COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

PRINCIPAL WORKING GROUP ON
CONFINEMENT OF ACCIDENTAL RADIOACTIVE RELEASES (PWG4)

SURVEY OF CONTAINMENT DESIGNS FOR
NEW/ADVANCED WATER REACTORS

FOR TECHNICAL REASONS, THIS DOCUMENT IS NOT AVAILABLE ON OLIS
SURVEY OF CONTAINMENT DESIGNS FOR NEW/ADVANCED WATER REACTORS

Report by a Group of Experts

Compiled by: R.L. RITZMAN

February 1994
OECD

The Convention establishing the Organisation for Economic Co-operation and Development (OECD) was signed on 14th December 1960.

Pursuant to article 1 of the Convention, the OECD shall promote policies designed:

-- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and this to contribute to the development of the world economy;

-- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and

-- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The current Signatories of the Convention are Australia, Austria, Belgium, Canada, Denmark, Finland, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Japan, Luxembourg, the Netherlands, New Zealand, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States.

NEA

The OECD Nuclear Energy Agency (NEA) now groups all the European Member countries of OECD and Australia, Canada, Japan, the Republic of Korea and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress.

NEA works in close collaboration with the International Atomic Energy Agency, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

CSNI

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and coordinate the activities of the OECD Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.
SURVEY OF CONTAINMENT DESIGNS FOR NEW/ADVANCED WATER REACTORS

Compiled By:
R. L. Ritzman

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FOREWORD

At a meeting held on 11th-12th April 1991, CSNI-FWG4’s Task Group on Severe Accident Phenomena in the Containment (SAC) decided to prepare a survey of ongoing efforts to design, qualify, and licence new containment concepts, highlighting considerations for severe accidents. Dr. R.L. Ritzman volunteered to co-ordinate this effort. After Dr. Ritzman’s retirement OECD awarded him a consultancy contract in order to allow him to continue and complete this work.

A preliminary draft was discussed by Principal Working Group No. 4 on Confinement of Accidental Radioactive Releases (FWG4) at a meeting held on 28th-29th September 1992. A revised draft was then prepared and submitted to SAC on 23rd-24th March 1994. These discussions led to the preparation of a final draft, which was examined by FWG4 on 27th-28th September 1993. A slightly modified version, dated 28th October 1993, was submitted on 1st-2nd December 1993 to CSNI who approved it for publication in the series of CSNI Reports subject to additional comments to be made before 31st January 1994. No additional comments were received.

The CSNI and the Secretariat thank Dr. Ritzman for his work on this survey.
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SURVEY OF CONTAINMENT DESIGNS FOR
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R. L. Ritzman

1.0 Introduction

The world is now in its fourth decade of commercial nuclear power production. Nuclear has become an important source of electricity in many countries and a major source in some. The water cooled reactor has been the main type of nuclear power plant and this is expected to continue into at least the early part of the next century.

For the last several years various public and private organizations have been working on improved water reactor designs to augment and eventually replace the present generation power plants. Safety improvements have received considerable attention in these design efforts with respect to accident prevention, gaining/maintaining control of accident situations, and mitigation of accident outcomes/effects. An important element of the safety envelope for a nuclear power plant is the containment system. The positive aspects of an effective containment system were demonstrated during the Three Mile Island (TMI) accident in 1979, while the negative impact of a lack of containment was clearly shown during the Chernobyl accident in 1986.

Most of the currently operating power reactors in the world are enclosed in strong containment buildings which have been designed to specific industry standards and which meet applicable regulatory requirements. These basically assure that the containment will perform its intended function in a variety of postulated accident situations. These are usually referred to as design basis accidents (DBAs). In many cases the containments are overdesigned such that they provide considerable protection in accidents which would be beyond the design basis level. Also, in a few countries regulators have, since TMI and Chernobyl, required additional equipment (such as hydrogen igniters or filtered vents) to mitigate the effects of severe accidents.* However, these were not specific objectives or features that were considered in the original containment design.

