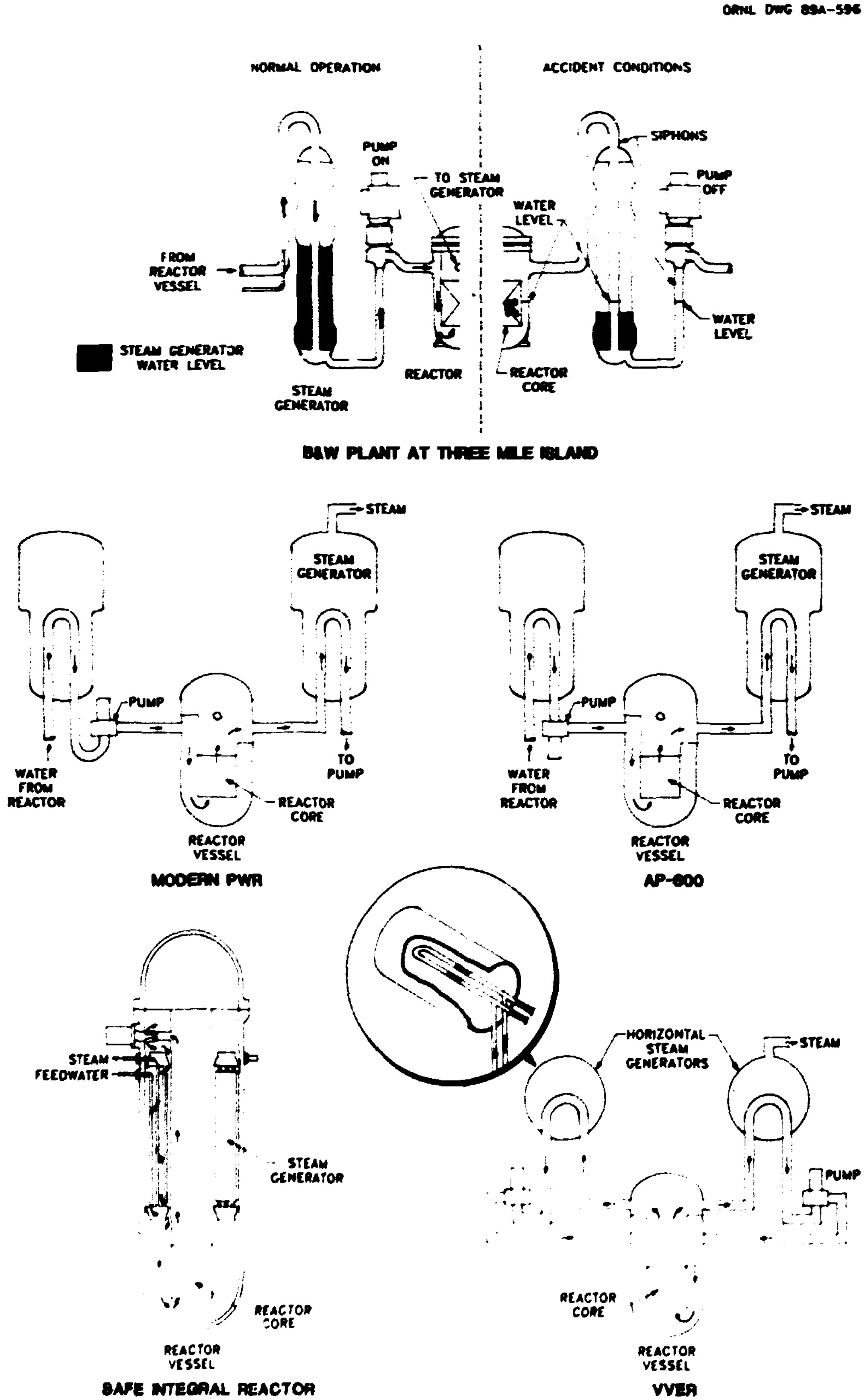

Fig. 1. Natural circulation flow paths in PWRs.

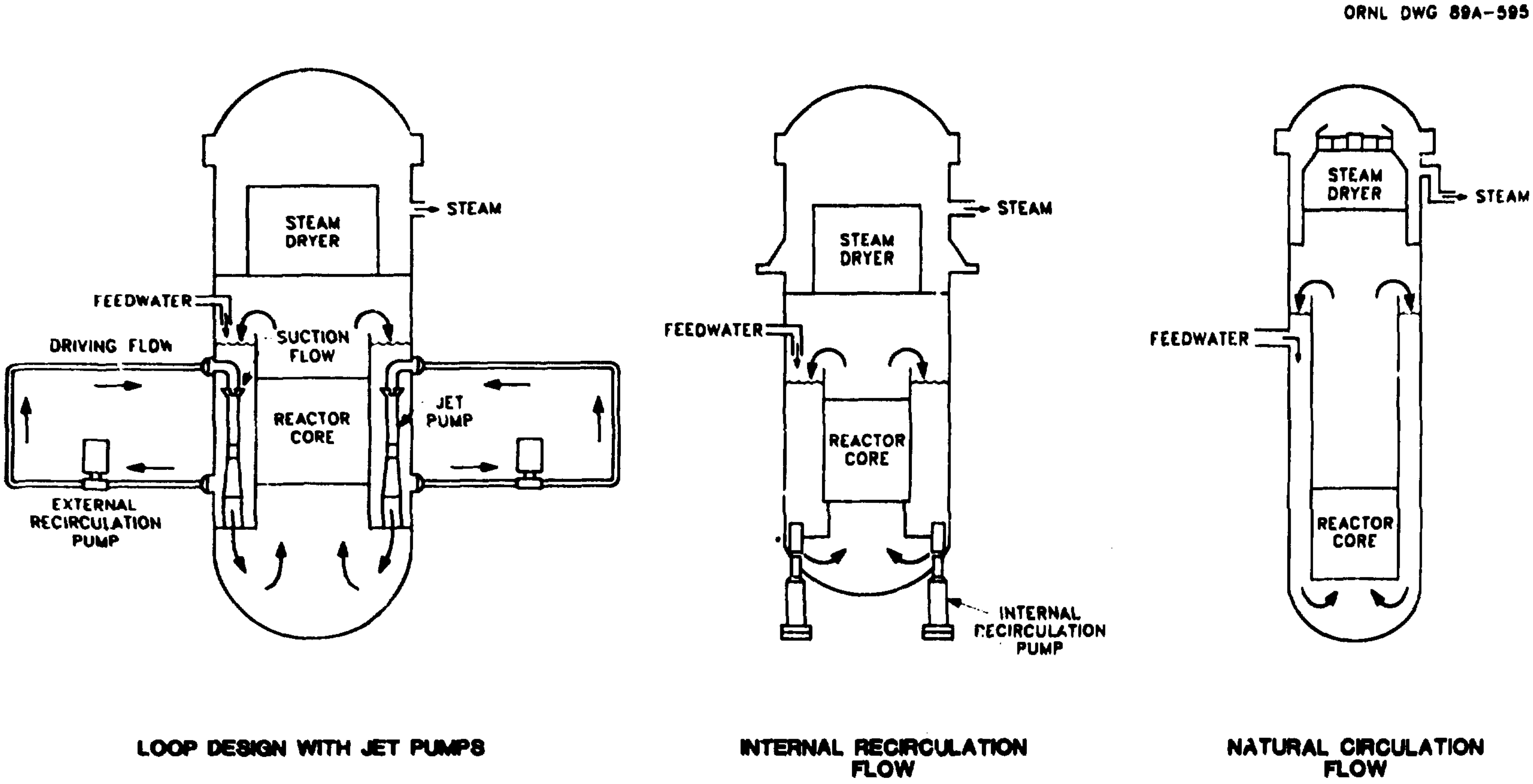

Fig. 2. Natural ci.culation flow paths in BWRs.

TITLE: DECAY HEAT REMOVAL BY NATURAL AIR CIRCULATION STEAM CONDENSERS

Functional Requirements: Level 1.2.3: Transport Heat to Ultimate Heat Sink

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Pressurized-water reactor (PWR)

Organization: Vendor (Multiple)

Examples of Implementation: None

Description: Natural air circulation steam condensers can be used to remove decay heat from a reactor after a postulated loss-of-feedwater transient that occurs after a station blackout with loss of emergency power. The primary function of the steam condensers is to passively cool and condense the steam from the secondary side of the reactor steam generators after a reactor transient with loss of all electrical power. The steam condensers then feed the condensed coolant back into the water volume at the base of the steam generators. Core cooling will be maintained by natural circulation of the reactor coolant.

The steam condensers could also provide core cooling during a small-break loss-of-coolant accident of the primary coolant circuit accompanied by loss of all electrical power. In addition, the passive system could assist in the normal shutdown cooling of a reactor being taken off-line.

Natural circulation steam condensers have been proposed for U.S. reactor designs such as the advanced Babcock & Wilcox (B&W) B-600 PWR [600 MW(e)] and for USSR reactor designs such as the advanced VVER-1000 PWR [1000 MW(e)]. Both of these steam condenser designs are similar in concept and operation. In the B&W B-600 reactor, which is still a conceptual design, the steam condensers are an integral part of the concrete shield building that encloses the steel containment dome. For the Soviet VVER-1000 reactors, steam condensers are being considered for both new plants and retrofit of older plants.

Figure 1 shows both of the steam condenser designs described above. The main operating parameters for the Soviet design are shown in Table 1. A typical natural air circulation steam condenser consists of heat exchangers connected to the secondary side of each steam generator. The heat exchangers should be located above the steam generators to ensure natural circulation of the coolant. The heat exchangers are placed within a duct system that is designed to ensure a passive flow of air due to density differences between the heated and surrounding air.

During normal reactor operations, gates limit the flow of air through the heat exchangers to prevent the undesirable loss of heat. The Soviet design holds the gates closed with electromagnetic latches so that during a loss of electrical power, the gates will open under the influence of gravity and activate the steam condensers. The resulting natural air circulation will remove the decay heat from the steam generators and, thus, the reactor core. An active system using motors is used to return the gates to their closed position.

Alternative Versions: Many variations in technology.

Status of Technology: The technology is currently being developed for advanced B&W B-600 and Soviet VVER PWRs. Modeling of reactor systems using natural circulation steam condensers has shown the effectiveness of the system to passively cool the reactor.

Advantages:
1. Passive decay heat removal is initiated upon loss of electrical power.
2. Steam condensers are currently used in many industrial applications and are well understood.
3. The device is passive in operation.

Additional Requirements: Natural coolant circulation in the primary coolant loop and the steam condenser loop must be assured. The B-600 design addresses this issue by proper placement of the steam generators above the core, placement of the steam condensers above the secondary coolant loop with its steam generators, and by increasing the primary piping diameters.

Comments: None

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Update/Compiler: May 1989/WJR

TABLE 1. MAIN PARAMETERS OF THE VVER-1000 NATURAL CIRCULATION STEAM CONDENSERS[3]

PARAMETER	VALUE
Total minimum thermal power (MW)	60.0
Minimum thermal power of one circuit (MW)	20.0
Nominal coolant pressure (MPa)	6.3
Nominal steam temperature at heat exchanger inlet (°C)	278.5
Maximum environmental air temperature (°C)	50.0
Minimum environmental air temperature (°C)	-40.0
Maximum time to reach nominal power (s)	120.0
Height of air duct measured from top of heat exchanger (m)	25.0
Effective surface area of one heat exchanger (m^2)	147.0
Maximum air temperature at heat exchanger outlet (°C)	221.0
Minimum air temperature at heat exchanger outlet (°C)	113.0
Average air velocity flowing across heat exchanger tubes (m/s)	3.3

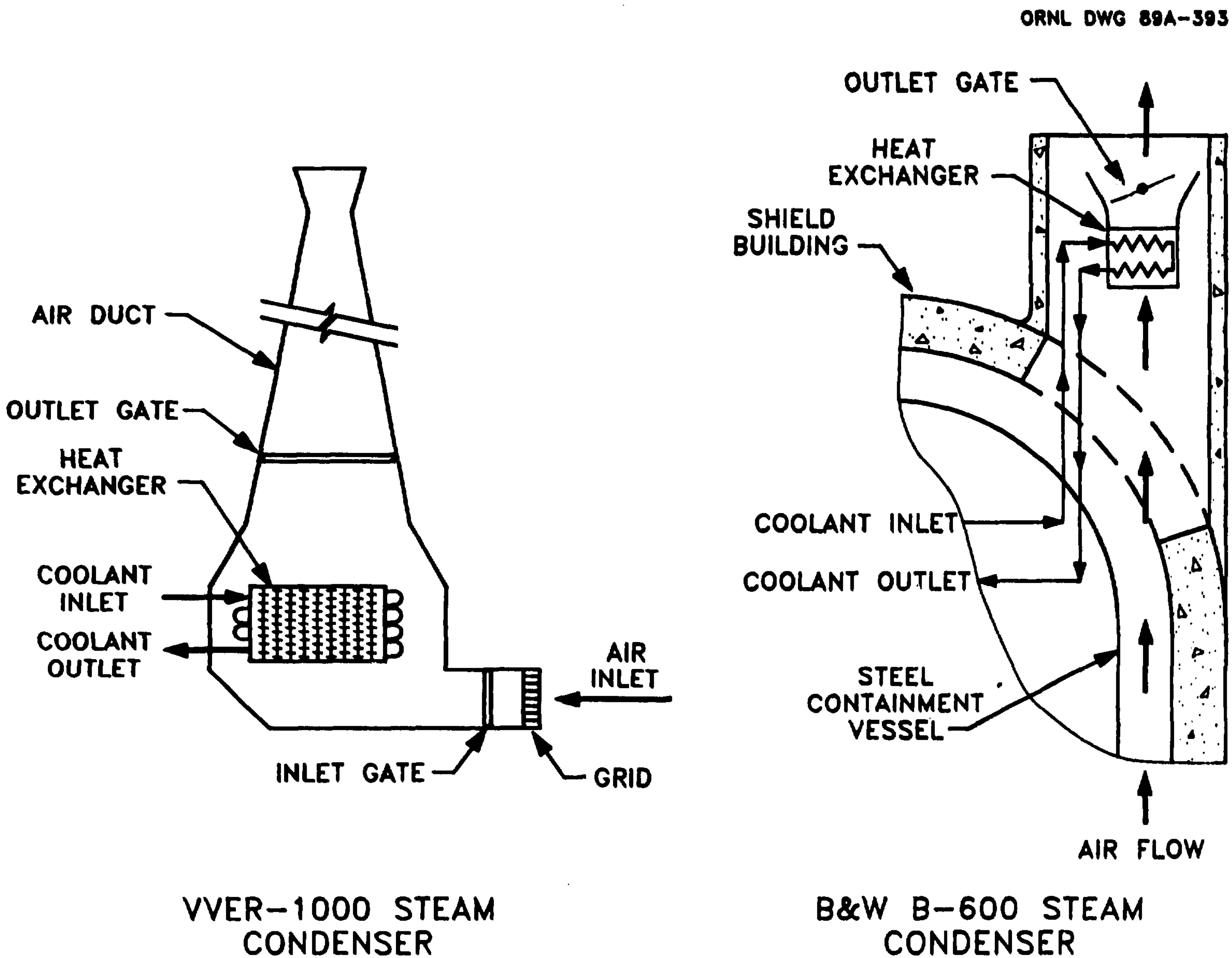

Fig. 1. Natural-air-circulation steam condensers.

TITLE: COOLING PONDS

Functional Requirements: 1.2.3 Transport Heat to Ultimate Heat Sink
3.2 Trap Radionuclides

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organizations: Multiple

Examples of Implementation: Multiple commercial power plants

Description: The ultimate heat sink for many power plants is a cooling pond where evaporation of water rejects heat to the atmosphere. Cooling ponds are passive in their operation. Cooling ponds are used for both normal operations and as heat sinks for emergency cooling systems. Cooling ponds can have high resistance to external threats and provide a low-temperature heat sink.

Alternative Versions: Modified cooling ponds have been proposed as passive ultimate heat sinks for passive emergency core cooling and containment cooling systems for various LWRs. These proposals incorporate two modifications of current pond designs.

1. The reactor is semiburied with respect to the cooling pond. This allows natural-circulation heat transfer loops to transfer heat from reactor core to pond. For example, if applied to the advanced simplified boiling-water reactor (Fig. 1), steam from the core can be condensed in a steam isolation condensor that is cooled by pond water with gravity flow of primary system water back to the reactor core. Similarly, suppression pool water (Fig. 1) could be cooled with hot, low-density suppression pool water entering near the top of a heat exchanger with cooling pond water and cold, dense suppression pool water leaving the heat exchanger.

2. The cooling ponds may be designed for multiple use and to withstand severe accidents. One example of a design is shown in Fig. 2. The cooling pond is located next to the reactor containment with cooling coils located near the bottom of the pool in a reinforced concrete box with holes in it. The bottom 9 m of the pool is filled with rock boulders and clean, crushed rock (no fines). The pool has at least 1 m of water over the top of the rock. The pool is at ground level. The pool may be lined, but also has a 1-m-thick clay layer below the pool.

   The water in the pool accomplishes the following:

   a. Heat is transferred by natural convection circulation of water from the hot coils to the rock and the surface of the pool where heat is rejected to the atmosphere.

   b. The water evaporates from the surface of the pool, removing pool heat.

   c. If the cooling coils contain radioactive material and fail, the pond water scrubs any aersols that are released, thus holding back release of radionuclides to the environment.

d. If the reactor containment overpressurizes after a postulated accident. the containment can be vented through the heat sink. The water scrubs any gases that are released to the atmosphere, thus significantly reducing release of radionuclides to the environment.

The rocks and boulders in the pond accomplish the following:

a. Provide a heat sink if water levels fall below the top layer of rock (normally, water evaporation maintains low pool temperatures).

b. Protects the ·ooling coils against extreme weather.

c. Protects cooling system against sabotage. In particular, rock prevents pumping the pool dry since there is no access to pool bottom.

d. If the cooling coils contain radioactive material and fail, the rock scrubs any gases that are released to the atmosphere, thus preventing release of radionuclides to the environment.

e. If the reactor containment overpressurizes after a postulated accident, the containment can be vented through the heat sink. The rock scrubs any gases that are released to the atmosphere thus holding back release of radionuclides to the environment. This is independent of whether there is water in the cooling ponds.

The toial pond provides a large heat sink that is available in extreme accidents, such as those involving fire. During a fire, thermal stratification assures hottest water on top of the pond, with the coldest water and rock available as a heat sink.

Status of Technology: Cooling ponds are a commercial technology.

Advantages: Passive, low-temperature heat sink.

Additional Requirements: None

Comments: None

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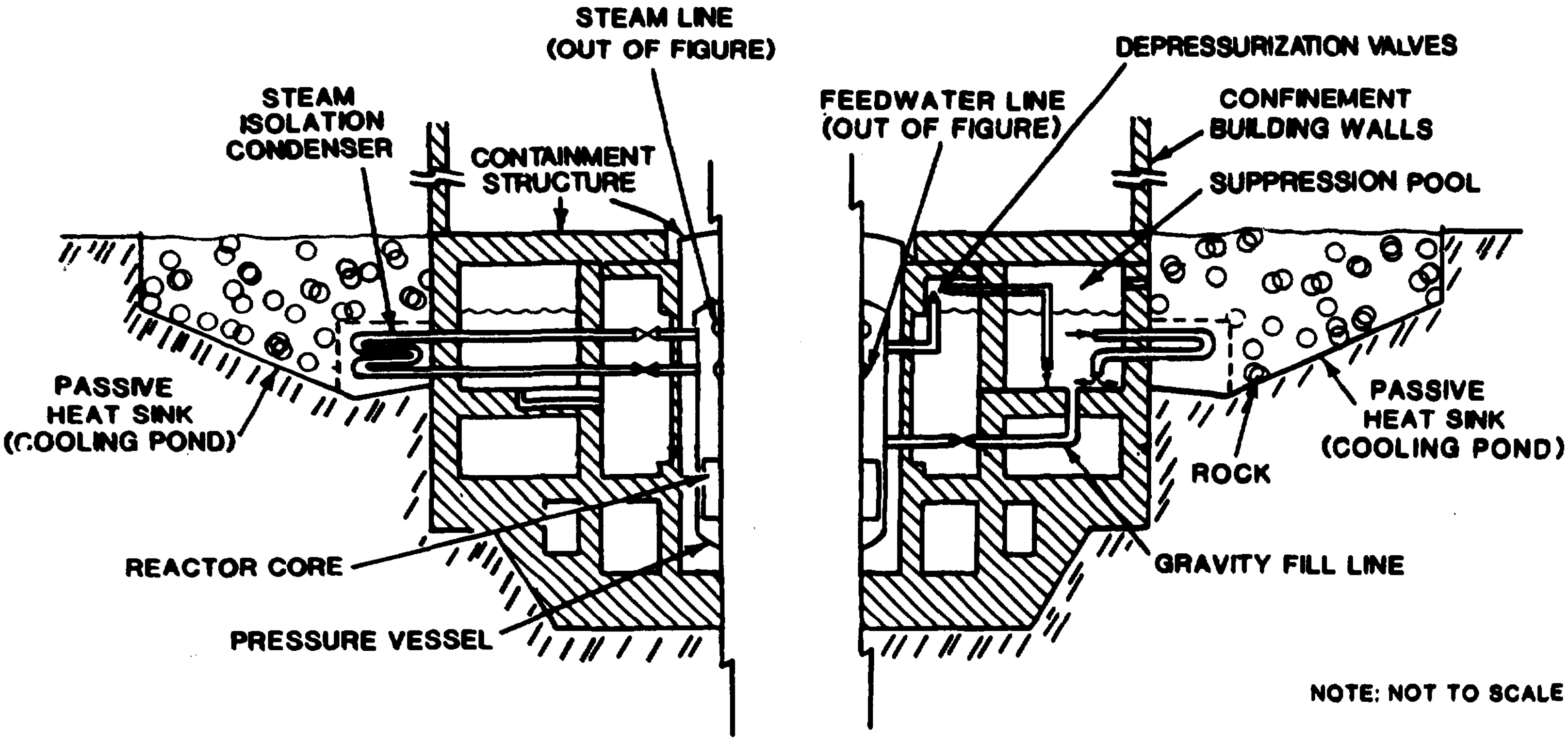
Update/Compiler: CWF/April 1989

ORNL DWG 88-975R



(a) PASSIVE HEAT SINK FOR STEAM ISOLATION CONDENSER

(b) PASSIVE HEAT SINK FOR SUPPRESSION POOL COOLING

Fig. 1. Schematic of passive heat sink applications for the advanced, simplified boiling-water reactor safety systems.

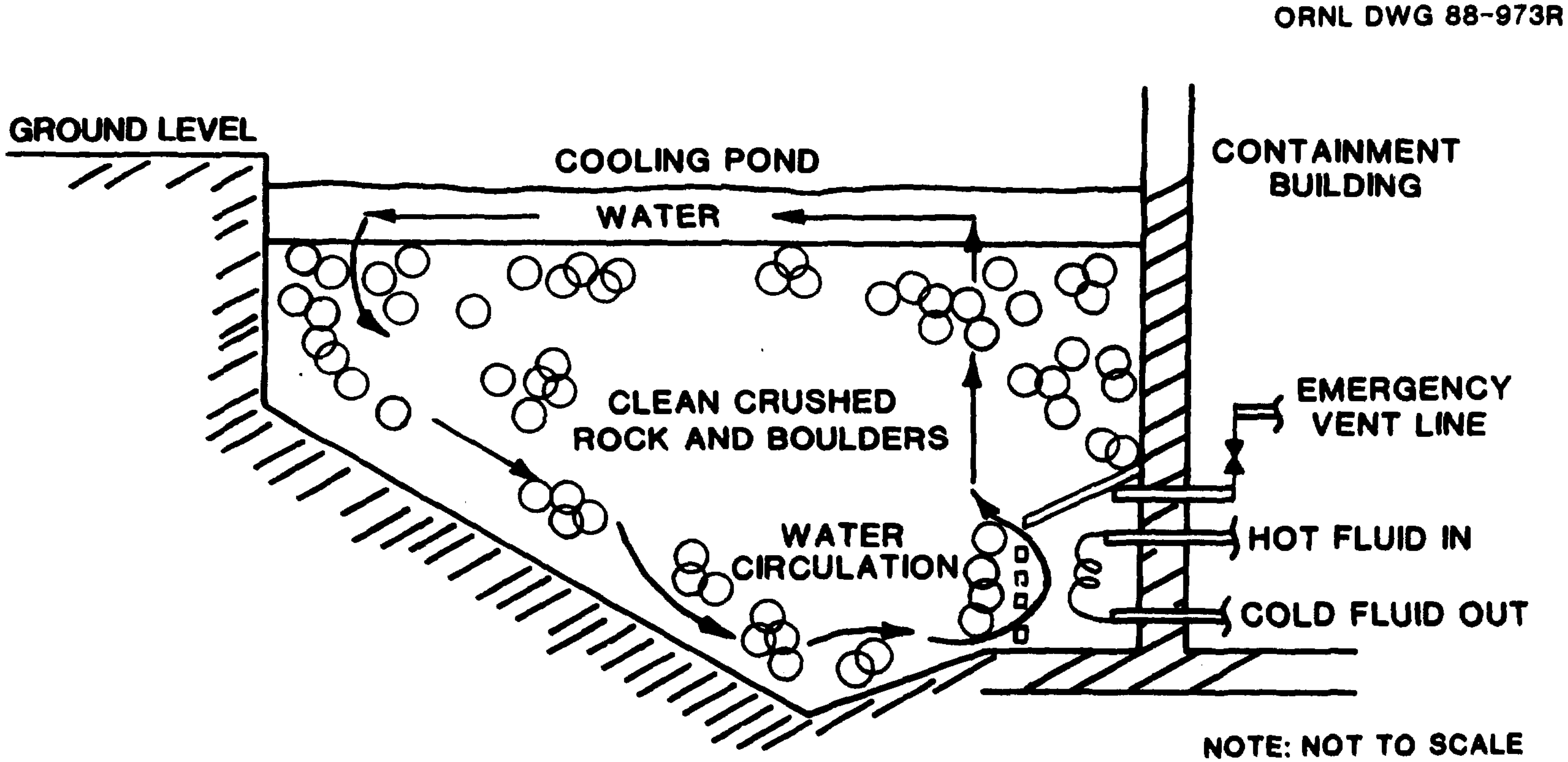


Fig. 2. Schematic of a passive heat sink.

TITLE: WATER QUENCH POOL WITH AIR COOLER FOR REACTOR ACCIDENT AND DECAY HEAT SINK

Functional Requirements: 1.2.3 Transport Heat To Ultimate Heat Sink
3.1.2 Control Energy in Containment

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (Mitsubishi)

Examples of Implementation: None

Description: An advanced concept for a reactor safety system heat sink is a water quench tank with air cooling system. Such a heat sink may be used for all reactor safety systems including decay heat removal, emergency core cooling, and containment cooling. It is a passive technology which can be used with active or passive systems. The components of the system, as shown in Fig. 1 are

1. A large quench tank filled with water inside containment which acts as the initial heat sink for all other safety systems.

2. A small, natural circulation heat transfer loop between the quench tank and the atmosphere outside containment which is used to cool the quench tank water. In a typical application, the evaporator in the quench pool would boil ammonia or water while cooling the quench tank water. The vapor would flow from the evaporator to an air cooled condenser at a higher building elevation outside containment. The condenser would dump the heat to the atmosphere and the condensed liquid would flow back by gravity to the evaporator in the quench pool.

The key technical characteristics of the system are

1. A large quench tank of water capable of absorbing heat at very rapid rates but with limited total heat capacity. This allows temporary storage of reactor blowdown energy after a reactor accident.

2. A natural circulation heat transfer loop with limited rate of heat dumping to the environment but with unlimited heat removal over time.

The combination of these two heat sinks eliminates the weaknesses in each individual heat sink system. This allows for a heat sink with the capability of both rapid adsorption of heat early in a postulated accident and long-term heat rejection to the environment. The quench tank and natural circulation heat transfer loop can be sized for any size of reactor or set of conditions. Larger quench tanks reduce the size of the natural circulation heat transfer loop and vice versa.

An example of this concept is the proposed Mitsubishi simplified, small, pressurized-water reactor (MS series) as shown in Figs. 2 and 3. In this particular example, the passive heat sink is coupled to both passive and active safety systems. The quench tank is a tall, annular tank around the inside of the reactor containment. The natural circulation heat transfer loop uses water and water vapor as the heat transfer fluid of choice. A secondary feature of this design is the use of the quench tank as a suppression pool to control containment pressure in a loss-of-coolant accident.

In modified form, this concept has been proposed for PIUS-type reactors [see: Process Inherent Ultimate Safety (PIUS) Reactor and Fluidic In-Vessel Emergency Core-Cooling System].

Alternative Versions: There are several alternative design options:

1. The natural circulation heat transfer loop can dump heat to the atmosphere or any body of water (ocean, lake, river, or cooling pond). Dumping heat to a body of water significantly reduces the cost and size of the condenser because of better heat transfer characteristics between condensing gases and liquids compared to condensing gases and air.

2. The natural circulation heat transfer loop can be replaced with heat pipes which operate on somewhat similar principles. (See: Heat Pipes for Reactor Containment Cooling or Reactor Core Decay Heat Removal.)

Status of Technology: Near commercial

Advantages: The system efficiently "matches" heat removal requirements from a reactor core after shutdown. Decay heat loads are initially high but rapidly decrease to relatively low levels.

Additional Requirements: None

Comments: None

References/Contacts:

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Update/Compiler: July 1989/CWF

ORNL DWG 89A-484

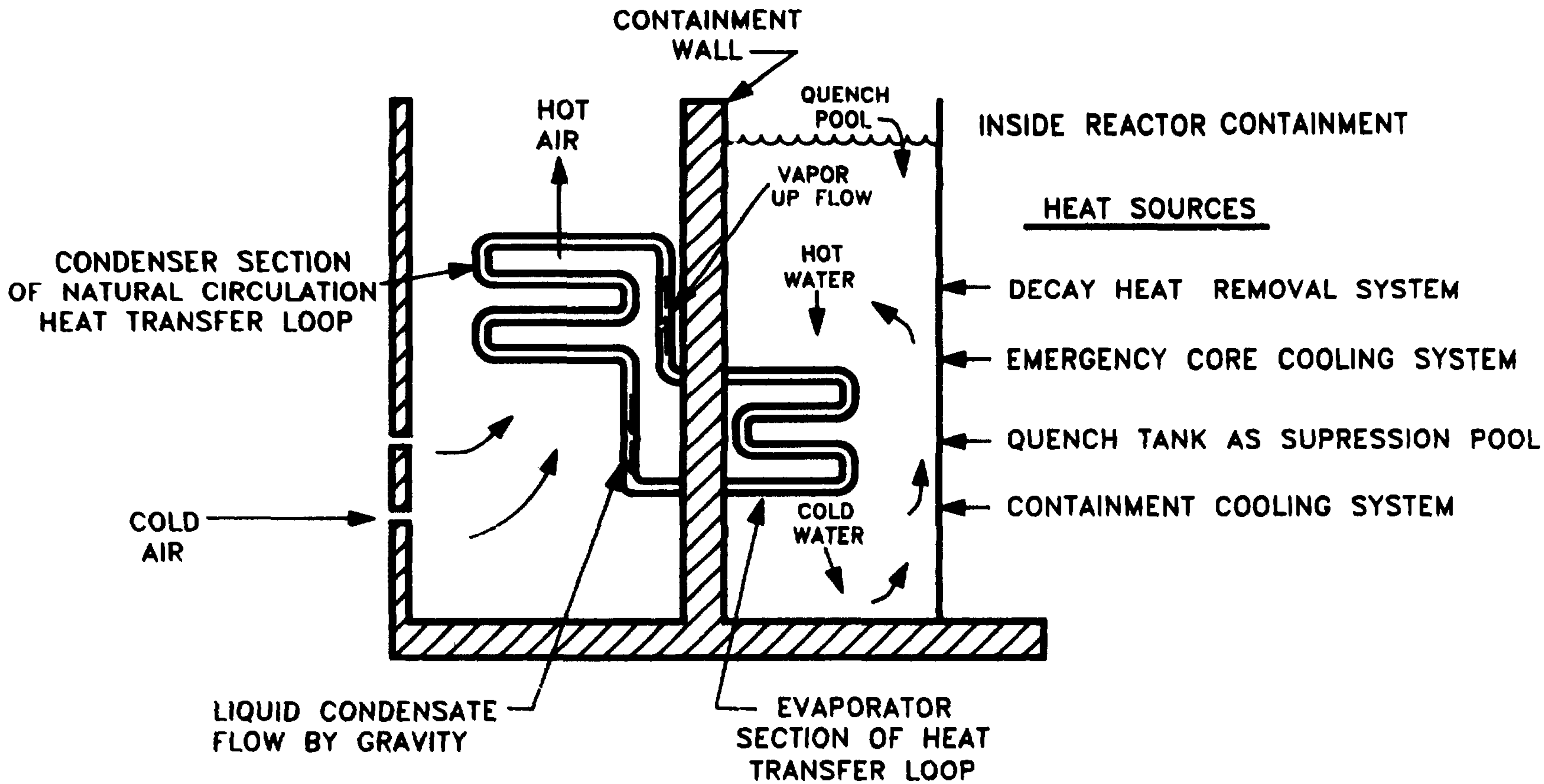

Fig. 1. Schematic of quench tank and natural circulation heat transfer loop for passive heat sink.

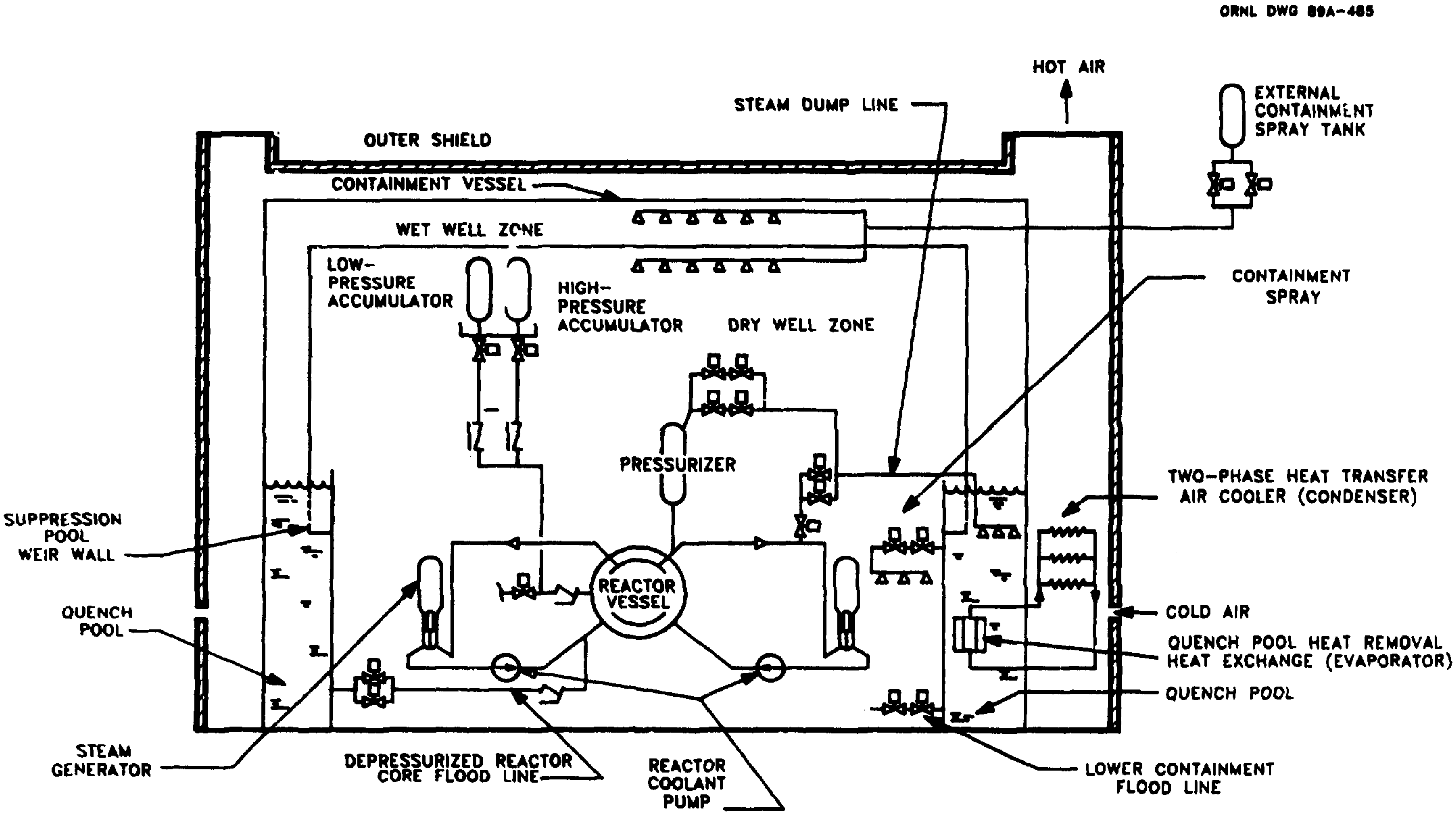

Fig. 2. Conceptual design of Mitsubishi midsize (~600 MWe) pressurized-water reactor with quench tanks and air cooler.

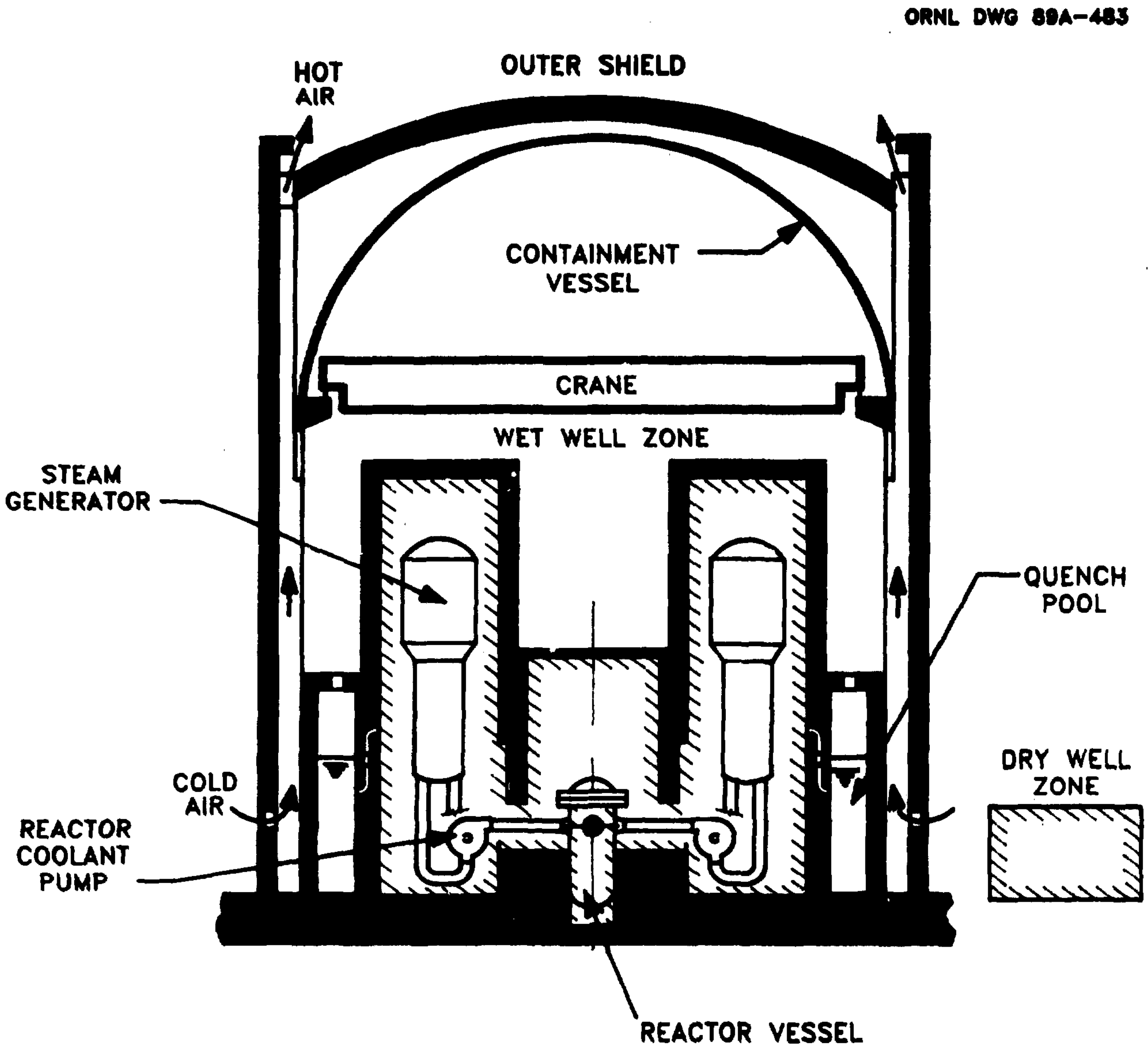

Fig. 3. Proposed Mitsubishi containment layout for midsized pressurized-water reactor.

TITLE: DECAY HEAT REMOVAL WITH SEAWATER

Functional Requirements: 1.2.3 Transport Heat to Ultimate Heat Sink

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Boiling-water reactor (BWR)

Organization: Vendor (Toshiba)

Examples of Implementation: None

Description: This system is similar to the General Electric "water wall" and is proposed for use in the Toshiba TOSBWR-900P reactor, a natural circulation BWR (Fig. 1). In an emergency such as a pipe break or other major failure, the pool water discharge valve opens, flooding the lower containment with gravity-driven water from the pool above. Cooling of the reactor core is accomplished by natural circulation between the pressure vessel and the flooded containment. The outer wall of the flooded containment is also the inner wall of a seawater-filled compartment used for heat exchange. The seawater compartment is connected to the sea by upper and lower cooling pipes. Decay heat is transferred from the flooded containment to the seawater compartment, where natural circulation of the seawater through the compartment provides heat removal.

Alternative Versions: None

Status of Technology: Proposed decay heat removal system for advanced Japaneese reactors

Advantages: 1. No pumps or emergency generators are needed.

2. Unlimited supply of cold water for cooling.

Additional Requirements: None

Comments: Compared to other heat sink alternatives, the use of seawater provides an unlimited cold heat sink for decay heat removal.

References/Contacts:

1. Y. Oka, "Research and Development of Next Generation Light-Water Reactors in Japan," Presented at the International Atomic Energy Agency Technical Committee Meeting on Passive Safety Features in Current and Future Water-Cooled Reactors, Moscow, USSR, March 21-24, 1989.

Update/Compiler: April 1989/EBL

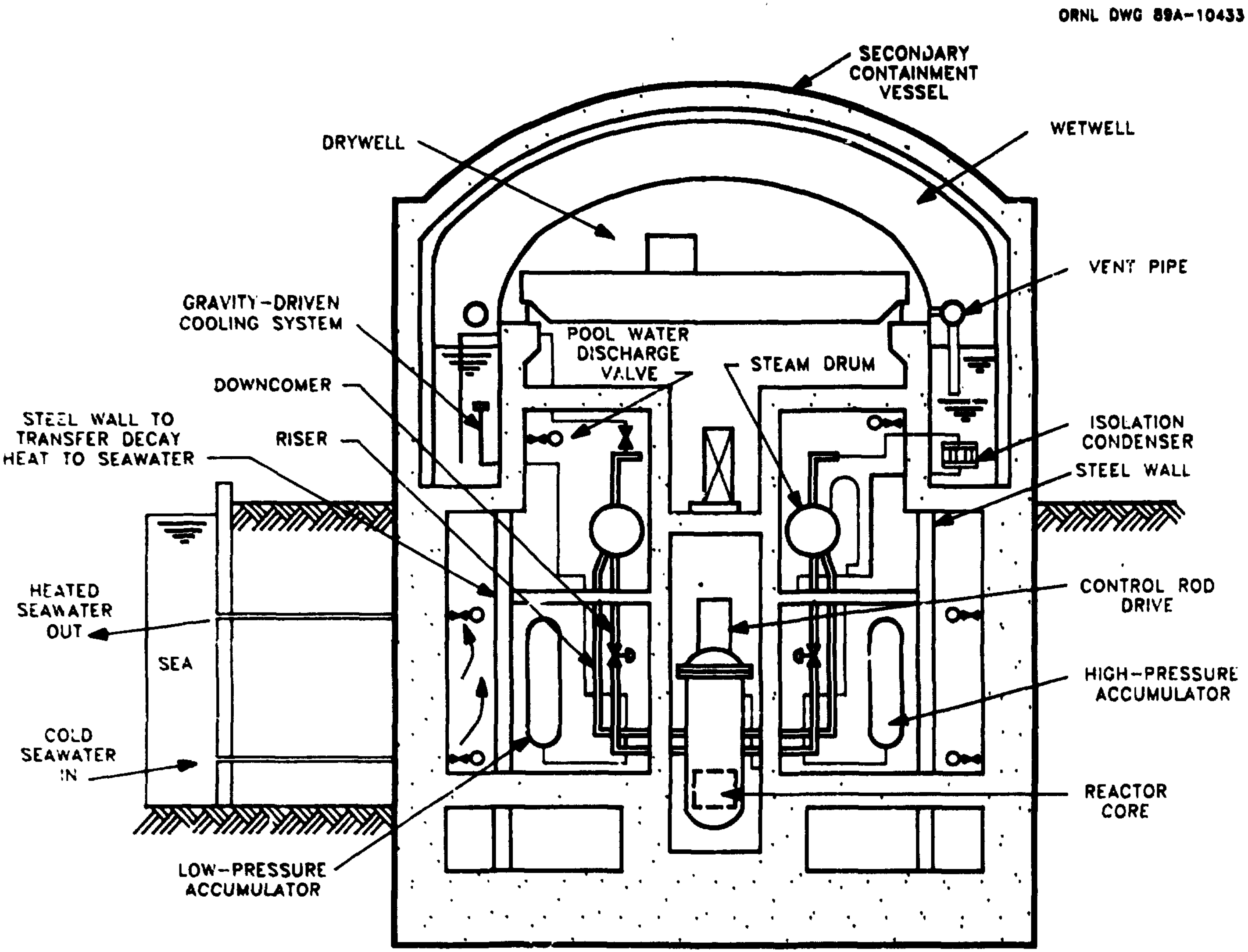

Fig. 1. Characteristics of Toshiba TOSBWR-900P with seawater post-accident decay heat removal system.

TITLE: CONDENSATION OF PRESSURIZED STEAM OR COOLING HOT WATER WITH BOILING WATER BATH

Functional Requirements: Level 1.2.3 Transport Heat to Ultimate Heat Sink

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (Multiple)

Examples of Implementation: None

Description: Several systems have been proposed to dump decay heat from the reactor core to an external bath of clean, nonradioactive water. Heat is removed from the water bath by boiling the water. Long-term heat removal is obtained by refilling the bath with water or condensation of exit steam in air coolers.

1. Boiling-Water Reactor (General Electric) - A water bath heat sink is proposed for use with a small, simplified BWR [600 MW(e)] with several passive safety systems (Fig. 1). In this concept, several safety systems dump reactor decay heat to a suppression pool (1,2). In turn, the suppression pool dumps heat to a refill water pool. The refill water pool dumps the heat to the environment by boiling of refill pool water. Decay heat is dumped to the suppression pool by two systems:

   - When it is necessary to isolate the reactor from the turbine, an isolation condenser dumps reactor decay heat to a suppression pool. Steam from the reactor is sent to the isolation condenser in suppression pool, is condensed to water, and the liquid water flows back to the reactor vessel by natural circulation.

   - In the event of a loss-of-coolant accident, depressurization valves vent steam from the reactor to a suppression pool positioned above the pressure vessel. When the pressure in the pressure vessel becomes sufficiently low, check valves in the suppression pool-pressure vessel lines open, allowing the suppression pool water to flow by gravity into the pressure vessel and provide core cooling. The reactor pressure vessel has no large pipes attached near or below core level, allowing the core to remain fully covered for all design basis events. Cooling is maintained by natural water circulation between the pressure vessel and the suppression pool.

   Operation of the above isolation condenser or emergency cooling system dumps the decay heat to the suppression pool and rapidly raises its temperature. The suppression pool dumps heat to a boiling water bath. One suppression pool wall is a steel annulus filled with water and vented via a refill pool to the atmosphere. Decay heat is transferred from the suppression pool water to the refill pool "water wall" and discharged to the atmosphere by the release of steam. This enables heat removal from the suppression pool water while the fluids remain contained. The fission products are retained in the suppression pool. The water volume of the refill pool enables cooling for three days without operator action and without containment venting of suppression pool water. If longer-term cooling is required, additional water can be supplied to the refill pool.

2. Pressurized-Water Reactor (Combustion Engineering, France) – A water bath condensor to condense decay heat steam from the steam generators during the loss of A.C. power or during other adverse conditions is proposed for the Safe Integral Reactor (SIR) by Combustion Engineering (3). The option has also been studied in France. The water bath condensor would be adjacent to the reactor containment and above the level of the steam generators. In operation: (1) steam from the steam generators flows to the water bath, (2) the heat is rejected through a condensor which boils bath water and condenses steam from the steam generator, (3) the condensed water flows by gravity back to the steam generators. In the SIR design, four of the twelve steam generators can be used for decay heat removal in this mode of operation (see: Integral PWR with no Primary System Piping). The water bath is sized to remove decay heat for 72 hours by water boiloff.

   The French design (4) of a steam condensor water bath is shown in Fig. 2. The condenser is composed of several series of interconnected vertical steam pipes with a common connector across the top and bottom of each series. The connector at the top serves as a steam inlet, while the one at the bottom collects the condensate formed. The condenser is totally immersed in water. Steam enters the condenser tube inlet and is cooled by the transfer of heat through the tubes to the surrounding water. The condensate that is formed runs down the inside of the tubes to the bottom connecter and is returned to the feed circuit for the steam generator. Since the cooling power of the condenser is practically proportional to the length of tubes free of condensate, the extent of cooling can be controlled by regulating the rate of condensate flow back to the steam generator feed circuit.

   The boiling of water at the outer tube surfaces produces a continuous circulation of cooling water from the bottom to the top of the tubes. Water contacting the tubes begins to form steam, which decreases the density of the fluid in contact with the tube walls. Consequently, the clean steam/water fluid rises rapidly along the tubes and is replaced by cooler, higher-density water. The clean steam exits the pool and contacts a grid just above the surface. The grid removes most of the water droplets entrained in the steam and returns them to the cooling volume, where they recirculate down the outer edges of the pool to the bottom of the cooling tubes. The clean steam then passes through a dryer in the exit chimney and is vented to the atmosphere. Water lost in the vented steam is replaced from a makeup volume. This design results in a very compact boiling water bath.

3. Pressurized-Water Reactor (Italy) - A modified water bath heat sink is proposed as part of the Multipurpose Advanced Reactor inherently Safe (MARS) pressurized water reactor decay heat removal system. MARS (5,6) has a passive decay heat removal system which is initiated on main reactor recirculation pump shutdown or failure due to steam in the primary system. (See: Main Recirculation Pump Failure Initiated Core Cooling and Shutdown Systems). In this system (Fig. 3), decay heat from the reactor primary circuit is transferred from the primary circuit to the water bath by an intermediate natural circulation heat transfer loop. The primary circuit coolant line to the natural circulation heat transfer loop lower heat exchanger has a one way valve. During normal operations, when the main reactor recirculation pumps are operating, the pressure balance in the primary system attempts to circulate water to the natural circulation heat transfer loop lower heat exchanger but

the one way valve prevents water flow. If the pump stops, the pressure balance in the primary reactor circuit changes and hot primary water or steam by natural convection circulates to the natural circulation heat transfer loop lower heat exchanger and dumps decay heat. Stopping pump operations reverses the fluid flow in primary system pipes to the decay heat system.

The natural circulation heat transfer loop dumps its heat to the water bath and the water bath loses heat by boiling of the water. The steam is dumped to the environment through natural circulation air coolers. The air coolers condense most of the steam to water which returns to the water bath. The combination of water bath with air-cooled steam condenser has several advantages:

- The air-cooled steam condensers provide an "infinite" long-term cooling capacity for the water bath.

- The water bath significantly decreases the air-cooled steam condenser size. After reactor shutdown, the decay heat load is initially high but rapidly decreases. The water bath initially loses a fraction of the water that is turned to steam when the decay heat load is high. It is capable of handling very high short-term heat rejection loads. The air condensers are designed to handle the smaller decay heat load hours after reactor shutdown. In effect, the water bath/air condensers can be "matched" to the changing decay heat loads with time.

Alternative Versions: None

Status of Technology:

1. General Electric System - under active development for the Advanced Simplified BWR.

2. Combustion Engineering System - under active development for the Safe Integral Reactor (SIR).

3. Italian System - under active study.

Advantages:

1. No pumps or emergency generators are needed.

2. With all of these systems, there is no need for containment venting of radioactive steam.

3. With the Italian system, decay heat removal is possible for extended times by condensation of steam.

Additional Requirements: With these concepts, the ultimate heat sink is boiling water. Water at atmospheric pressure boils at 100° C. For heat to be transferred from the condensing steam or hot water to the boiling water bath, steam condensation and hot water temperatures must exceed 100° C with corresponding above-atmospheric pressures. This implies that pressure integrity of the coolant system or containment system is required. While pressure integrity is not necessary for reactor safety a priori, it is required for these systems.

Comments: None

References/Contacts:

1. General Electric Company, Preliminary Conceptual Design Study for a Small LWR, EPRI NP-5150M, (June 1987).

2. "Small and Simplified BWR with Passive Safety Systems Looks Promising," Nuclear Engineering International, (June 1987).

3. R. A. Matzie, J. Longo, R. B. Bradbury, K. R. Teare, and M. R. Hayns, Design of the Safe Integral Reactor, Combustion Engineering, TIS-8471 (1989).

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Update/Compiler: October 1989/CWF

ORNL DWG 89A-169

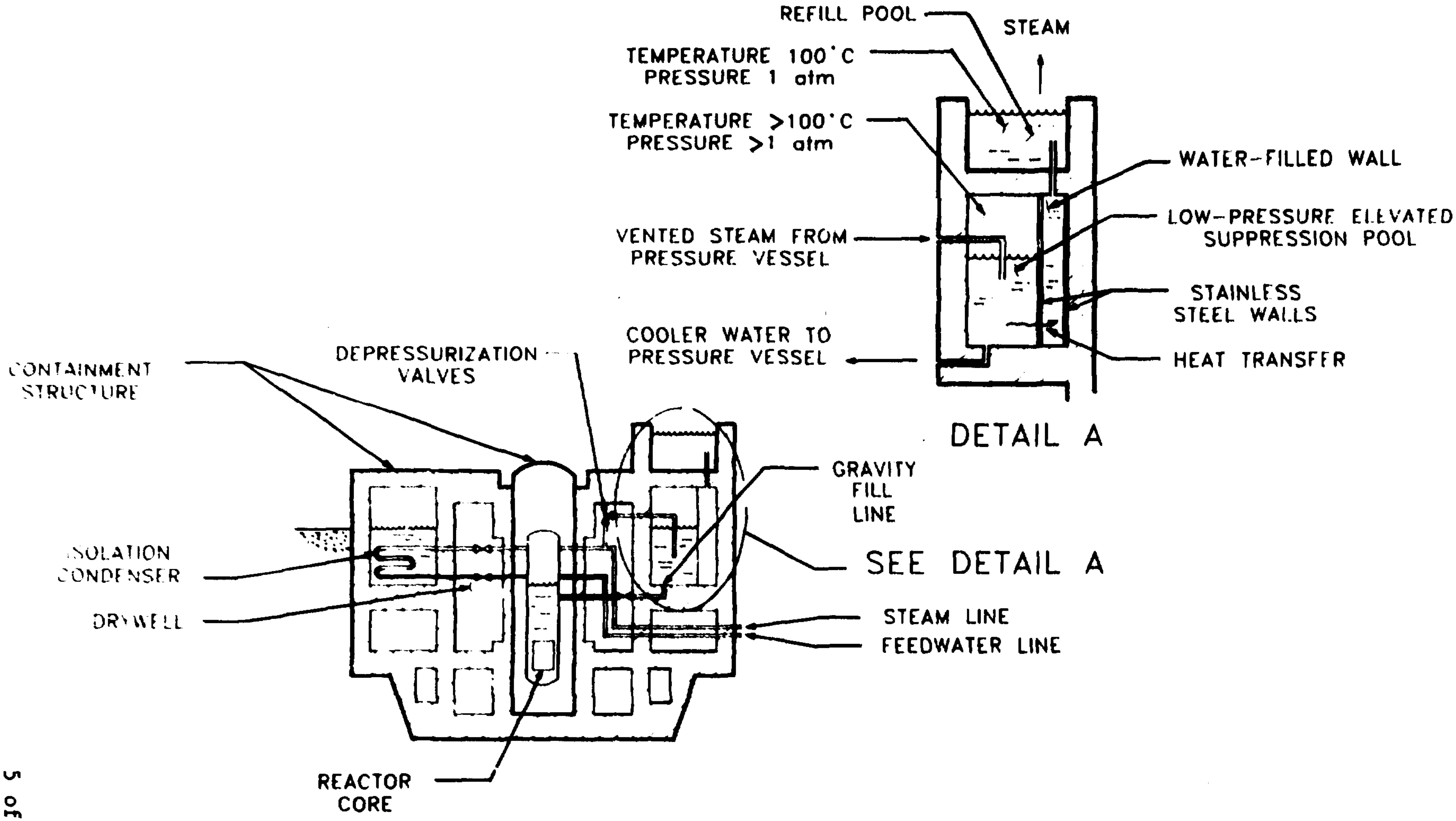

Fig. 1. General Electric 600-MWe BWR concept.

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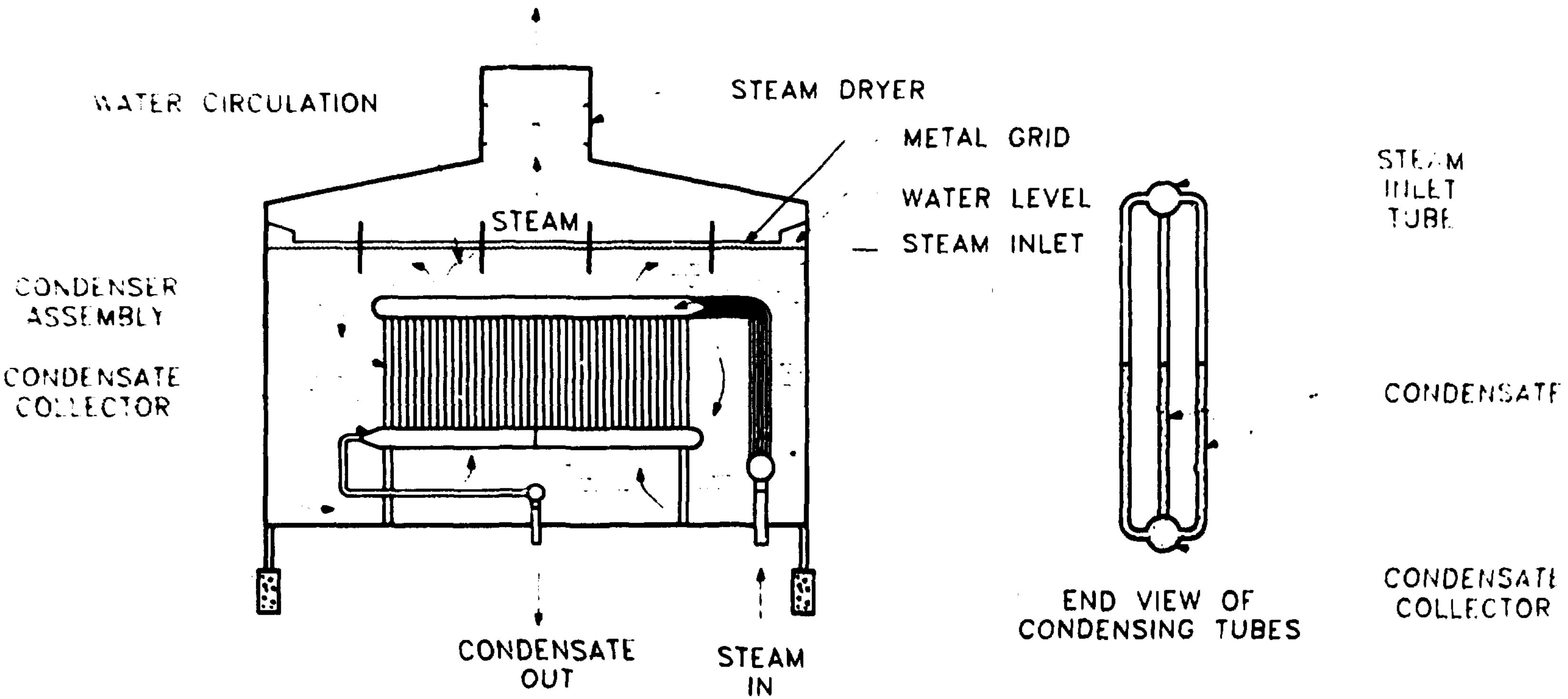

Fig. 2. French passive steam condenser design.

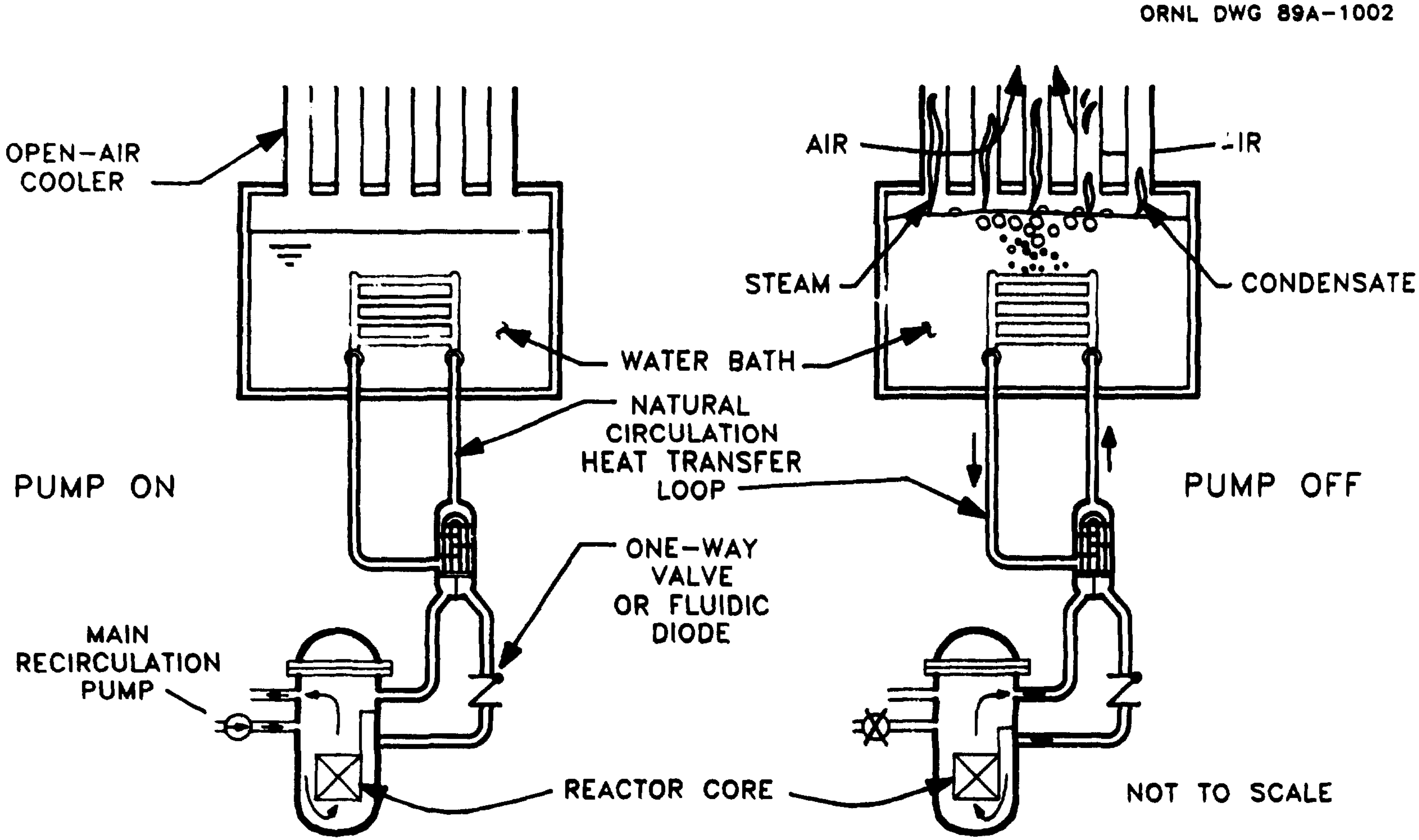

Fig. 3. Schematic of pump-failure-initiated decay heat removal system with air cooler for pressurized water reactor.

CHAPTER 7

Structures, Systems and Components to Control Chemical Attack of Clad

Function 1.3

Table 7.1 Structures, Systems, and Components to Control Chemical Attack of Clad: Function 1.3

Description Title	Function	Location of Description	Page
Reduction of Coolant/Clad Chemical Reactions Under Severe Accident Conditions	1.3	Chapter 7	7-3
Chemical Getters to Protect Interior of Fuel Pins	1.3	Chapter 7	7-7

TITLE: REDUCTION OF COOLANT/CLAD CHEMICAL REACTIONS UNDER SEVERE ACCIDENT CONDITIONS

Functional Requirement: Level 1.3: Control Chemical Attack of Clad
Level 1.2.3: Transport Heat to Ultimate Heat Sink
Level 3.1.2: Control Energy Release In Containment

Safety Type: Inherent/Passive

Developmental Status: 4 Analyzed

Reactor Type: Light-Water Reactor (LWR)

Organization: Electric Power Research Institute

Examples of Implementation: None

Description: The major materials for reactor core structural applications are several types of Zircaloy and stainless steels. Both materials have excellent mechanical properties. Historically, stainless steel was first used in LWR cores, but more recently, Zircaloy has replaced stainless steel for most in-core applications. Zircaloy has better nuclear properties (lower neutron cross sections) than stainless steel and, hence, has lower nuclear fuel costs than stainless steel. Both of these materials have characteristics that make them excellent structural materials during normal reactor operations.

The most important and difficult structural requirement in the reactor core is the cladding of the uranium oxide fuel. This material is exposed to the highest temperatures and radiation levels of any structural materials in the reactor core. The primary fuel cladding material used today is Zircaloy.

Under severe accident conditions, these materials deteriorate.[1]

1. In the event of a loss-of-cooling accident (LOCA) in an LWR, the reactor core heats up from decay heat. Eventually, the metal fuel cladding will balloon. When ballooning occurs, cooling channels in the fuel assemblies may be blocked and it may no longer be possible to stop an accident by addition of cooling water to the reactor core. Furthermore, fuel clad will begin to fail. The maximum time available for an emergency core-cooling system (ECCS) to start operating after an LOCA initiation depends on how high the clad temperature can go before fuel failures. The higher the acceptable clad temperatures before failure, the more time available to stop an accident. Zircaloy begins to lose its tensile strength at ~600°C. Stainless steel loses its strength at somewhat higher temperatures. Therefore, higher-temperature fuel clad materials will provide more time in an accident before a core meltdown.

2. Zircaloy and stainless steel react with water to form metal oxides and hydrogen. This has multiple adverse consequences.

   a. These reactions are highly exothermic. Analysis of core melt scenarios indicates that during certain parts of an accident, more heat may be generated by these chemical reactions than from radioactive decay heat.

b. The steam metal reaction generates hydrogen. The hydrogen creates the potential to pressurize the containment or create hydrogen/oxygen explosions if the containment is not inerted. In many accident scenarios, this hydrogen is the major threat to reactor containment integrity.

Figure 1 shows the reaction rate[2] of Zircaloy and stainless steel with steam. Zircaloy reacts faster with steam at lower temperatures, but stainless steel reacts faster at high temperatures.

The replacement of Zircaloy clad with better materials has the theoretical potential for gains in safety and costs. Better fuel clads could reduce requirements (and hence costs) for ECCSs and reactor containments.

There are two sets of options to reduce or eliminate coolant and clad chemical interactions. The first set of options (passive safety) is to choose metals or metal coatings for clad that do not react as rapidly with high-temperature steam as Zircaloy. The second set of options (inherent safety) is to use ceramics, which do not react with water, as cladding materials. Most metals can react with water to produce hydrogen. This does not normally happen because the kinetics of the chemical reactions are very slow. For example, from a thermodynamic perspective, aluminum should react rapidly with water or air. This does not occur because aluminum forms a protective oxide layer when exposed to air or water. In contrast, a ceramic such as aluminum oxide is thermodynamically more stable than water and, hence, cannot react with water.

A recent Electric Power Research Institute (EPRI) report[3] reviewed possible alternative metal cladding materials and possible coatings for Zircaloy for LWRs. The report conclusions are as follows:

1. "The steam oxidation resistance of Zircaloy-clad rods may be improved by use of silicide, aluminide, flame-sprayed zirconia, or zirconium-beryllide coatings."

2. "Alloys of molybdenum and niobium exhibit improved oxidation resistance and high temperature strength compared to different types of Zircaloy." They are possible replacements for Zircaloy.

3. There are significant economic penalties with these alternative clad materials.

4. Large research and development efforts would be required to develop any alternative LWR cladding materials to a commercial status.

The second option is to replace Zircaloy cladding with ceramic clad fuels. The long-term potential of such materials is great. Some known nuclear ceramic materials, such as those used in high-temperature gas-cooled reactors (HTGRs), can survive temperatures in excess of 1600°C[4] before onset of fuel failure. However, because current ceramics are difficult to fabricate and brittle, they probably cannot be used in the form of long tubes. The use of ceramic clad fuel may imply the need to change the fuel form – probably to small (<10 cm) spheres. This is a radical change in technology and core design. Proposals for such LWR fuel/reactor designs have been made and limited experimental work has been done. (see SSC: High-Temperature Ceramics, Clads, and Fuels for LWRs.

Alternative Versions: None

Status of Technology: Limited theoretical and experimental studies have been conducted to identify and develop new fuel cladding materials for LWRs.

Advantages: Reduction or elimination of chemical interactions between clad and coolant offers the potential for major gains in economics and safety. These include the following:

(1) Higher temperature clad materials allow use of ECCSs with slower response time after initiation of an accident. This may be a safety and cost advantage.

(2) Elimination of hydrogen generation from clad/coolant interactions in a serious accident reduces potential for reactor containment failure due to overpressure or a hydrogen/air explosion.

(3) Elimination of hydrogen generation from clad/coolant interactions in a serious accident may reduce containment design requirements and, therefore, costs.

Additional Requirements: None

Comments: Development of a new clad material for LWR fuel would require a major, long-term program. There are major economic issues associated with alternative metal clad materials. Most alternative materials imply high neutron absorption loses to clad with the need for more expensive, more enriched uranium. No detailed studies exist that quantify the benefits if a much higher-temperature, less chemically reactive fuel clad could be developed. From a research perspective, the incentives for better clad materials needs to be quantified.

References/Contacts:

1. H. Ocken, "An Improved Evaluation Model for Zircaloy Oxidation," Nucl. Technol., 47, 343 (Feb. 1980).

2. H. C. Brassfield, J. F. White, L. Sjodahl, and J. T. Brittel, Recommended Property and Reaction Kinetics Data for Use in Evaluating A Light-Water-Cooled Reactor Loss-of-Coolant Incident Involving Zircaloy-4 or 304SS-Clad UO_2, GEMP-482, April, 1968.

3. L. Goldstein, O. Reyes, and A. A. Strasser, Evaluation of Fuel-Cladding Properties at High Temperatures, EPRI NP-5427, April, 1988.

4. W. Katscher, "Coated Particle Fuel Element For Pressurized-Water Reactors," Nucl. Technol., 35, 557 (Sept. 1977).

Update/Compiler: Dec. 1988/CWF

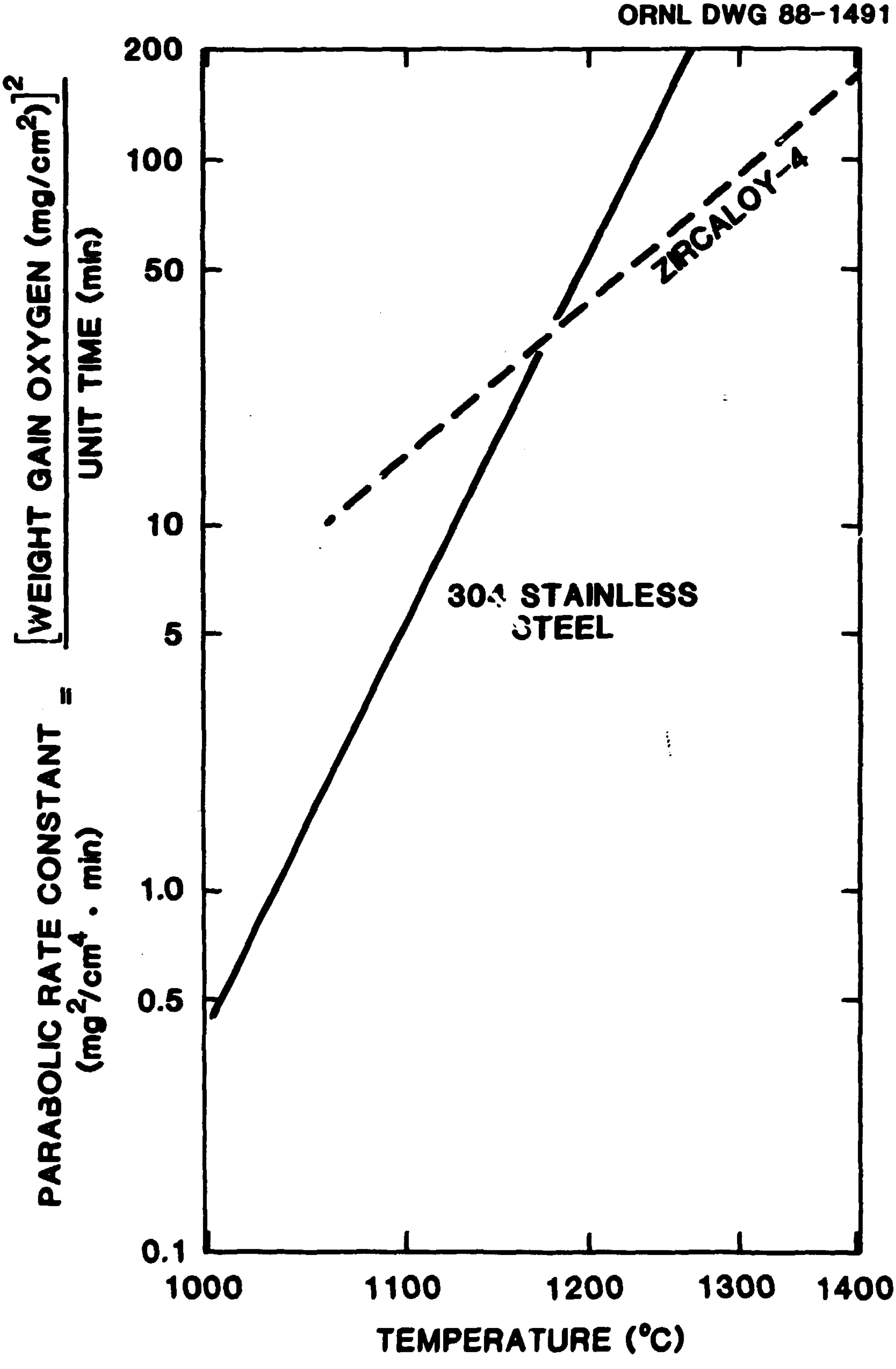

Fig. 1. Reaction rates of type 304 stainless steel and zircaloy-4 in steam.

TITLE: CHEMICAL GETTERS TO PROTECT INTERIOR OF FUEL PINS

Functional Requirements: 1.3 Control Chemical Attack of Clad

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organizations: Vendors (Multiple)

Examples of Implementation: None

Description: In a nuclear reactor, fission products produced in the fuel elements and impurities from fuel fabrication can react with and corrode the fuel rod cladding. To protect against corrosion, getters can be used to react with the fission product or other impurity responsible for the corrosion. For example, fission-produced iodine can corrode the zirconium oxide normally present on zircaloy fuel cladding and react with the free zircaloy underneath. Since the oxygen partial pressure in a fuel rod is probably extremely low, not enough oxygen is present to heal the corroded zirconium oxide.

The chemical getter, e.g., copper oxide or nickel oxide, can be fabricated in various shapes and attached to the fuel rod by several different means. In the example shown in Fig. 1, pellets of the getter are located at the top of the fuel rod. The fuel rod has a central channel and an outside channel that communicate with each other via transverse passages. Due to the temperature differences inside the fuel rod, fission product gases circulate up the central channel, through the getter pellets, and back down through the outer channel. In this example, the copper oxide or nickel oxide getter would remove iodine from the gas stream and release oxygen to help heal any corroded zirconium oxide in the cladding.

Alternative Versions: Chemical getters can be used for control of impurities, such as moisture, from manufacturing.

Status of Technology: Commercial technology in Boiling-Water Reactor fuel fabrication for hydrogen control. Proposed but not demonstrated for control of other elements.

Advantages: Fuel rod corrosion is prevented by passive means.

Additional Requirements: None

Comments: None

References/Contacts:

1. J. S. Armijo, Fuel Rod for Nuclear Reactors, Swedish Patent Application No. 8300424-2, (August 1983): ORNL/OLS-89/6, (1989).

2. S. Junkrans and G. Vesterlund, Fuel Rod for Nuclear Reactor, Swedish Patent No. 7501252-6, (April 1978): ORNL/TR-89/16, (1989).

3. Japanese Patent No. 63-275989, English abstract (No Translation Available)

Update/Compiler: May 1989/EBL

ORNL DWG 89A-11580

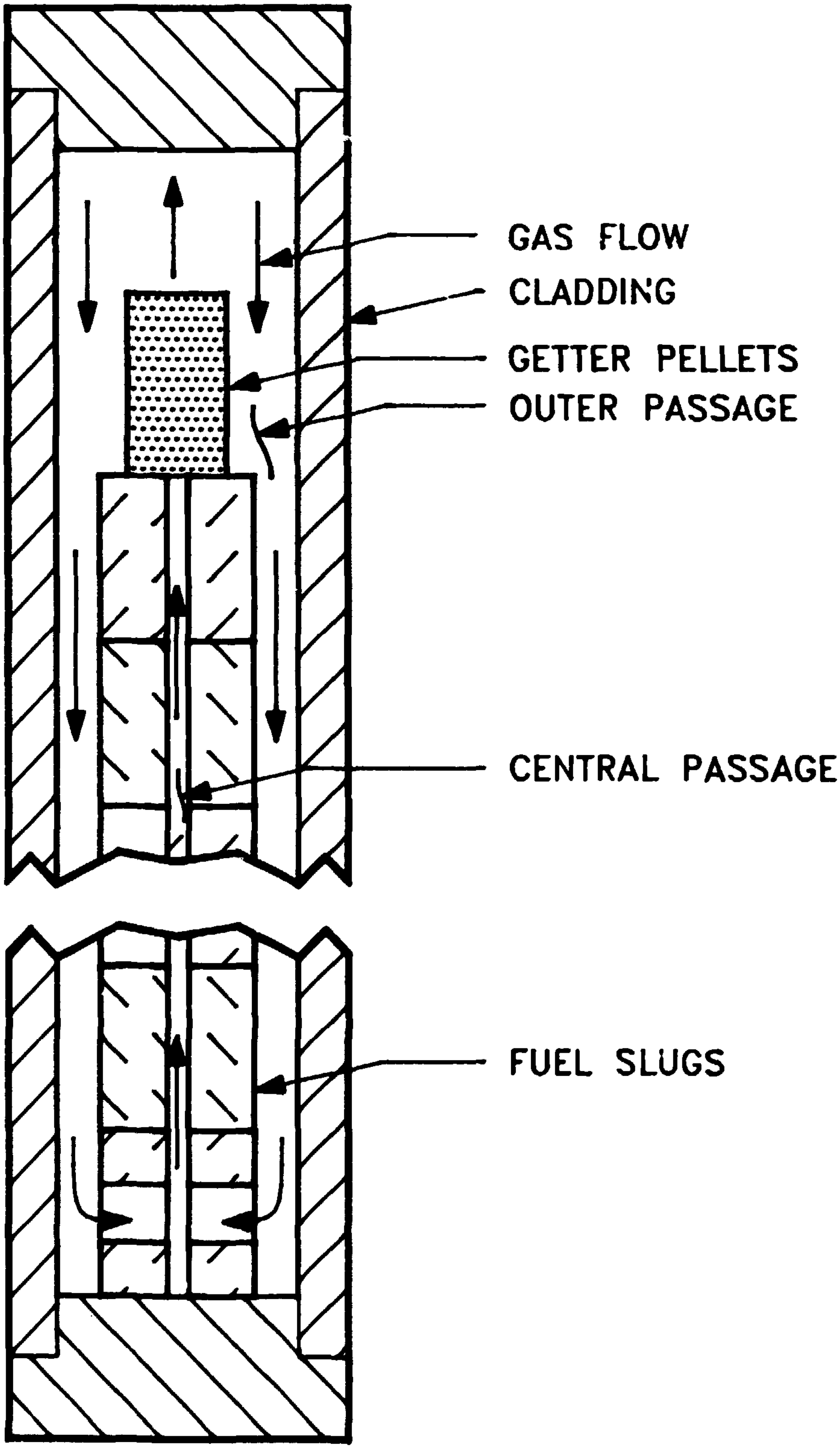

Fig. 1. Design of fuel rod using getters to prevent corrosion.

CHAPTER 8

Structures, Systems and Components to Maintain Coolant Boundary Integrity

Function 2.1

Table 8.1 Structures, Systems and Components to Maintain Coolant Boundary Integrity: Function 2.1

Description Title	Function	Location of Description	Page
Double Pressure Vessel	2.1	Chapter 8	8-3

TITLE: DOUBLE PRESSURE VESSEL

Functional Requirement: Level 2.1 Maintain Coolant Boundary Integrity
Level 1.2.1 Maintain Core Coolant Boundary Integrity

Safety Type: Passive

Development Status: 4 Analyzed

Reactor Type: Pressurized-water reactor (PWR)

Organization: University [University of Rome (Italy)]

Examples of Implementation: None

Description: A major type of postulated accident for LWRs is failure of the primary system pressure boundary such as a pipe break. One proposed solution is a double pressure boundary where the primary system is inside a secondary pressure boundary (Fig. 1). The key design characteristics of this system include the following:

- The primary system is designed to handle full system pressure.
- The secondary pressure boundary is designed to handle full reactor system pressure and normally operates at primary system pressure.
- The space between the primary and secondary pressure boundaries is filled with pressurized water at 70°C. The exterior of the primary pressure boundary is insulated to minimize heat losses to the cool water.

During normal operations, the primary pressure boundary for this system is under near-zero strain since the inside and outside pressures are approximately equal. A leak in the primary system would be identified by trace radioactivity in the cooler water outside the primary system.

The secondary pressure boundary is at full reactor system pressure. If a failure occurs, only cold water exits the secondary pressure boundary with no release of steam and no large energy release from depressurization of the cool water. The primary pressure boundary then takes over the function of maintaining coolant boundary integrity.

Alternate Versions: Both steel and prestressed concrete are options for the secondary pressure boundary.

Status of Technology: This option is currently being investigated at the University of Rome (Italy).

Advantages:
1. Passive double pressure boundary to protect against single pressure boundary failure.
2. Cold water zone between two pressure boundaries implies low energy release to containment after a postulated pressure boundary failure.

3. Secondary pressure boundary with cold water results in very low radiation levels near the reactor inside containment during normal operations. The water acts as a shield.

Additional Requirements: None

Comments: The double pressure vessel is one component of a PWR reactor concept called the Multipurpose Advanced Reactor Inherently Safe (MARS).

References:

1. Department of Energetics of La Sapienza University of Rome (Italy), Multipurpose Advanced Reactor Inherently Safe, Draft (July 1989).

2. M. Caira, M. Cumo, and A. Naviglio, "MARS Reactor: A Proven PWR Technology Combined with Advanced, Safety Requirements," Energia Nucleare 23 (May/September 1987).

Update/Compiler: October 1989/CWF

ORNL DWG 89A-977

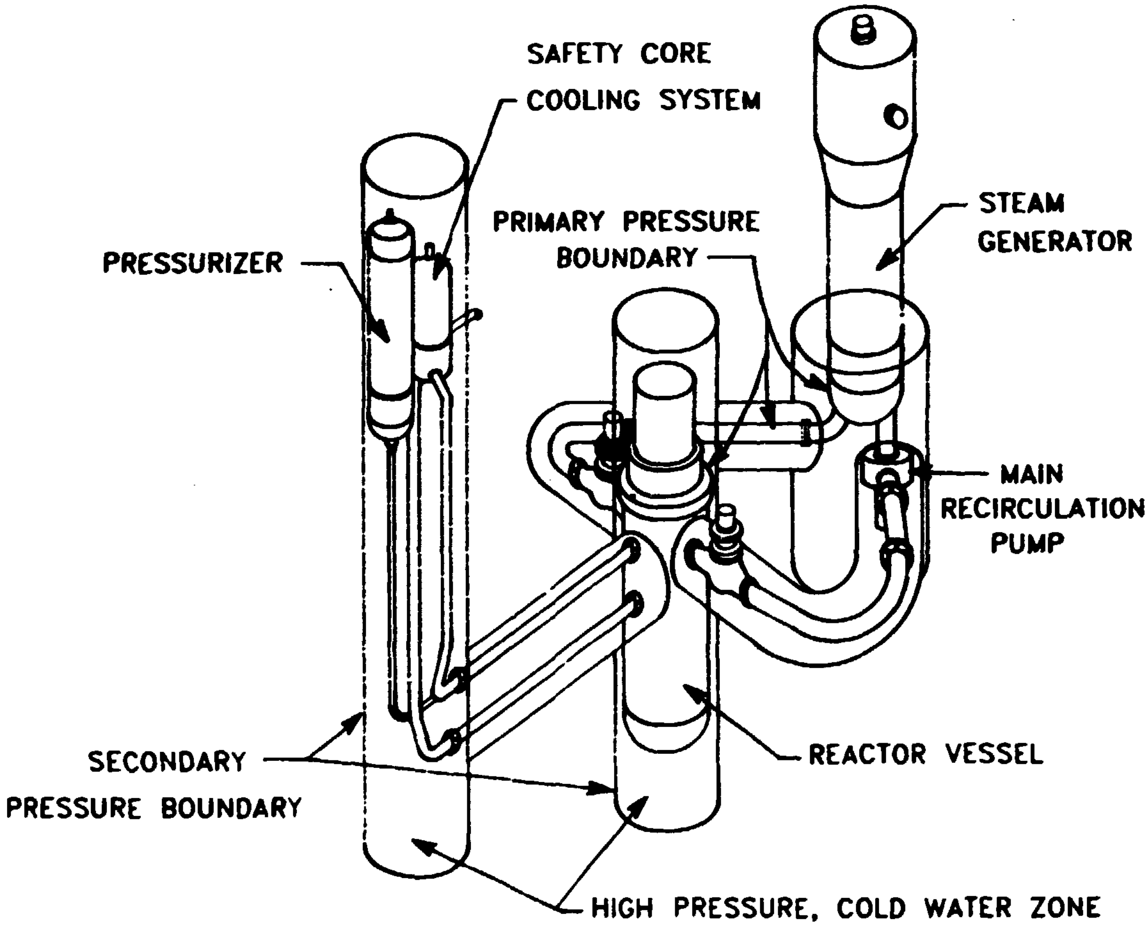

Fig. 1. Double pressure vessel system for pressurized water reactor.

CHAPTER 9

Structures, Systems and Components to Control Primary Circuit Pressure

Function 2.2

Table 9.1 Structures, Systems and Components to Control Primary Circuit Pressure: Function 2.2

Description Title	Function	Location of Description	Page
In Reactor Core Power Fuses	2.2	Chapter 5	5-49
Safety and Relief Valves	2.2	Chapter 9	9-3
Continuous/Simultaneous Pump Pressurizer	2.2	Chapter 9	9-7
Passive Pressurizer Spray	2.2	Chapter 9	9-10

TITLE: SAFETY AND RELIEF VALVES

Functional Requirements: Level 2.2 Control Primary Circuit Pressure

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Multiple

Examples of Implementation: Pressurized-Water Reactors (PWRs) - Main steam pressure relief, pressurizer, pressurizer relief tank, surge tank, feedwater systems for turbines, various high pressure gas storage tanks, etc.

Boiling-Water Reactors (BWRs) - Reactor vessel and main steam lines (code safety), turbine related steam systems, storage tanks, etc.

Description: The ASME Boiler and Pressure Vessel Code requires that all pressure vessels be provided with a means of overpressure protection. In nuclear and chemical plants, safety and relief valves are frequently used to satisfy this safety requirement. Safety valves (Fig. 1) are spring loaded, actuated by pressure, and usually intended for gas or vapor. They are designed to be fast-acting, "popping" open when over pressure conditions exist and, except for rupture disks, reseating after the pressure drops below limits. Relief valves (Fig. 2) may also be spring loaded and operated by pressure, but are usually used for liquids and have a more modulated actuating response to overpressure. The nuclear industry commonly uses versions of both valves in liquid or vapor systems.

Code safety valves are proven performers in large volume applications in the nuclear industry. In PWRs, the pressurizer and each steam generator uses multiple safety valves in a parallel configuration where they are sized to handle the steam load under all plausible accident conditions. (The non-power operated relief valves are used in only smaller plant applications and a code design does not exist for relief valves.) In BWRs, code safety valves are used directly on the reactor vessel and main steam lines. The safety valves used in LWRs are very reliable in actuation, but are known to experience occasional problems in fully reseating.

All ASME Code safety and relief valves are required to be of the direct, spring-loaded type. Pilot-operated valves on pressure vessels must be pressure actuated and main pressure relief valves must open automatically at no more than their set pressure, discharging full rated capacity even if part of the pilot should fail. In brief, the code also requires that the disk and seat be of noncorrosive materials and that the seat be securely fastened so that it cannot be lifted by the valve disk. The full discharge capacity cannot be lost by the failure of any valve component. The code also restricts the type of valves (no weight or lever types) to ensure that it cannot be easily tampered with by operators. The Code-required actuation tolerances of safety valves are as follows:

1. Between 71 and 300 psi, ±3%

2. Between 301 and 1000 psi, ±10 psi

3. Over 1000 psi, ±1%

Safety valves include a class of nonreclosing-type protection such as that provided by fusible plugs, rupture disks, and breaking pin devices. Rupture disks utilize pressure-relief rupture diaphragms or membranes of metal or other sheet material that burst when the pressure reaches the set-to-operate or bursting pressure. Rupture disks are high-reliability, nonreclosing pressure safety devices that, in conventional usage, are generally the final level of protection when all else fails.