Now the situation is somewhat different. Challenges from low

*Such accidents, which are very unlikely because they occur only if multiple safety systems fail, involve gross core damage and extensive fuel melting (i.e., core meltdown). This can result in breach of the reactor pressure vessel and direct attack of the containment boundary.
probability events, including severe accidents, are being more actively considered in the design of containment systems for the next generation of water-cooled reactor power plants. The purpose of the present document is to provide an overview of what is currently taking place in this technical area. Information of a non-proprietary nature was solicited from the international community regarding new/advanced containment concepts that may be in various stages of development (1a). The emphasis is on complete containment systems rather than on individual subsystems or concepts, and also on systems which are intended to accommodate severe accident challenges, regardless of whether or not they may be required for licensing purposes.

The following sections of the report present the information on specific systems that was received in response to the solicitation by the end of May, 1993. All-together information was obtained on thirteen different systems. This included data on some mature designs as well as on several newer or emerging concepts. As a result of this mix it was decided to group the discussion of systems according to their current stage of development.

Four stages or phases of development were selected as follows: (1) the commercially available phase which consists of systems that are ready for commercial deployment, (2) the detailed engineering phase which is the final design phase that precedes commercial status, (3) the basic engineering phase which encompasses standard safety analysis report (SSAR) preparation and early first-of-a kind engineering work, and (4) the conceptual design phase which is the earliest stage of development and thus involves minimal engineering support.

Of the thirteen systems indicated above two were assigned to stage (1), two to stage (2), three to stage (3), and the remaining six were placed in the 4th stage. The discussion of each system begins with a description of the general characteristics of the reactor and the containment. This is followed by comments regarding how particular severe accident challenges to containment integrity have been addressed in the design. Each discussion concludes with a brief summary of the current status of the design and development effort.

The last section of the report provides a summary table which attempts to compare in abbreviated format the important features of the various containment designs and the methods they employ for coping with severe accident phenomena/challenges.
2.0 Commercially Available Designs

2.1 ABWR

General Characteristics (2a,2b)

The ABWR (Advanced Boiling Water Reactor) has a rated power of 3926MWe with a gross electrical power output of about 1356MWe. The plant utilizes a General Electric designed single-cycle, forced-circulation, boiling water reactor (BWR) in a reactor pressure vessel (RPV) which features use of internal coolant recirculation pumps. Thus the vessel has no large coolant nozzles or piping below the top of the core. The reactor is equipped with a diverse combination of high and low pressure emergency core cooling systems and power sources including an independent automatic depressurization system (ADS) capability which actuates if high-pressure injection systems cannot maintain reactor water level in loss-of-coolant accident (LOCA) events.

The general layout of the ABWR is shown in Figure 2-1 and a simplified containment schematic is given in Figure 2-2. The ABWR has a containment system that is similar to previous BWR designs but with added or improved features. The primary containment is a low-leakage vessel having a design internal pressure of 45psig (0.31MPa) which incorporates the following main features:

1. the drywell, a cylindrical steel-lined reinforced concrete structure with a free volume of about 260,000 cu ft (7350 cu m) surrounding the reactor pressure vessel;

2. a suppression pool equipped with horizontal entrance vents connected to the drywell. The pool is filled with about 126,000 cu ft (3570 cu m) of water which serves as a heat sink during normal operations and under accident conditions;

3. the air space above the suppression pool with free volume of about 210,000 cu ft (5950 cu m);

4. the diaphragm floor, a reinforced concrete circular slab which serves as a barrier between the upper part of the drywell and the suppression chamber;

5. the reactor pedestal, a composite steel and concrete structure which provides support for the reactor pressure vessel, the reactor shield wall, the diaphragm floor, access tunnels, horizontal vents, and the lower drywell (region under the reactor pressure vessel) access platforms. There are ten channels connecting the upper drywell to the lower drywell and the horizontal vents to the suppression pool; and

6. the reactor building which is structurally integrated with the concrete primary containment structure. This includes the containment foundation mat which is 18 ft (5.5 m) thick.
Figure 2-1 General Layout of ABWR

Figure 2-2 ABWR Containment Schematic
The reactor building is constructed of reinforced concrete with a steel frame roof. It has four stories above ground level and three stories below. It is about 56 m wide, about 59 m long, and about 58 m high. The outer portions of this structure serve as a secondary containment which permits monitoring and treating all potential radioactive leakage from the primary containment.