Rupture disks are often used in series with relief valves, providing protection to the relief valves when corrosive liquids or gases are involved (Fig. 3). In this type of application, the ASME Code requires that the space between the relief valve and the disk be monitored by a suitable indicator to show whether the disk is cracked or leaking. Sufficient backpressure in this area may prevent the rupture of the disk at its design pressure. When used alone, the disk exhibits the following features: (1) leak tightness, (2) low cost, (3) low maintenance, (4) fixed setting, (5) nonreclosing (but readily replaceable), and (6) disposable. Rupture disks are available in a wide variety of materials, sizes, shapes, rupture pressures, tolerances, and maximum operating temperatures. Standard rupture disk materials are aluminum, silver, nickel-200, monel-400, inconel-600, and 316 stainless steel. Larger rupture disks may be prebulged and are generally installed in a safety head in which the disk is secured by several bolts.

Rupture disks are available for a wide range of temperatures and pressures. While the maximum recommended temperatures of aluminum and silver disks are 250°F, Inconel-600 and 316 stainless steel are suitable up to 900°F Special materials such as platinum are used at even higher temperatures. Rupture pressures range from a few psig to 3,000 psig and higher, with the expected rupture tolerance between -5 and +5% (for rupture pressures above 15 psig).

The reliability of rupture disks is excellent and out-of-specification ruptures due to mishandling, poor installation, aging, and corrosion are in the safe direction (i.e., at pressures below the set-to-operate pressure). Material selection criteria should include considerations of creep, fatigue, and corrosion as well as those operating conditions contributing to each (e.g., temperature, pressure, pulsating or fluctuating service, and chemical reactivity). Material fatigue can be reduced through the use of thicker disks and certain disk shapes. Scheduled inspections and/or maintenance can be performed to increase rupture disk reliability and burst accuracy if necessary.

Alternative Versions: Rupture disks can be advantageously used in conjunction with safety and relief valves. For instance, the use of a relief valve in parallel with the rupture disk can provide a smooth, carefully modulated response to overpressure, preserve the disk, and provide automatic resealing. When the relief-valve set pressure is 5 to 10 % below that of the rupture disk, the disk serves in a backup capacity in case of relief valve failure. Rupture disks, in series with and on the inlet side of safety and relief valves, can improve their reliability by isolating them from corrosive substances prior to rupture. Rupture disks may also be used on the outlet side of safety valves to prevent leakage of the system through the safety valve and to protect the valves from corrosion due to exposure to atmosphere or other corrosive outlet environments.

Status of Technology: Safety and relief valves have been widely used in industry for decades. The chemical and nuclear industries are predominant users of each.

Advantages:

1. Safety and relief valves are actuated by pressure and are simple, reliable valves.
2. The full-actuation response of safety valves to impinging overpressure is instantaneous.
3. Relief valves can provide gradual pressure relief, as required, below the set pressures of safety valves.
4. Rupture disks cannot fail to operate beyond their burst pressure.
5. Rupture disks have no moving parts except for fragments of disk after failure.
6. Rupture disks boast low cost and low maintenance.
7. The leak tightness of a rupture disk is excellent (exceeding that of metal-seated safety and relief valves).

Additional Requirements: None

Comments: None

References/Contacts:

1. L. R. Harris, "Select the Right Rupture Disk," Hydrocarbon Processing, 62, (5) (May 1983).
2. Anthony L. Kohan, Pressure Vessel Systems, McGraw Hill Book Company, New York (1987).
3. "Hydraulic Pressure and Flow Valves," Machine Design, 56, (22) (September 27, 1984).
4. E.R. Cunningham, "Keeping Fluid Handling Systems Safe with Overpressure Protection," Plant Engineering (February 14, 1985).

Update/Compiler: April 1989/KHS and GAM

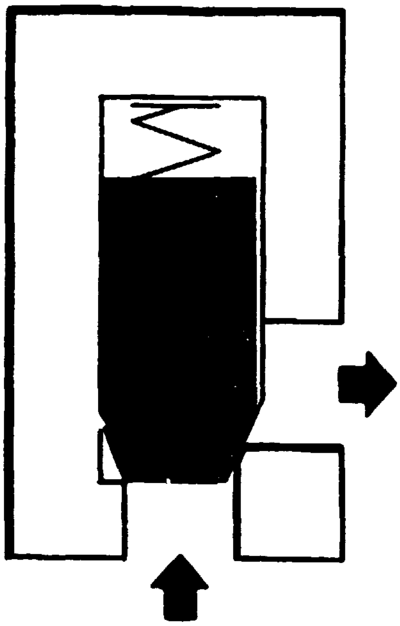

Figure 1 Safety Valve

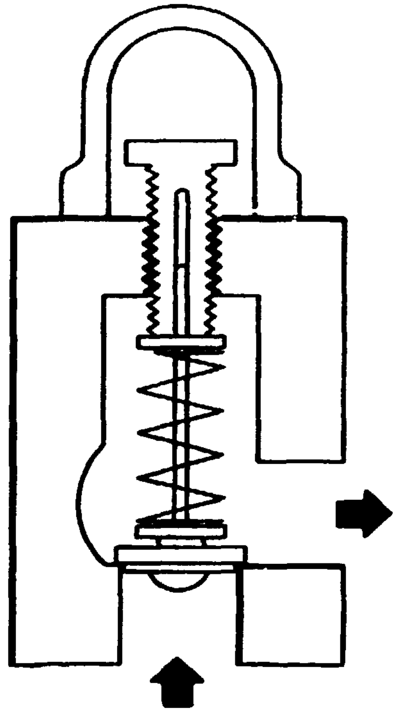

Figure 2 Adjustable Relief Valve

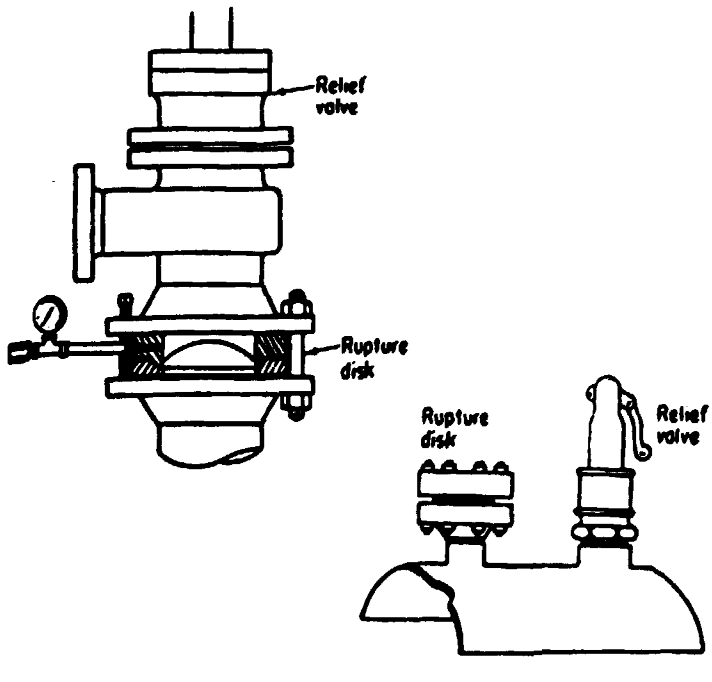

Fig. 1. Rupture disk used in series with and in parallel with relief valve.

TITLE: CONTINUOUS/SIMULTANEOUS PUMP PRESSURIZER

Functional Requirement: Level 2.2: Control Primary Circuit Pressure
Level 1.2.2: Maintain Core Coolant Makeup

Safety Type: Passive (See Comments)

Developmental Status: 5 Concept

Reactor Type: Pressurized-water reactor (PWR)

Organization: University (Pennsylvania State University)

Examples of Implementation: None

Description: A continuous pump pressurizer (CPP) could be used to control the primary coolant system pressure and water inventory. The CPP would replace the more conventional vapor pressurizer and associated relief valve found in existing PWR primary coolant systems.

A continuous pump pressurizer system[1-4] consists of a centrifugal injection pump, an atmospheric water tank, a regenerative heater, and a letdown system as shown in figure 1. This particular design is proposed for the Pennsylvania State University 600 MWe Light Water Ultra-Safe Plant Concept.[1] The in-containment water tank contains approximately 1.2 million pounds of 140° F water at atmospheric pressure. The passive tank cooler is used to maintain this temperature during operation. The water tank is connected to the primary coolant system through an injection pump and a modified letdown system. The regenerative heater is used to cool the water flowing from the hot leg of the reactor into the atmospheric tank and to warm the water flowing out of the atmospheric tank to the cold leg of the primary coolant system.

During steady-state reactor operation, the injection flow into the cold leg of the reactor and the letdown flow from the hot leg of the reactor are equal, with a 150 gpm flow rate. A 450 gpm flow rate returns to the atmospheric tank through the isolation valve so that the full rated injection pump flow is 600 gpm. During pressure transients, the primary coolant system pressure and coolant inventory are stabilized by the combined effects of the injector pump and the letdown system flow rates. A primary system pressure increase will cause the centrifugal injector pump flow rate to decrease and simultaneously cause the letdown flow rate to increase and therefore pressure stabilization is achieved. Alternately, a primary system pressure decrease will cause the pump flow rate to increase and the letdown flow to decrease and pressure stabilization will again be achieved. These flow characteristics of the pump and letdown system are passive in nature and require no external controlling mechanisms.

The injector pump bypass system is needed to prevent the conventional centrifugal pump from overheating during low-flow or no-flow conditions. A new type of centrifugal pump, the Cool Pump[5] may offer a simpler alternative for the design of this pressurizer system. The Cool Pump is designed with two parallel impellers with mismatched hydraulic characteristics so that during normal operation the pump functions similar to a conventional pump but during low-flow or no-flow conditions the Cool Pump will redirect the flow back to the atmospheric tank. This design eliminates isolation valves and may be more efficient. This technology is in an earlier state of development.

Alternative Version: A centrifugal pump is open to flow through the pump even if the pump is not operating. Placement of the atmospheric tank above the primary coolant system ensures gravity flow of the water to the core after a primary system depressurization whether by operator action or primary system pipe break. In effect, a gravity flow emergency core cooling system (ECCS) exists without mechanical block valves. The pump acts as a fluidic (block) valve requiring power to remain closed but opening upon the loss of power. If a major coolant pipe breaks and the system depressurizes, cooling water is provided to the reactor core passively.

Status of Technology: Some preliminary modelling of the CPP has been done. There are significant uncertainties. The use of CPP as a gravity flow ECCS system has not been investigated.

Advantages:
1. Controls primary circuit pressure and inventory during a pressure transient.
2. Possible passive gravity flow ECCS.

Additional Requirements: None

Comments: This system allows the reactor to depressurize in a controlled manor without active components and allows for gravity flow of water into the core after depressurization. The "active" components are needed for operation but not safety.

References:

1. E. Klevans, M. A. Schultz, G. Robinson, A. Baratta, E. S. Kenny, M. Edlund, I. McMaster, R. M. Edwards, J. Borkowski, J. Helsel, R. Schaffer, T. Schearer, K. Smith, L. Wang, and J. Zardas, "The Pennsylvania State University Light-Water Ultra-Safe Plant Concept: Final Report," Pennsylvania State University Department of Nuclear Engineering (May 25, 1989).

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5. S. C. Chang, J. H. Kim, "Cool Pump: A New Centrifugal Pump," Presented at the Second International Symposium on Transport Phenomena, Dynamics, and Design of Rotating Machinery, Hawaii, April 3-6, 1988.

Update/Compiler: May 1989/WJR.

ORNL DWG 88A-413

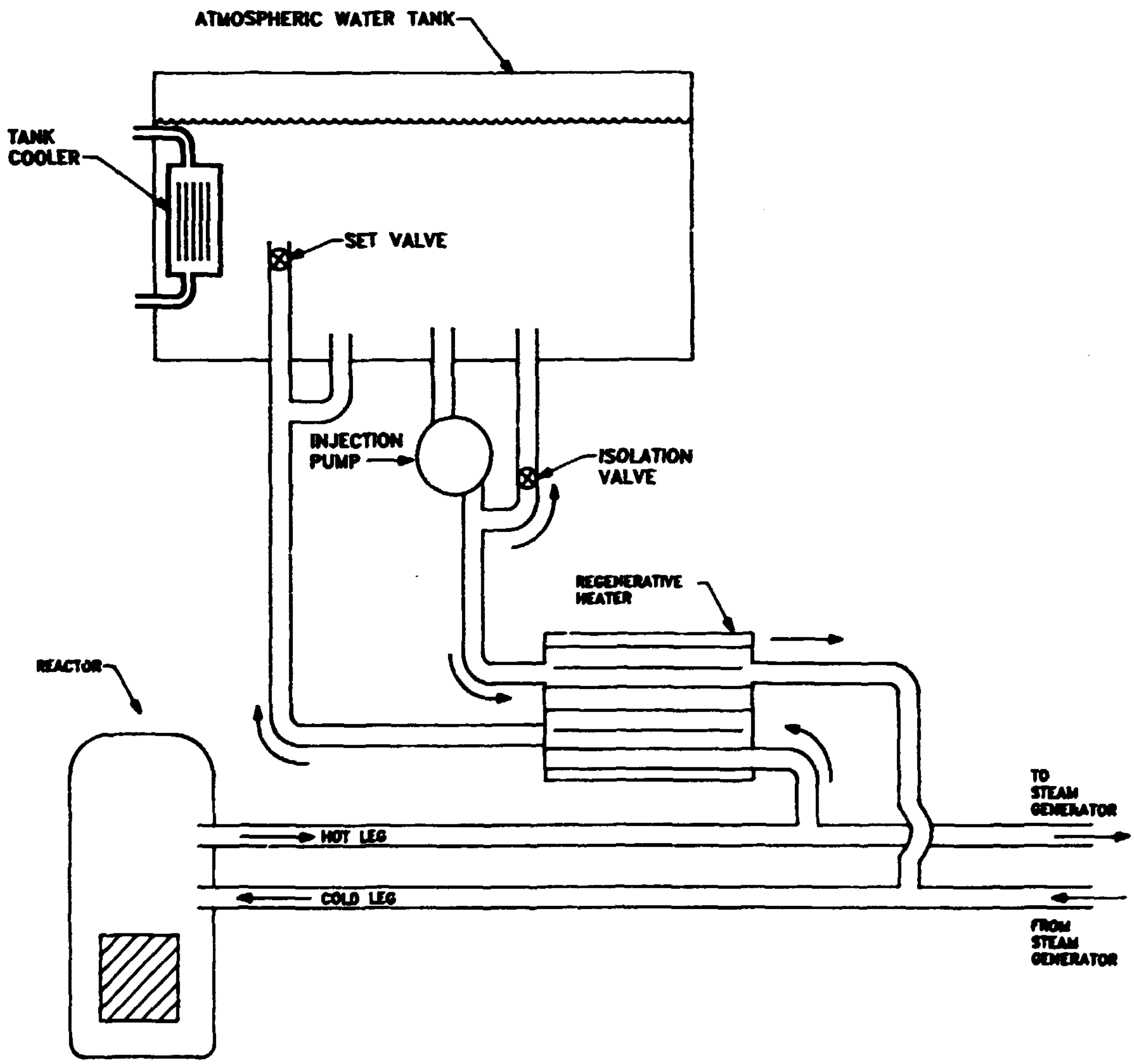


Fig. 1. Continuous pump pressurizer system.

TITLE: PASSIVE PRESSURIZER SPRAY

Functional Requirements: Level 2.2 Control Primary Circuit Pressure

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Pressurized-water reactor (PWR)

Organization: Vendor (Combustion Engineering)

Examples of Implementation: None

Description: PWRs have pressurizers to control reactor pressure. The conventional pressurizer is a tank connected to the primary system, with the top half of the tank filled with steam and the bottom half filled with primary-system water in contact with the rest of the primary system. If it is desired to increase the system pressure, electric heaters heat the water in the pressurizer and create additional steam. This increases the pressure. The pressurizer water in contact with the steam volume is at the boiling point, but the water at the bottom of the pressurizer is at the primary-system temperature, which is much lower. To decrease system pressures, colder water from the primary circuit is injected into the steam volume in the pressurizer (which condenses some of the steam).

Following decreased-heat-removal events in a PWR (e.g., loss of flow, loss of feedwater, loss of condenser vacuum, and feedwater line breaks), the increase in primary circuit pressure from thermal expansion of water in PWRs is reduced by pressurizer sprays. In conventional plants, the spray flow requires pumps. If the reactor coolant pumps (RCPs) are operating, the spray flow is driven by the pressure difference from the RCP discharge to the pressurizer. If they are not operating, a pump such as the charging pump is required to provide spray flow.

The use of fluidic diodes, a type of one-way valve with no moving parts, provides a passive means of supplying spray flow (see: Fluidic Diodes). The proposed Combustion Engineering Safe Integral Reactor (SIR) uses fluidic diodes to provide a passive internal spray system to limit reactor pressure under normal and accident conditions (Fig. 1). The SIR pressurizer is within the upper head of the reactor vessel (RV). The upper head coolant is separated from the primary circuit coolant in the lower RV by a steel plate. Penetrations in the plate include the fluidic diodes, control-rod shrouds, and pipes to spray headers. The fluidic diodes are equivalent to the surge line on conventional PWRs.

If the reactor pressure is too low, the fluidic diodes allow water to leave the pressurizer quickly as more steam is created by the pressurizer heaters. If the primary circuit water level is low, the fluidic diode allows drainage of water from the pressurizer to the reactor core.

If the reactor begins to overpressurize, the cooler primary circuit water (295 to 318°C) enters the pressurizer via the pressurizer spray and control-rod guide tubes. This cooler water is sprayed into the steam, condenses some of the steam, and lowers the pressure. During normal operations, the steam temperature is (~360°C), corresponding to a saturation pressure of 15.5 MPa (155 bar). The fluidic diode prevents large flows of water into the bottom of the pressurizer where liquid water is located and forces incoming water into the pressurizer spray lines and spray nozzles. In short, water only enters the pressurizer via spray nozzles into the steam volume and only leaves by the fluidic diodes at the bottom of the pressurizer where the coldest liquid water is located. A passive pressure control system is created.

Alternative Versions: None

Status of Technology: SIR conceptual design completed

Advantages: Passive pressurizer spray, which provides passive protection against over-pressure conditions.

Comments: None

References:

1. R. A. Matzie, J. Longo, Jr., R. B. Bradbury, K. R. Teare, and M. R. Hayns, Design of the Safe Integral Reactor, Combustion Engineering, TIS-8471 (1989).

Update/Compiler: October 1989/CWF

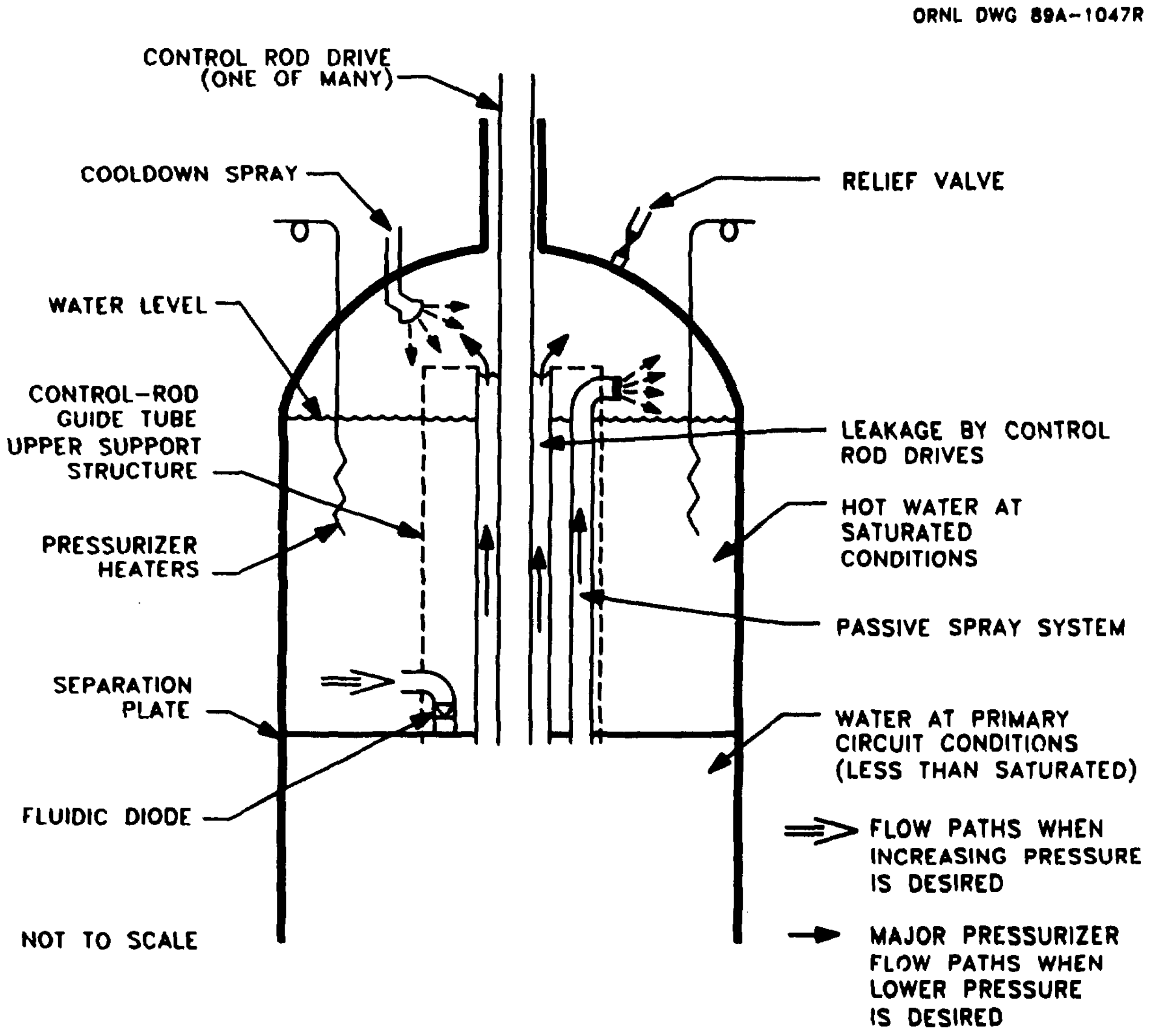


Fig. 1. Schematic of safe integral reactor pressurizer flow.

CHAPTER 10

Structures, Systems and Components to Isolate Primary Circuit From Balance of Process

Function 2.3

10. STRUCTURES, SYSTEMS, AND COMPONENTS TO ISOLATE PRIMARY CIRCUIT FROM BALANCE OF PLANT

No passive devices were identified to isolate the primary circuit from the balance of plant. The basic functional requirement is isolation of the radioactivity that may be in the primary circuit; however, in practice, this usually implies isolation of the fluids in the primary circuit containing the radionuclides. In current reactors, this task is accomplished by: (1) sensors throughout the plant, (2) a control system that determines when the primary system should be isolated, and (3) mechanical closure of the primary circuit isolation valves.

From one perspective, the steam generator in a PWR isolates the primary system from the balance of the plant. However, there are other systems that bypass this isolation, including water cleanup systems and pressure relief valve systems.

The difficulty of identifying passive devices for this functional requirement reflects, in part, a distinction between systems designed to prevent core damage (Function 1: Maintain Core Integrity) and those designed to mitigate accident consequences (Function 2: Control Transport from Primary Circuit and Function 3: Control Transport from Containment). To prevent core damage, there are a single set of conditions that define the safest state (core shutdown at cold temperatures). Many passive ways exist to ensure this condition if the reactor core is threatened. In accident mitigation, the preferred safe conditions for most (but not all) postulated accident situations can be defined and remain unchanged with time. The exceptions can impose constraints on the type of systems for accident mitigation. For example, it is desired to depressurize the reactor in certain accidents to maintain primary system integrity, but depressurization is a controlled form of temporarily eliminating isolation of the primary circuit from the containment. In effect, the preferred action by the process isolation system depends not only on local conditions, but on conditions elsewhere in the reactor. The safe direction for action may change with time.

The above problem is difficult to address at the level of passive structures, systems, and components. It can be addressed at a higher level in terms of total plant design with the use of passive devices to avoid situations in accident mitigation where the preferred action depends on conditions throughout the plant that vary with time.

Table 10.1 Structures, Systems and Components to Isolate Primary Circuit From Balance of Process: Function 2.3

None Found

CHAPTER 11

Structures, Systems and Components to Maintain Containment Boundary Integrity

Function 3.1

Table 11.1 Structures, Systems and Components to Maintain Containment Boundary Integrity: Function 3.1

Description Title	Function	Location of Description	Page
Water Quench Pool with Air Cooler for Reactor Accident and Decay Heat Sink	3.1.2	Chapter 6	6-79
Reduction of Coolant/Clad Chemical Reactions Under Severe Accident Conditions	3.1.2	Chapter 7	7-3
Reactor Containment Buildings	3.1.1	Chapter 11	11-3
Self-Sealing Ice Containment Structure	3.1.1	Chapter 11	11-11
Filtered, Vented Containment	3.1	Chapter 11	11-14
Vacuum Containment	3.1.2	Chapter 11	11-19
Passive Hard Vacuum Containment System	3.1.2	Chapter 11	11-24
Containment Pressure Control-Suppression Pool and Bubbler Condensers	3.1.2	Chapter 11	11-29
Steam Condensation and Vacuum Creation by Suppression Pools	3.1.2	Chapter 11	11-37
Water Coolable Core Catcher	3.1.2	Chapter 11	11-44
Containment Pressure Control by Pressure Activated Water Sprinklers	3.1.2	Chapter 11	11-49
Temperature Activated Water Sprinklers for Steam Condensation and Accident Mitigation	3.1.2	Chapter 11	11-54
Containment Pressure Control: Ice Condensers	3.1.2	Chapter 11	11-57
Containment Cooling with Existing and Augmented Heat Capacity of Containment	3.1.2	Chapter 11	11-62
Heat Pipes for Reactor Containment Cooling or Reactor Core Decay Heat Removal	3.1.2	Chapter 11	11-67
High Heat Conductivity Reactor Containment for Containment Cooling	3.1.2	Chapter 11	11-72
In-Containment Post-Accident Water Collection	3.1.2	Chapter 12	12-5

TITLE: REACTOR CONTAINMENT BUILDINGS

Functional Requirements: 3.1.1 Effect Containment Isolation
Control Radiation from Materials/Equipment in Storage

Safety type: Passive (See Comments)

Developmental Status: 1 Commercial

Organization: Vendor (Multiple)

Reactor Type: Light-water reactor (LWR)

Examples of Implementation: U.S. boiling-water reactors (BWRs) use the "pre-MARK I" and MARK I, II, and III containment designs, while pressurized-water reactors (PWRs) primarily use the large, dry containment designs. (PWRs also use the ice condenser and subatmospheric designs.)

Description: Containments are the structure that houses the reactor system, protects the reactor system against external environmental conditions (tornadoes, etc.), and protects the environment against postulated reactor accidents. Nuclear structures must be designed to sustain various stress loading scenarios that may be expected to occur in accidents during the lifetime of the plant. Containments are designed to withstand single or multiple static and dynamic loads, which are caused by accidental and environmental conditions. Containments must be able to retain internal pressures that could result from the rapid vaporization of the coolant (with accompanying fission products and hazardous gases) during the maximum credible accident.

The containment cannot be viewed simply as a tough, oversized pressure vessel and then designed in a conventional manner. It is a structure that may be subjected to various random, undefined loads of undefined duration. For instance, an earthquake may represent a load with statistical variations for time, intensity, or duration; therefore, the containment structural resistance and related secondary failures cannot be accurately determined because the containment materials themselves exhibit a statistical variation in properties. Such uncertainties have required a probabilistic approach and hardware verification efforts such as pressure-to-failure testing of reduced-in-scale containments in the Nuclear Regulatory Commission (NRC) Containment Integrity Program.[1]

The design of LWRs built in the U.S. vary by type of reactor (BWR or PWR) and by containment shell liner and reinforcement type (e.g., full steel, rebar reinforced, and post tensioned). The number of each type of containment constructed in U.S. plants is summarized in Table 1.

The four basic structural features of the conventionally reinforced concrete containment are the reactor sump, the base mat, the cylinder, and the dome. The containment receives its strength in tension from the heavy use of high-strength rebar embedded in the walls, dome, and base. A steel liner on the inside surface of the concrete serves primarily as a gas-tight barrier. Although the liner must transmit pressure loads to the concrete, it is not relied on as a structural member.

The prestressed concrete containment shown in Fig. 1 makes use of post-tensioning tendons embedded in the containment shell to provide design pressures as high as 60 psig. The cylindrical portion of the containment is prestressed by horizontal and vertical tendons. The dome portion uses a three-way post-tensioning system where three groups of tendons are anchored at the vertical face of the dome ring girder. The structure is lined with 1/4-in. welded steel plate for a vapor barrier. In the prestressed containment, sufficient post-tensioning is used on the cylinder and dome to more than balance the internal pressure so that a margin exists beyond that required to resist the design accident pressure load. In BWRs, prestressed concrete may only be used in the walls of the drywell, which is a smaller structure fully enclosed by the containment building walls. These designs are, of course, based on the fact that concrete is strong in compression and weak in tension.

The full steel containments shown in Fig. 2 make use of a large pressure vessel with a typically torospherical or ellipsoidal steel shell bottom, cylindrical walls, and hemispherical dome. The design makes use of well-developed steel pressure vessel design codes and a double containment leakage concept where the steel containment is surrounded by a controlled volume annulus. The freestanding containment vessel has no structural links to the concrete containment building and is separated from it by a narrow annular space, except at the bottom where the shell sits on a concrete fill slab which, in turn, rests on the structural slab. Other major containment components are the concrete cylindrical walls and the elliptical or torospherical dome.

PWR containments utilize the reinforced, prestressed, and/or all-steel designs described above. They are of three basic designs (See Fig. 3): large dry; ice condenser (See: Containment Pressure Control: Ice Condensers); or subatmospheric (See Vacuum Containment). The large dry containments are by far the most common (78% of all PWRs), utilizing their large volume to limit the pressure developed during a loss-of-coolant accident (LOCA) blowdown. In contrast, the ice condenser containment is constructed so that steam produced by the primary coolant system blowdown after an LOCA is directed through an ice condenser to cool and condense it and thus limit the maximum containment pressure. The advantage of the ice condenser containment over the large, dry containment is that by limiting the energy released to the containment, the containment may be ~50% smaller.

BWR containments feature special systems such as drywells in the primary containment and pressure suppression chambers (See Fig. 4) or "suppression pools" (See: Containment Pressure Control-Suppression Pool and Bubbler Condenser). The drywell encloses the reactor vessel and, when sealed, acts as the first containment barrier. It is tied to a suppression pool via a connection vent system. The suppression pool stores a large volume of water and, in the event of an LOCA, condenses the blowdown steam that is channeled through it. In the MARK I design, the pool is in the shape of a torus encircling the bottom of the drywell; in the MARK II, it is located directly below the drywell in the same structure; and in the MARK III, it located in the bottom of the dry containment along the outer containment building wall. The secondary containment in BWRs is the containment concrete shield building, which is not unlike those used in PWRs. Each of the MARK containments is an update of a previous version and thus the MARK I is unlikely to see use in future plant designs.

Alternative Versions: New, alternate designs worth noting include the vented annulus containment and the large, passive containment. The Germans have developed "overdimensioned" containments similar to the steel design in that there is a large reactor containment shell and an outer reinforced concrete shell (1.8-m thick). The concrete structure contains multiple compartments to house ESF and major primary and secondary components. The detached steel shell is constructed of 15 MnNi 63, is 56 m in diameter, has a 38 mm wall thickness, and is designed for 6.3 bar at 145°C. Although it has always been assumed that leakage would occur in this type of reactor containment at 8.5 bar, competent experts in the German nuclear industry today believe that the release would be at > 14 bar.[2] The annulus between the containment shell and concrete is vented through particulate air and iodine filters for deposition of radioiodine and radioactive particles in accidents. The venting, in combination with the large-volume reactor containment, shows promise in reducing the source terms in core meltdown accidents.

Status of Technology: Used in all LWRs in the United States

Advantages:
1. Containment buildings are passive (except for isolation of penetrations) and highly reliable in their design limits.
2. Provide vapor containment at high pressures (See Comments) and shielding.
3. Provide a passive high level of protection to the reactor system from external events such as earthquakes, high winds, and tornadoes.

Additional Requirements: None

Comments: Containment buildings are passive; but, to fully isolate the reactor system from the environment, various isolation systems (such as valves in process lines through the containment building) must also work. These systems usually contain active components. Some containments may also require other active systems to avoid excessive temperatures or pressures which could threaten containment integrity. The large volume and other characteristics of a containment building would significantly limit releases in a postulated accident even with the loss of isolation by providing holdup time after an accident for various mechanisms to trap radionuclides inside containment (See function 3.2: Trap Radionuclides).

References/Contacts:

1. T. E. Blejwas, W. A. von Riesemann, et al., "The NRC Containment Integrity Program," Structural Mechanics in Reactor Technology, Transactions from the International Association for Structural Mechanics in Reactor Technology, J (August 1983).

2. W. K. E. Braun, K. Hassmann, et al., "The Reactor Containment of Standard-Design German Pressurized-Water Reactors," Nucl. Technol., 72 (March 1986).

Table 1. U.S. pressurized-water reactors (PWRs) and boiling-water reactors (BWRs)[a]

	Steel	Reinforced	Post Tensioned	Total
			PWPs	
Large Dry	9	8	33	50
Ice Condenser	6	2	-	8
Sub-atmospheric	-	6	-	6
PWR TOTAL	**15**	**16**	**33**	**64**
			BWRs	
Pre-MARK I	4	-	-	4
MARK I	22	2	-	24
MARK II	1	3	2	6
MARK III	2	1	-	3
BWR TOTAL	**29**	**6**	**2**	**37**
LWR TOTAL	**44**	**22**	**35**	**101**

[a]Information from: R. F. Sammataro, "Containment Long-Term Operational Integrity - A 1988 Status Report," Fourth Workshop on Containment Integrity, U.S. Nuclear Regulatory Commission, NUREG/CP-0095, SAND88-1836 (June 1988).

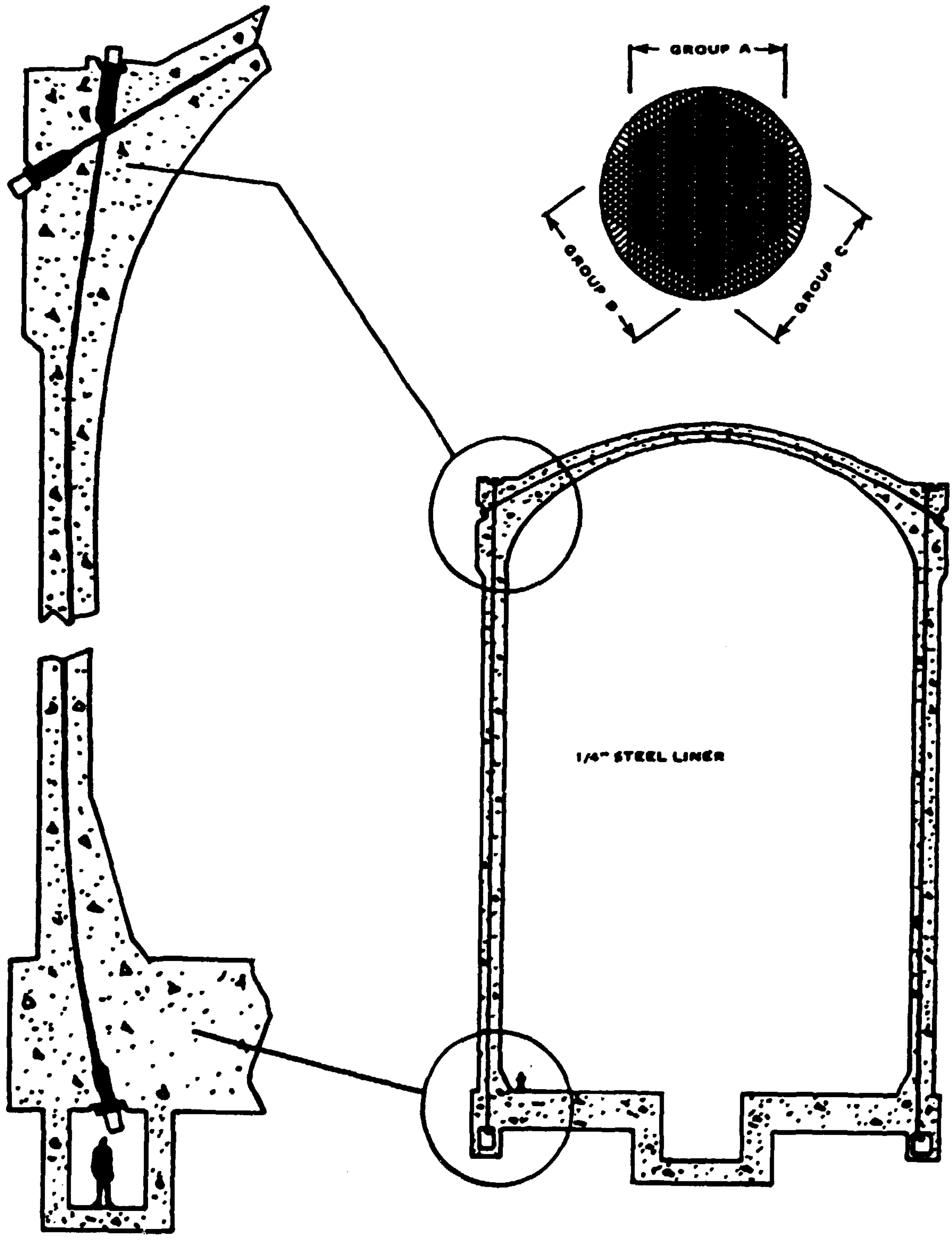


Fig. 1. Prestressed concrete containment with shallow dome.

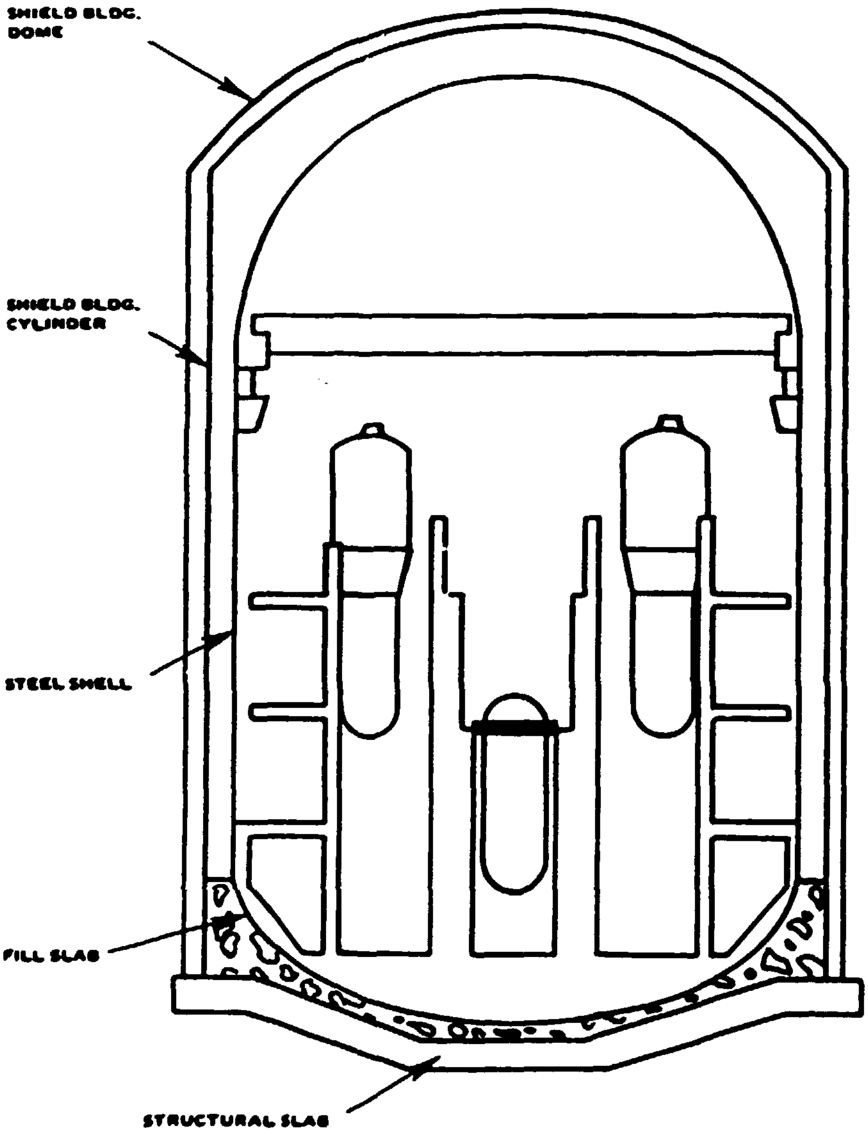
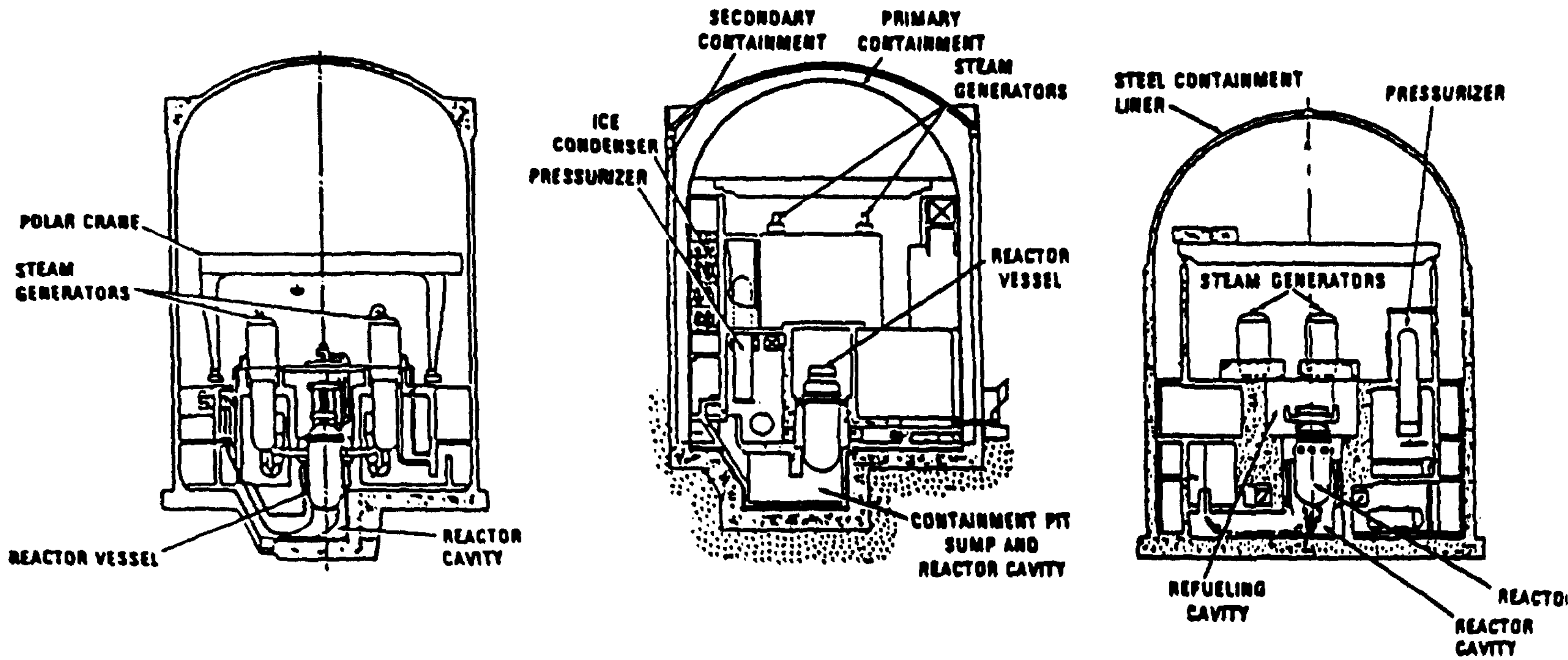


Fig. 2. Typical full steel containment structure.



PWR Large Dry Containment

PWR Ice Condenser Containment

PWR Sub-Atmospheric Containment

Fig. 3. PWR containment configurations.

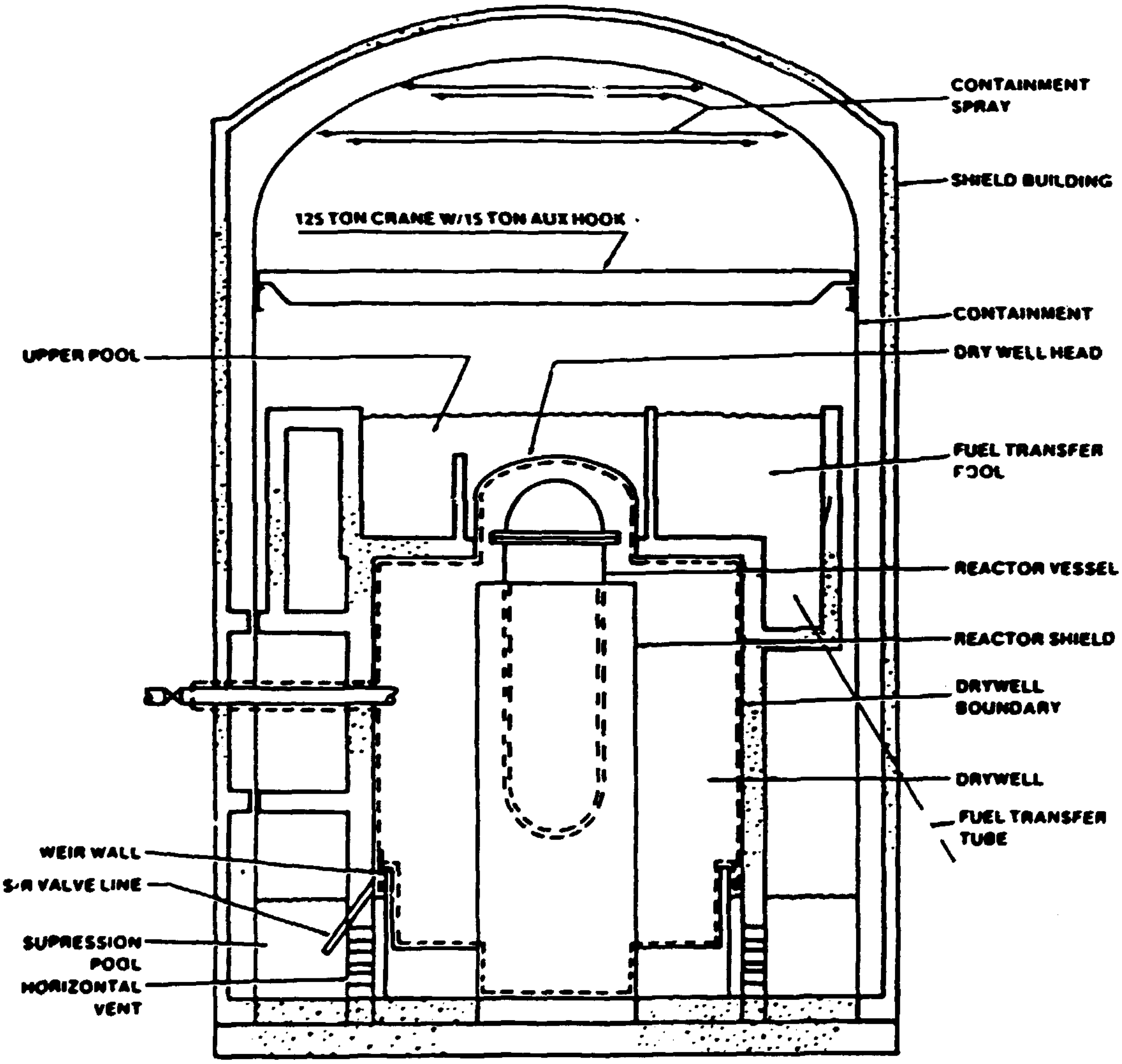


Fig. 4. Primary containment system (Mark III).

TITLE: SELF-SEALING ICE CONTAINMENT STRUCTURE

Functional Requirements: 3.1.1 Effect Containment Integrity

Safety Type: Passive

Developmental Status: 4 Analyzed

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (Asea Brown Boveri)

Examples of Implementation: None

Description: Underground installation of a nuclear reactor in an area where groundwater is the source of drinking water poses the risk of radioactive contamination of the water supply in the event of a major reactor accident. Increasing the thickness of the concrete confinement does not eliminate this risk, as cracks could still occur in the concrete as a result of an explosion or earthquake.

The probability of leakage of radioactive water from an underground reactor to the surrounding groundwater can be substantially reduced by cooling the soil or rock adjacent to the concrete confinement to a temperature below the freezing point. This could be accomplished by installing cooling coils along the walls and ground surface of the soil and rock mass that surrounds the containment. Each cooling coil would contain several hydraulic, series-coupled cooling elements, with each element inserted into a hole drilled into the soil and rock mass (Fig. 1). The cooling coils would be parallel-coupled with each other and connected to an absorption-type cooler. This cooler would obtain at least some of the heat required by the cooling apparatus from flow-through water heated by the reactor. The maximum temperature of the cooling medium circulated through the coils would be 0° C and the minimum thickness of the frozen soil and rock layer would be 1 m; the preferred cooling medium temperature would be -40° C, and the preferred thickness of the frozen soil and rock layer would be 10 m.

If radioactive water leaked from the containment it would not immediately reach the surrounding groundwater, but would be frozen in the frozen soil and rock layer. The retention time would depend on the size of the cooling system. The frozen layer would be self-sealing; any cracks that occurred would reseal when the reactor water penetrating them froze. If desired, the cooling capacity of the soil and rock mass could be increased by the addition of a number of cold accumulators, e.g., plastic bags containing a salt solution with a freezing point of -20° C.

Alternative Versions: None

Status of Technology: Not currently in use.

Advantages: 1. Radioactive water leaking from the containment would be retained in the frozen soil and rock mass surrounding the containment.

2. Even if an accident prevented the cooling machinery from functioning, the soil and rock mass would still remain frozen for an extended period, allowing time to carry out decontamination procedures.

3. The large frozen mass provides a large heat sink for the reactor decay heat.

Additional Requirements: None

Comments: None

References/Contacts:

1. K. Hannerz, Inneslutningsanordning för en Kärnreaktor (Containment Arrangement for a Nuclear Reactor), Swedish Utlaggningsskrift No. (B) (21) 7508757-7, (August 1975): ORNL/TR-89/15 (1989).

Update/Compiler: March 1989/EBL

ORNL DWG 89A-10437

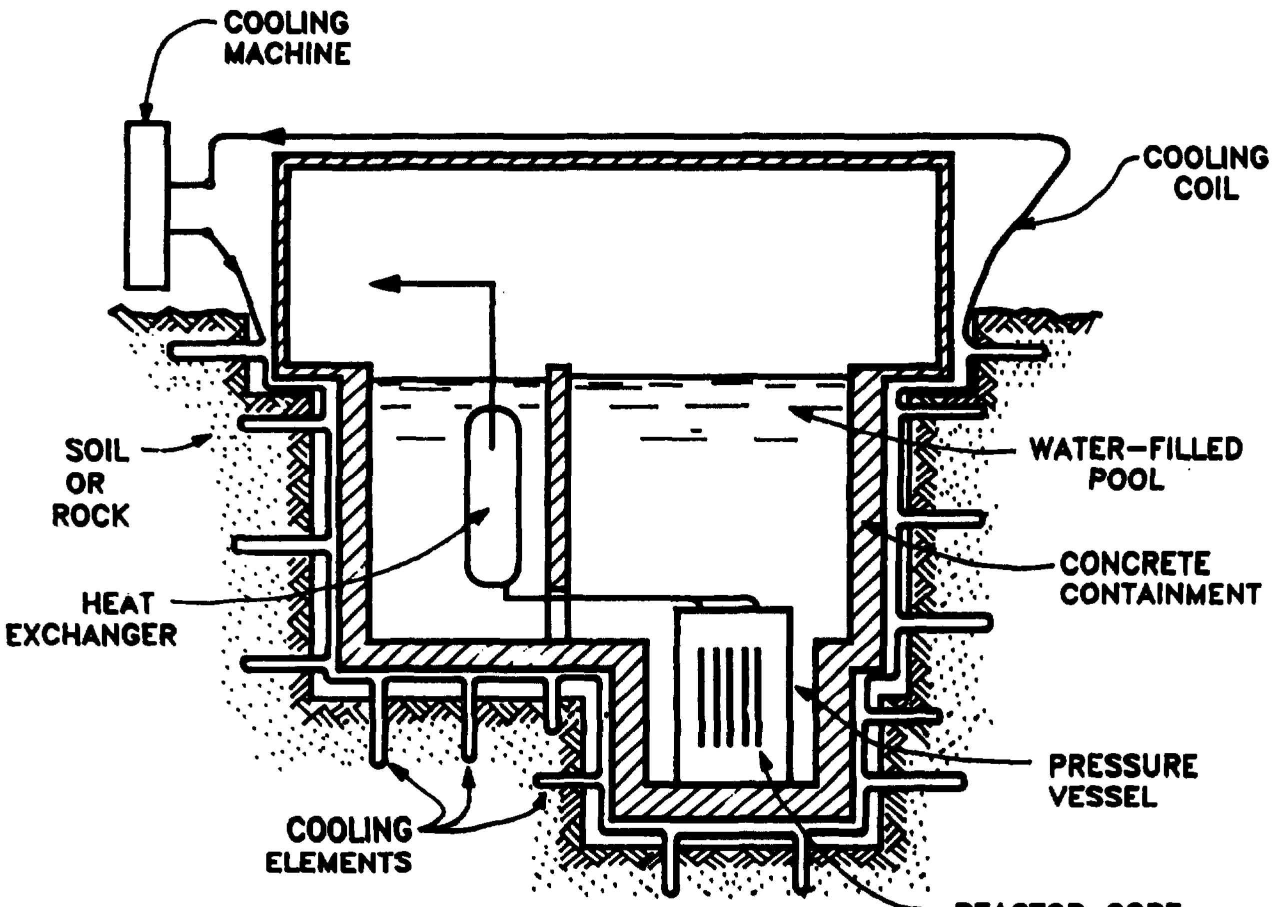


Fig. 1. System to retain radioactive water leaking from the reactor containment.

TITLE: FILTERED, VENTED CONTAINMENT

Functional Requirement: Level 3.1 Maintain Containment Boundary Integrity

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-Water Reactor (LWR)

Organization: Vendor (Asea Brown Boveri and KWU)

Examples of Implementation: The FILTRA gravel-bed system was put into operation at the Swedish Barseback Nuclear Power Station in 1985.

Description: A filtered, vented containment is a standby system connected to a nuclear power plant containment structure. Its function is to provide pressure relief for the reactor containment in order to avoid containment rupture following a postulated reactor accident, while limiting radioactive release to the atmosphere. Many types of filtered, vented, containment systems (FVCS) have been proposed: dry filters (fibrous mats, gravel beds, and sand beds), wet scrubbers (water pools, submerged gravel beds, washed fibrous mats, and submerged venturi), and combinations of different types of these filters. An extensive review of FVCSs can be found in Ref. 1. Table 1 lists a variety of FVCSs now being used or planned on LWRs throughout the world.

The two-stage FILTRA-MVSS is an example of one of the FVCSs listed in Table 1.[2,3] It is being installed on all of the Swedish LWR nuclear power stations, except the boiling-water reactors (BWRs) at Barseback. FILTRA-MVSS includes a pressure relief system, a system for collecting iodine and particulate matter, and a moisture separation system (see Fig. 1). The FVCS is housed in a separate concrete building whose interior is in the form of a pressure vessel provided with a stainless steel liner. A radiation-shielded concrete culvert connects the venting system to the reactor containment. If a serious reactor accident should cause an excessive pressure rise in the reactor containment, the steam and gas are ducted via the pressure relief system through a central pipe into the filter chamber, which is partially filled with water. The central pipe leads to a number of horizontal distribution pipes that feed to multiple venturi nozzles. A series of water seals automatically match the number of venturi nozzles in operation to the required off-gas flow rate. In the venturi nozzles, a mixture of steam, gas, and water is accelerated and discharged into the water pool. The gas then bubbles up through the pool in the filter chamber, trapping the particulate matter and dissolving elemental iodine. After the gas leaves the pool, it goes through a dry gravel bed where entrained water droplets are separated from the gas. It then exits the plant into the atmosphere via the discharge stack.

Filter vent systems are typically designed to remove 99 to 99.9% of the fission products, excluding noble gases.

Alternative Versions: Passive activation of some FVCSs could be achieved by including appropriate devices in the pressure relief system, such as rupture disks (used in some FVCS's parallel to conventional pressure relief values) or hydraulic water locks (see Fig. 2).

Status of Technology: Many European countries have opted for FVCSs to avoid the possibility of land contamination, to reduce the planned evacuation radius, and to provide an additional option for severe accident management. Although the first European FVCS installation was a large, expensive, gravel-bed filter shared by two Swedish BWRs, the trend is toward the installation of small, lower-cost FVCS such as the sand beds or stainless steel fiber filters located at French and German PWRs, and multistage filters, consisting of a submerged venturi scrubber followed by a demister/filter at German BWRs and Swedish LWRs. The U.S. Nuclear Regulatory Commission does not require FVCSs on commercial plants that it licenses. Several U.S. utilities have proposed the installation of FVCSs at BWRs. A listing of the various types of FVCSs installed or being installed at LWRs throughout the world can be found in Table 1.

Advantages:

1. Prevention of containment rupture because of overpressurization following a postulated reactor accident.
2. Control of radioactive release to the atmosphere following a postulated reactor accident.

Additional Requirements: Pressure integrity of containment structure and successful isolation of containment after an accident.

Comments: The various FVCSs require different levels of operator attention after an accident. Morewitz's analysis[1] concluded that "so far, the Barseback FILTRA system is the only FVCS designed for totally passive operation.[4] All of the other systems appear to require" various degrees of support. The FILTRA system is a 20-m diam, 40-m high (10,000-m^3 volume) crushed gravel bed filled with 25-mm crushed quartzite (a type of granite). The bed is designed to remove 99.9% of the radioactivity, except from noble gases and organic iodine.

FVCSs have generally been add-on systems to existing reactors. The preferred system or systems for a totally new reactor design may be significantly different than current designs when integrated into the reactor containment system.

References/Contacts:

1. Harry A. Morewitz, "Filtered Vented Containment Systems for Light-Water Reactors," Nucl. Technol., 83, 117-33 (November 1988).

2. Kjell Elisson and Tore Walterstan, "Sweden Employs a Multi Venturi Scrubber for Containment Venting," Nucl. Eng. Int., 33, (405), (March 1988).

3. "FILTRA MVSS," Progress Report No. 5, April 1987, ABB-Atom AB, S-721 63, Västerås, Sweden.

4. Ake H. Persson, "The Filtered Venting System Under Construction at Barsebäck," Nuclear Technology, 70, 158-60 (August 1985).

Update/Compiler: Jan. 1989/RLP

Table I. Various LWR Filtered Vented Containment Systems

Filter name	Type (Sequence of filters)	Reactor	Country	Footnote
Single Stage System				
FILTRA	Filtered vents	2 BWRs (Teollisuuden Voima Oy Utility)	Finland	a
	Gravel beds	Barseback BWRs	Sweden	b
COSA	Thermal-type hydrogen recombiner	Leibstadt Mark-III BWR	Switzerland	c
UNKNOWN	Sand beds	All 900 MW(e), 1300 MW(e), and 1400 MW(e) PWRs operated by Electricite de France	France	d
UNKNOWN	Water pools	Muhleberg Mark-I BWR	Swtizerland	c
Multistage System				
EFAD	(1) a moisture separator and air heater; (2) prefilter; (3) dry HEPA filter; (4) activated charcoal filter; (5) HEPA filter; (6) motorized fan	All CANDU heavy-water-moderated power reactors operated by Ontario Hydro	Canada	c
FILTRA-MVSS	(1) Multi-venturi scrubber; (2) gravel bed	All LWRs except at Barseback	Sweden	d
UNKNOWN	(1) Multi-venturi scrubber; (2) filter module	Brunsbuttel and Krumnel KWU Series '69 BWRs	Germany	c
UNKNOWN	2 and 3 stage stainless steel filber filter units	Brokdorf, Isar II, Emsland and Neckarwestheim 1300 MW(e) PWRs 7 more 1300 MW(e) PWRs	Germany	c

[a]To be installed in 1989.
[b]Installed in 1985.
[c]Installed.
[d]Installation in progress.
[e]Design modifications underway.
[f]To be installed.

ORNL DWG 88-14941

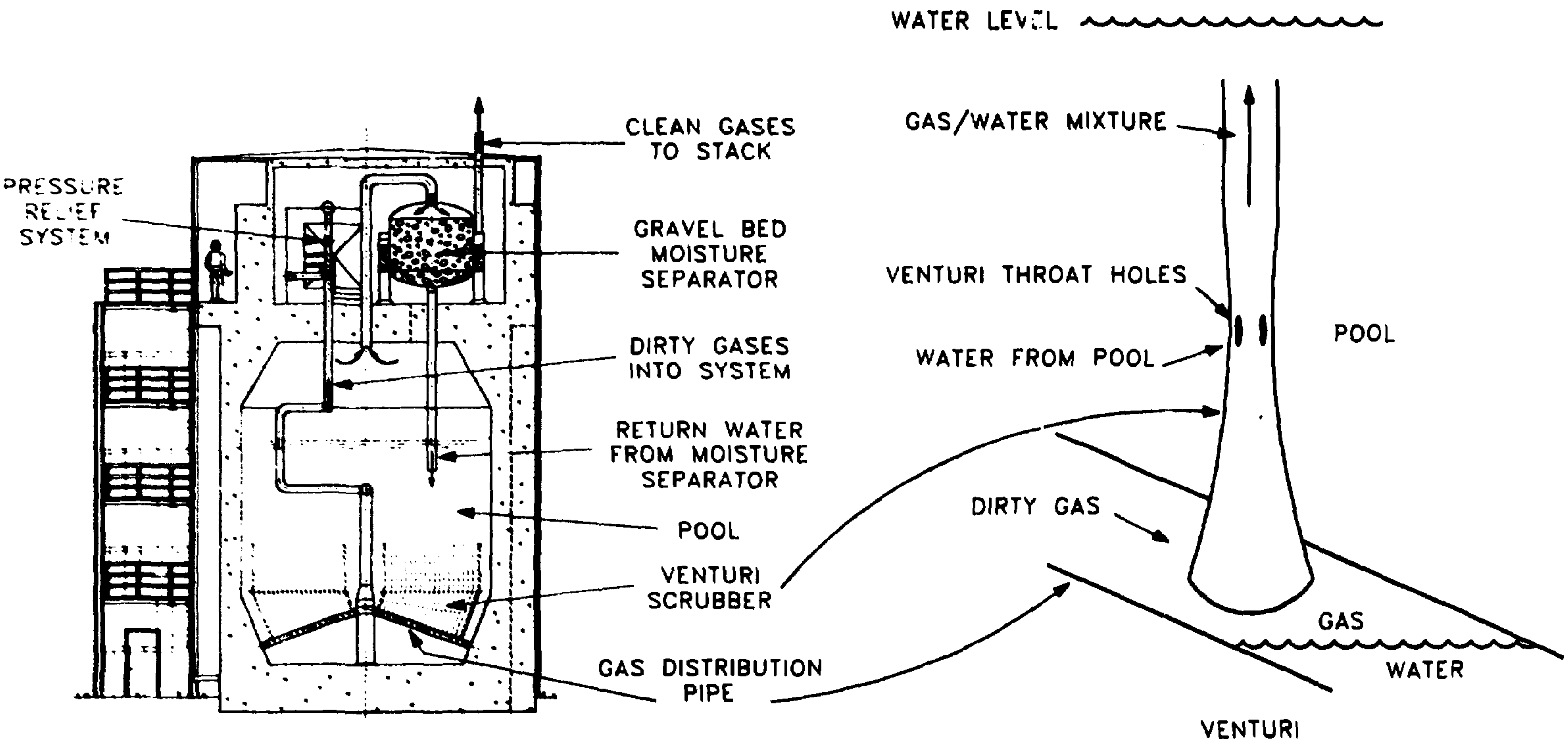


Fig. 1. Schematic of FILTRA-MVSS system.

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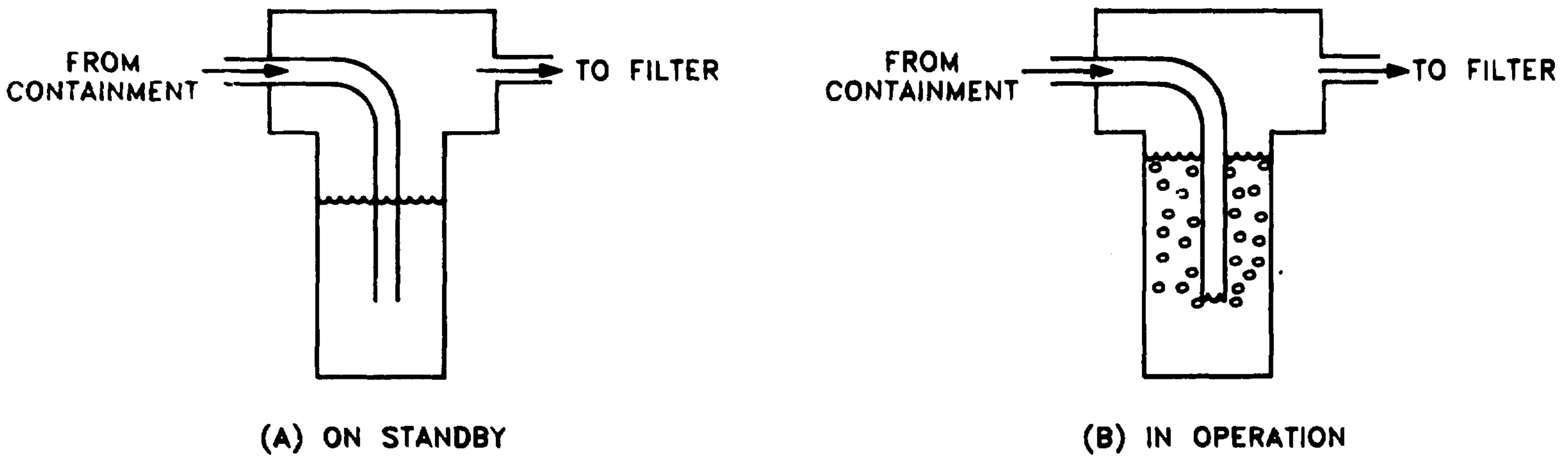


Fig. 2. Sketch of a hydraulic water lock.

TITLE: VACUUM CONTAINMENT

Functional Requirement: Level 3.1.2: Control Energy in Containment
Level 3.2 Trap Radionuclides

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-Water Reactor (LWR)

Organization: Canada

Examples of Implementation: Vacuum containment systems are in use at the Pickering Nuclear Generating Station (NGS), the Bruce-A NGS, Bruce-B NGS, and the Darlington NGS in Ontario, Canada.

Description: A vacuum containment can prevent and contain the release of energy and radioactivity after a postulated reactor accident. A vacuum containment's function is to terminate overpressure transients and thereafter maintain the containment at a subatmospheric pressure for several hours or days after an accident, thus preventing or significantly delaying the possibility of radioactive releases. The currently used vacuum containments have the main containment near atmospheric pressure and a separate building under hard vacuum. In an accident, pressure relief valves between the two connecting buildings open and cause the overpressure in the main containment building to be lowered.

Canada's Ontario Hydro has incorporated a vacuum containment system for all of its CANDU reactors. A common feature of the Pickering NGS, Bruce-A NGS, Bruce-B NGS, and Darlington NGS power stations is a single vacuum building that is connected by self-activating pressure relief valves (PRVs) to a common, large volume containment shared by all the reactor buildings (see Fig. 1). A description of this system is provided in Refs. 1-3. In the event of a loss of coolant accident (LOCA), the resultant pressure rise in the containment beyond a preset value opens the self-activating PRVs, allowing the air-steam mixture to enter the vacuum building. A schematic of the PRV in the open and closed position can be seen in Fig. 2. One side of the piston within the PRV is kept at atmospheric pressure. The other side of the piston is exposed to the pressure within the reactor plant containment. Normally, the pressure within the containment is kept at a negative pressure, about 5 kPa(g) less than atmospheric pressure, and the PRVs remain closed. But if the containment pressure should suddenly exceed atmospheric pressure due to an accident, the PRVs will quickly open, allowing the air-steam mixture to enter the vacuum building that is normally maintained at 7 kPa. A water spray is initiated and maintained by the resulting pressure difference between the main vacuum building chamber and the upper vacuum chamber normally pumped down to 1.5 kPa (see SSC description: Containment Pressure Control by Water Spray). The dousing system's primary function is to condense and cool the steam-air mixture that enters the vacuum building due to the accident; this terminates the short overpressure transient and thereafter maintains the reactor containment at a subatmospheric pressure (as illustrated in Fig. 3).

Alternative Versions: Many options exist.

Status of Technology: Multi-unit CANDU nuclear generating stations utilizing a common vacuum building were installed at the Pickering NGS, the Bruce-A NGS, the Bruce-B NGS, and the Darlington NGS in Ontario, Canada in the early part of the 1980s.

Advantages:
1. Vacuum containment systems can be designed to reduce containment overpressure transient within minutes after a reactor accident and to maintain a subatmospheric pressure for several days after.
2. Radioactivity can be contained.
3. The system is passive in operation.

Additional Requirements: None

Comments: None.

References/Contacts:

1. J. Huterer et al., "Pressure relief structures of multi-unit CANDU nuclear power plants," Nuc. Eng. and Des., 100, (1), 21-39 (February 1987).

2. Walter W. Koziak, "Bruce-B NGS - Containment Commissioning Test," Nuclear Studies and Safety Department, Ontario Hydro, Toronto, Ontario, Canada, October, 1983.

3. Z. M. Beg and R. S. Ghosh, "Evolution of CANDU vacuum building and pressure relief structures from Pickering NGS A to Darlington NGS A," Transactions of the 9th International Conference on Structural Mechanics in Reactor Technology, H, Concrete and Concrete Structures, CONF-870812, 467-73 (1987).

Update/Compiler: Jan. 1989/RLP

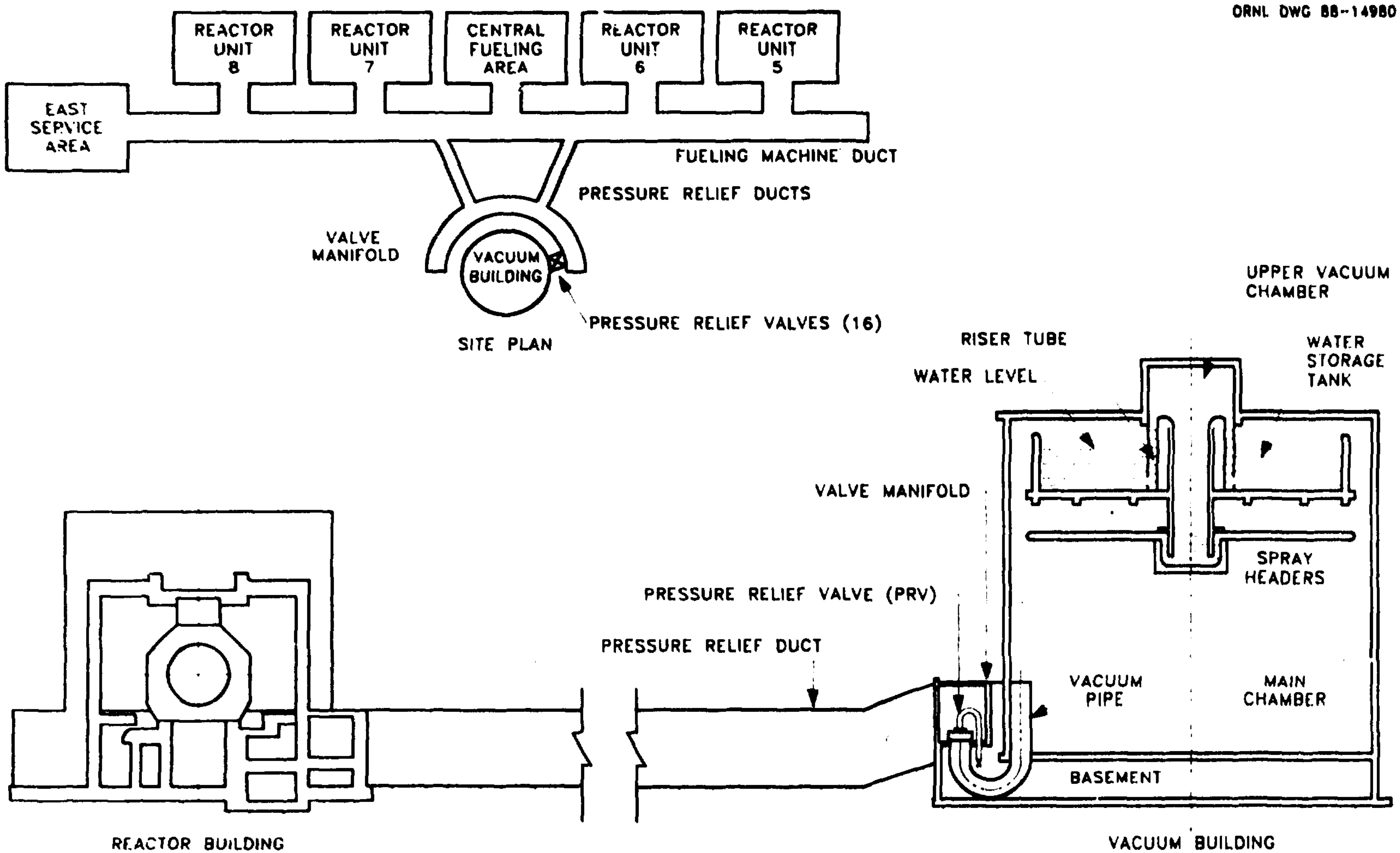


Fig. 1. Schematic of reactor building and emergency pressure relief system.

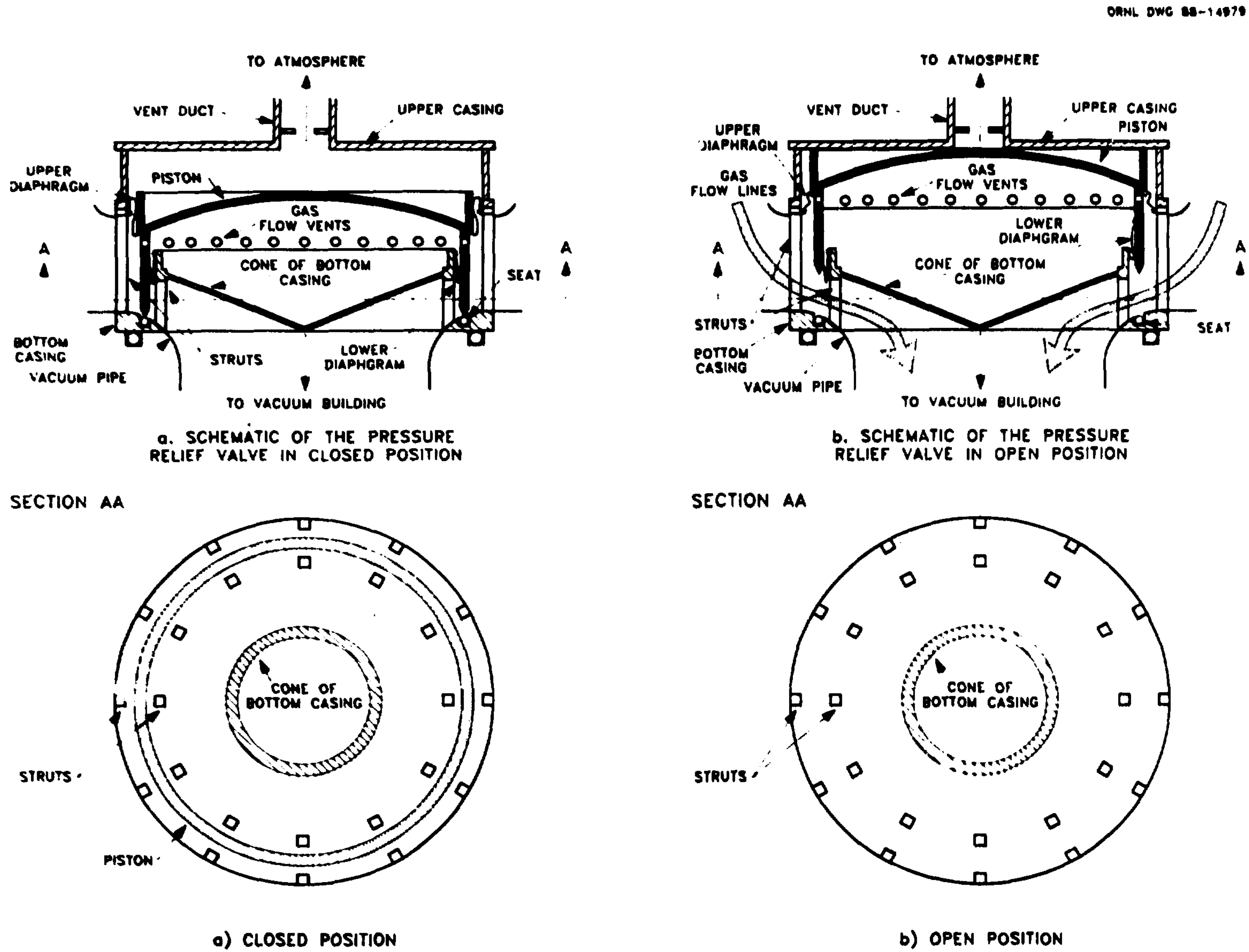


Fig. 2. Schematic of the pressure relief valve.

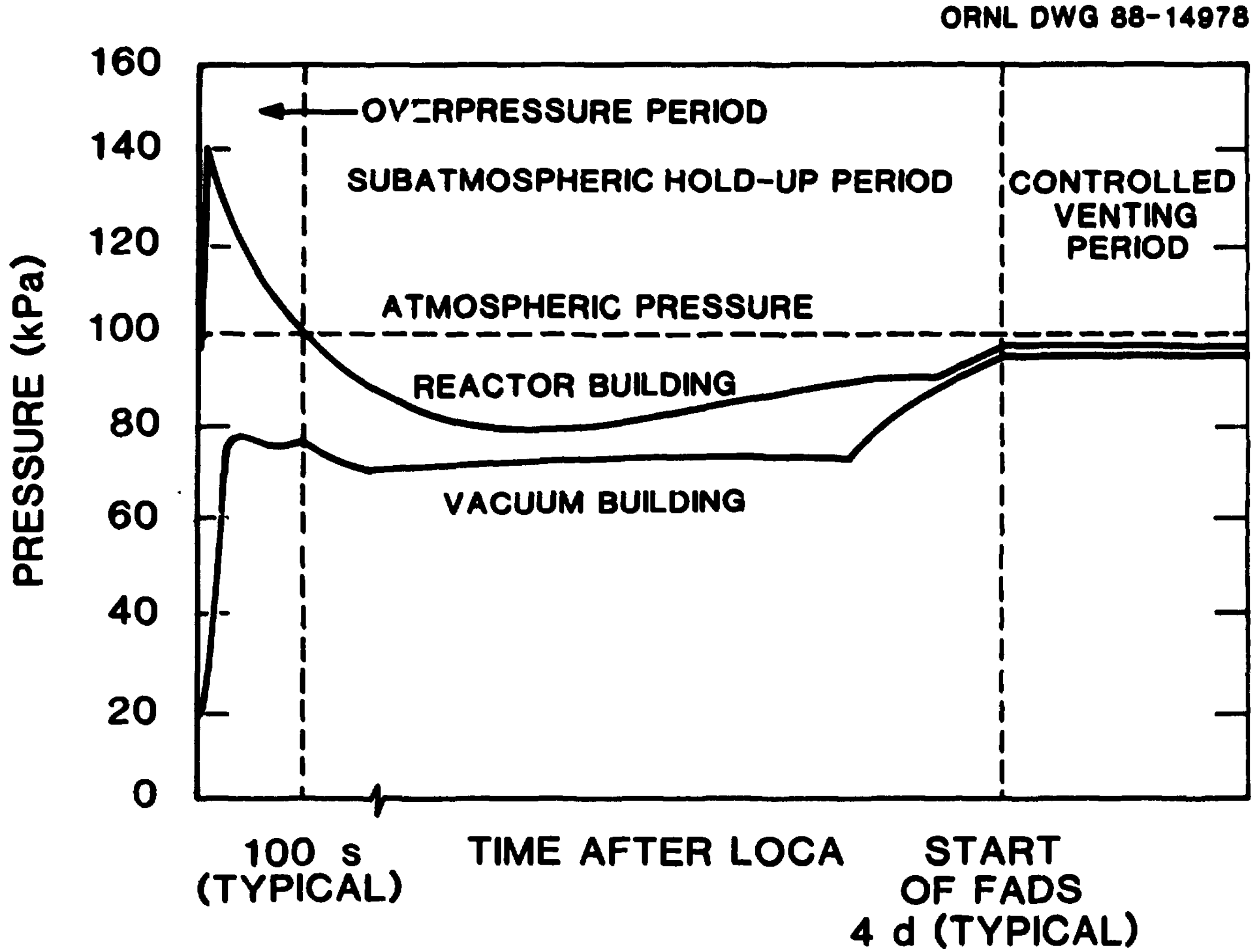


Fig. 3. Containment pressure following LOCA at Bruce NGS B.

TITLE: PASSIVE HARD-VACUUM CONTAINMENT SYSTEM

Functional Requirement: Level 3.1.2: Control Energy in Containment
Level 3.2: Trap Radionuclides

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Light-Water Reactor (LWR)

Organization: Industrial (Nucle Dyne Engineering Corporation)

Examples of Implementation: None

Description: A vacuum containment can prevent and contain the release of energy and radioactivity after a postulated reactor accident. A vacuum containment's function is to terminate overpressure transients and thereafter maintain the containment at a subatmospheric pressure for several hours or days after an accident, thus preventing or significantly delaying the possibility of radioactive releases.

The Passive Containment System (PCS-3) is a hard-vacuum containment system (1-3) where the containment structure is divided into three zones: primary, auxiliary, and secondary. The primary containment encloses the reactor, steam generators, pressurizer, circulation pump heads, and engineered safety system components (see Fig. 1). It consists of multiple interconnected cylindrical steel cells surrounded by reinforced concrete biological shield walls. During normal operations, it operates at vacuum conditions. Reinforced-concrete encloses the auxiliary containment structure, which operates near atmospheric pressure. The reinforced-concrete reactor building provides a secondary containment.

Major containment safety systems inside primary containment include depressurizer vessels and refill, deluge, and quench tanks. Each of these contains temperature-controlled, stored water with boron and other dissolved chemicals. Their heat sink capacities are given in Table 1. The primary containment structure, containing 250,000 ft^3 of free volume, is designed for normal operation at a 2.22-psia vacuum and for a 75-psia peak accident pressure pulse.

The auxiliary containment enclosures house equipment required for (a) normal operation, including (but not limited to) reactor coolant auxiliaries, component cooling and service water systems, instrumentation and controls, and electrical equipment; and (b) refueling, including handling and storage of new and spent fuel (outside primary containment), spent fuel cooling, and (c) handling and temporary storage of solid, liquid and gaseous radwaste. Select enclosures are steel-lined to prevent leakage of radioactivity in an adverse event. These include both the refueling enclosure designed to handle new and spent fuel and the enclosure for gaseous radwaste.

The reactor building houses environmental control systems for the primary and auxiliary containment structures therein. The building is constructed with roof closures above the reactor cooling system and other major components. These closures provide for the installation and ready replacement of components without upending them inside containment.

In the event of a pipe break in containment, steam travels through variable orifice vents between the interconnected cells in the primary containment to the deluge and quench tanks. The tanks act like suppression pools (see Containment Pressure Control-Suppression Pool/Bubbler Condensers). Steam enters through vent pipes and is discharged below the water level and is condensed. At about 8.66-psia pressure, the total orifice area exposed is 204 ft^2, limiting the primary containment peak pressure to <75 psia in the large RCS pipe break. Continued steam carryover into the deluge and quench tanks passively reduces the containment pressure below atmospheric within 1 min after the pipe break. In the small-diameter break, the primary containment pressure remains below atmospheric during RCS blowdown and for the term of the accident.

In the loss-of-coolant accident (LOCA), borated water from the depressurizer vessels and refill and deluge tanks passively floods the primary containment, including the reactor vessel cavity, to an elevation above postulated RCS pipe breaks. The temperature of the water flooding the containment, as well as that of the stored water retained within the quench tanks, gradually increases as a result of decay heat absorption.

A postaccident decay heat transfer system becomes passively functional during gravity flooding of the primary containment, submerging heat exchangers within the primary containment, which in turn are interconnected to heat exchangers in cooling pond(s). At about 6 h into the LOCA, the heat transfer rate of this system equals the prevailing decay heat generation rate and maintains the containment at a subatmospheric pressure.

The vacuum in the primary containment keeps the oxygen in containment below combustible limits, preventing hydrogen/oxygen reactions and fires. Containment flooding provides radionuclide scrubbing of gases and the absorption of fission products.

Alternative Versions: Multiple design options exist.

Status of Technology: Detailed analysis of the system has been made.

Advantages:

1. Vacuum containment systems can be designed to reduce the containment overpressure transient within very short times following a reactor accident and to maintain a subatmospheric pressure for several days thereafter. Radioactivity is fully contained.
2. The vacuum eliminates the possibility of fire in containment due to the lack of oxygen.
3. Containment integrity is continually checked during normal operation. The existence of a vacuum is proof of containment integrity.
4. The system can be designed for high heat loss in the event of an accident. High-vacuum containments eliminate the need for primary system insulation. In a postaccident environment, as the containment is flooded, the reactor system can be partly cooled by heat transfer through the primary system walls (which have no insulation). Similarly, passive heat removal systems can be activated by the presence of steam, water, or higher gas pressures.
5. The system is passive in operation.

Additional Requirements: None

Comments: None

References/Contacts:

1. O. B. Falls, Jr., and F. W. Kleimola, "A Passive Containment System for Advanced Light-Water Reactors," Nucl. Safety 29(3) (July-September 1988).

2. O. B. Falls, Jr., and F. W. Kleimola, "Passive Containment System: An Advanced Light-Water Reactor Plant Having a Zero Source Term," NEC-13, Nucle Dyne Engineering Corporation, 728 West Michigan Avenue, Jackson, Michigan, June 1987.

3. F. W. Kleimola and O. B. Falls, Jr., "Recommissioning – An Alternative to Decommissioning," Trans. Am. Nucl. Soc. 30 555-56 (1978).

Update/Compiler: Sept. 1989/CWF, FWK

Table 1. Heat sink capacity of stored water within primary containment

Component	Total volume		Initial temperature		Heat sink capacity[a]			
					200°F, 11.5 psia [b] 93.3°C, 7.9×10^4 Pa		212°F, 14.7 psia 100°C, 1.0×10^4 Pa	
	ft^3	m^3	°F	°C	10^6 Btu	10^9 J	10^6 Btu	10^9 J
Containment [c]	250,000	7,080	50	10	8	8	11	12
Depressurizer vessels [d]	3,600	102	120	49	18	18	18	19
Refill tanks [d]	23,200	657	50	10	209	220	221	233
Deluge tanks [d]	57,177	1,619	50	10	535	564	578	610
Quench tanks [e]	49,100	1,393	50	10	460	490	491	518
Totals					1229	1300	1319	1392

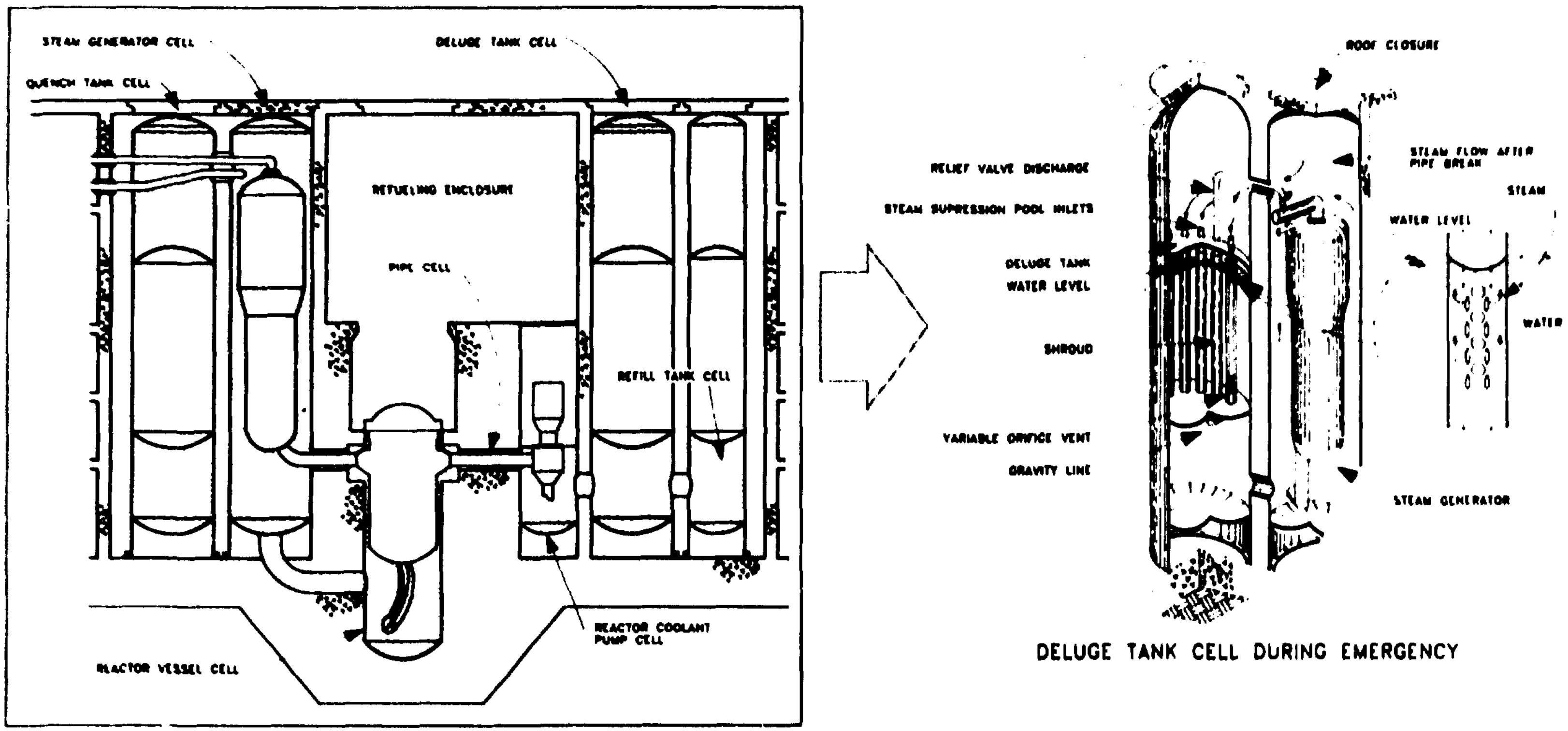
[a] Added energy when heated to the given temperature and pressure in adverse event; does not include heat sink capacity of structures.