The containment design includes various active and passive systems for maintaining containment integrity and controlling radioactive discharges during design basis and severe accident events. The containment isolation system uses a variety of isolation valves that automatically close fluid penetrations for all systems not required for emergency operation upon receipt of signals of high drywell pressure or low water level in the reactor vessel. Fluid penetrations supporting engineered safety feature systems have remote manual isolation valves which can be closed from the control room if required.

Containment heat removal is normally achieved with a portion of the residual heat removal system which in effect circulates suppression pool water through heat exchangers connected to the reactor building cooling water system and finally to the reactor service water system. The system includes drywell and wetwell sprays to promote heat removal from these gas spaces via drainage to the suppression pool. A passive system is also in place to flood the lower drywell with water if high temperature core debris should enter that region which is below the reactor pressure vessel.

Combustible gas control in containment is achieved using a nitrogen inverting system to maintain primary containment oxygen concentrations at or below 3.5 volume percent during normal operations via a purge and makeup process. Oxygen analyzer and recombiner systems are provided to measure and control the concentration of oxygen produced by radiolysis in the primary containment. In addition overpressure protection to relieve excessive containment pressure through a pathway from the wetwell airspace to the stack is provided by a passive vent system equipped with dual rupture disks.

The suppression pool acts as an efficient scrubber of radioactive material (aerosols and vapors) from gas flows that pass through it from the drywell regions during LOCAs or from the reactor vessel during safety relief valve discharges. It is in effect a passive radioactivity retention system. The ABWR is also equipped with a standby gas treatment system (SGTS) to minimize exfiltration of contaminated air from the secondary containment to the environment following an accident or abnormal condition that could result in high airborne radiation in the reactor building. This is a safety grade active system containing both HEPA and activated charcoal filters which discharges treated effluent to the plant stack.
Severe Accident Challenges Addressed

Numerous potential severe accident challenges to containment integrity can be accommodated in this design. The ADS capability should reduce the probability of significant direct containment heating (DCH) events to negligible levels. Operating with a nitrogen inerted atmosphere in the primary containment should provide protection against combustion of hydrogen that could accumulate in the containment during a severe accident. The lower drywell flooding system is provided to promote quenching of molten core debris that might enter this region and thus limit thermal attack of the concrete basemat and consequent generation of noncondensable gaseous decomposition products. The robust structures surrounding this region should protect the containment boundary from missiles that might be generated by energetic fuel coolant interaction (FCI) events that could accompany debris quenching. In the unlikely event that active means for removing heat from the primary containment should fail, the passive venting system should prevent overpressure rupture of the structure. The vent lines can be reclosed manually to restore containment isolation upon completion of pressure relief. Radioactivity present in the drywell atmosphere that undergoes venting would be attenuated by scrubbing in the intervening suppression pool. As noted earlier the suppression pool should be a major passive radioactivity retention site throughout a severe accident. Other natural processes, i.e., surface plateout and aerosol settling, will operate to reduce suspended radionuclide concentrations in the primary containment as well as in the multi-compartmented secondary containment. In some severe accident scenarios the active SGTS may also provide significant retention of radioactivity that might be released into the secondary containment.

Design/Development Status

The ABWR program has been underway for several years. Construction of the first two units is currently in progress in Japan and commercial operation of the two is scheduled to begin in 1996 and 1997. The U. S. Nuclear Regulatory Commission (USNRC) has been reviewing the ABWR design documentation for some time and final design approval is now scheduled for May, 1994. Actual certification as a standardized design would occur at a later date.