[b] Containment at subatmospheric pressure.

[c] Water vapor alone.

[d] Contains boron and other chemicals in solution.

[e] Contains dissolved chemicals.



VOLUME UNDER VACUUM DURING NORMAL OPERATION

Fig. 1. Schematic of inner connected vacuum cells of a pressurized water reactor.

TITLE: CONTAINMENT PRESSURE CONTROL–SUPPRESSION POOLS AND BUBBLER CONDENSERS

Functional Requirements: Level 3.1.2 Control Energy in Containment
Level 3.2 Trap Radionuclides

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (Multiple)

Examples of Implementation: All BWRs designed and licensed by General Electric (GE) include a pressure suppression pool. Examples include Commonwealth Edison Company's two-unit Dresden Nuclear Power Station (Mark I type containment), Philadelphia Electric Company's two-unit Limerick Generating Station (Mark II type containment), and System Energy Resources, Inc.'s two-unit Grand Gulf Nuclear Station (Mark III type containment) with outputs of 772 MW(e), 1055 MW(e), and 1142 MW(e) respectively.

There are 13 Russian 440 MW(e) PWR nuclear units incorporating bubbler condenser pressure suppression systems now in operation and 13 more are under construction.

Description: Suppression pools are used to rapidly cool and condense the steam-air mixture after a loss of coolant accident, thus acting as a passive heat sink to absorb the released energy from the break. Suppression pools accomplish this function by bubbling the higher-pressure steam-air mixture through a bath of cool water. This condenses the steam and cools the air. This heat sink is a large volume of water stored in either pressure suppression chambers or a bubbler condenser compartment inside the primary containment, or in a bubbler condenser tower next to the reactor building. A secondary function is to prevent gaseous (especially halogens) and particulate radionuclides from entering the containment atmosphere and, potentially, the environment. This is done by scrubbing escaping gases from a reactor coolant break with pool water.

All BWR primary containments consist of a drywell that encloses the reactor vessel and recirculation system, a pressure suppression chamber that stores the large volume of water, and the main containment volume. A schematic of a typical BWR Mark III type containment system can be seen in Fig. 1. The pressure suppression pool is contained within the volume between the drywell wall and the lower containment wall. The suppression pool contains approximately one million gallons of water at its maximum pool level. The drywell and suppression pool are connected by 135 circular vents through the lower drywell wall below the pool's water level, which, under accident conditions, conduct the steam-air mixture into the suppression pool. When steam and/or other gases escape the reactor during an accident, the pressure in the drywell increases, the water level between the weir wall and the drywell wall decreases, and each of the three rows of vents are progressively uncovered. As each row is cleared, the steam-air mixture enters the suppression pool below the surface where the steam is cooled and condensed, thus reducing the overall pressure within the primary containment. The inert gases are vented from the suppression pool to the containment volume. The design characteristics for a typical Mark III type primary containment are shown in Table 1.

The Union of Soviet Socialist Republics (USSR) has in operation thirteen 440 MW(e) PWR nuclear units designated as the VVER-440. (In addition, 13 more units of larger sizes are now under construction.) A number of older units and all newer units include a pressure suppression system. The VVER-440 model typically incorporates a large number (~1960) of water trays that serve as suppression pools. For each unit, the set of suppression pool trays is located inside a bubbler condenser tower either in the containment (Fig. 2) or constructed adjacent to the reactor building in a connected building. A variant of this standardized design is now under construction at Juragua, Cuba. Its design makes use of internal suppression pools and is similar in layout to Westinghouse-designed, ice-condenser-equipped PWR. Instead of using an ice condenser, the intermediate compartment contains passive bubbler condensers (see Fig. 3 and SSC description: Containment Pressure Control: Ice Condensers). The multilevel bubbler condenser trays contain approximately 200,000 gal of water. When the pressure rise in the lower compartment, which contains the reactor, exceeds the head of water initially filling the bubbler tubes, the steam-air mixture is forced downward through the bubbler tubes (see Fig. 2). Steam is condensed as the steam-air mixture that is exiting the bottom of the tubes bubbles up through the surrounding water layer. The air and uncondensed gases passing through the water exit the intermediate compartment through several orifices into the upper compartment. Condensation of the steam-air mixture is complete when the pressure in the lower compartment is no longer sufficient to force the steam through the bubbler condenser unit.

A modular pressure suppression containment system is proposed for the Combustion Engineering (3) Safe Integral Reactor (SIR). SIR is a pressurized-water reactor (See: Integral PWR with no Primary System Piping). The pressure suppression containment system consists of: (1) a reactor vessel compartment which houses the reactor pressure vessel and support structure, (2) eight cylindrical steel suppression tanks with external fins, each containing a pool of water, and (3) a vent system connecting the reactor vessel compartment with pressure suppression tanks (Fig. 4).

Each shop-fabricated cylindrical steel pressure suppression tank operates as a conventional suppression pool with the incoming steam-air mixture directed below the water level in the tank to condense the steam. Each tank has a finned exterior surface to promote heat transfer to the ambient air for long-term cooling. The pressure suppression tanks are housed within a reinforced concrete structure which has outside air intakes and discharge provisions for circulating ambient air. The design pressure of the containment is 241 KPa (34 psig) while the design temperature of the reactor vessel containment is 171°C (340°F) and that of the suppression tanks is 116°C (240°F).

Alternative Versions: There are many design variations of suppression pools.

Status of Technology: Suppression pools are installed in all three versions of the GE-designed BWR containments. Bubbler condenser towers sit adjacent to the USSR-designed VVER-440 model 213 PWRs. Two of these same VVER-440s with bubbler condensers within the primary containment are now being constructed in Cuba.

Advantages: 1. Containment overpressure due to an LOCA is reduced within a few minutes to a few psi, thus reducing the probability of fission product release from the containment.

2. The suppression pools/bubbler condensers are static devices, do not require any power source, and their operation during an accident does not depend upon the functioning of other systems.

3. Radionuclides are scrubbed from escaping gases and steam from the reactors.

Additional Requirements: None

Comments: The Advanced Boiling-Water Reactor currently under development has a suppression pool similar to that of the MARK III containment, except that it is not open to the containment.

References/Contacts:

1. Updated Final Safety Analysis Report, Grand Gulf Nuclear Station, Docket 50-416, Chapters 1-6.

2. U.S. Department of Energy, Overall Plant Design Descriptions, VVER Water-Cooled, Water-Moderated Energy Reactor, DOE/NE-0084, Rev. 1 (October, 1987)

3. R. A. Matzie, J. Longo, R. B. Bradbury, K. R. Teare, and M. R. Hayns, Design of the Safe Integral Reactor, Combustion Engineering, TIS-8471 (1989).

Update/Compiler: May 1989/WEK

Table 1: Mark III Containment

Design Parameters			
	Drywell		Containment
Internal pressure (psig)	30		15
External design pressure differential (psdi)	21		3
Design temperature (°F)	330		185
Net free volume (ft^3)	270,000		1,400,000
Suppression pool water volume (ft^3)-total, max		138,700	
Pool depth-normal (ft)		18'7"	
Vent System:			
Number of vents		135	
Nominal vent diameter		28"	
Net vent area (ft^2)		552	
Vent centerline elevations (ft above pool bottom)			
Top row		11'4"	
Middle row		7'2"	
Bottom row		3'0"	

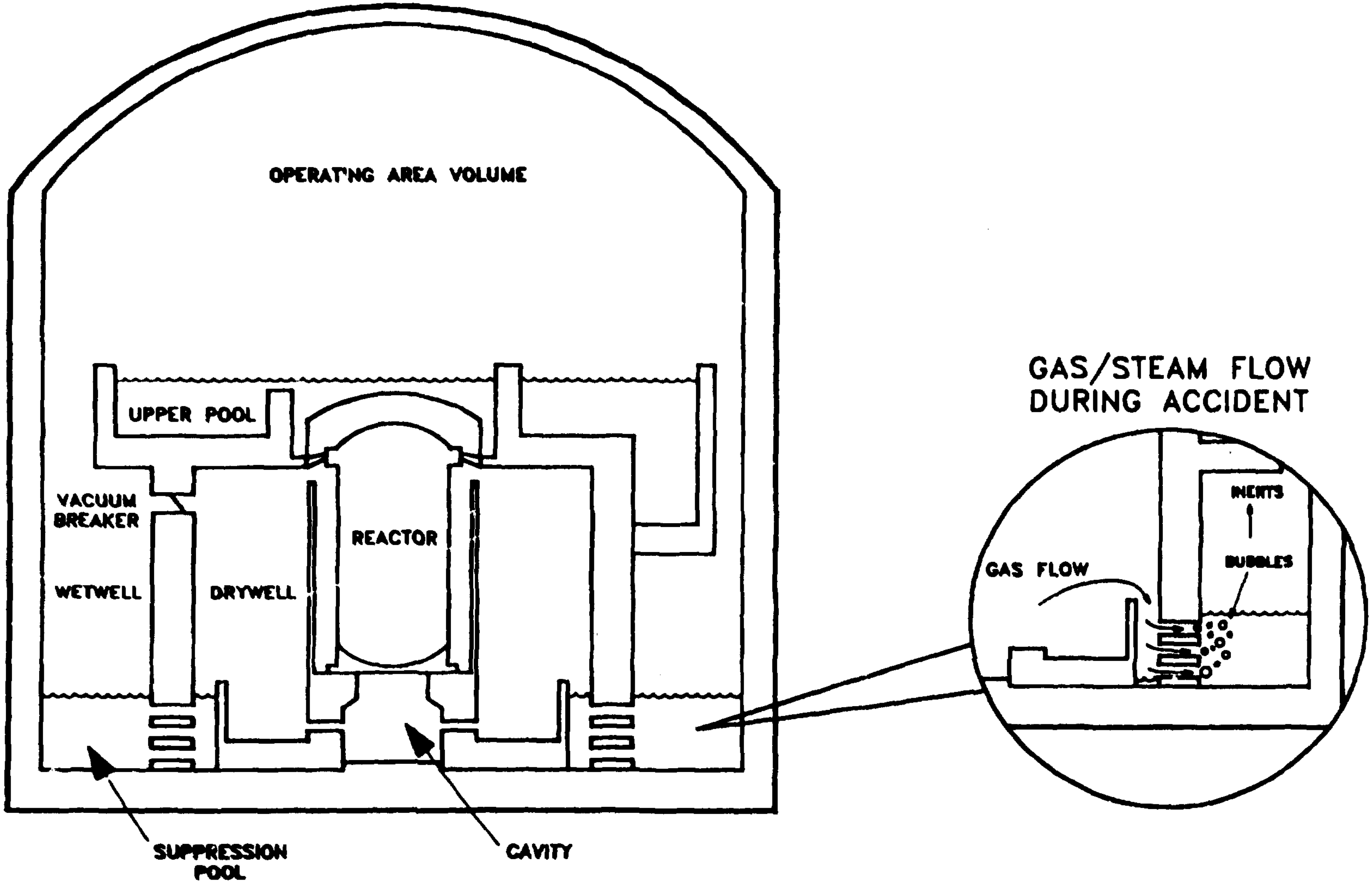


Fig. 1. Operation of BWR Mark III suppression Pool.

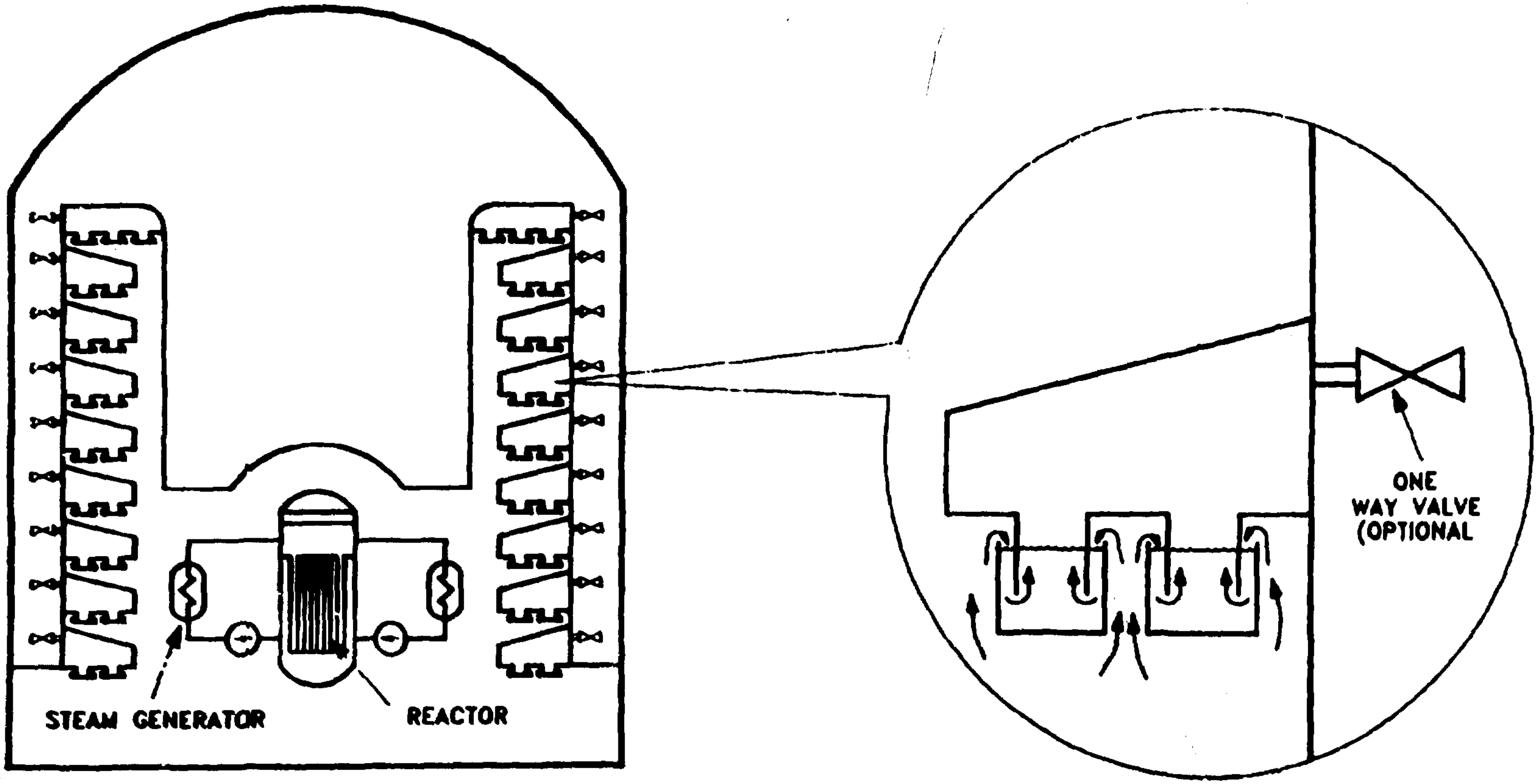


Fig. 2. VVER bubbler/condenser tower.

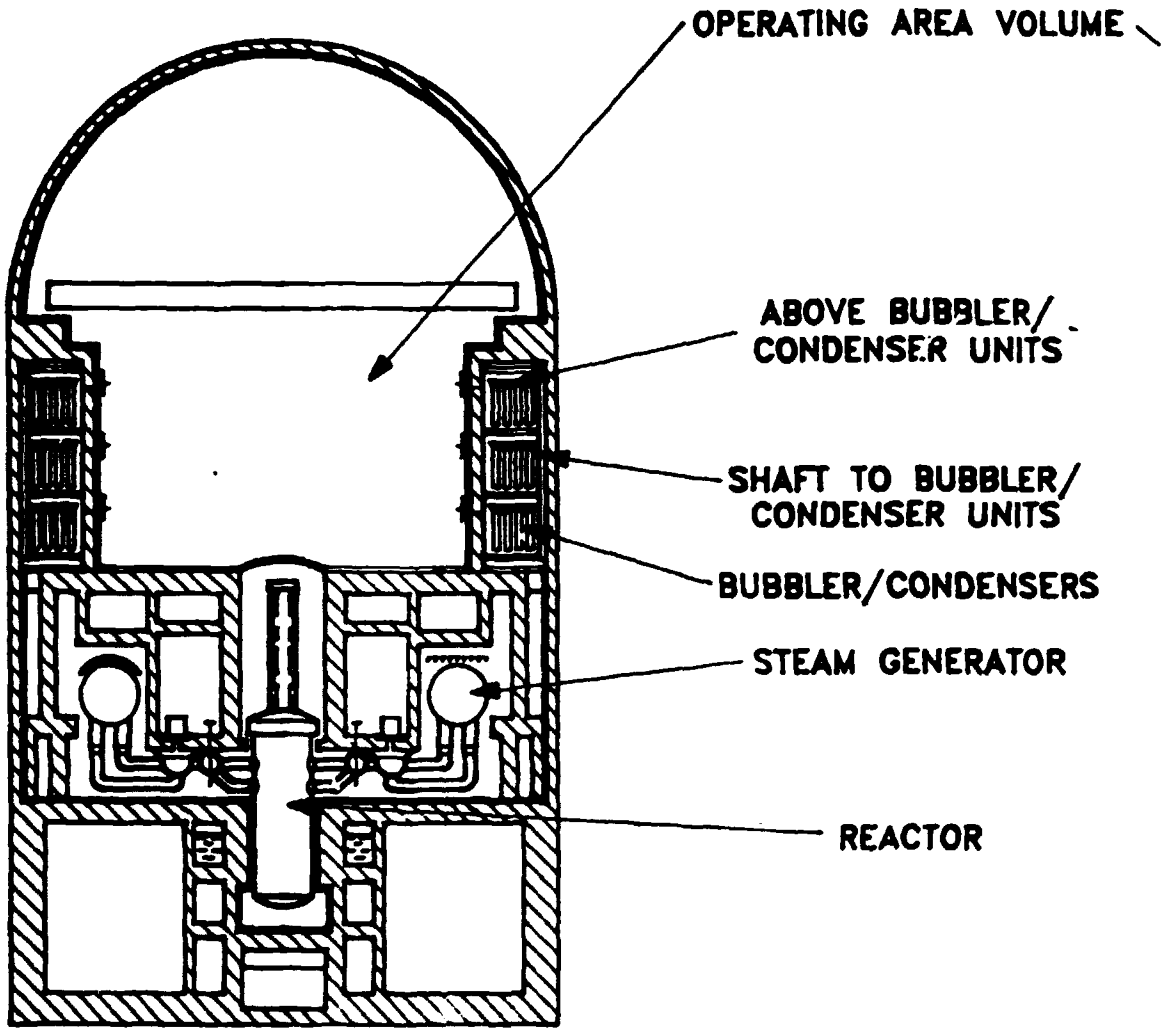


Fig. 3. Soviet VVER-440, model 213 reactor bubbler/condenser containment.

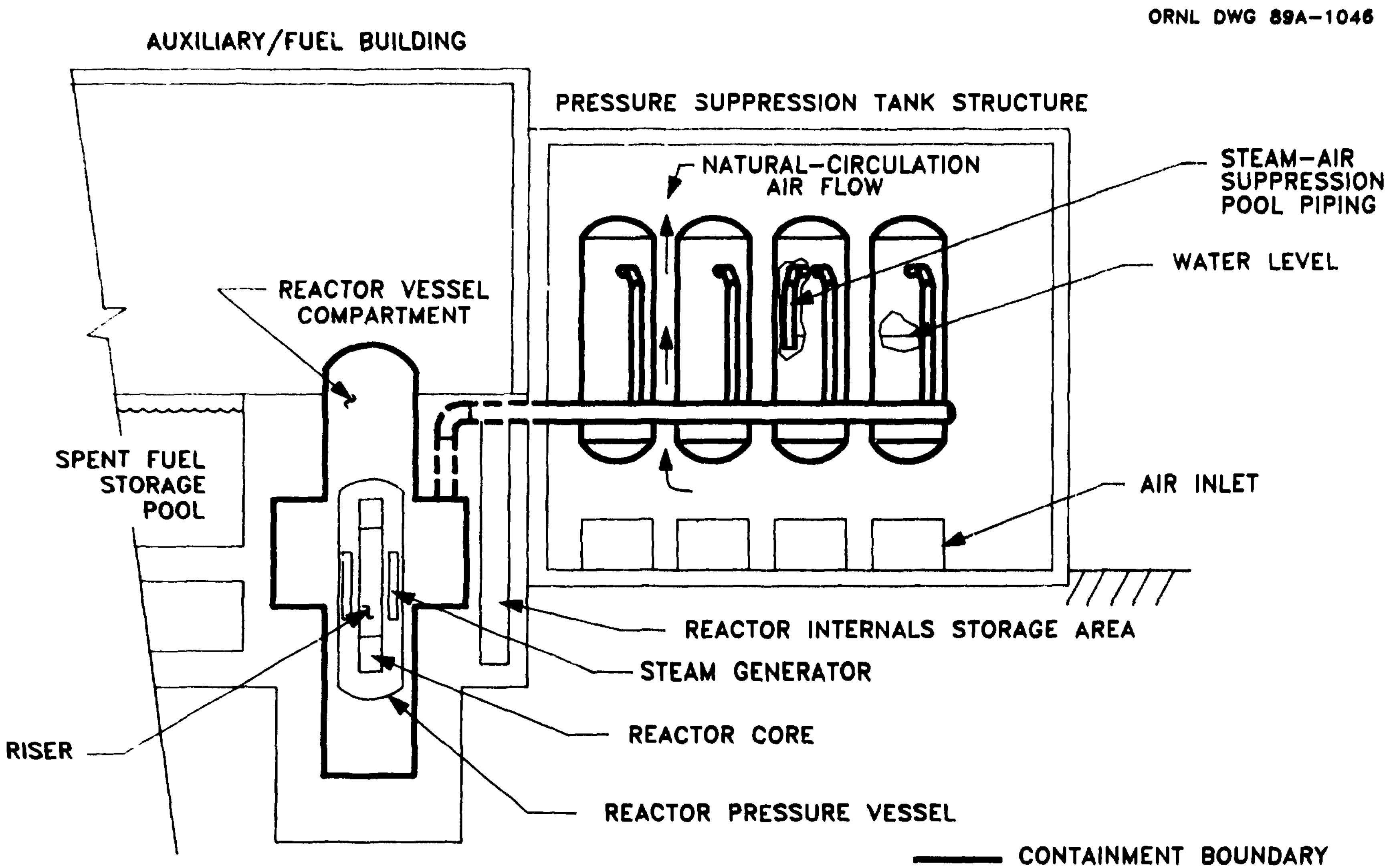


Fig. 4. Safe integral reactor containment boundary and pressure suppression system.

TITLE: STEAM CONDENSATION AND VACUUM CREATION BY SUPPRESSION POOLS

Functional Requirements: Level 3.1.2 Control Energy in Containment

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (USSR)

Examples of Implementation: Certain Soviet VVER (pressurized water) reactors

Description: In a loss-of-coolant accident (LOCA), flash vaporization of the leaking reactor coolant produces a rapid pressure increase in the reactor containment building. Several systems have been developed to use this initial postaccident pressure surge to move the relatively nonradioactive gases, which were originally in containment and near the reactor, to a secondary containment volume, then condense steam and, finally, create a vacuum in the containment volume near the reactor core. Since radioactive gases and aerosols are not immediately released in an LOCA, sufficient time is available to create vacuum conditions in containment immediately after the accident. Vacuum containment prevents leakage of any subsequent radioactive gases and aerosols.

Each of the systems used divides the containment into two or more volumes: one volume containing the reactor and primary system, the other volume(s) containing the remaining space in containment. To reduce the pressure of the containment volume near the reactor after initiation of an accident, a secondary containment area is connected to the primary containment by a conduit containing a passive vapor condenser such as a suppression pool. The suppression pool is a water-filled trough covered by a housing that is designed to channel the high-pressure steam-air mixture into the water (Fig. 1, left side). The housing inlet pipe is submerged in the pool so that the steam-air mixture bubbles through the water. The cooler water of the pool condenses most of the steam (thus reducing the pressure), while the air proceeds through the housing outlet pipe and into the secondary containment (Fig. 1, right side). Several arrangements of this system are possible, including individual pools venting to individual secondary containments (Fig. 2, top), groups of pools venting to a secondary containment common to the group (Fig. 2, middle), and multiple pools venting to a common secondary containment (Fig. 2, bottom).

The pressure surge that forces the steam-air mixture to the secondary containment ends when the reactor is fully depressurized. As the steam in primary containment condenses against the colder walls it creates a negative pressure in the primary containment, preventing the escape of any fission products to the outside atmosphere. Water sprinkler systems are used to condense any steam subsequently produced, and the negative pressure is maintained until the pressure in secondary containment becomes high enough to (1) overcome the pressure exerted by the water in the suppression pool (ΔH in Fig. 1) and (2) force air back into the primary containment. To prevent the higher pressure in secondary containment from forcing a backflow of air, a one-way valve can be installed at the inlet to the secondary containment (Fig. 1, bottom right).

Alternative Versions: Figure 3 shows an advanced design that does not use a valve but is still expected to provide excellent protection against backflow from secondary to primary containment volumes. This eliminates concern about valve failure due to defects, damage, or the presence of foreign objects. Following an LOCA, the initial pressure surge drives the steam-air mixture through openings in the wall of the primary containment, up into an intermediate chamber, and down through a long vertical conduit into the suppression pool in secondary containment below. The vertical conduit is designed to provide a water seal against backflow of air from the secondary containment, with the vertical conduit height (in meters) being no less than 10 times the ratio of the primary containment volume to the secondary containment volume. As with current designs, the pressure needed to force gas from the primary to secondary containment is proportional to the depth of water in the suppression pool (ΔH_1). However, if the secondary containment volume gas pressure is larger, it forces water up the vertical conduit. The pressure needed to force gas from secondary to primary containment volumes is proportional to the height of the vertical conduit (ΔH_2). In effect, a small pressure difference moves gas from primary to secondary containment, but a much larger pressure difference is needed to move gas in the reverse direction.

A number of different configurations of this advanced design have been proposed[2,4] for different reactor plant layouts.

Status of Technology: The multiple suppression pools with check valve systems are used in Soviet VVER pressurized-water reactors. One version not requiring a valve will be used in the VVER-440 Model V-213 being constructed at Juragua, Cuba, as shown in Fig. 4 and described in Table 1.

Advantages: Negative pressure in primary containment prevents escape of fission products to the outside environment.

Additional Requirements: None

Comments: None

References/Contacts:

1. A. M. Bukrinsky, G. V. Matskevich, J. V. Rzheznikov, et al., System for Mitigating the Effects of an Accident at a Nuclear Power Plant, U. S. Patent 4,056,436 (November 1, 1977).

2. A. M. Bukrinsky, J. V. Rzheznikov, J. V. Shvyryaev, et al., System for Mitigating Consequences of Loss-of-Coolant Accident at Nuclear Power Station, U. S. Patent 4,362,693 (December 7, 1982).

3. U.S. Department of Energy, Overall Plant Design Descriptions, VVER Water-Cooled, Water-Moderated Energy Reactor, DOE/NE-0084 (October 1987).

4. D. Sykora, [illegible]pressed Air Vacuum System for Localization of Accidents in Nuclear Reactors, Ge[illegible]an Democratic Republic Patent 160,446: (November 4, 1980) ORNL/TR-89-12 (1989).

Update/Compiler: March 1989/EBL

Table 1. Safety Features of VVER-440 Model V213 - Cuban Variant

Containment	Steel-lined cylindrical reinforced concrete
Sprinkler systems	Active and Passive
Suppression pool design	Multilevel passive pools
Suppression pool water additives	Boric acid, sodium thiosulfate, and potassium hydroxide
Aggregate volume of suppression pools	~780 m^3
Estimated pressure differential required to return air to primary containment	0.038 MPa (5.5 psi/water head of 13 ft)
Estimated time for primary containment to reach negative pressure following a cold leg pipe break	100 s

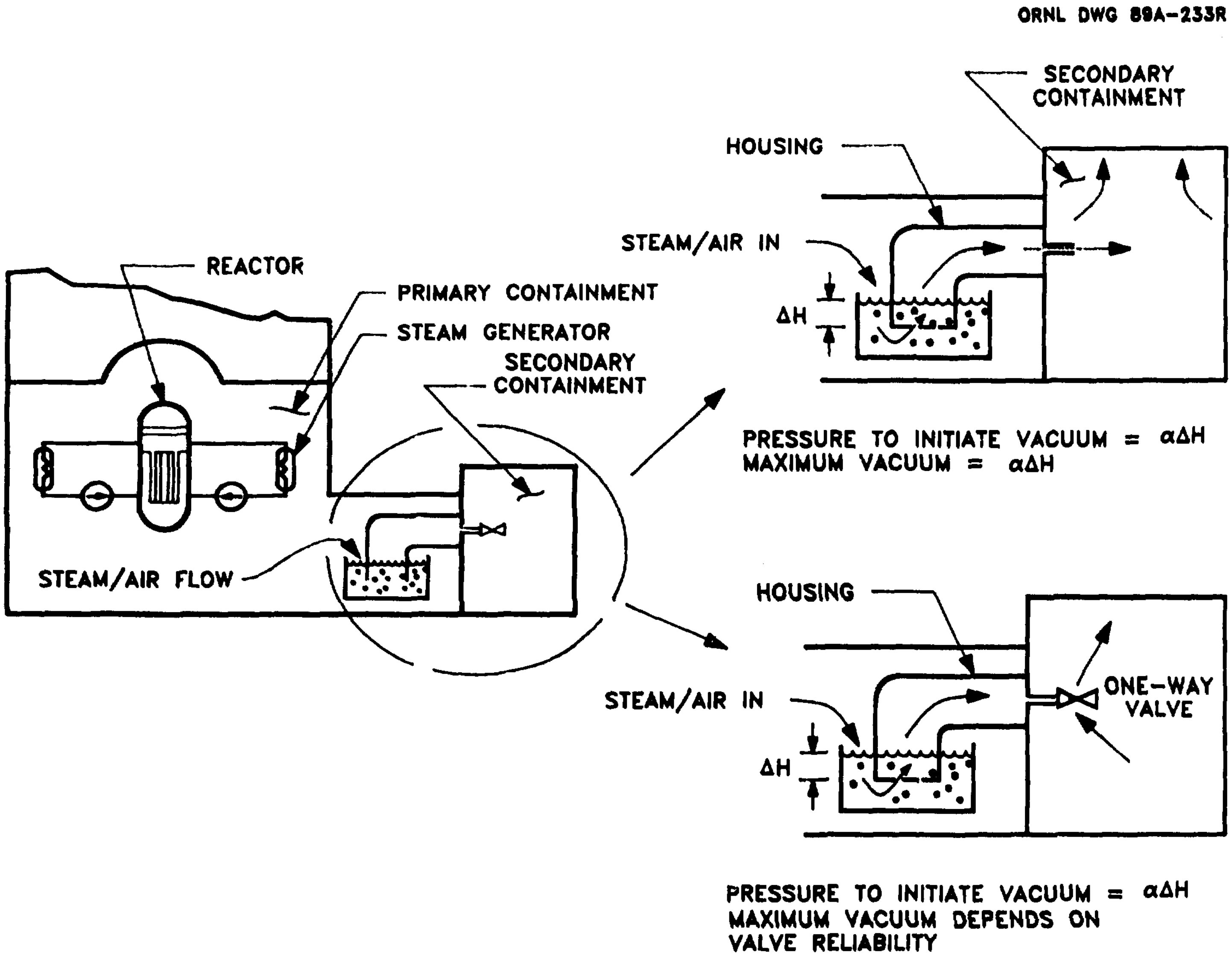


Fig. 1. Current designs for generating vacuum with suppression pools.

ORNL 89A-234

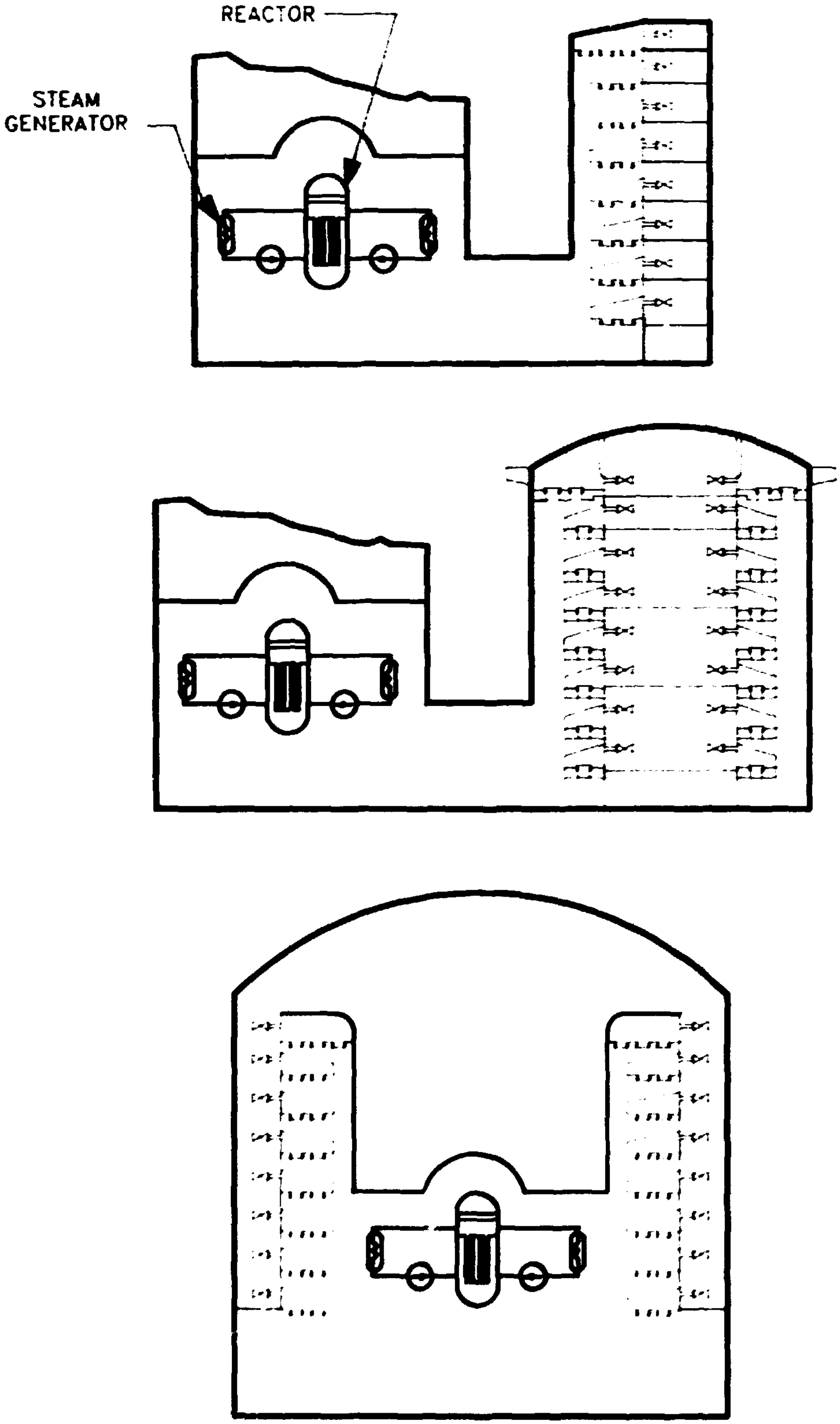


Fig. 2. Alternative suppression pool arrangements.

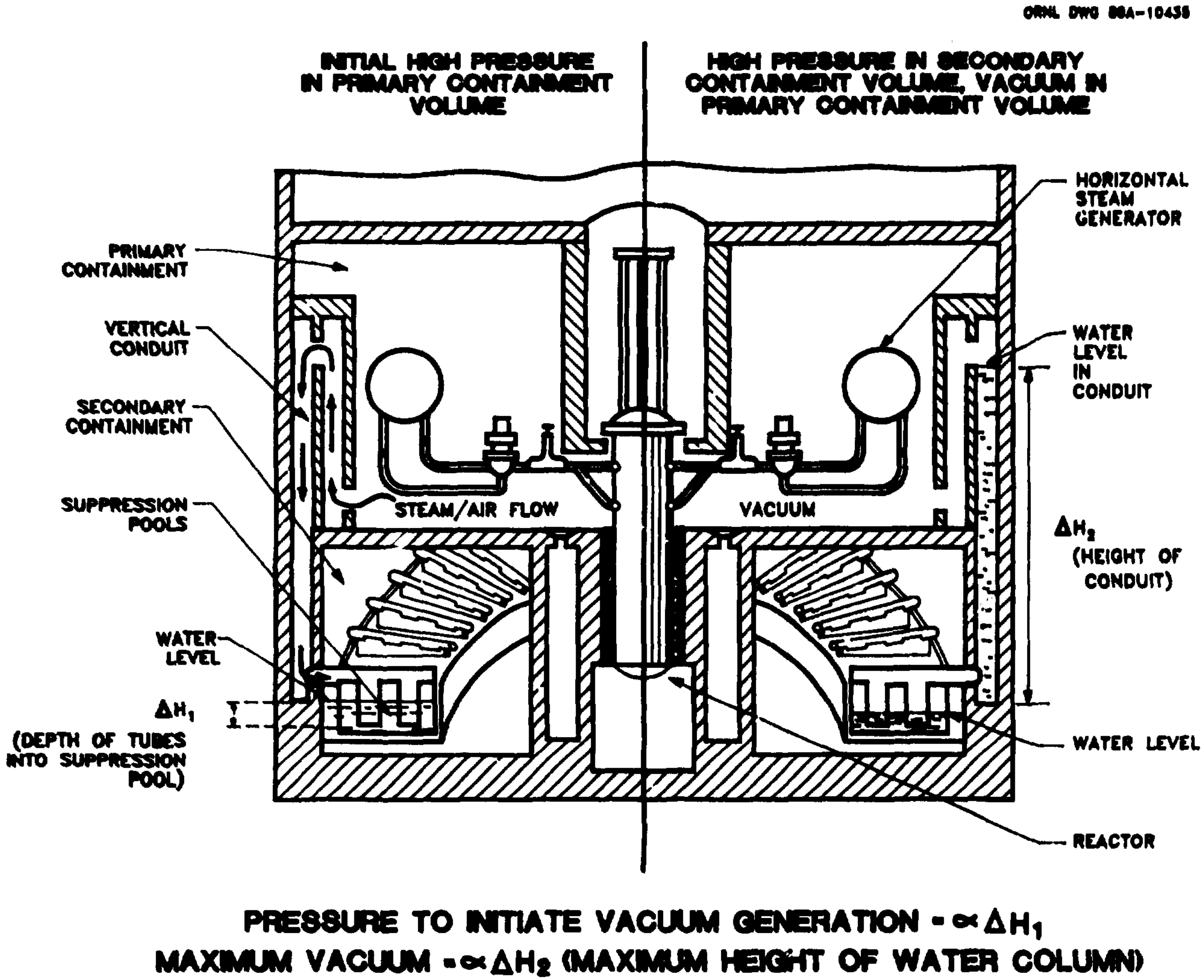


Fig. 3. Advanced option for generating vacuum with suppression pools during operation.

ORNL DWG 89-00052

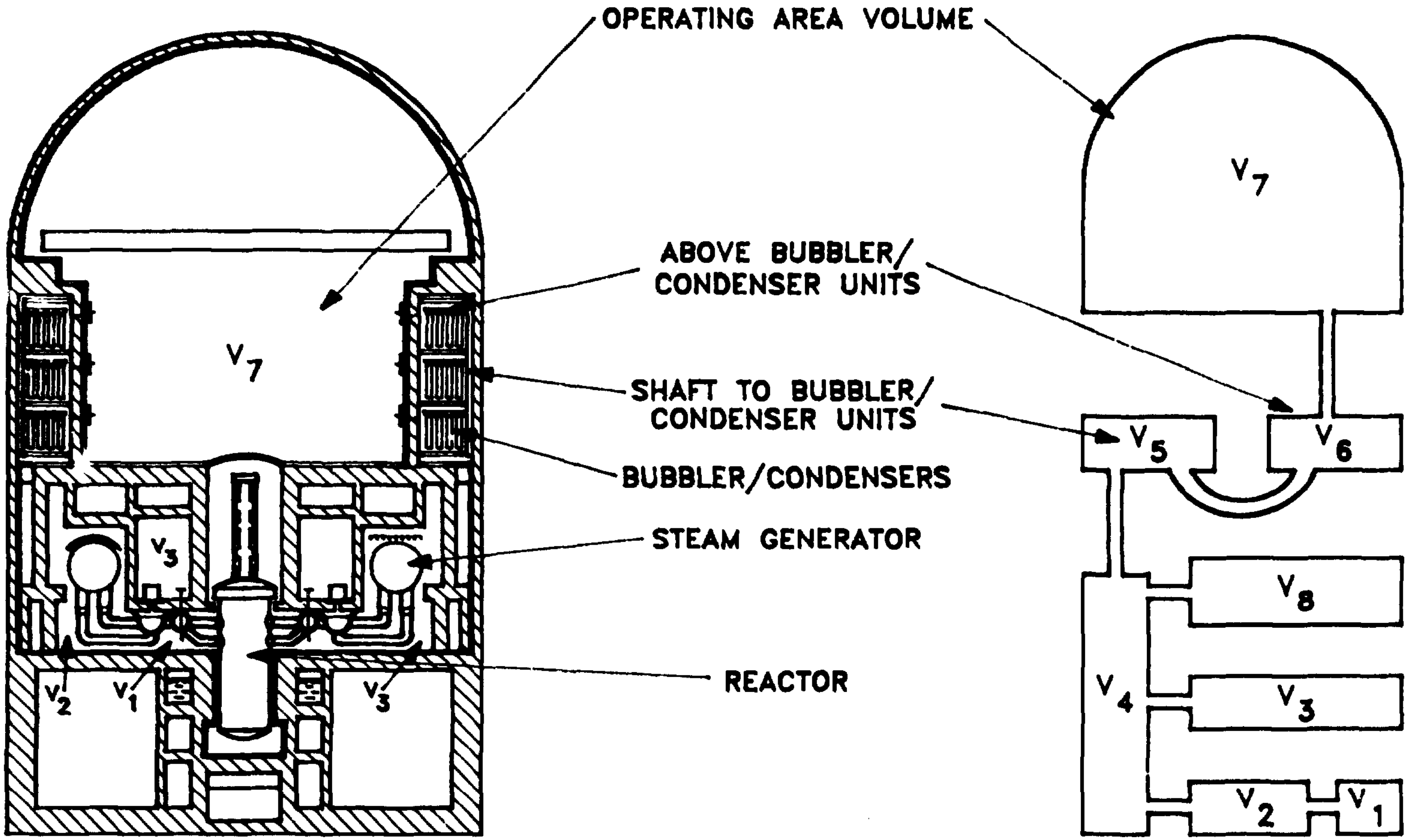


Fig. 4. Schematic of bubbler/condenser containment concept (elevation view) in Soviet VVER-440, model V213 reactors.

TITLE: WATER-COOLABLE CORE CATCHER

Functional Requirement: Level 3.1.2: Control Energy in Containment
Level 3.2: Trap Radionuclides

Safety Type: Passive

Developmental Status: 3 Evaluated

Reactor Type: Light-water reactor (LWR)

Organization: Multiple

Examples of Implementation: None

Description: A water-coolable core catcher can be used to contain the molten debris generated by a nuclear reactor core meltdown after a postulated reactor accident. The primary function of a core catcher is to retain and cool the molten core debris and keep the debris in a long-term coolable geometry to prevent damage to the integrity of the reactor containment structures. Interaction of molten or hot solid core debris with the concrete containment structure may generate explosive gases, overpressurize the containment structure, result in accelerated aerosol production, enhance heat transport to containment structures, and could ultimately breach the primary containment and release radioactive debris to the environment.

The two main types of core retention materials that have been considered for use in a core catcher are (1) sacrificial materials and (2) refractory materials. Sacrificial materials, such as borax, will interact with the molten core to absorb and dissipate the heat. Refractory materials, such as firebrick, alumina, and oxides of magnesium and thorium, are very resistant to chemical and thermal attack by the molten core and will act to physically contain the debris, similar to a crucible. The retention materials can be formed into bricks, cast as a solid crucible, or layered in various fashions in a particle bed. Many proposed designs place the core catcher outside and underneath the pressure vessel, yet inside the primary containment building, but designs have been proposed that place the core catcher within the pressure vessel (see Alternate Versions).

A typical water coolable core catcher is shown in Fig. 1. Many designs for particle-bed core catchers have been proposed and offer advantages over other designs. Particle beds have a lower overall thermal conductivity than solid beds and thus offer greater protection to the catcher basin and containment structure. Particle beds can be layered with particles of varied size and composition to address different design criteria. For example, the top layer of a particle bed is often composed of large particles (1-3 cm) of high-heat-resistant material (such as ThO_2) to allow rapid penetration and subsequent dilution of molten core materials. These large particles also prevent the top layer from being expelled from the catcher by a high velocity stream from the initial breach of the vessel. Middle layers of similar heat resistant materials are usually composed of small particles (~ 1 mm) in order to contain the molten core using surface tension effects and conduction freezing. Bottom layers usually contain much larger particles to allow for passage of coolant around the trapped core debris. Alumina has been used as a bottom layer where the more costly, heat resistant materials are not required.

Other designs have proposed homogeneous particle beds composed only of graphite. If axial particle-size variation is undesirable, another option[4] is to use stick-like graphite pieces to enhance the top layer's resistance to expulsion by high pressure streams and yet retain the ability to trap the molten core.

The initial trapping of the molten core is the first stage of operation of a core catcher. It has been shown[1,4] that cooling water has no effect on the initial penetration depth or penetration rate of the molten core debris and may be undesirable in the initial stage of core debris and catcher contact. In the second stage of core catcher operation, after the initial trapping of the molten core, heat buildup in the catcher materials will result in remelting of the debris and further penetration. Thus, cooling of the core catcher in the second stage is necessary. Either cooling water could be pumped through the bed or a more passive system relying on gravity could be used. (See: Temperature Activated Water Sprinkler for Steam Condensation and Accident Mitigation.)

In addition to protecting the containment structures, the core catcher will help to immobilize releases of radionuclides to the containment as detailed in Ref. 7. A summary of the results is given in Fig. 2,[7] which shows the mass fraction of radionuclides released to containment as a function of time after core meltdown. Releases are shown with and without water under the pressure vessel. It can be seen that flooding the core catcher with water reduces the releases of a number of radionuclides.

Alternative Versions: The core catcher described in Ref. 5 is referred to as a collecting tray. This tray, placed underneath the core and inside the pressure vessel, is a conical-shaped container with a number of attached heat pipes that allow cooling of the core debris. At the base on the inside of the pressure vessel wall is an attached skirt to direct molten core debris into the collecting tray. Cooling is totally passive by natural convection. The tray has been proposed for PWR, BWR, and Liquid-Metal Fast Breeder Reactor designs. The tray and cooling pipes would be arranged to ensure that the molten core would not form a critical mass.

Status of Technology: Small-scale and prototype demonstrations on sacrificial and refractory core catcher materials have been performed. Models have been developed to extrapolate experimental results to reactor-scale core catcher devices. Use of reactor core catchers has been proposed for future European reactor designs[3,6].

Advantages:
1. Damage to the primary containment structure, and in some designs the pressure vessel, is prevented.
2. Radionuclide releases to containment after core meltdown are reduced.
3. No moving mechanical components are required.
4. Cooling is totally passive.

Additional Requirements: None

Comments: None

References:

1. J. D. Fish, M. Pilch, F. E. Arellano, "Demonstration of Passively-Cooled Particle-Bed Core Retention," Proceedings of the L.M.F.B.R. Safety Topical Meeting, Lyon-Ecully, France, July 19-23, 1982, American Nuclear Society, Vol. 3, pp. 327-36.

2. D. A. Powers, "A Survey of Melt Interactions with Core Retention Material," Proceedings of the International Meeting on Fast Reactor Safety Technology, Seattle, WA, August 19-23, 1979, American Nuclear Society, Vol. 1, pp. 379-88.

3. L. Fogelström (Västeras, Sweden) and M. Simon (Federal Republic of Germany), "Development Trends in the Area of Light-Water Reactors," Atomwirtschaft, August/September, 1988, pg. 423-426: ORNL/TR-88/38, Oak Ridge National Laboratory. Oak Ridge, Tennessee (1988).

4. M. J. Driscoll and F. L. Bowman, Core Catcher for Nuclear Reactor Core Meltdown Containment, United States Patent No. 4,113,560 (September 12, 1978).

5. D. Broadley, Collecting Tray for Nuclear Reactor Core Debris, United Kingdom Patent Office, London Patent No. 1 464 425, (February 16, 1977).

6. R. Adinolfi and L. Noviello, "Passive Systems Considerations for Light-Water Reactors," presented at the International Atomic Energy Agency Technical Committee Meeting on Passive Safety Features in Current and Future Water-Cooled Reactors, Moscow, USSR, March 21-24, 1989.

7. H. P. Nourbakhsh, M. Khatib-Rahbar, and R. E. Davis, Fission Product Release Characteristics Into Containment Under Design Basis and Severe Accident Conditions, NUREG/CR-4881 (BNL-NUREG-52059) (March 1988).

Update/Compiler: April 1989/WJR

ORNL DWG 89A-320

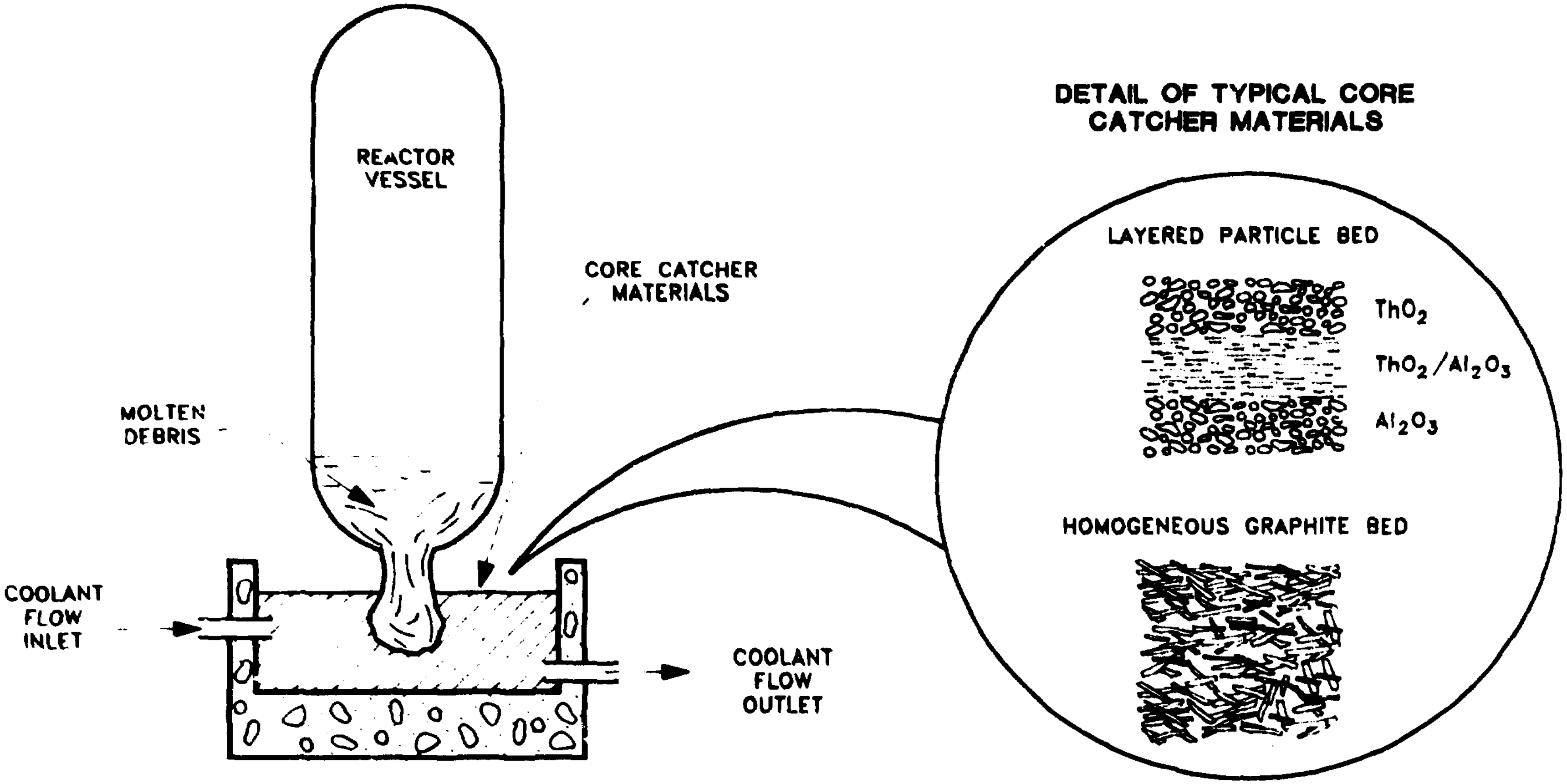
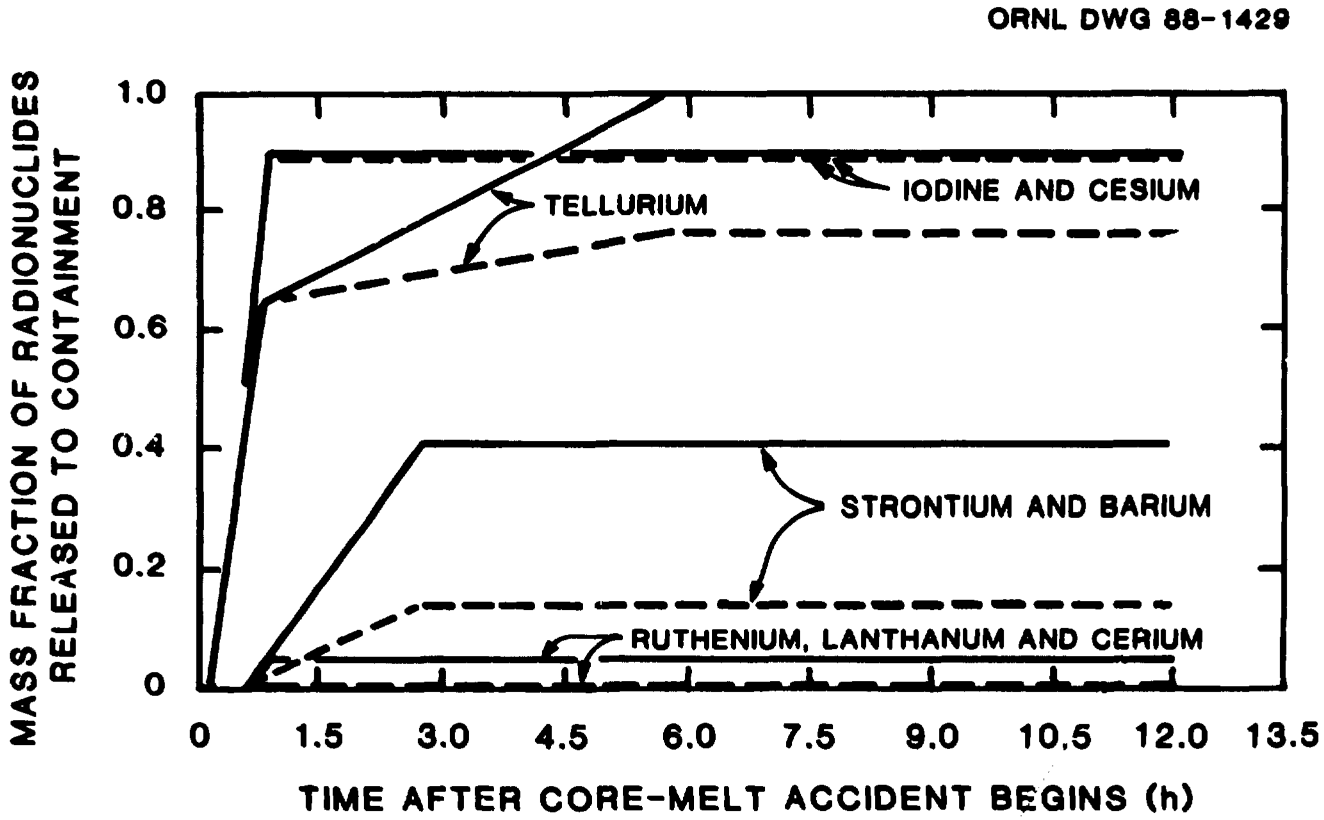


Fig. 1. Typical water-coolable core catcher.



VARIABLE
- - - WATER UNDER PRESSURE VESSEL, DF = 3
—— NO WATER UNDER PRESSURE VESSEL

CASE SPECIFICS
PRESSURIZED WATER REACTOR
LOW REACTOR COOLANT SYSTEM PRESSURE BEFORE STEEL VESSEL MELTTHROUGH
LIMESTONE CONCRETE

Fig. 2. Cumulative release fractions of radionuclides to containment by chemical group and amount of water under pressure vessel after core-melt accident.

TITLE: CONTAINMENT PRESSURE CONTROL BY PRESSURE ACTIVATED WATER SPRINKLERS

Functional Requirement: Level 3.1.2: Control Energy in Containment
Level 3.2: Trap Radionuclides

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-Water Reactor (LWR)

Organization: Vendor (Canada and USSR)

Examples of Implementation: Water sprinkler dousing systems designed to control containment pressure are in use at the Pickering Nuclear Generating Station (NGS), the Bruce-A NGS, the Bruce-B NGS, and the Darlington NGS in Ontario, Canada.

Description: Water sprinklers can be used to suppress containment pressure after a postulated loss-of-coolant accident (LOCA). The dousing system's primary function is to rapidly cool and condense the steam-air mixture, thus acting as a passive heat sink to absorb the released energy. Water drawn from an emergency storage tank (located above the sprinklers and drained by gravity) can act as the heat sink as it is sprayed into the compartment or building to which the steam-air mixture has been vented. Sprinklers also provide a secondary function: they wash out radionuclides in the containment environment. Sprinkler operation can be initiated by pressure difference, pressure pulse, or rise in containment pressure.

Sprinkling systems can be passively activated by a pressure difference and can be designed so that water flow is interruptible or continuous (see Fig. 1). Canada's Ontario Hydro has incorporated a dousing system in all of its CANDU reactors.[1,2,3] The dousing system is initiated and maintained in operation by a pressure difference between the main vacuum building chamber and an upper vacuum chamber, as shown in Fig. 2. (see SSC description: Vacuum Containment). The main vacuum building containment is normally maintained at 7 kPa and the upper vacuum chamber is normally pumped down to 1.5 kPa. If an LOCA occurs, the sudden rise in pressure in the main vacuum chamber pushes cold water in the emergency water storage tank up sixteen riser pipes, over a flow regulating weir, and into spray headers containing 896 spray plates. Water will continue to spray into the vacuum building until all of the steam is condensed and the vacuum is reduced to normal. This type of device can either operate based on continuous pressure difference or short-term pressure rise. As shown in Fig. 2b, extension of the siphon below the bottom of the water tank assures continued sprinkler operation after initiation until the tank is dry.

Some types of Soviet LWRs contain passive sprinklers designed to lower containment pressure within minutes of initiation of an accident and act as "suppression pools." The sprinkler systems can be connected to large, elevated tanks of water or multiple, smaller tanks of water. Three design options[3] are shown in Fig. 3 and described below.

Sprinklers can be activated by pressure surges (see Fig. 3a). Immediately after an LOCA accident, containment pressure rises rapidly. The resultant steam-air mixture is pushed backwards through the sprinkler system and siphon into the sealed sprinkler water tank. The steam is condensed while the gas pressure in the water tank is increased by noncondensable gases. As steam begins to condense in the containment, the containment pressure begins to drop. At this time, the higher pressure in the sealed water tank pushes water into the sprinkler siphon and sprinkler action is initiated. Once the siphon starts operation, the sprinkler continues until the water tank is empty.

Other methods to rapidly activate sprinklers by rising pressure in containment are shown in Fig. 3b and Fig. 3c. In Fig. 3b, after reactor accident initiation, containment pressure rises, and the resultant steam-air mixture is pushed backwards through the sprinkler system and siphon into the sealed sprinkler water tank. The normal siphon rapidly pressurizes the gas volume in the sprinkler tank. In contrast, the second siphon contains a large gas volume and a throttle. Steam condenses in the gas volume and the pressure rise is very slow in this siphon. As a consequence, this siphon has a lower pressure than either the containment or the water tank. The higher gas pressure in the water tank pushes water into the siphon, initiating operation of the sprinkler. The sprinkler lowers containment pressure, which forces operation of the first siphon. The pressure-rise activation sprinklers in Fig. 3c operate in a similar fashion.

Alternative Version: None

Status of Technology: Passive water sprinklers are in use in Canadian CANDU reactors and certain Soviet LWRs. Details of the Soviet designs are not currently available.

Advantages:
1. Water sprinkler systems can be designed to reduce containment overpressure transients within minutes after an accident.
2. The systems are completely passive in operation.
3. Sprinklers wash radionuclides from containment atmosphere.

Additional Requirements: None

Comments: None

References/Contacts:

1. Z. M. Beg and R. S. Ghosh, "Evolution of CANDU Vacuum Building and Pressure-Relief Structures from Pickering NGS A to Darlington NGS A," 9th International Conference on Structural Mechanics in Reactor Technology, H. Concrete and Concrete Structures, Lausanne, August 17-21, 1987.

2. W. W. Koziak, "Bruce-B NGS - Containment Commissioning Test," Nuclear Studies and Safety Department, Ontario Hydro, Toronto, Ontario, Canada, (October 1983).

3. J. Huterer, D. G. Brown, M. A. Osman, and E. C. Ha, "Pressure Relief Structures of Multi-unit CANDU Nuclear Power Plants," Nuclear Engineering and Design, 100, 21-39 (1987).

4. A. M. Bukrinsky, J. V. Rzheznikov, J. V. Shvyryaev, V. P. Tatarnikov, A. L. Lapshin, V. I. Sanovich, D. A. Zlatin, J. A. Kuznetsov and E. A.Babenko, System for Mitigating Consequences of Loss-of-Coolant Accident at Nuclear Power Station, U.S. Patent 4,362,693 (Dec. 7, 1982).

Update/Compiler: Feb. 1989/RLP

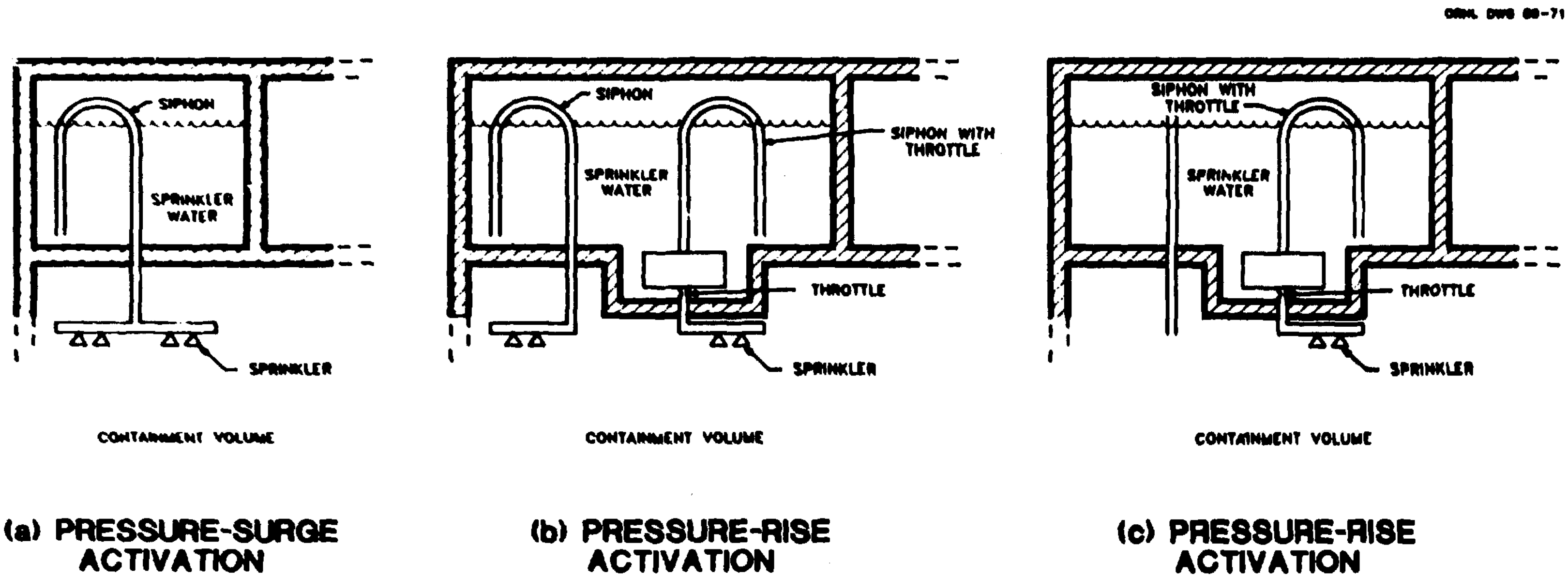


Fig. 1. Schematic of passive-pressure-activated sprinkler systems.

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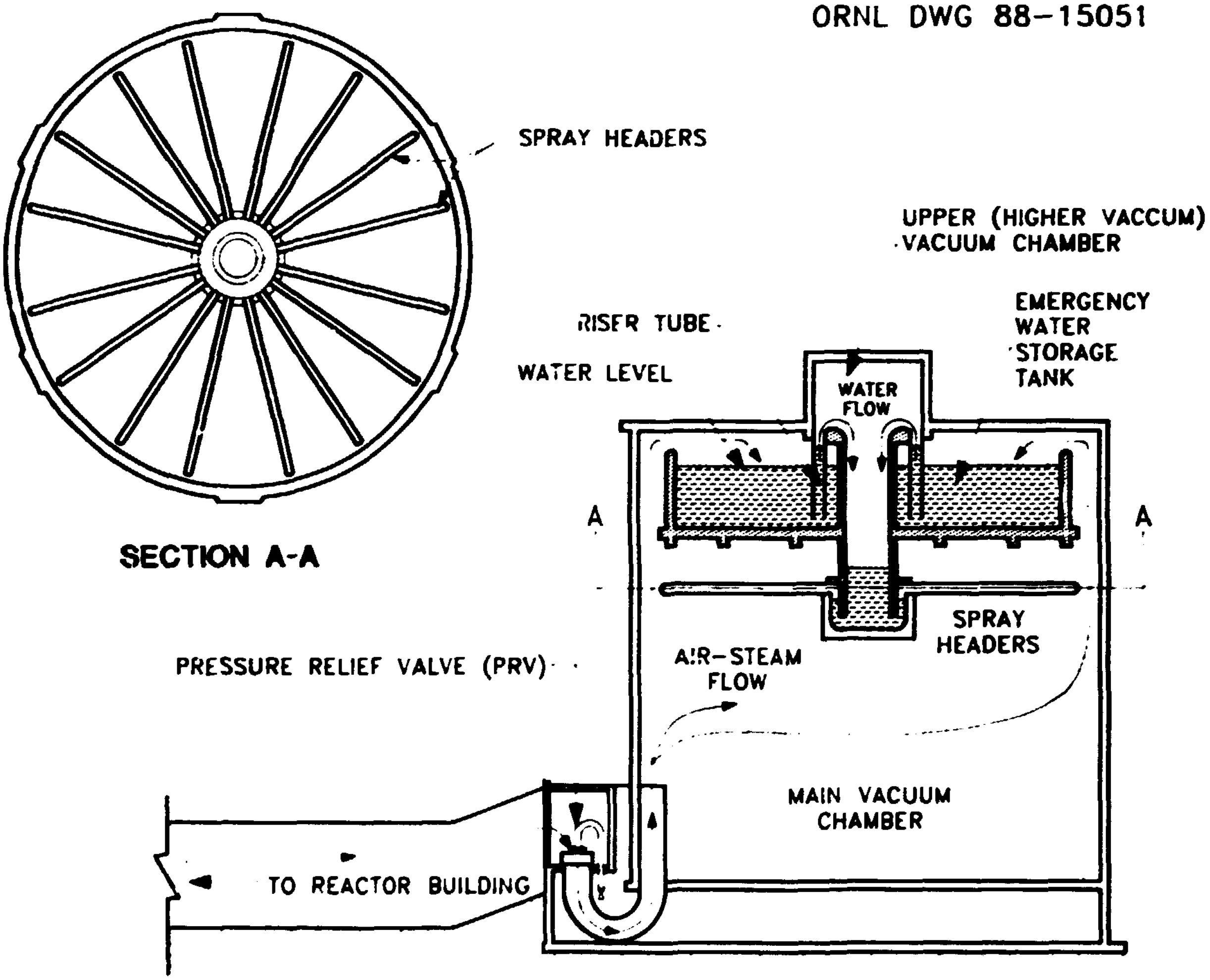
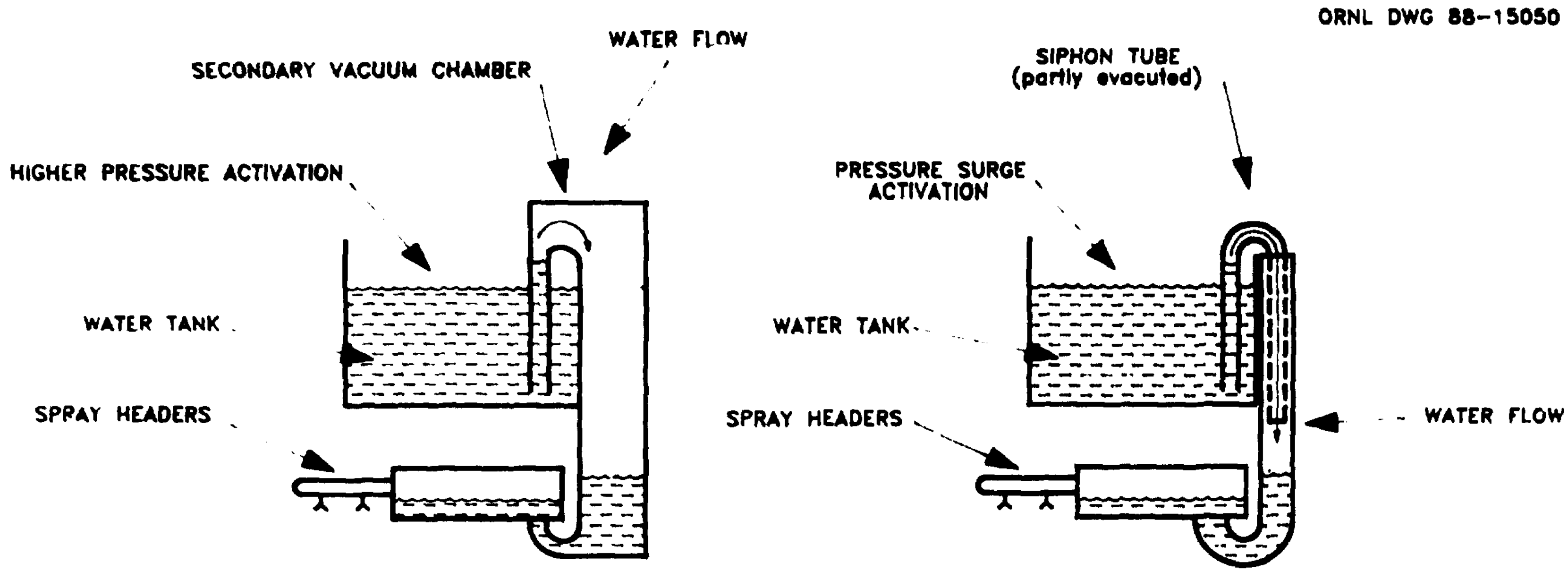


Fig. 2. Vacuum building dousing system.



(a) INTERRUPTIBLE-FLOW DESIGN
(PREOPERATION)

(a) CONTINUOUS-FLOW DESIGN
(PREOPERATION)

Fig. 3. Pressure-activated containment spray systems.

TITLE: TEMPERATURE ACTIVATED WATER SPRINKLERS FOR STEAM CONDENSATION AND ACCIDENT MITIGATION

Functional Requirement: Level 3.1.2 Control Energy Release Into Containment
Level 3.2 Trap Radionuclides

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Light-Water Reactor (LWR)

Organization: Industrial (Multiple)

Examples of Implementation: None

Description: Water sprinklers can be used to suppress containment overpressure after a postulated loss-of-coolant accident (LOCA). The sprinkler system's primary function is to rapidly cool and condense the steam-air mixture, thus acting as a passive heat sink to absorb the released energy. Water drawn from an emergency storage tank (located above the sprinklers and drained by gravity) can act as the heat sink as it is sprayed into the compartment or building to which the steam-air mixture has been vented. Sprinklers can also provide other functions: they washout radionuclides in the containment environment and may be used to flood the area below the pressure vessel in a postulated core melt accident to prevent loss of containment.

Sprinkling systems can be passively activated by the ambient temperature. There are several types of temperature-sensitive automatic sprinklers used in buildings for fire protection (see Fig. 1).[1] Under normal conditions, the discharge of water from an automatic sprinkler is restrained by a cap or valve held tightly against the orifice by a system of levers and links pressing down on the cap and anchored firmly by struts on the sprinkler. The common fusible-link automatic sprinkler operates upon the fusing of a metal alloy with a predetermined melting point. The frangible-bulb sprinkler has a small glass bulb operating element. The bulb contains a liquid that does not completely fill the bulb, as there is a small gas bubble entrapped in it. When the bulb is heated, the gas is absorbed by the liquid. Then the pressure rises rapidly and the bulb shatters, releasing the valve cap. The exact operating temperature is regulated by adjusting the amount of liquid and the size of the bubble when the bulb is sealed. The frangible-pellet sprinkler has either a pellet of solder or other eutectic metal under compression that melts at a predetermined temperature and allows movement of the releasing elements. Automatic sprinklers have various temperature ratings that are based on standardized tests. The maximum safe room temperature is closer to the operating temperature for bulb and fusible pellet sprinklers than for sprinklers having soldered fusible elements. Solder begins to lose its strength somewhat below its actual melting point. National Fire Protection Association sprinkler standards are listed in Table 1.

Alternative Version: Each of the temperature-sensitive automatic sprinklers shown in Fig. 1 will initiate a continuous discharge of water. If an interruptable flow of water is desired, development of an automatic sprinkler with a new type of activator such as a bimetallic element will be required.

Status of Technology: Automatic temperature-sensitive sprinkler systems have been used by the building industry for fire protection for many decades, but have not been used in reactor containments. For the Advanced Boiling-Water Reactor, General Electric is planning to use sprinklers to flood the area below the pressure vessel in the event of a core melt accident (2) to stop the accident.

Advantages:
1. Water sprinkler systems can be designed to reduce containment overpressure transients within minutes after an accident.
2. The systems are completely passive in operation.
3. The technology for construction of automatic sprinklers is available.
4. Sprinklers wash radionuclides from containment atmosphere.

Additional Requirements: None

Comments: A very large industrial experience base exists with this technology.

References/Contacts:

1. Fire Protection Handbook, 14th Ed., National Fire Protection Association, Boston, Massachusetts (1976).

2. C. D. Sawyer, private communication to C. W. Forsberg (October 13, 1989).

Update/Compiler: Feb. 1989/RLP

Table 1. Standard temperature ratings of automatic sprinklers

Rating	Desired operating temps. (°F)	Max. safe ceiling temp. °F
Ordinary	135°, 150°, 160°, 165°	100°
Intermediate	175°, 212°	150°
High	250°, 280°, 286°	225°
Extra high	325°, 340°, 350°, 360°	300°
Very extra high	400°, 415°	375°
Very extra high	450°	425°
Very extra high	500°	475°

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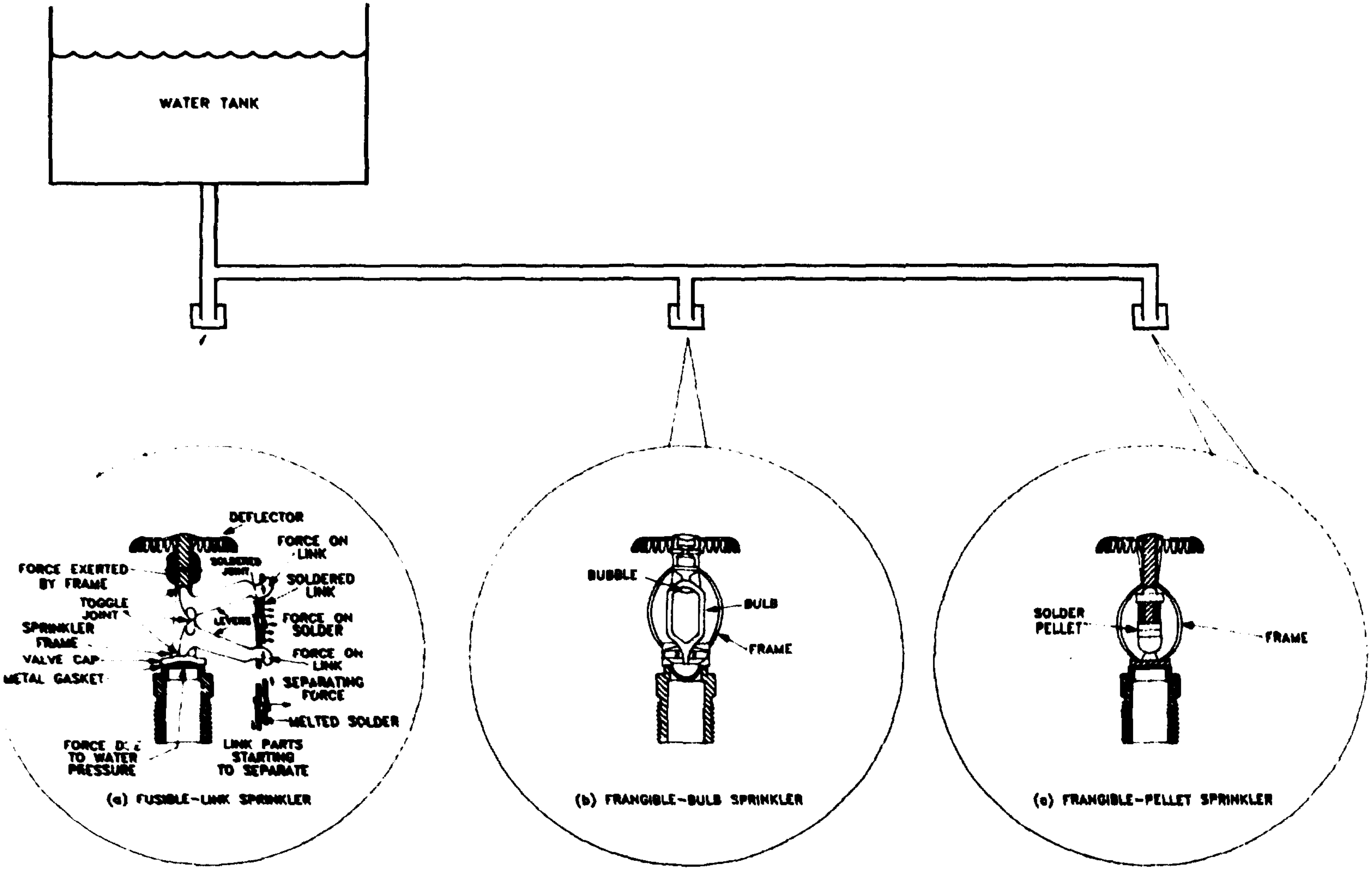

Fig. 1. Schematic of temperature-activated automatic sprinklers.

TITLE: CONTAINMENT PRESSURE CONTROL: ICE CONDENSERS

Functional Requirement: Level 3.1.2 Control Energy in Containment

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Pressurized-Water Reactor (PWR)

Organization: Vendor (Westinghouse)

Examples of Implementation: Ice condensers have been used in a number of Westinghouse-designed PWR reactors, such as the two-unit, Tennessee Valley Authority (TVA) Sequoyah Nuclear Power Plant. Each of these units is rated at 3595 MW(t).

Description: An ice condenser can be used to suppress containment overpressure after a postulated loss-of-coolant accident (LOCA). Its primary function is to rapidly cool and condense the steam-air mixture, thus acting as a passive heat sink to absorb the released energy. This heat sink, located inside the primary containment, consists of a suitable quantity of borated ice in a cold storage compartment.

In a typical application using an ice condenser[1,2] the containment building is divided into three volumes: a lower volume that houses the reactor and the reactor coolant system, an intermediate volume housing the ice condenser, and an upper volume that accommodates the air displaced from the other two volumes during an LOCA (see Fig. 1).

The ice bed, which is the heat sink of the ice condenser, is a completely enclosed compartment located around the perimeter of the upper compartment of the containment. It penetrates the operating deck so that a portion extends into the lower compartment of the containment (see Fig. 2). The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment that are designed to remain closed during normal plant operation . At the top of the ice condenser are another set of doors exposed to the atmosphere of the upper compartment, which remain closed during normal plant operation.

The ice is held within the ice condenser in baskets that are arranged to promote heat transfer from the steam to the ice. A refrigeration system maintains the ice several degrees below the freezing temperature, and insulation surrounding both ice baskets and refrigeration ducts serves to minimize the heat conduction through the ice condenser enclosure.

In the event of an LOCA or a secondary-steam system rupture, the door panels located below the operating deck open almost immediately due to the increased pressure in the lower compartment caused by the escaping steam. The open doors allow the air and steam to flow from the lower compartment into the ice condenser. The resulting buildup of pressure in the ice condenser causes the door panels at the top of the condenser to open and allows the air to flow out of the condenser into the upper compartment. The steam is condensed in the ice condenser compartment, thus limiting the peak pressure in the containment. Condensation of steam within the ice condenser results in a flow of steam from the lower compartment to the condensing surface of the ice, thus reducing the time that the containment is at an elevated pressure. The dividing barrier separates upper and lower compartments and ensures that the steam is directed into the bottom of the ice condenser.

The ice is made in ice machines that are installed in an auxiliary building. It is generated in flake form for ease in handling and is moved into the containment vessel through a normally closed penetration by a pneumatic conveying system. The system feeds ice to a loading head, which is positioned by the bridge crane, through removable tubes in the plenum. The ice condenser design parameters for TVA's Watts Bar nuclear power station (now under construction) are shown in Table 1.

Alternative Versions: None

Status of Technology: Ice condensers are installed in many Westinghouse-designed PWRs, including Units 1 and 2 at TVA's Sequoyah Nuclear plant, which began operation in 1980.

Advantages:

1. Containment overpressures due to an LOCA are reduced within a few minutes to a few psi, thus reducing the probability of fission product release from containment.
2. The ice condenser performance is only slightly sensitive to blowdown rate and blowdown energy within the ranges of interest.
3. Large reductions in heat transfer surface area of the ice do not significantly affect ice condenser performance.
4. Using ice as a heat sink instead of water is more mass-efficient because melting ice requires an extra 79.7 cal/g.
5. The ice condenser is a static device, does not require any power source, and its operation during an accident does not depend upon the functioning of any other systems.

Additional Requirements: None

Comments: None

References/Contacts:

1. Preliminary Safety Analysis Report, Sequoyah Nuclear Plant, Tennessee Valley Authority, Docket-50327--1, Volume 1 (1968).

2. Preliminary Safety Analysis Report, Watts Bar Nuclear Plant, Tennessee Valley Authority, Docket-50390--3, Volume 2 (1971).

Update/Compiler: Feb. 1989/RLP

Table 1. Ice Condenser Design Parameters for Westinghouse Ice Containment

MSSS power, MWt	3596
Design Energy Release to Containment	
Initial blowdown mass release, lb	516,000
Initial blowdown energy release, Btu	351.7×10^6
Allowance for undefined energy release in addition to core residual heat, Btu	50×10^6
Reactor Containment Volume (net free volume)	
Upper Compartment, ft^3	716,000
Ice Condenser, ft^3	126,940
Lower Compartment (active), ft^3	301,250
Total Active Volume, ft^3	1.14419×10^6
Lower Compartment (dead-ended), ft^3	98,930
Total Containment Volume, ft^3	1.243×10^6
Reactor Containment Air Compression Ratio*	1.43
Ice Condenser parameters	
Weight of ice in condenser, lb	2.45×10^6
Dimensions of ice condenser	
O.D., ft	115
I.D., ft	89
Average length, ft.	267
Depth (less insulation panels), ft	11
Ice bed height, ft	48
Inlet door flow area, ft^2	1000
Ice condenser flow area, ft^2	1326
Ice condenser inlet door opening pressure, lb/ft^2	1.0
Ice boron concentration, ppm boron	2000
Refrigeration cooling capacity	
Installed cooling capacity for compartment, tons	50
Maximum compartment heat gain, tons	35
Total cooling capacity for plant, tons (Capacity shared by two units)	150

*Defined as (Total Active Volume)÷(Upper Compartment Volume plus 0.645 x Ice Condenser Volume).

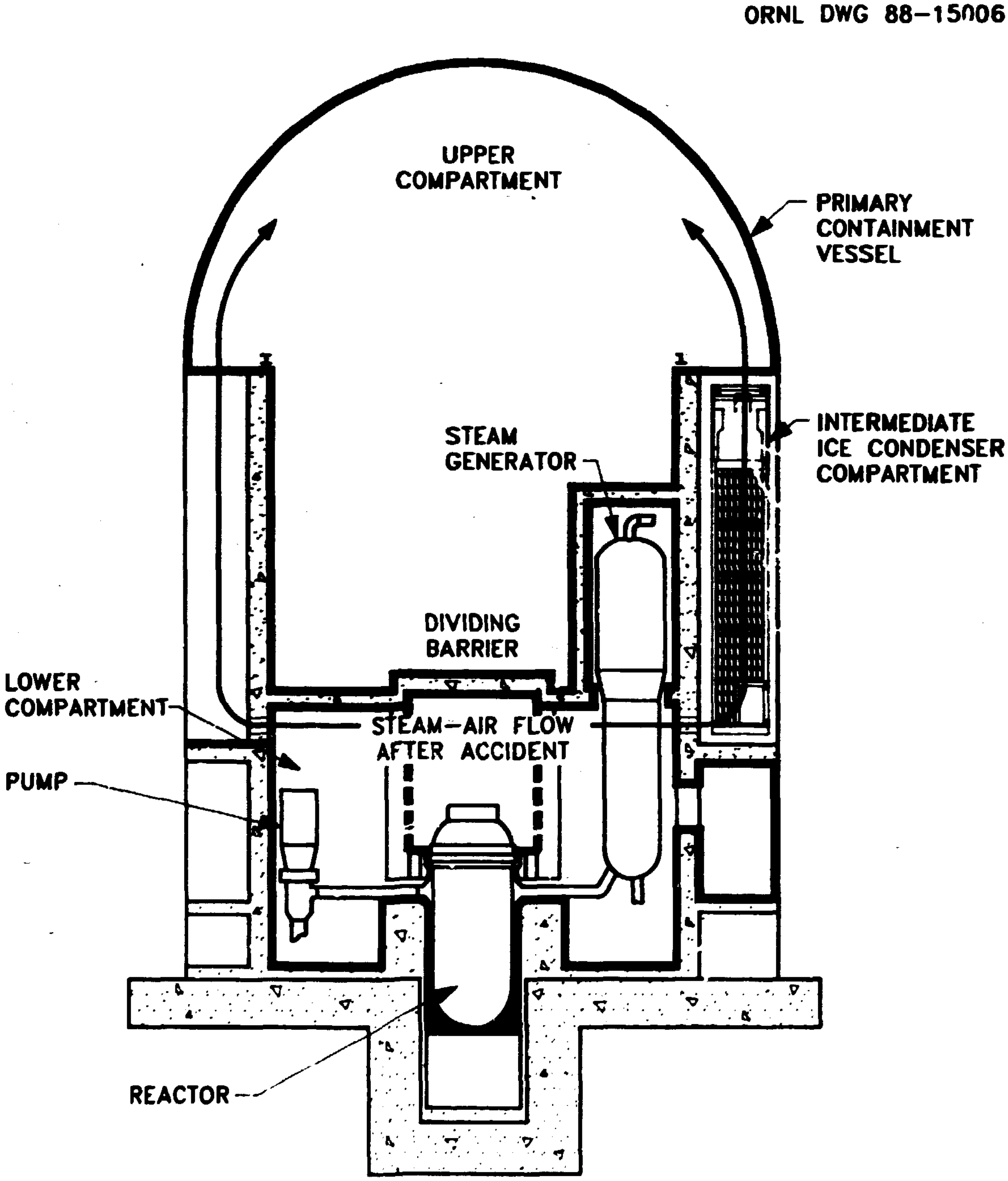

Fig. 1. Interior design of primary containment vessel.