2.2 BWR90

General Characteristics (2c)

The BWR90 represents the ABB ATOM advanced boiling water reactor (BWR) nuclear plant design for the 1990s. The BWR90 is available
in two standard sizes, 2350MWe (830MWe) and 3300MWe (1170MWe). The reactor design is characterized by the use of internal recirculation pumps and fine motion control rod drives. Extensive redundancy and separation of safety-related systems are employed.

The BWR90 pressure-suppression containment consists of a cylindrical prestressed concrete structure with an embedded steel liner. The containment vessel, including the pressure-suppression system and other internal structural parts as well as the pools above the containment, forms a monolithic unit and is statically free from the surrounding reactor building, except for the common foundation slab. The main data for the BWR90 containment are as follows:

- Design Pressure: 0.6MPa abs
- Total gas volume: 9700 cubic meters
- Suppression Pool volume: 3200 cubic meters
- Design leak rate: 0.5Vol-%/24 h at max calculated pressure

The seismic effects caused by a design basis earthquake is taken into account in the BWR90 design.

Figure 2-3 illustrates the important design features of the containment. The essential features are:

1. The upper drywell floor is stiffly connected to the cylindrical containment wall structure whereby the sealing at this joint as well as the possible bypass of the pressure-suppression function have been eliminated.

2. Horizontal openings between drywell and wetwell for blowdown of steam to the suppression pool.

3. The relief pipes from the safety/relief valves are routed into the suppression pool via the lower drywell rather than penetrating the drywell-wetwell intermediate floor.

4. The number of penetrations through the drywell-wetwell intermediate floor has been minimized.

5. A pit is provided at the bottom section of the lower drywell for the purpose of collecting and confining fuel melt debris. The cylindrical wall and pit form a pool which is filled with water in the event of a severe accident.

6. The penetrations in the lower drywell area are equipped with mechanical protection devices.

The BWR90 also encompasses a number of features of importance for severe accidents that has been standard for ABB Atom design for a long time such as:

1. Nitrogen inerted containment atmosphere during operation.
Figure 2-3 Schematic of the BWR90 Containment
(2) Diversified insertion principle of the control rods as well as an upgraded boron system to minimize the probability of ATWS events.

(3) An Automatic Depressurization System actuated upon suitable chosen actuation signals to avoid high pressure RPV melt-through.

The above discussed arrangements improve the reliability of the pressure-suppression system and reduce the probability of containment leakage during an accident. In addition, the BWR90 is equipped with a filtered containment venting system which is connected both to the drywell and the wetwell gas phase. The filter which is a steel pressure vessel version of FILTRA-MVSS, is located inside the reactor building. The system can be actuated both automatically (by means of a rupture disc burst) and manually by the operator. The system can then be in operation for more than 24 hours without any operator actions.

The BWR90 is also equipped with an independent containment spray system which provides the possibility to transfer water to the containment from either mobile water sources or the fire water system via the containment spray system spray nozzles. This supply is also beneficial for the purpose of increasing the heat sink in the containment and for washing out radioactive matter from the containment atmosphere.

Severe Accident Challenges Addressed

When the design review of the BWR90 was initiated, regulatory developments indicated a need to strengthen the capability of the reactor containment to withstand the effects of a severe accident. Requirements in this are now codified in Finland and Sweden. The containment of the BWR90 plant has therefore been modified to achieve enhanced environmental safety during a degraded core accident, compared with the containment in the reference plant. As indicated earlier an ADS capability has been installed on the reactor coolant system to avoid high pressure RPV melt-through and thus DCH type events. The use of a nitrogen inerted atmosphere eliminates concern about hydrogen combustion as long as the containment remains isolated. The pressure-suppression feature provides an effective short-term heat sink for energy liberated from stored heat, decay heat, and exothermic chemical reactions, and thus protects against early over-pressurization of containment. Long-term protection is achieved by the ability to add water to containment (independent spray system) and by the presence of the filtered venting equipment (FILTRA/MVSS). The BWR90 also depends on a large reactor cavity and water flooding to disperse and quench molten core debris that might drain from the reactor vessel in a severe accident. Quenched debris would not attack the concrete basement nor ini-
tiate generation of noncondensable gases. Internal shield walls in the containment should provide protection for the outer containment boundary from missiles that might be generated if energetic FCI events should occur. The suppression-pool should act as an efficient scrubber of radioactive materials (aerosols, vapors, etc.) that might be carried by transporting steam and fluid flows. The independent spray system could also wash out suspended material from the sprayed region. If the FILTRA/MVSS equipment should come into use it has been designed to provide a decontamination factor (DF) of at least 500 for aerosols and 100 for iodine.

Design/Development Status

The BWR90 is currently ready for commercial use. It is being considered for deployment as a fifth nuclear power plant in Finland.
3.0 Detailed Engineering Phase Designs

3.1 System 80+

General Characteristics (3a,3b)

The System 80+ standard plant is an advanced PWR with a power output of about 3800 MWe (1300 MWe). It is a product of ABB Combustion Engineering which represents evolutionary improvements to the standard System 80 design. It is a complete power plant, including nuclear island, turbine island, and balance of plant. The reactor coolant system has a two-loop configuration and various improvements have been made to components to enhance performance and safety including larger pressurizer volume, upgraded steam generator features, and larger secondary feedwater inventory. Active, dedicated, four-train safety systems provide emergency core cooling and feedwater and decay heat removal. The safety injection system consists of a passive portion (pressurized tanks) and an active portion (pumps aligned with an in-containment refueling water storage tank [IRWST]) which deliver borated water directly to the RPV. A safety depressurization system (manually activated) which discharges to the IRWST is provided which, in combination with the safety injection system, permits decay heat removal through feed and bleed. The instrumentation and control systems for the plant employ the latest technology to enhance both normal and off-normal operations.

The System 80+ uses a spherical containment structure as depicted in Figure 3-1. The steel vessel is 200ft (61 m) in diameter, has a design pressure of 49 psig (0.34MPa), and a free volume of about 3.4 million cu ft (96,000 cu m). It is surrounded by a reinforced concrete shield building composed of a right circular cylinder with a hemispherical dome having 3ft (1 m) thick walls and a common foundation basemat with the surrounding nuclear system annex buildings. The containment is equipped with a number of engineered systems to maintain containment integrity and remove radioactivity in accident situations. This includes a containment isolation system, a containment spray/heat rejection system, an annulus ventilation system, and combustible gas control systems.

The isolation system acts to close all non-safety related fluid lines penetrating containment in response to isolation actuation signals. The containment spray system has two independent trains of pumps, heat exchangers, and spray headers to deliver borated water from the IRWST to the upper region of the containment space for use in LOCAs to condense steam, extract heat, and remove airborne radioactivity from the internal atmosphere. The annulus ventilation system processes air from the annulus between the steel containment and the concrete shield building to maintain a negative pressure zone and provide fission product removal capa-
bility by filtration in accident situations. Combustible gas control is accomplished for design basis accidents using a hydrogen recombiner system, and for degraded core accidents using a dual system of distributed hydrogen igniters. The latter is designed to prevent the average hydrogen concentration in containment from reaching 10 volume percent in case of 100% reaction of the active fuel cladding with steam during a degraded core accident. The igniters have diverse power supplies including batteries via DC to AC inverters.

Figure 3-1 System 80+ Spherical Steel Containment
Water storage and management in containment utilizes two major reservoirs; the IRWST and the Holdup Volume Tank (HVT). The IRWST serves as the water supply for the various safety systems as described above, provides the primary heat sink for primary coolant system pressure relief discharges, and is the source of water for the cavity flooding system (used to flood core debris that could discharge from the reactor vessel in a severe accident). The HVT collects water that drains from high elevations in the containment and is connected to the IRWST via a series of spillways. It also serves as a holding volume for the cavity flooding system.