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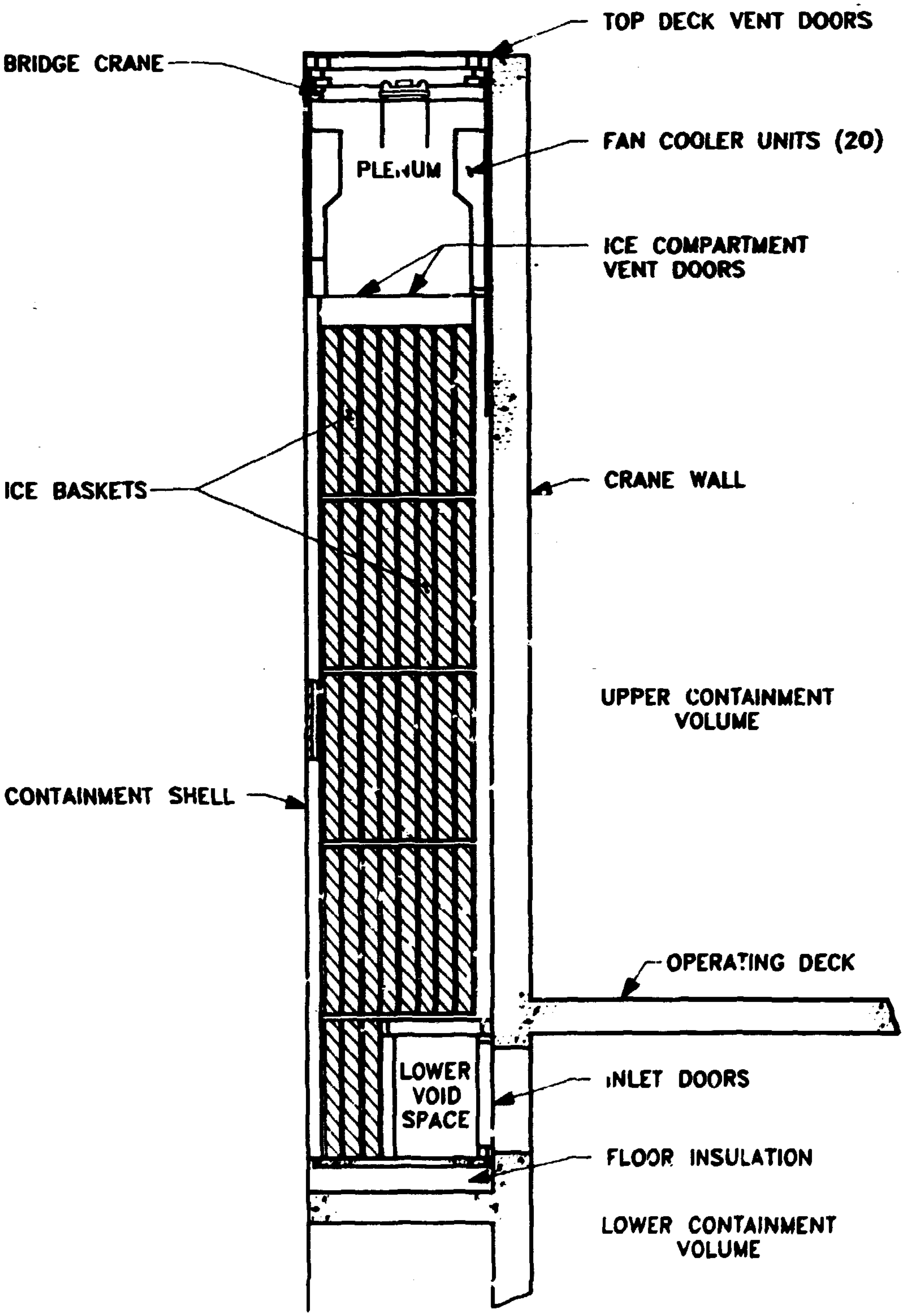

Fig. 2. General arrangement of ice condenser.

TITLE: CONTAINMENT COOLING WITH EXISTING AND AUGMENTED HEAT CAPACITY OF CONTAINMENT

Functional Requirements: 3.1.2 Control Energy in Containment

Safety Type: Inherent

Developmental Status 1. Commercial
2. Demonstrated (See comments)

Reactor Type: Light-water reactor (LWR)

Organization: Vendor (B&W) and National Laboratory (Oak Ridge National Laboratory)

Examples of Implementation: All LWRs

Description: In a postulated reactor accident, such as a loss-of-coolant accident (LOCA), large quantities of steam and other gases are generated that may threaten containment integrity by overpressurization of containment. To protect the reactor containment, heat must be removed from the containment atmosphere to condense steam and cool inert gases.

Containment atmosphere cooling can be done by transferring heat from containment to the environment (multiple methods are described in this report) or by using the heat capacity of the equipment and containment structures to adsorb heat.[1,2] All reactor containment systems use both approaches to cool containment gases.

Radioactive decay heat decreases rapidly with time; thus, if the heat capacity of the reactor structure can absorb heat quickly, the size of the containment cooling system needed to dump heat to the environment can be small. At sufficiently low rates of decay heat removal from containment to environment, the heat can be conducted through containment walls. For a typical 1000 MW(e) pressurized-water reactor (PWR), the heat capacity of 20% of the power plant concrete, if heated to 50°C, equals 800 MWh. This is sufficient heat capacity to absorb total nuclear decay heat for hours after a postulated reactor accident.

The actual heat absorbed by the heat capacity of the facility depends upon the particular facility design. In many cases, much of the containment heat capacity is not available. The containment is composed of many reinforced concrete walls from 1 to 8 ft in thickness. The rate of heat conduction into these walls is slow. It takes from 2 to 140 h (see Table 1) for the center of these walls to reach half the temperature rise of that on the surfaces of the walls. In practical cases, heat transfer, not heat capacity, limits the use of building heat capacity to absorb heat.

Alternative Versions: A number of design innovations have been proposed to better utilize the heat capacity of the facility by improving heat transfer from containment atmosphere to containment structures. Two examples are described herein:

1. Improved Heat Transfer with Mass Concrete

   Heat transfer between containment atmosphere and building structures can be improved as shown in Fig. 1. The design innovation consists of (1) replacing the vertical steel rebar in the concrete walls with steel pipe, (2) connecting the tops of the pipe with pipes going to the surface of the concrete walls, and (3) connecting the bottoms of the pipe with pipes going to the surface of the concrete walls[3,4].

The steel pipe provides the structural strength in the place of the rebar. In the case of an accident, hot air/steam would (1) enter the tops of the pipes, (2) be cooled by the cold concrete, and (3) exit as an air/water mixture from the bottom openings in the pipe. In effect, there would be natural circulation flow of gases through the tubing, with hot gases entering at the top and cool gases leaving at the bottom.

With this approach, all walls appear to be thin with respect to heat conduction. Instead of heat being conducted through 1 to 4 ft of concrete to reach the center of the wall, it needs to be conducted through less than a foot of concrete. This results in rapid cooling of containment air by heating of the concrete.

There is some relevant experience in a related field that uses a somewhat similar technique. In building very large dams such as Glen Canyon Dam (concrete > 15-m thick), it is necessary to control concrete temperatures while the concrete cures to obtain the desired strength. This is done by embedding piping in the concrete and pumping cold water through the tubing to control the temperature as the concrete cures. The tubing acts as both a short-term cooling coil and a long-term rebar in the concrete.

2. Pebble Bed

For some advanced reactors, it has been proposed to include gravel beds within containment to condense steam after an accident.[5] Figure 2 shows the Babcock and Wilcox B-600 PWR conceptual design. In this design, gravel beds surround the primary reactor system. Gravel size is chosen to maximize heat transfer and allow rapid condensation of any escaping steam. The gravel beds can serve multiple purposes: condense steam, trap radionuclides, and provide radiation shielding of the primary system.

Advantages:
1. Inherent safety found in all reactors.
2. Multiple approaches exist to upgrade the capabilities of building heat capacity to cool containment air.

Additional Requirements: None

Comments: Several studies have been done on the heat capacity effects of building structures on containment temperature control. Only limited work has been done on how to further augment this phenomena. For small reactors, available heat capacity is sufficient to cool containment. As the reactor becomes larger in size, this is no longer true because internal containment surface areas for heat transfer are less per unit of power output for larger reactors.

References/Contacts:

1. I. Haarala and T. Kukkola, "Passive Safety Features in VVER-440 Type Plant Design for Finland," IAEA Technical Committee Meeting on Passive Safety Features in Current and Future Water-Cooled Reactors, Moscow, USSR, March 21-24, 1989.

2. P. S. Ayyaswamy, J. N. Chung, and K. K. Niyogi, "Reactor Containment Heat Removal by Passive Heat Sinks Following A Loss-of-Coolant Accident," Nuclear Technology, 33, p. 243 (May 1977).

3. C. W. Forsberg, "Report of Possible Invention or Discovery: Passive Short-Term Cooling for Nuclear Reactor Containment," Martin Marietta Energy Systems, Inc.; Oak Ridge National Laboratory (Oct. 8, 1987).

4. U.S. Bureau of Reclamation, Design of Arch Dams: Design Manual for Concrete Arch Dams, U.S. Department of the Interior, 1977.

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

6. G. E. Kulynych and J. E. Lemon, "Babcock and Wilcox Advanced PWR Development," Nuclear Europe, 4, p. 17 (1986).

Update/Compiler: May 1989/CWF

Table 1. Time for concrete centerline wall temperature to increase by half that of the surface wall temperature[a, b] after instantaneous surface temperature rise

Thickness of wall (ft)	Time to respond (hr)
1	2.1
2	8.5
3	19.1
4	33.9
5	53.0
6	76.3
7	104.0
8	136.0

[a]Physical properties: $Cp = 0.156$ Btu/lb°F; $\rho = 150$ lb/ft^3; $k = 1.05$ Btu/hr-ft°F.
[b]Equation: Hours = 2.12 x (wall thickness in ft)^2.

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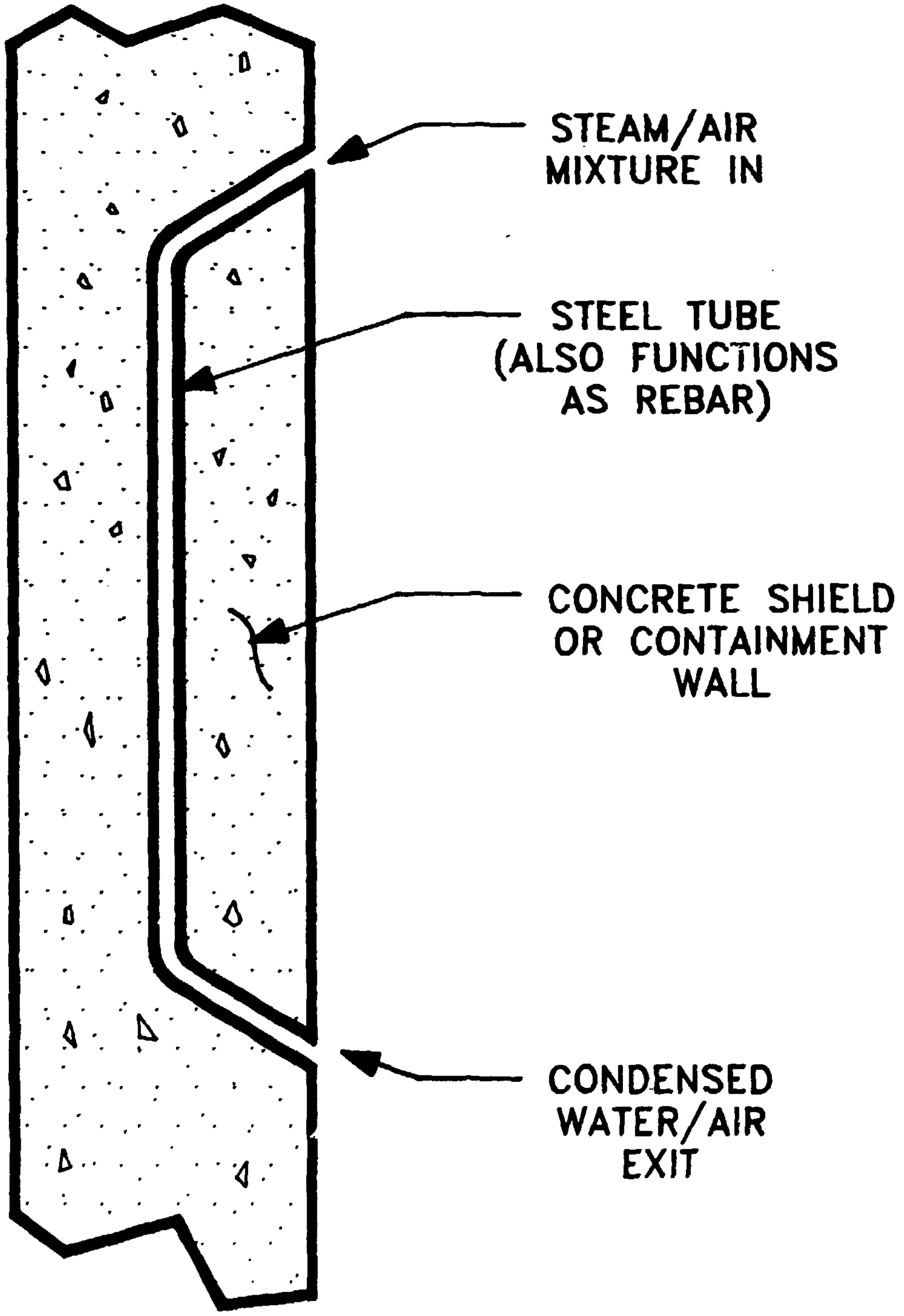

Fig. 1. Shield/containment wall heat capacity cooling system.

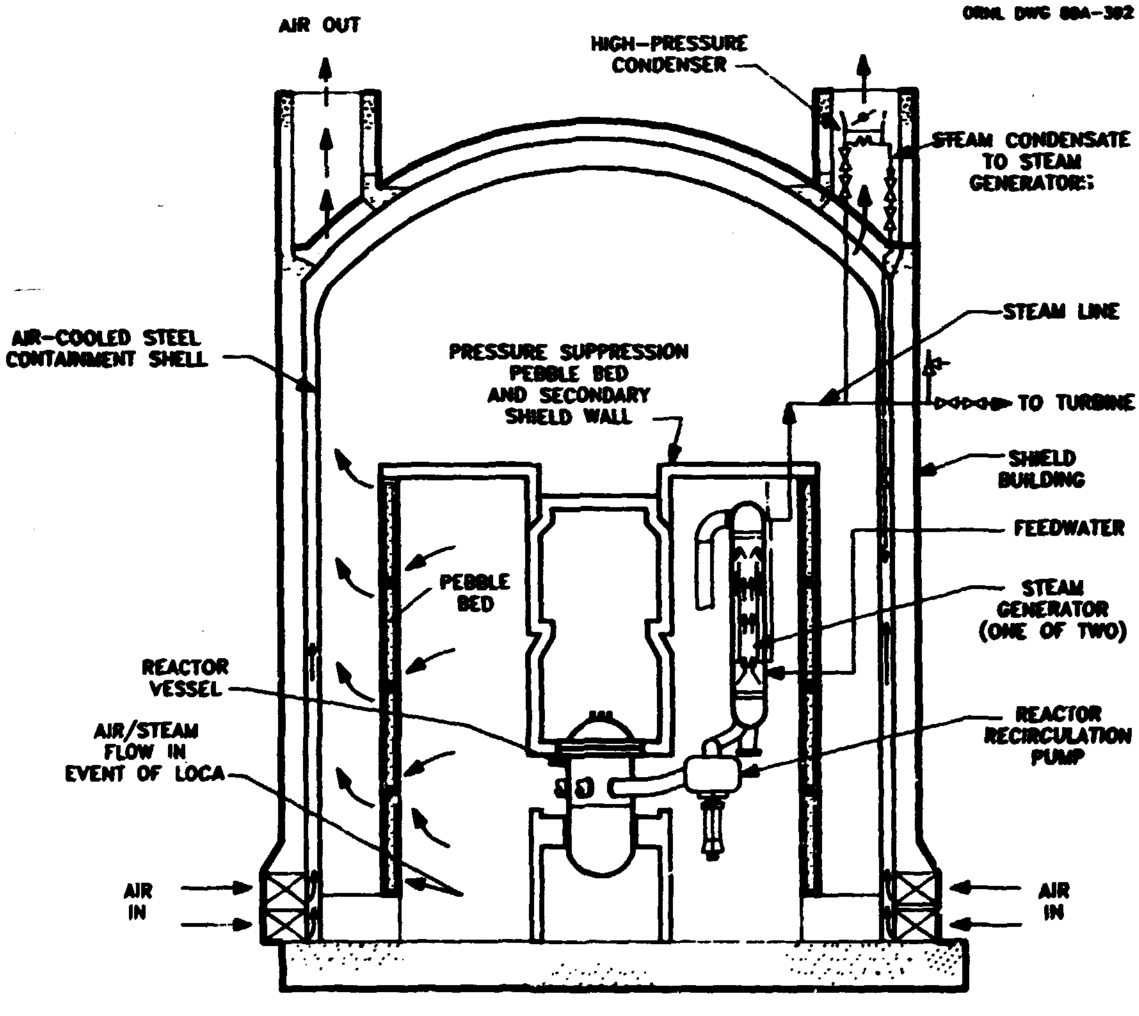

Fig. 2. Schematic of B-600 containment safety systems.

TITLE: HEAT PIPES FOR REACTOR CONTAINMENT COOLING OR REACTOR CORE DECAY HEAT REMOVAL

Functional Requirements: Level 1.2.3 Transport Heat to Ultimate Heat Sink
Level 3.1.2 Control Energy in Containment

Safety Type: Passive

Developmental Status: 2 Demonstrated

Reactor Type: Light -Water Reactor (LWR)

Organization: University (University of California)

Examples of Implementation: None

Description: Heat pipes are devices that transfer heat from one portion of a sealed pipe to another portion by a process of evaporation and condensation using a circulating fluid (See Fig. 1a).[1] The phase change from liquid to gas in the evaporator section of a heat pipe absorbs heat. The gas inside the pipe flows from the evaporator zone to the condenser zone because of the difference in the gas pressures. The heat is then rejected as the gas is condensed back to a liquid in the condenser section of the heat pipe. In order not to limit the rate of heat transfer, the liquid is returned from the condenser section to the evaporator section of the heat pipe by capillary forces through a porous wick and/or by gravity assistance. When the heat sink is parallel or below the heat source, wick action is used to move the liquid from the condenser to the evaporator section (see Fig. 1b). The preferred type of wick is a screen-covered surface with small capillary grooves cut into the wall. Gravity-assisted heat pipes, which have the condenser above the evaporator, can be used whenever the heat source is at or below the heat sink, as shown in Fig. 1c. Heat transfer is improved by placing fins on the evaporator and condenser surfaces.

For reactor containment cooling, the evaporator section of the heat pipe must be exposed to the containment atmosphere. There are three possible heat sinks for the condenser section of a heat pipe: (1) the earth,[2,3] (2) water (see SSC description: Cooling Ponds), and (3) the atmosphere as described herein. A water sink requires the smallest surface area for the heat pipe condenser section; the earth requires the largest surface area for the heat pipe.

Conceptual designs of reactor containments have been developed using heat pipes for post-accident containment cooling (see Fig. 2).[4] Table 1 summarizes one of these designs. Heat pipe technology may also be used to remove decay heat from the reactor core or any other source of heat inside the reactor containment.

Alternative Versions: Heat pipes can be designed to remove heat only above a set temperature. This is important in LWRs located in colder climates where it is undesirable to remove containment heat in the winter during normal operation of the plant. The principle of operation is as follows: the heat pipes are loaded with a quantity of gas that will not condense at the normal operating temperature. This noncondensable gas is swept into the condenser section of the heat pipe where it forms a gas plug. This gas plug represents a diffusion barrier to the flowing vapor and shuts off the portion of the condenser that it fills. When the vapor temperature and the corresponding vapor pressure are increased, such as would occur in a reactor accident, the noncondensable gas is compressed into a smaller volume. Under such circumstances, more of the condenser surface becomes available for heat transfer.

Heat pipes can be a transport mechanism to an ultimate heat sink for most types of emergency core cooling systems. They are a generic heat removal system where the evaporator zone of the heat pipe can be incorporated into any heat exchanger.

Status of Technology: Heat pipes have been used for many industrial applications. For example, they have been used for over a decade as part of the vertical support system of the Trans-Alaska Pipeline. In this application, they were designed to transfer heat from the tundra to the aboveground portion of the pipeline. Heat pipes have not been used for LWR applications.

Advantages:

1. Heat pipes can be designed to begin removing heat at a preset temperature and to operate under specific containment conditions. They can be designed to remove a ~ accidental heat load.

2. Heat pipes have no moving parts and require no external power for operation.

3. Heat transfer systems using heat pipes require two breaks before a failed heat pipe would provide a route for escape of containment gases into the environment.

4. The technology for construction of the heat pipes needed for this application is available.

Additional Requirements: None

Comments: None

References/Contacts:

1. S. W. Chi, Heat Pipe Theory and Practice, a Sourcebook, McGraw Hill Book Company, New York (1976).

2. V. C. Mei, Horizontal Ground-Coil Heat Exchanger Theoretical and Experimental Analysis, ORNL/CON-193, Oak Ridge National Laboratory, Oak Ridge, Tennessee (December 1986).

3. L. H. Goldish and R. A. Simonelli, Ground-Source and Hydronic Heat Pump Market Study, Electric Power Research Institute, EM-6062, November, 1988

4. A. A. Ahmad, I. Catton, et al., PWR Severe Accident Delineation and Assessment, NUREG/CR-2666 (UCLA-ENG-8284), University of California, Los Angeles, California (January 1983).

Update/Compiler: Dec. 1988/RLP

Table 1. Conceptual design of a heat-pipe-cooled containment building

Parameter	Value
Reactor type	PWR
Reactor power	3000 MW(t)
Total heat rejection rate	22.5 MW(t) (0.75% of full power)
Heat sink	Air
External containment conditions	
Ambient air temperature	25°C
Wind velocity	3 m/s
Internal containment conditions	
Temperature	136°C (47 psi steam)
Atmosphere	
Air	20%
Water (steam)	80%
Total heat pipe containment penetration area	1 m^2
Number of heat pipes	100
Design	See Fig. 1c
Heat pipe characteristics	
Fluid	Water plus noncondensible gas
Vapor temperature	100°C
Material of construction	Austenitic steel (Type 304 SS)
Pipe dimensions	
Inside diameter	11.3 cm
Wall thickness	3 mm
Evaporator Length	13.3 m
Condenser length	11.35 m
Evaporation fin characteristics	
Number of fins	36
Material of construction	Aluminum
Rectangular fin dimensions	
Length	13.3 m
Height	3.73 cm
Thickness	5 mm
Condenser fin characteristics	
Fin Frequency	1.5 fins/cm
Material of construction	Aluminum
Annular fin dimensions	
Outside diameter	34 cm
Inside diameter	11.9 cm
Thickness	2.5 mm

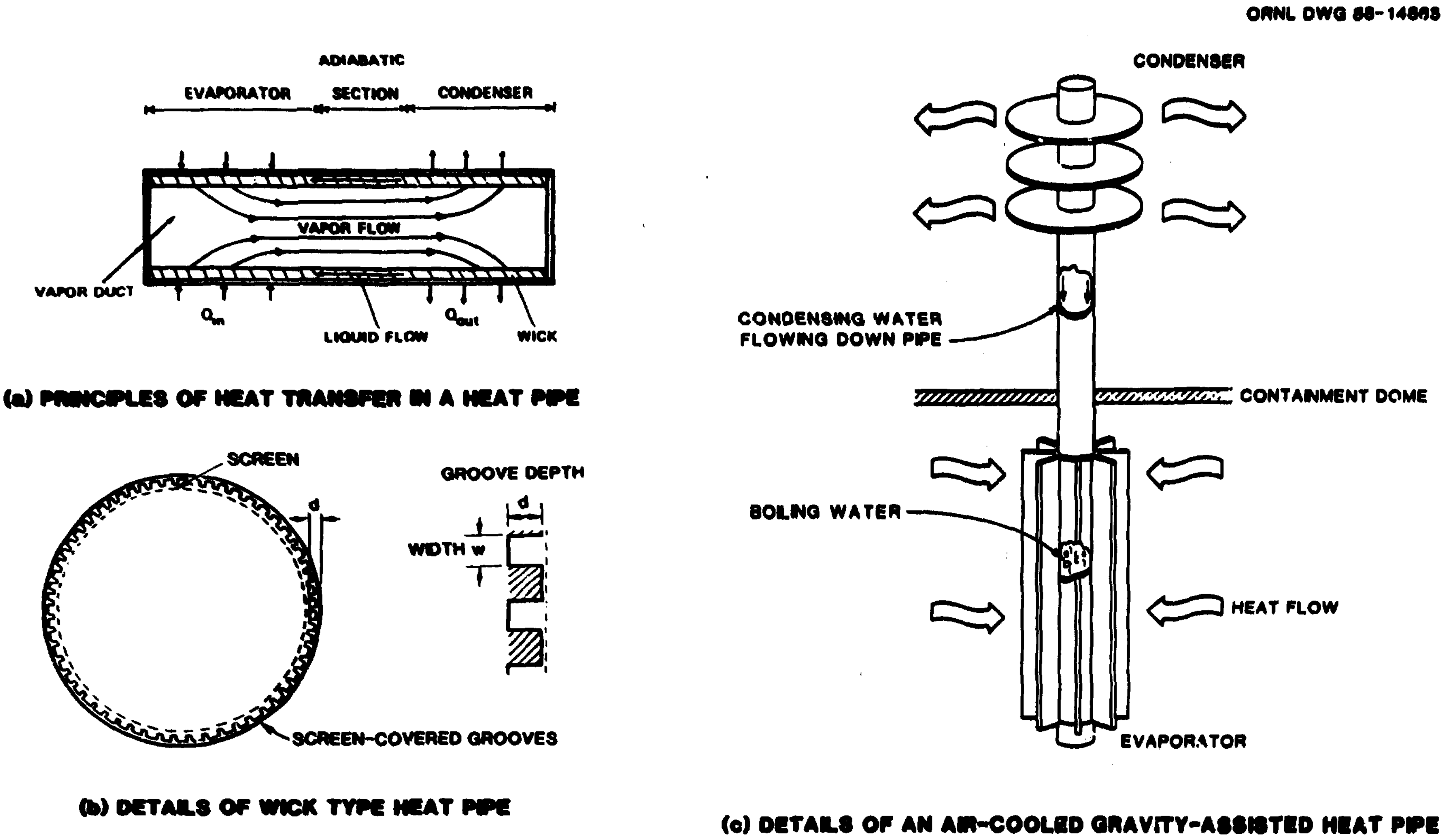

Fig. 1. Heat pipe configuration.

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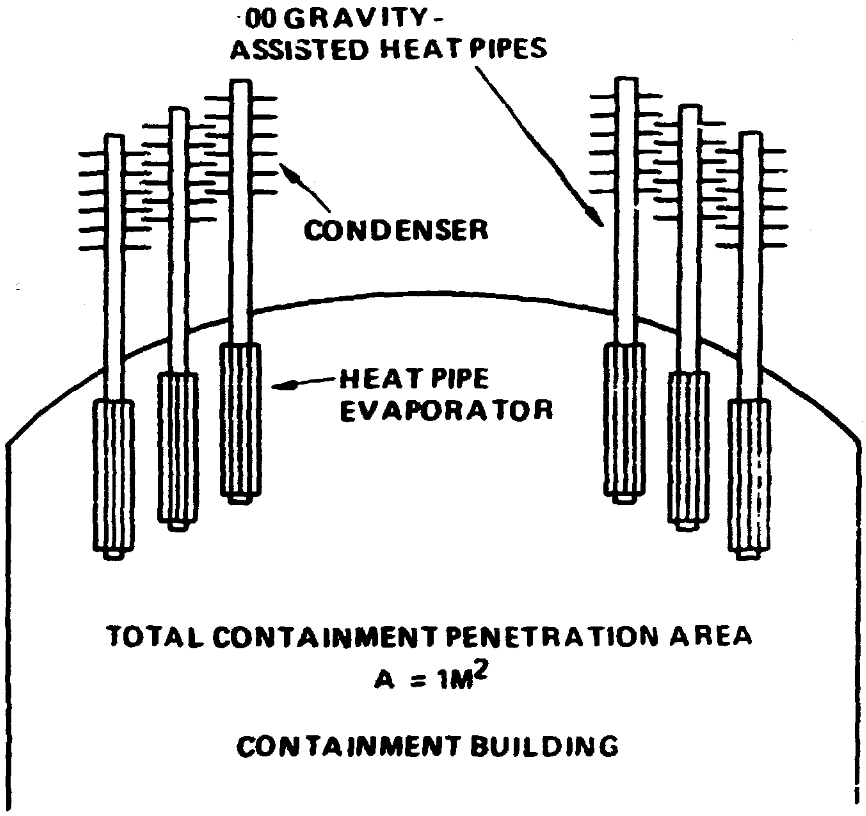

Fig. 2. Overall schematic of a LWR passive heat removal system based on air-cooled gravity assisted heat pipes.

TITLE: HIGH-HEAT CONDUCTION REACTOR CONTAINMENT FOR CONTAINMENT COOLING

Functional Requirement: Level 3.1.2: Control Energy in Containment

Safety Type: Passive (See comments)

Developmental Status: 1. Commercial (See comments)
2. Demonstrated (See comments)

Reactor Type: Light-water Reactor (LWR)

Organization: Vendor (Multiple)

Examples of Implementation: None

Description: A high heat conduction reactor containment cooling system can provide an ultimate heat sink to remove heat from containment after a postulated reactor accident. This helps assure containment integrity by lowering containment temperatures and by reducing containment pressure. Containment pressure reduction occurs by both condensation of steam and cooling of noncondensable gases.

All containments lose heat by heat conduction from the inside of the containment to the atmosphere. For smaller reactors (several hundred megawatts), this is sufficient to cool the containment after a postulated reactor accident. As power reactor size increases, the surface area of the containment for typical designs does not increase as rapidly as the reactor power levels. For larger reactors, either other methods of reactor containment cooling must be used or the reactor containment design must be modified to maximize heat transfer from containment to environment. Several methods to enhance containment cooling by heat conduction through the containment have been proposed.

One passive method to enhance containment cooling is to fabricate the containment structure out of a material with a high thermal conductivity such as a steel. The use of a metallic containment structure with natural convection air cooling has been considered for various reactor designs[1]. Some early reactors[2] also had containments where significant quantities of heat were removed by natural air circulation over a steel containment. For example, Indian Point 1 has a spherical steel containment with an external concrete biological shield. This annulus is vented to a stack with good natural draft cooling.

A second method for use with concrete containments is to take advantage of the variation in the thermal conductivity of concrete with composition to build a containment structure with higher thermal conductivity. Improvements in the thermal properties of concrete are possible by the use of metallic materials and other special aggregates.

An example of a passive containment cooling system is shown in figure 1. Air inlet vents near the top of the shield building direct air around the baffles and over the metal containment to transfer the heat from the reactor containment to the environment as shown in figure 1. The system is designed to operate by natural convective flow of the air.

Early in an accident, the heat load on the containment will be high. This heat load decreases with time. Short-term containment cooling can be increased with a water storage tank placed over the reactor containment vessel at the top of the shield building with a gravity feed system that directs cooling water over the steel containment structure. The water spray provides additional steel containment shell cooling by evaporation of water on the steel containment shell lowering its temperature early in a postulated accident when heat loads are particularly high. As the reactor decay heat decreases, the water flow is reduced using a passive device in the water storage tank that takes advantage of the water level drop in the tank. Various techniques exist to activate such sprinkler systems.

Alternative Version: Many options.

Status of Technology: Heat removal from containment by conduction is an SSC that exists for all reactor containments. Work being done at Westinghouse and elsewhere is developing methods to increase this mode of heat removal so it meets total containment cooling requirements for larger reactors.

Advantages:
1. Heat conduction is passive in operation.
2. Containment overpressure can be passively reduced.

Additional Requirements: None

Comments: For small reactors, heat conduction out of containment is an inherent safety feature. For larger reactors proper air flow by the containment is needed to remove the larger quantities of heat. The need for air flow channels makes the technology a passive technology for larger reactors. For very large reactors, insufficient surface area exists to passively cool the containment with this approach. Cooling can be augmented with passive heat pipes or other devices.

References:

1. R. Adinolfi, L. Noviello, "Passive Systems Considerations for Light Water Reactors," International Atomic Energy Agency, Technical Committee Meeting on Passive Safety Features in Current and Future Water-Cooled Reactors, Moscow, USSR, March 21-24, 1989.

2. Consolidated Edison, Preliminary Safety Analysis Report for Indian Point I, Docket 50-3

3. P. S. Ayyaswamy, J. N. Chung, K. K. Niyogi, "Reactor Containment Heat Removal by Passive Heat Sinks Following a Loss-of-Coolant Accident," Nuclear Technology, Vol. 33, Pg. 243-247, May 1977.

Update/Compiler: May 1989/WJR

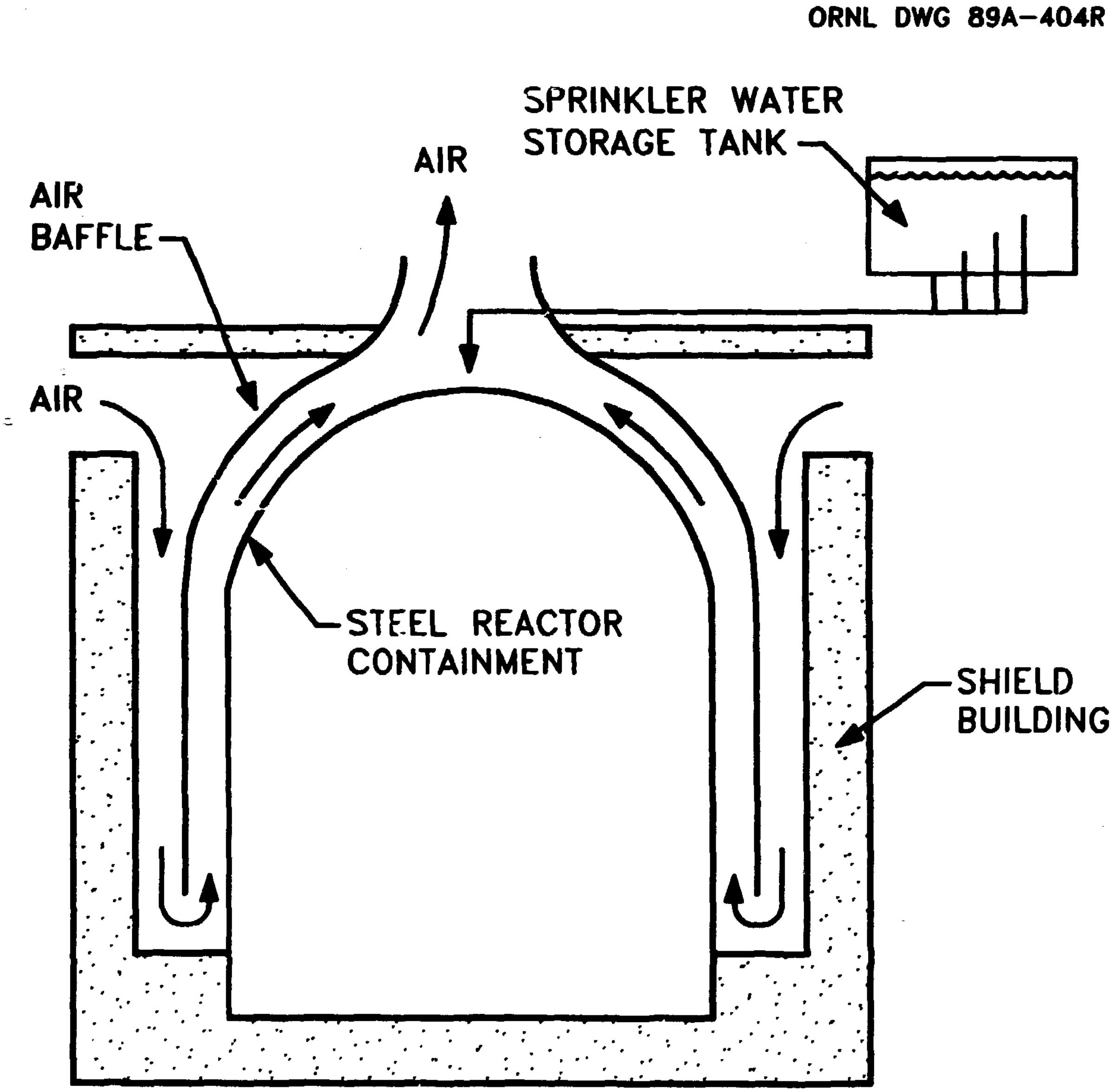

Fig. 1. Passive containment cooling.

CHAPTER 12

Structures, Systems and Components to Trap Radionuclides

Function 3.2

Table 12.1 Structures, Systems and Components to Trap Radionuclides: Function 3.2

Description Title	Function	Location of Description	Page
Metal Pressure Vessel in Pool	3.2	Chapter 6	6-38
Cooling Ponds	3.2	Chapter 6	6-75
Vacuum Containment	3.2	Chapter 11	11-19
Passive Hard Vacuum Containment System	3.2	Chapter 11	11-24
Containment Pressure Control-Suppression Pools and Bubbler Condensers	3.2	Chapter 11	11-29
Water-Coolable Core Catcher	3.2	Chapter 11	11-44
Containment Pressure Control by Pressure Activated Water Sprinklers	3.2	Chapter 11	11-49
Temperature Activated Water Sprinklers for Steam Condensation and Accident Mitigation	3.2	Chapter 11	11-54
Water/Steam in Post-Accident Environment	3.2	Chapter 12	12-3
In-Containment Post-Accident Water Collection	3.2	Chapter 12	12-5
Core Melt Source Term Reduction System (COMSORS)	3.2	Chapter 12	12-7

TITLE: WATER/STEAM IN POST-ACCIDENT ENVIRONMENT

Functional Requirements: 3.2 Trap Radionuclides

Safety Type: Inherent

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Multiple

Examples of Implementation: All LWRs

Description: In a postulated reactor accident, the release of radionuclides from the core to containment depends upon the local environment. In a water-cooled reactor, the post-accident environment will contain steam and water. The accident at Three Mile Island (TMI) and experiments since then indicate that the presence of water in a post-accident environment reduces the release of all radionuclides except noble gases by several orders of magnitude compared to expected releases of radionuclides in a postulated accident without water. The accompanying references summarize these observations. The mechanisms responsible for this effect include dissolution of particular radionuclides (such as iodine) in water, sedimentation aided by growth of hygroscopic particles, and chemical reactions aided by the presence of water.

The importance of this inherent safety mechanism was not fully recognized until after the TMI accident, where the radionuclide releases to the environment from the accident were, for some radionuclides, orders of magnitude lower than assumed for some radionuclides, and the radiation levels in containment were lower than expected for such an accident.

Alternative Versions: This inherent safety mechanism is a characteristic of LWRs. Certain design features such as sprinklers in containment can (see: Temperature Activated Water Sprinklers for Steam Condensation and Accident Mitigation) augment this inherent safety characteristic. Traditionally, devices such as sprinklers in containment functioned to condense steam and reduce containment pressures. Design changes to augment this safety characteristic are possible.

Status of Technology: Commercial

Advantages: Inherent characteristic of LWR

Additional Requirements: None

Comments: Water in containment reduces the source term; but, if there is insufficient containment cooling, the water is converted to steam and can over pressurize the containment.

References/Contacts:

1. F. J. Rahn, The LWR Aerosol Containment Experiments (LACE) Project, Electric Power Research Institute, NP-6094D, November, 1988.

2. Ame.ican Nuclear Society, "Transactions: International Conference on Nuclear Fission: Fifty Years of Progress in Energy Security and Topical Meeting on TMI - 2 Accident: Materials Behavior and Plant Recovery Technology," 57, Oct. 30 - Nov. 4, 1988, Washington, D.C.

3. M. F. Osborne, J. L. Collins, and R. A. Lorenz, "Experimental Studies of Fission Product Release from Commercial Light-Water Reactor Fuel Under Accident Conditions," Nucl. Techol., 78, 157 (Aug. 1987).

4. J. Jokiniemi, "The Growth of Hygroscopic Particles During Severe Core Melt Accidents," Nucl. Techol., 83, 16 (Oct. 1988).

5. M. F. Albert, J. S. Watson, and R. P. Wichner, "The Adsorption of Gaseous Iodine by Water Droplets," Nucl. Technol., 77, 161 (May 1987).

6. A. W. Cronenberg and D. J. Osetek, "Reaction Kinetics of Iodine and Cesium in Steam/Hydrogen Mixtures," Nucl. Techno'., 81, 347 (June 1988).

7. R. D. Spence and A. L. Wright, "The Importance of Fission Production/Aerosol Interactions in Reactor Accident Calculations," Nucl. Technol., 77, 150 (May 1987).

8. Nucl. Technol. (Entire issue), 53, (May 1981).

Update/Compiler: May 1989/CWF

TITLE: IN-CONTAINMENT POST-ACCIDENT WATER COLLECTION

Functional Requirements: 3.1.2 Control Energy in Containment
3.2 Trap Radionuclides
1.2.2 Maintain Core Coolant Makeup

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Vendors

Examples of Implementation: This system exists in many reactors. In many cases, it was not designed as a safety system but later was recognized as a desirable design feature.

Description: A containment post-accident water collection system is designed to collect water that is in containment after an accident and is used for emergency core cooling and accident mitigation. In a postulated LWR accident, water and steam would fill the containment. Much of the steam would condense to water due to condensation on cool surfaces and from in-containment air coolers. Emergency systems such as sprinklers and emergency core-cooling systems may add additional water to containment.

With appropriate design, this water can

1. Provide a water source for emergency core-cooling systems.

2. Flood the area below the reactor vessel. If a core meltdown occurs, the pool of water will quench the molten core material flowing from the reactor vessel and reduce the release of fission products to the containment atmosphere.

The design of such a system consists of several sets of components:

- Sloping floors, floor trenches, and floor drains divert water on the floors anywhere in containment to a central holdup drain tank.

- The holdup drain tank is a relatively small tank which is normally empty. It is used to collect water from normal spills and leaks during operations. Water from the tank can be pumped to water cleanup systems. The tank size is a small fraction of the water volume expected in containment in a post-accident environment. In the event of an accident, water flows to the holdup tank, which quickly fills and overflows to a larger water storage tank.

- The larger water storage tank is a multipurpose tank. In a pressurized-water reactor (PWR), it may be the in-containment refueling water storage tank that is used as a water source for emergency core-cooling systems. In a boiling-water reactor (BWR), this tank may be the suppression pool in a design such as the Mark III containment (see: Reactor Containment Systems). The tank is normally filled with water.

In a post-accident environment, water from the holdup drain tank refills the storage tank if it is being emptied to provide water for emergency core-cooling systems or other purposes. If tank water is not being used, the storage tank overflows and floods the reactor cavity under the pressure vessel.

- With waterflow from the large water storage tank, the reactor cavity is flooded in a post-accident environment. This pool of water quenches any molten core debris from the reactor vessel. (see: Water-Coolable Core Catcher). Note: The system allows for a normally dry reactor cavity during regular operations.

Alternative Versions: Many design options

Status of Technology: Commercial

Advantages:
1. Provides passive method to flood under core location after extreme accident conditions.
2. Very low cost, passive safety option.
3. Provides source of water for other emergency systems.

Additional Requirements: None

Comments: None

References/Contacts:

1. Electric Power Research Institute, Advanced Light-water Reactor Requirements Document (Draft).

Update/Compiler: July 1989/CWF

TITLE: CORE MELT SOURCE REDUCTION SYSTEM (COMSORS)

Functional Requirements: Level 3.2 Trap Radionuclides

Safety Type: Inherent

Development Status: 2 (5, See Comments) Demonstrated

Reactor Type: Light-Water Reactor (LWR)

Organization: National Laboratory (Multiple)

Examples of Implementation: See Comments

Description: Core Melt Source Reduction System (COMSORS) refers to a set of concepts to limit maximum release of aerosols and gases to containment from a reactor core melt accident. If a reactor core meltdown occurs, the molten core material will eventually contact and begin to melt the concrete foundation structure. The chemical reactions and molten core/concrete temperatures will determine the rate and quantities of radioactive gases and aerosols generated by the core/concrete interactions and released to containment. The generation of gases can pressurize the containment and increase the potential for containment failure. If containment fails, the quantities of radioactive aerosols and gases determine the maximum accident potential. If the containment does not fail, large quantities of radioactive aerosols and gases in containment will not only slow efforts to stop an accident, but also cleanup after an accident. Use of a COMSORS, either as a separate engineered device or by selection of appropriate aggregate in the concrete, may allow the creation of a method to limit the maximum possible source term (radioactive gases and aerosols) by incorporation of molten core and other materials into a stable high-level waste (HLW) matrix. This concept is based on two sets of experimental observations.

1. The U.S. Nuclear Regulatory Commission, its contractors, and others[1-4] have been investigating the physical and chemical mechanisms of a reactor core meltdown. This has included experiments where molten core materials have been poured onto various types of concrete used in nuclear power plants under the reactor core. The experimental studies show that the quantities of radioactive gases and aerosols generated and released by the molten core/concrete interactions vary widely depending upon concrete chemistry (see Fig. 1). For example, concrete containing limestone aggregate causes high rates of radioactive gas and aerosol generation because the limestone decomposes at high temperatures and releases carbon dioxide gases. This gas generation creates aerosols and strips the more volatile fission products from the concrete/core molten bath. In contrast, concrete containing basaltic (volcanic) and granitic aggregates do not generate large quantities of gases and, hence, releases less radioactivity to the containment atmosphere when reacting with core melt materials.

2. There are major programs in the United States, Europe, and Japan for the solidification of HLW from reprocessing plants into stable, low-leach glasses. There are multiple requirements to solidify HLW.[5] The glass must incorporate uranium, plutonium, and fission products into a stable chemical form. The glass must allow easy processibility and minimize generation of radioactive aerosols and gaseous fission products. Excessive aerosol or gas generation during solidification processes would result in operating difficulties and high costs for treating HLW plant off-gas streams. In

principle, the requirements to make HLW glass from reprocessing plant HLW, and the requirements to stop and solidify materials from a molten reactor core meltdown, are similar.

Detailed designs and experimental studies of COMSORS have not been done for LWRs. To date, LWR studies have emphasized "core-catchers," where the objective has been to prevent melt-through of molten core through the bottom of the reactor containment. COMSORS has this as a subobjective, but includes other objectives. Developmental work has been done on COMSORS for gas-cooled fast reactors[6] for 1000 MWe power plants. A description of the proposed COMSORS for gas-cooled fast reactors is included below to provide an understanding of the concept. The design would require modification for specific LWR conditions.