Severe Accident Challenges Addressed

The variety of features described above were incorporated to delay or prevent challenges to containment integrity in the rare case that a severe accident should develop. It appears that the reactor depressurization systems could be effective in avoiding significant DCH events. Mixing in the large containment volume plus the presence of igniters should avoid the buildup of appreciable hydrogen concentrations. The large volume and internal heat sinks will delay long term pressure buildup in station blackout cases, thus providing more time for restoring power or connecting extra heat sink capability. The large reactor cavity combined with the cavity flooding system should promote spreading and cooling of severe accident core debris and thus limit basemat attack and noncondensable gas generation from MCCI. Thick walls in this location and the presence of intervening structures should protect the containment boundary from missiles that might result from FCI events during the core quenching process. The containment spray and the annulus ventilation systems represent active means for achieving control of radioactivity. Natural plateau and aerosol settling processes will also help to remove and immobilize airborne radioactivity during accidents. The available documentation indicated no plans or provisions for intentional venting of the containment.

Design/Development Status

The System 80+ Standard Safety Analysis Report is complete. Final design approval from the USNRC is expected in August, 1994 and design certification at a later date. Some of the System 80+ design features are included in plants currently under construction at sites in the Republic of Korea.
3.2 CANDU-3

General Characteristics (3c)

CANDU-3 is the latest and smallest version of the CANDU Pressurized Heavy Water Reactor (PHWR) with a net electrical output in the range of 450 MWe. All key components of CANDU-3 (steam generators, pressure tubes, fueling machines) are identical to those proven in service in operating CANDU stations except only half as many of these are needed, enhancing maintenance capability and reducing maintenance cost. CANDU-3 has a modular design. The internal layout is as shown in Figure 3-2. As with any other CANDU reactors fueling is on-line but single-ended. Two independent shutdown systems (fail-safe, spring-assisted shut-off rods and neutron-absorbing poison injection system) are used. Emergency core cooling system operation is in two parts; short-term injection consists of high pressure and low pressure injection stage while long-term cooling is through recirculation of light water/heavy water coolant from the reactor floor back into the primary system via heat exchangers and pumps.

CANDU-3 containment building is a 41 m outside diameter, 55 m high, reinforced concrete structure with a wall thickness of 1.2 m. It has a low carbon steel liner, 6 to 8 mm thick, attached to it with studs. The design pressure of the building for the Service Load category is 230 kPa(g). Building penetrations such as air locks, essential piping for core heat removal and electrical penetrations are equipped with isolation valves that close automatically when an increase in containment pressure or radioactivity level is detected. Air coolers in the building that are used to control temperature during normal operation provide a heat sink for containment atmosphere. These coolers are environmentally qualified for operation after a LOCA. Another feature is an on-line gross-leakage monitoring system to detect any significant breach of containment while the reactor is in normal operation. Hydrogen control is planned through dispersal and recombination.

Severe Accident Challenges Addressed

This containment has been designed to maintain its integrity in the event of a Design Basis Earthquake that can lead to ground accelerations up to 0.3 g. Analysis has further indicated that the containment integrity will be maintained for bounding primary-side LOCA coincident with a loss of emergency coolant injection (LOECI), and for secondary side main steam line break accident coincident with a loss of air coolers. The design also accounts for ignition of hydrogen that may be released in LOCA plus LOECI accidents. As for more severe accidents, estimates of fre-
Figure 3-2  A Schematic of the CANDU-3 Reactor Containment and Internal Arrangement
quencies are lower (<10-5/yr) for CANDU reactors in view of the automatic reactor control system, the two, simple, independent, rigorously tested, shut down systems and the liquid moderator system that acts as a dispersed emergency heat sink. CANDU-3 is designed to reduce this frequency to a target that is lower than 10-6/yr. Further, CANDU plant response to increasingly severe accidents is gradual. In a severe LOCA plus LOECl accident, the liquid moderator that is present in large amounts in the calandria vessel surrounding the fuel channels, prevents fuel melting and maintains fuel channel integrity even if the channels contain no coolant at all. If, in turn, the moderator is lost, water-filled shield tanks that surround a large part of the calandria vessel can (in most failure sequences) prevent or delay for many hours melt-through of the shielding system.