The gas cooled fast reactor is a helium cooled reactor where the helium flows downward through a core of stainless steel clad UO_2. A prestress concrete reactor vessel (PCRV) is used. Figure 2 shows the reactor and COMSORS.

In this specific case, COMSORS consists of cooling coils in the liner of the reactor vessel, a thin layer of graphite, and seven layers of stainless steel boxes filled with borax ($Na_2B_4O_7$). In a reactor core melt accident, (1) the borax will melt, (2) dissolve uranium oxide and fission products to produce a "glass," and (3) absorb large quantities of heat. The borax/uranium oxide/fission product molten glass (with melting points between 742.5 and 966°C, depending upon chemical composition) spreads the heat load across the bottom of the pressure vessel. This reduces heat fluxes to manageable levels and allows cooling of the molten core materials. The graphite acts to eliminate any localized hot spots. The stainless steel boxes prevent rapid penetration of molten uranium oxide to the graphite by providing higher temperature barriers. This provides the necessary time for the borax to dissolve and dilute the fuel.

Experimental studies have been conducted on this particular system. For an LWR, the preferred material would probably be borax with a mixture of several other glass formers. After an accident in an LWR, the probable availability of water used for cooling on top of the molten core materials may affect selection of COMSORS materials.

Alternative Versions: A conceptual description of an advanced COMSORS incorporated into the concrete structure is described herein. Under the reactor vessel (Fig. 3), a portion of the concrete mat has a specially controlled concrete mat chemical composition. The concrete contains a mixture of different aggregates. The aggregates are chosen so that when the various aggregates, cement, steel rebar, and core materials melt, a waste glass that incorporates the core materials is created. The glass contains one or more aggregates containing neutron poisons to prevent any possibility of a criticality accident. The glass chemical composition is chosen to have a very high affinity for volatile fission products. The aggregates are chosen to minimize gas generation upon melting and, hence, minimize aerosol formation. The glass also has a high surface tension to minimize aerosol generation.

The depth and width of the concrete mat with the special concrete aggregate is chosen to contain the reactor core. A heat balance exists between radioactive decay heat and (1) heat needed to melt the concrete and (2) heat conducted out or removed by other mechanisms from the molten core/concrete matrix. Eventually, heat conduction out of the waste matrix will exceed heat generation and the core/concrete matrix will begin to solidify. The special aggregate concrete mat is sized to exceed the maximum volume of the molten core/concrete matrix and the area is chosen to maximize cooling. In particular, the top surface area is large enough so that if water is available, it will cool and solidify the waste matrix if it is poured on top of the waste matrix.

The concrete aggregate is a relatively low-melting aggregate (400-900°C). Low melting points are desirable for multiple reasons:

1 A low melting waste matrix will quickly spread the molten core/concrete material over a wide area under the reactor. This improves heat transfer and cools the matrix to quickly form a solid.

2 A low melting waste matrix minimizes gas and aerosol generation by two mechanisms. First, the rate of release of semivolatile radioactive gases is temperature dependent. Lower temperatures imply less gas release. Second, the rate of release of semivolatile radioactive gases is dependent on the concentration of those materials in the waste matrix. Diluting the core material reduces the fractional release of radioactive materials.

Status of Technology: Based on experimental evidence, existing concretes have widely differing responses if a molten core/concrete interaction occurs. Reactor source terms from core/concrete interactions vary widely. Clearly, some existing concretes under reactors are preferable to others. The technology can be said to be already applied, although its application was unintended.

For a number of advanced LWRs,[7] the use of core-catchers is planned. For example, the Asea Brown Boveri-Atom, [a 1000 MWe boiling-water reactor (BWR)], Model BWR-90, incorporates a core-catcher into the design. The above technology is directly applicable.

There are multiple COMSOR options, such as the specially engineered structures that have been proposed for the gas-cooled fast reactor or for modification of concrete aggregate. No studies have been done to identify preferred options.

Advantages:

1. Reduces maximum possible source term with reduced maximum possible accident in the event of reactor meltdown and containment failure.

2. Reduces maximum possible source term with reduced challenge to containment (lower pressure and temperature) and lowered probability of containment failure.

3. Reduces challenge to containment (lower pressure and temperature) and hence, may al'ow use of lower-cost containment buildings.

Additional Requirements: None

Comments: The historical design basis accident for LWRs was the so-called double-ended pipe-break accident. Reactor containments were designed to withstand this accident where the reactor containment filled with steam, but the emergency core-cooling system worked and there was no core meltdown. After the Chernobyl accident, a number of European countries adopted policies that reactor containments should withstand core melt accidents. With these regulatory requirements, the core melt accident source term may (but does not necessarily) control design of the reactor containment. Under such circumstances, there may be large economic incentives to reduce maximum accident source terms.

For advanced LWRs that use PCRVs, this technology may be particularly attractive.

References/Contacts:

1. H. P. Nourbakhsh, M. Khatib-Rahbar, and R. E. Davis, Fission Product Release Characteristics Into Containment Under Design Basis and Severe Accident Conditions, NUREG/CR-4881 (BNL-NUREG-52059), (March 1988).

2. D. A. Powers, "A Survey of Melt Interactions With Core Retention Materials," Proceedings of the International Meeting on Fast Reactor Safety Technology, Seattle, Washington, August 19-23, 1979.

3. A. Skokan and H. Holleck, "Chemical Reactions Between Light -Water Reactor Core Melt and Concrete," Nucl. Technol., 46 255 (Dec. 1979).

4. M. Peehs, A. Skokan, and M. Reimann, "The Behavior of Concrete in Contact with Molten Corium in the Case of a Hypothetical Core Melt Accident," Nucl. Technol., 46, 192 (Dec. 1979).

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Update/Compiler: Dec. 1988/CWF

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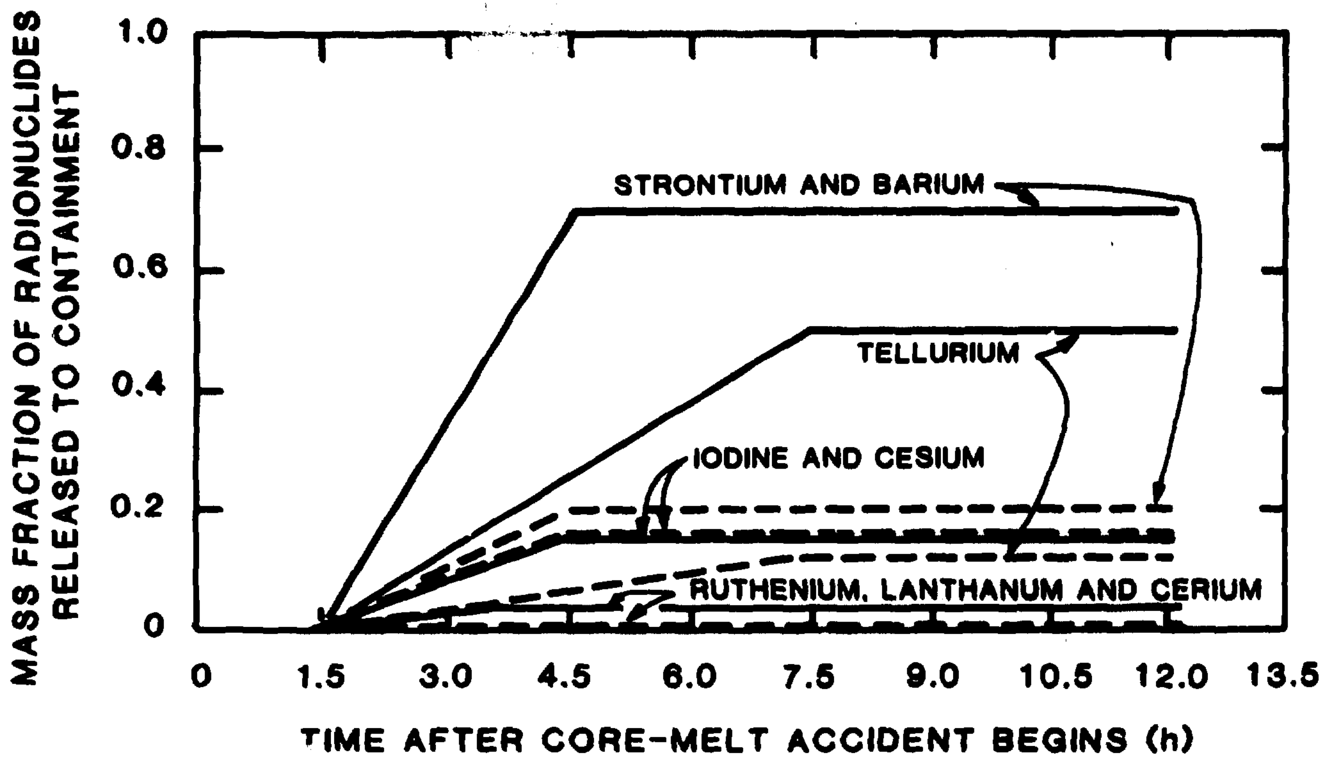

VARIABLES
---- BASALTIC CONCRETE/CORE INTERACTIONS
—— LIMESTONE CONCRETE/CORE INTERACTIONS
CASE SPECIFICS
BOILING WATER REACTOR
HIGH REACTOR COOLANT SYSTEM PRESSURE BEFORE STEEL VESSEL MELTTHROUGH
DRY CAVITY

Fig. 1. Cumulative release fractions of radionuclides to containment by chemical group and concrete type after core-melt accident.

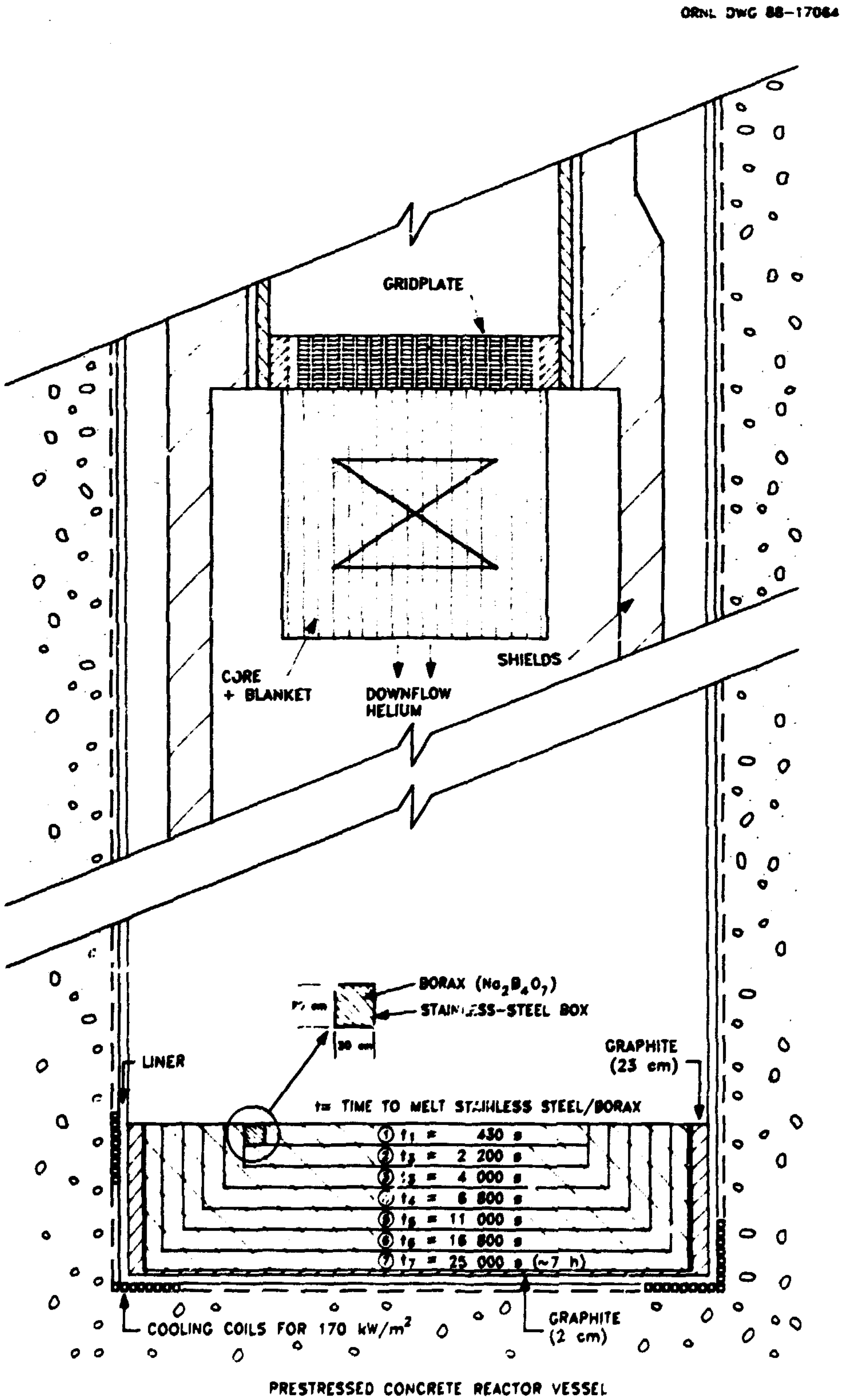

Fig. 2. Borax internal core catcher for gas-cooled fast reactor.

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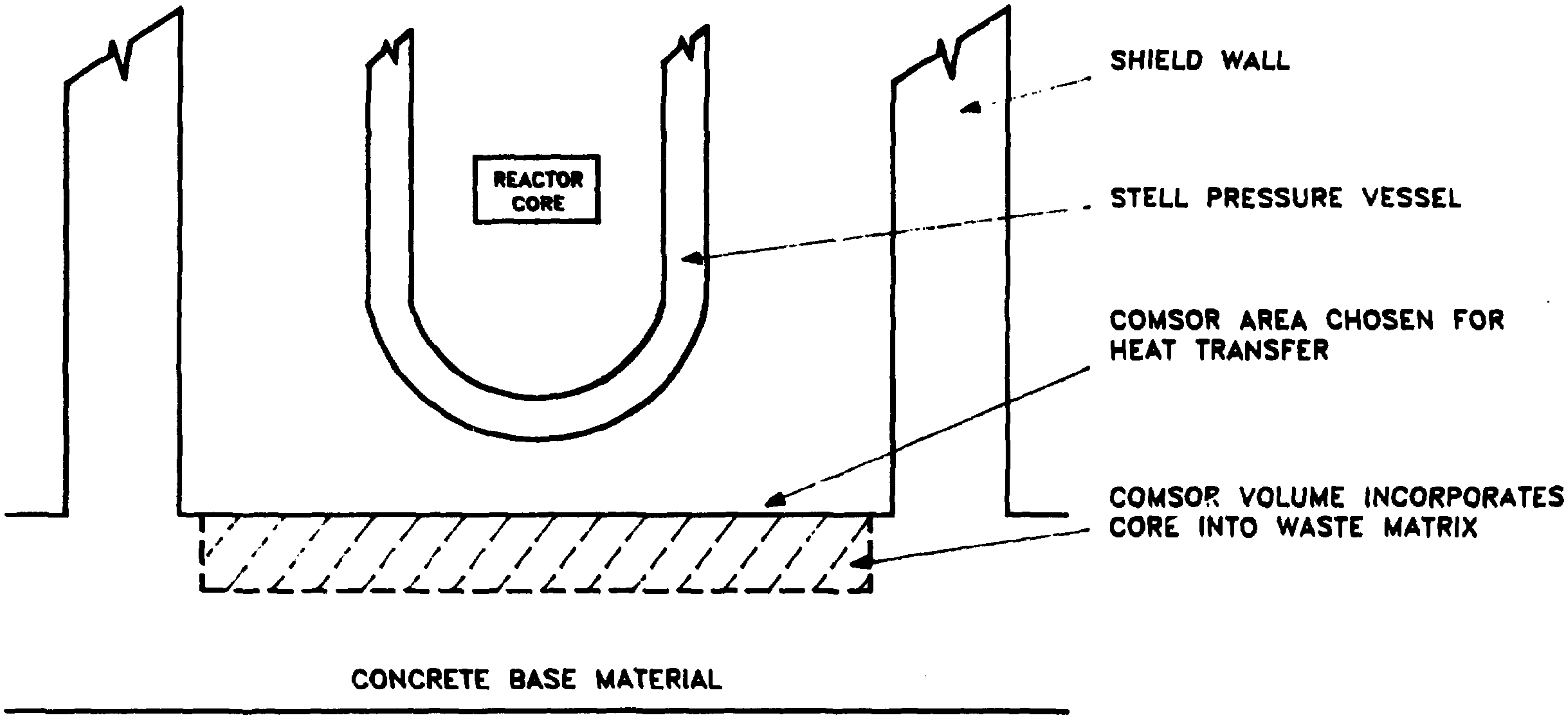

Fig. 3. Core melt source reduction system (COMSORS).

CHAPTER 13

Structures, Systems and Components: Other

13. STRUCTURES, SYSTEMS, AND COMPONENTS: OTHER

This chapter includes SSCs of functions that are identified in Fig. 1.3 but are not associated with radioactive materials from the reactor core. This includes storage of spent fuel at the reactor. Spent fuel is the second largest source of radioactivity at a nuclear plant. The inventories of spent fuel may be sufficient to be the dominant source term when the reactor core is shut down.

Table 13.1 Structures, Systems and Components: Other

Description Title	Function	Location of Description	Page
Concrete Shielding		Chapter 13	13-4
Vault (Canyon) Storage or Spent Nuclear Fuel		Chapter 13	13-6
Sealed Cask Storage of Spent Nuclear Fuel		Chapter 13	13-9
Subsurface Drywell Storage of Spent Nuclear Fuel		Chapter 13	13-13

TITLE: CONCRETE SHIELDING

Functional Requirements: Control direct radiation

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Multiple

Examples of Implementation: Concrete shielding currently used on all PWRs and BWRs.

Description: Reactor shielding in commercial LWRs can be divided into three general categories: (1) reactor vessel shielding, (2) secondary shielding enclosing the primary coolant system, and (3) biological shielding that forms the reactor building walls and floors and also surrounds the containment building. In boiling-water reactors (BWRs), drywell walls and floors serve as secondary shielding. The secondary and biological shields are generally massive concrete walls several feet in thickness. The concrete shields, excluding penetrations (which are not considered here), are passive and require no external power.

Since concrete is a mixture of elements, its composition can vary. Concrete shielding is produced using either "ordinary" or "heavy" mixtures. Ordinary concrete is made using portland cement; it primarily contains oxygen, and either silicon or calcium depending on whether the aggregates are granite, sandstone, or limestone. It has a density of 2.2 to 2.4 g/cm^3 and is an especially effective attenuator when used in conjunction with steel or other heavy material. Heavy concrete uses aggregates such as limonite, magnetite, barytes (barium sulfate), and steel in such forms as shot, punchings, sheared bars, etc. Heavy concrete, although considerably more expensive and requiring special pouring procedures, has a density of 3 to 6 g/cm^3, allowing smaller-volume shielding structures for a given level of attenuation.

Shielding is used to provide protection from neutrons and gamma radiation. It may be used to (1) slow down fast neutrons, (2) capture neutrons whether fast or slowed down, and (3) attenuate primary and secondary gamma radiation. Secondary radiation is generated by interactions between neutrons and nuclei in the shield. Whereas water is an excellent fast neutron attenuator but a poor gamma ray attenuator, and lead is an excellent gamma ray attenuator but a poor neutron attenuator, concrete, because of its mixture of elements, provides useful shielding from both neutrons and gamma radiation. Low cost and ease of construction have made it a primary shielding material for commercial power plants.

Extensive industry experience with concrete shielding has made it possible to design shields that provide the required attenuation in a minimum volume using analytic and semianalytic procedures, experimental data, and comparisons to other reactor and shield designs.

Alternative Versions: Since there is no "standard" ordinary or heavy concrete composition, the type of aggregate, cement, filler, and additives used may be varied to attain the desired structural and shielding properties.

Boron is used to increase the neutron shielding effectiveness of concrete, especially for low-energy neutrons. It also acts as a suppressor of secondary gamma rays. Boron may be added to the concrete in the aggregate (i.e., colemanite, pyrex glass, or boron frits), the cement, or the mixing water.

In ordinary concrete, the water used for the neutron shielding is lost at temperatures exceeding 200ºF. For that reason high temperature concrete was developed where the mineral serpentine preserves the hydrated water to temperatures exceeding 900ºF. Also at high temperatures, lumnite cement is stronger than portland.

Status of Technology: A mature technology in common use.

Advantages:

1. Shielding is passive, requiring no energy or activating mechanisms.
2. Effective attenuator of neutron and gamma radiation.
3. Highly reliable and durable.
4. Ordinary concrete is relatively inexpensive.
5. Adaptable to block or monolithic construction.
6. Tolerant of particle bombardment.
7. Excellent structural properties - especially useful for containment buildings.

Additional Requirements: None

Comments: None

References/Contacts:

1. N. M. Schaeffer, Reactor Shielding for Nuclear Engineers, U.S. Atomic Energy Commission, Office of Information Services, TID-25951 (1973).
2. R. G. Jaeger, E. P. Blizard et al., Engineering Compendium on Radiation Shielding, "Shielding Materials- Vol. 2 ", Springer-Verlag, New York (1975).
3. S. Glasstone and A. Sesonske, Nuclear Reactor Engineering, D. Van Nostrand Company, Inc., (1967).

Update/Compiler: April 1989/RHS and GAM

TITLE: VAULT (CANYON) STORAGE OF SPENT NUCLEAR FUEL

Functional Requirement: Control Radiation from Materials and Equipment in Storage

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Multiple

Examples of Implementation: A commercial vault system is being used at the Carolina Power and Light Co. H. B. Robinson Nuclear Power Plant in Hartsville, South Carolina. A similar vault system will be used at the Duke Power Company Oconee and the Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plants. The Irradiated Fuel Storage Facility at the Idaho Chemical Processing Plant has been using a canyon dry storage facility to store high-temperature, gas-cooled reactor (HTGR) spent fuel and fuel from the Rover Nuclear Rocket Program since 1975.

Description: Vault or canyon storage can be used to control the transport of radionuclides from spent nuclear fuel to the surrounding environment. The primary function of vault storage is to provide adequate barriers to the release of radionuclides while passively cooling the spent nuclear fuel. The storage system will also provide radiation shielding, protect the spent fuel from damage, and allow easy retrieval of the spent fuel.

A vault storage system consists of a sealed metal basket or canister located inside an aboveground concrete vault or "canyon." The spent fuel assemblies are placed in a steel basket or canister while under water in the reactor storage pool. The canister is then drained, sealed, dried, and leak tested. The canister is then placed in some type of a storage vault or canyon that is usually constructed of steel-reinforced concrete.

The commercially available vault system used at the H. B. Robinson Nuclear Power Plant is known as the NUTECH HOrizontal Modular Storage (NUHOMS) system. This system, shown in Fig. 1, uses concrete storage modules to contain the stainless steel spent fuel canisters. Canisters have been designed that will hold up to 24 PWR and 48 BWR fuel assemblies. The vault is designed with adequate air flow passages to provide passive convective cooling of the steel canisters. The NUHOMS system can accept fuel with a maximum burnup of 35,000 MWd/MTU and a minimum cooling time of 5 years. This would result in a maximum acceptable heat load for PWR fuel of approximately 1 kW/assembly.

Alternative Versions: Several commercial systems are available.

Status of Technology: Vault storage of spent fuel has been demonstrated. Commercial systems of various design are available.

Advantages:
1. The spent fuel can be easily monitored and retrieved.
2. Cooling is passive.

Additional Requirements: None

Comments: The technology is currently being used to expand spent fuel storage at existing reactors. The technology may be of interest for advanced reactors where limited spent fuel may be stored within the reactor pressure vessel. Out-of-reactor storage of spent fuel in vaults for aged spent fuel may eliminate the need for spent fuel storage ponds.

References:

1. D. D. Orvis, C. Johnson, R. Jones, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, Electric Power Research Institute, EPRI NP-3365, February 1984.

2. J. V. Massey, J. M. Rosa, M. Shamszad, S. S. Wang, and J. A. Maly, "Design of a Horizontal Concrete Module and a Dry Shielded Canister for Use in an Irradiated Fuel Storage System," Proceedings of the Third International Spent Fuel Storage Technology Symposium/Workshop, Vol. 2, CONF-860417, NUTECH, Inc., San Jose, California, April 1986.

3. P. A. Anderson and H. S. Meyer, Dry Storage of Spent Nuclear Fuel: A Preliminary Survey of Existing Technology and Experience, U. S. Nuclear Regulatory Commission, NUREG/CR-1223, April 1980.

Update/Compiler: April 1989/WJR

ORNL DWG 89-292

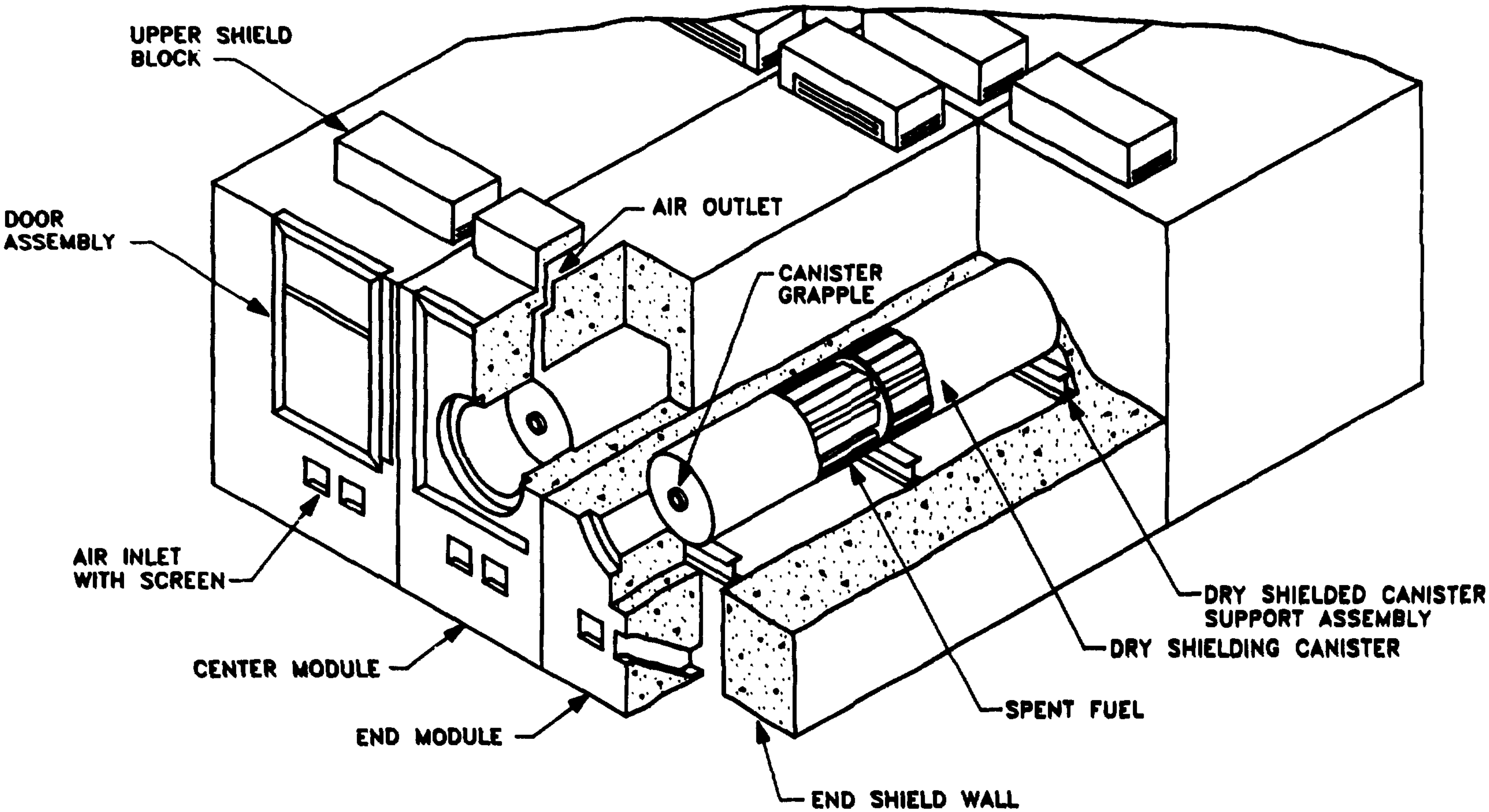

Fig. 1. Nutech Horizontal Modular Storage system.

TITLE: SEALED CASK STORAGE OF SPENT NUCLEAR FUEL

Functional Requirements: Control Radiation from Materials and Equipment in Storage

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Multiple

Examples of Implementation: Sealed cask storage is used worldwide. In the United States, storage casks are used at the Virginia Electric Power Company's Surry Plant. Canada has storage cask facilities for spent CANDU fuel. In West Germany, storage casks are used at the Gorleben Monitored Retrievable Storage Facilities. The Gorleben site has a capacity of 420 casks, with equal size facilities at two other locations.

Description: Sealed cask or silo storage can be used to control the transport of radionuclides from spent nuclear fuel to the surrounding environment. The primary function of sealed cask storage is to provide adequate barriers to the release of radionuclides while passively cooling the spent nuclear fuel. The storage system will also provide radiation shielding, protect the spent fuel, and, for certain cask designs, allow off-site shipment of the spent fuel. Spent fuel stored in a sealed cask system is easily retrievable.

A typical cask storage system (as shown in Fig. 1) is similar in design to a rail-transported shipping cask, except that a cask storage system places greater emphasis on capacity. Off-site transport of the storage casks may be possible by placing the storage cask in an approved overpack, transferring the spent fuel to a transport cask, or by initially using a dual-purpose storage and shipping cask.

Storage casks are loaded with spent fuel while under water in the reactor storage pool. The cask is then drained, sealed, dried, and leak tested. Table 1 summarizes the storage and storage/transport casks available in the United States.[4]

Alternative Versions: Many commercial versions are available (see Table 1).

Status of Technology: The technology is commercially available.

Advantages:
1. Storage cask design can be integrated with shipping cask design.
2. The spent fuel can be easily monitored and retrieved.
3. The costs to expand the system are lower than for other storage systems such as a spent fuel pool.
4. Cooling is totally passive.

Additional Requirements: None

Comments:
1. A number of commercial storage and shipping and storage-only casks are available and are detailed in the report by Johnson and Notz.[4]

2. The technology may be of interest for advanced LWR reactor designs where limited spent fuel may be stored within the reactor pressure vessel. Out-of-reactor storage of spent fuel in sealed casks for aged spent fuel may eliminate the need for spent fuel storage pools.

References/Contacts:

1. D. D. Orvis, C. Johnson, R. Jones, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, Electric Power Research Institute, EPRI NP-3365, February 1984.

2. P. A. Anderson and H. S. Meyer, Dry Storage of Spent Nuclear Fuel: A Preliminary Survey of Existing Technology and Experience, U. S. Nuclear Regulatory Commission, NUREG/CR-1223, April 1980.

3. I. E. Wall and Z. S. Beallor, "Management of Irradiated Components from the Pickering Units 1,2 Retube," p 17.19, Proceedings of the 6th Annual Conference of the Canadian Nuclear Society, ISSN 0227-1907, Ottawa, Canada, 1985.

4. E. R. Johnson and K. J. Notz, Shipping and Storage Cask Data for Spent Nuclear Fuel, ORNL/TM-11008, Oak Ridge National Laboratory, November 1988.

5. J. O. Blomeke, A Review and Analysis of European Industrial Experience in Handling LWR Spent Fuel and Vitrified High-Level Waste, ORNL/TM-10696, Oak Ridge National Laboratory, June 1988.

Update/Compiler: April 1989/WJR

Table 1. Storage and storage/transport casks for spent fuel

	CASTOR-IC	CASTOR V/21	TN-24P	TN-24	TN-RBG/ BRP	MC-10	REA-2023	NAC S/T	C-E Dry Cap	NUPAC CP-9D[h]	NUHOMS 0708
CASK TYPE ->	Storage	Storage	Storage/ Transport	Storage	Storage/ Transport	Storage	Storage	Storage/ Transport	Storage/ Transport	Storage	Storage
Capacity (assemblies)											
Intact SF	16 BWR	21 PWR	24 PWR	24 PWR 52 BWR	40 PWR 85 BWR[b]	24 PWR	24 PWR 52 BWR	26 PWR[f]	24 PWR 60 BWR	9 PWR	7 PWR
Consolidated fuel rods	21 BWR	42 PWR	48 PWR	48 PWR 104 BWR	-	48 BWR	48 PWR 104 BWR	56 PWR 122 BWR	48 PWR 100 BWR	18 PWR	-
Weight-loaded (tons)[a]	81	117	100	100	103/98	120	98	100	111/113	88	11[c]
Design heat rejection (kW)[a]	14.4	28	20.6	24	5	13.5	44.6	26[g]	24	5.4	7
Peak clad temp. (°C)[a]	365	365	221	375	375	336	304	360	355	290	380
Mat'ls. of construction											
Cask body	Nodular cast iron	Nodular cast iron	Forged steel	SA350, Gr. LF1	Forged steel	Forged steel	Pb/SS	Pb/SS	Steel	Concrete	Concrete[d]
Basket	Borated SS	SS/Borated SS	Al/B	SS/B, Cu plate	SS/B	SS Grid/ SS cell w/boral plates	Ss w/ boral inserts	Al	SS	Steel	SS/Pb in ends[e]
Neutron shield	Polyethylene	Polyethylene	Resin-sides; lid-polypropylene	Polyester resin	None	BISCO N-3	Glycol and water	BISCO NS4FR	BISCO	Concrete/ RX-277	Concrete[d]
Cavity atmosphere	He	He	He	N or He	Inert	He	He	He	He	Inert	He
Licensing status	Licensed	Licensed	Applied for-storage	Applied for	Applied for-transport	Licensed	Applied for	Applied for-storage	Applied for	Applied for	Licensed
No. in existence in U.S. (or on order)	1	6	1	0	2	1	1	1	0	0	3

[a]For intact fuel assemblies.
[b]Short assemblies, double stacked.
[c]Weight of canister only - - not concrete storage module.
[d]Refers to NUHOMS concrete storage module.
[e]Refers to NUHOMS canister.
[f]31 PWR w/burnup credit.
[g]16 kw for burnup credit version.
[h]Applies for CP-9W.

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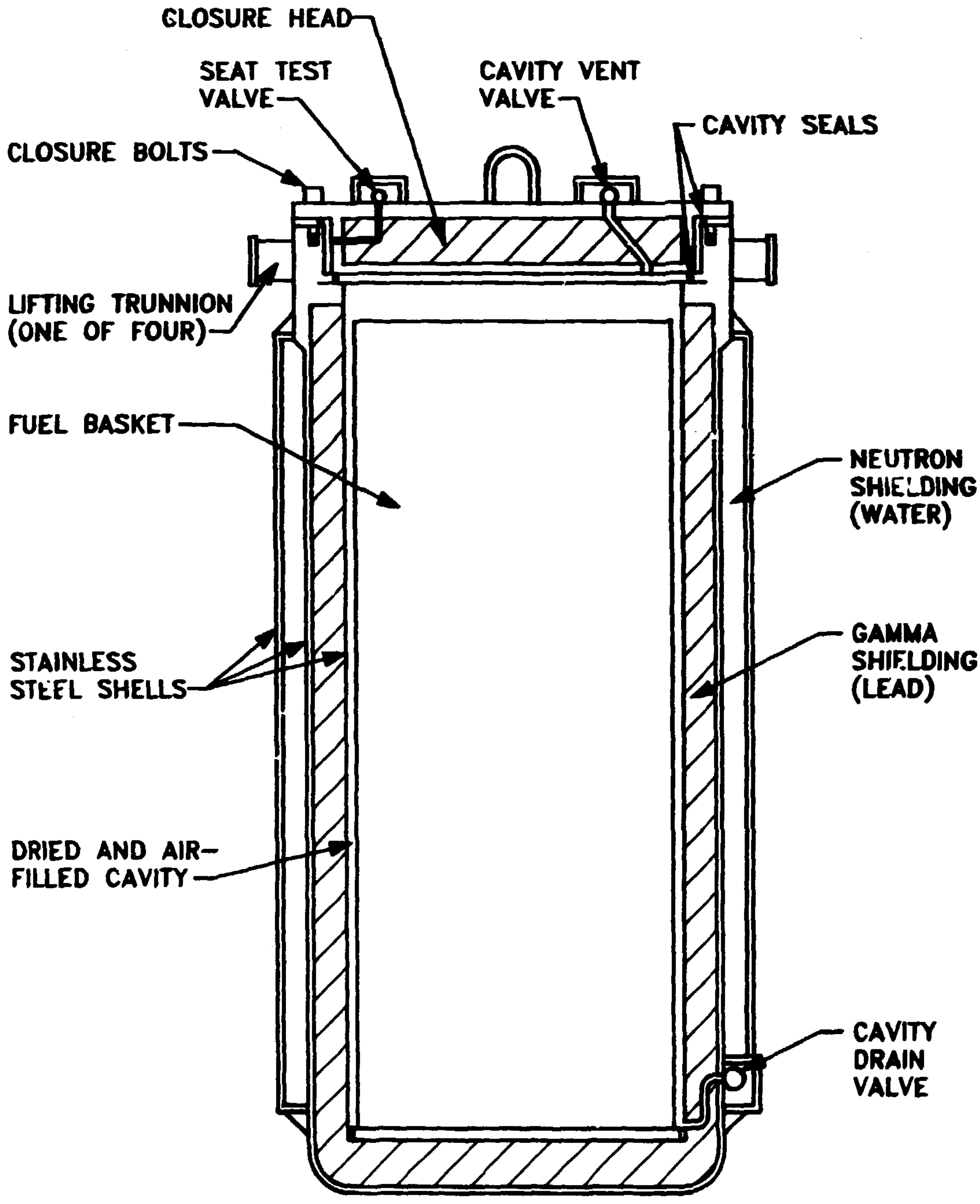

Fig. 1. Sealed cask storage of spent nuclear fuel.

TITLE: SUBSURFACE DRYWELL STORAGE OF SPENT NUCLEAR FUEL

Functional Requirements: Control Radiation from Materials and Equipment in Storage

Safety Type: Passive

Developmental Status: 1 Commercial

Reactor Type: Light-water reactor (LWR)

Organization: Multiple

Examples of Implementation: A large drywell facility exists at the Idaho Chemical Processing Plant (ICPP) at the Idaho National Engineering Laboratory. A drywell demonstration facility was tested by Rockwell Hanford Operations in the late 1970s. Drywell storage has been demonstrated for pressurized-water reactor (PWR) and boiling-water reactor (BWR) fuel assemblies at the Nevada Test Site.

Description: Drywell or dry caisson storage can be used to control the transport of radionuclides from spent nuclear fuel to the surrounding environment. The primary function of drywell storage is to provide adequate barriers to the release of radionuclides while passively cooling the spent nuclear fuel. The storage system will also provide radiation shielding and protect the spent fuel from accidental and intentional damage. Spent fuel stored in a drywell system will be easily retrievable.

A typical drywell storage system[1,2] consists of a steel or steel-lined concrete caisson with the bottom typically ~25 ft below ground. The spent fuel assemblies are placed in a steel containment overpack that is then placed in the underground caisson as shown in Fig. 1.[1] Next, a shielding plug is fitted above the fuel assembly overpack. Then, a steel or concrete cover is placed on top of the caisson. The caisson, typically 20 to 30 in. in diameter, is not considered a containment vessel and thus may have a vented or open bottom. The top cover is provided only for safety and safeguard purposes. The fuel cladding and overpack are considered the primary barriers to radionuclide release.

The amount of heat dissipation is inversely proportional to the caisson diameter. The thermal conductivity of the soil will also affect the amount of heat dissipated. The decay heat from the fuel will dissipate through the caisson barriers into the surrounding earth and, subsequently, to the atmosphere. The air gaps between the overpack, the caisson, and the ground do not cause large temperature increases because heat transfer at higher temperatures is primarily by radiation. The temperature, pressure, and other parameters can be monitored through the access tube provided in the design.

Heat loads of 1-3 kW/caisson are considered acceptable. Assuming a heat load of about 2.4 kW, it has been shown[2] that a caisson spacing of 25 ft center-to-center will cause no significant interaction between adjacent caissons.

Alternative Versions: None

Status of Technology: Subsurface drywell storage of spent fuel has been demonstrated and probabilistic risk assessment studies have been performed.

Advantages: 1. Cooling is passive.
2. The spent fuel can be easily monitored and retrieved.
3. The degree of protection afforded the spent fuel is high and the risks of a radioactive release are low.
4. The costs to expand the system are much lower than for other storage systems such as a spent fuel pool.

Additional Requirements: None

Comments: The technology may be of interest for advanced LWR reactor designs where limited spent fuel may be stored within the reactor pressure vessel. Out-of-reactor drywell storage of spent fuel for aged spent fuel may eliminate the need for spent fuel storage pools.

References/Contacts:

1. D. D. Orvis, C. Johnson, R. Jones, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI NP-3365, Electric Power Research Institute, February 1984.

2. L. M. Richards and M. J. Szulinski, "Subsurface Storage of Commercial Spent Nuclear Fuel," Nucl. Technol., 43, 155-64, (Mid-April 1979).

3. P. A. Anderson and H. S. Meyer, Dry Storage of Spent Nuclear Fuel: A Preliminary Survey of Existing Technology and Experience, NUREG/CR-1223, U.S. Nuclear Regulatory Commission, April 1980.

Update/Compiler: April 1989/WJR

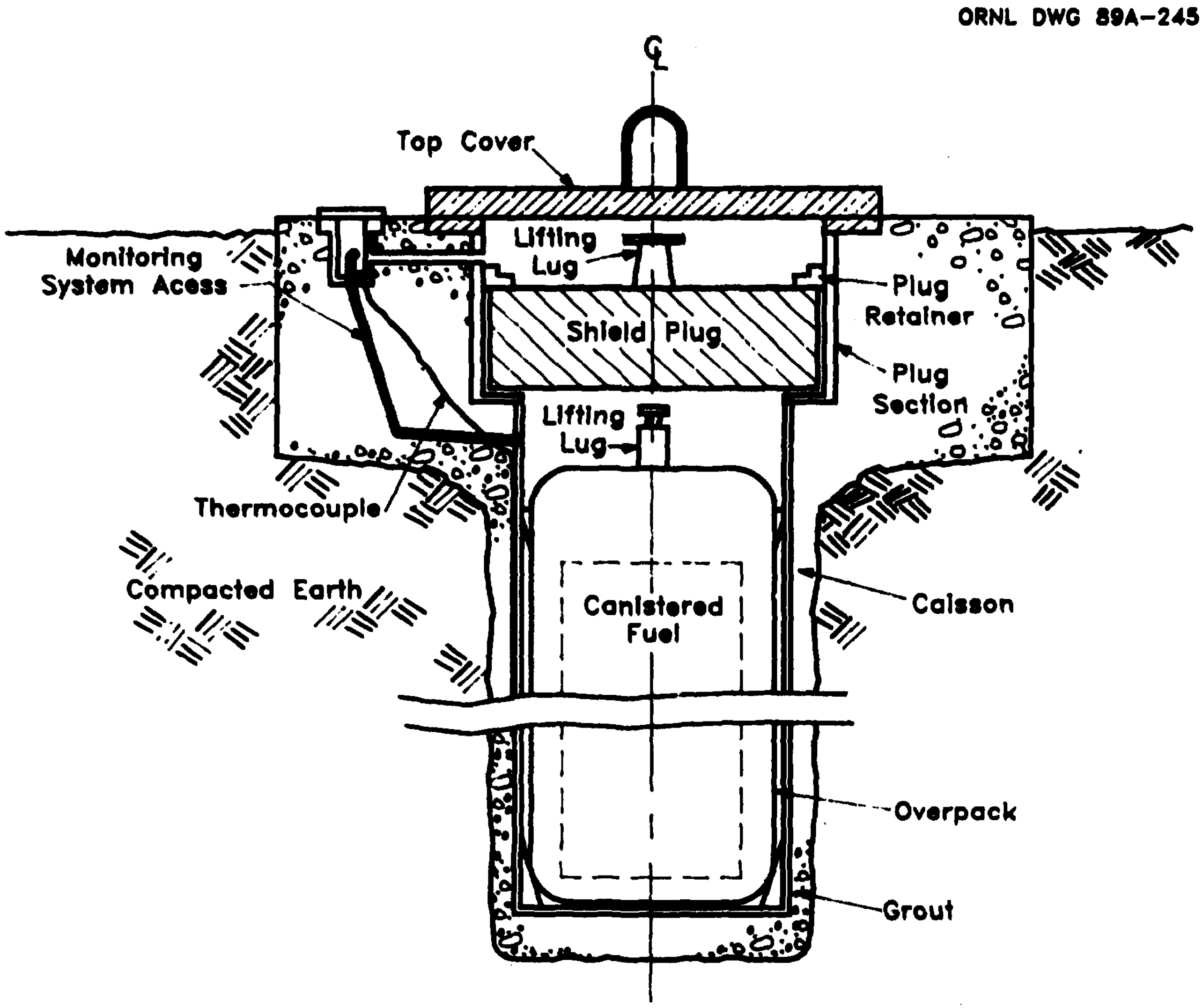

Fig. 1. Subsurface drywell storage of spent nuclear fuel.

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