Design/Development Status

Conceptual design of all systems was completed in 1990. Standard product detailed design has been underway for the past two years and is scheduled for completion in 1996. Conceptual designs for future containment-related developments for CANDU reactors, aimed at enhancing passive safety, are being considered to evaluate their impact. These include:

a vacuum tank occupying a significant fraction of containment volume to provide sufficient pressure relief to maintain sub-atmospheric pressure during a LOCA, and

a steel containment with a water jacket along the inner wall. The thermal capacity of the stored water is intended to provide sufficient cooling for the first day or two after a LOCA. A concrete shroud will surround the containment serving as a shield for radiation and shield the containment vessel from external events. The annulus between the containment and shroud will provide a flow path for natural convection of air for eventual heat rejection from the water jacket.

An initial evaluation of these concepts has shown the capital cost to be competitive and the core-melt frequency to be significantly reduced. A more detailed evaluation is in progress.
4.0 Basic Engineering Phase Design

4.1 AP600

General Characteristics (4a)

The AP600 is a simplified passive light water reactor plant having a power output of 1940 MWe (600 MWe). The nuclear steam supply system is a Westinghouse-designed pressurized water reactor (PWR) consisting of a reactor pressure vessel, two vertical steam generators, a large pressurizer, and associated pumps, valves, and piping. The reactor core is located low in the pressure vessel to minimize core temperatures during loss-of-coolant accidents. There are no reactor vessel penetrations below the top of the core. Passive core cooling systems are provided which are designed to maximize use of natural driving forces such as pressurized nitrogen, gravity flow, and natural circulation flow. An automatic reactor coolant system depressurization feature (ADS) is included to enable effective operation of the gravity driven core reflood system.

The containment building is a cylindrical steel containment vessel with elliptical upper and lower heads. It has a diameter of 130 ft (39.6 m), a height of 189 ft-10 in (57.9 m), a design pressure of 45 psig (0.31 MPa), and a design temperature of 280 F (138 C). The bottom head is embedded in concrete up to 39.5 ft (12.0 m) above the bottom of the concrete basement. Above this it is effectively an independent, free-standing structure. The containment building is surrounded by a seismic Category 1 reinforced concrete shield building (see Figure 4-1). These two buildings along with an auxiliary building, form the nuclear island. The foundation for the nuclear island is an integral basement which supports these buildings. The cylindrical section of the shield building serves as shielding and as an external missile barrier for the containment vessel. It also structurally supports the shield building roof.

Both the containment vessel and the shield building are integral parts of the passive containment cooling system (PCS). The other parts of the PCS are the passive containment cooling system water storage tank (PCCWST), the air baffle (located between the steel containment and the concrete wall of the shield building), air inlets (located at the top of the shield building wall), an air diffuser/exhaust (located in the center of the conical-shaped roof), and a water distribution system (mounted on the outside surface of the steel containment vessel).

In the event of an accident or spurious actuation the PCS drains the PCCWST through the distribution system. The water runs down the outside of the containment vessel and is carried out of the
Figure 4-1 Schematic of Major Structures for AP600 Containment
shield building by a drainoff system, thus removing heat that was picked up during contact with the steel containment vessel. The combination of air inlet, air baffle, and air exhaust also provide a pathway for natural circulation of cooling air through the annulus between the containment vessel and shield building for the further extraction of heat from the containment. In this case heat is transferred to the air between the baffle and the steel containment. The heated air rises and exhausts through the air diffuser in the shield building roof. Cooler outside air is drawn in through the air inlets and travels down between the air baffle and the shield building wall to complete the natural convection loop.

The PCS operation is automatically initiated upon receipt of a high containment pressure signal. It is capable of removing sufficient thermal energy including subsequent decay heat from the containment atmosphere following a design basis event resulting in containment pressurization such that the containment pressure remains below the design value with no operator action required for three days. The PCS is designed to reduce containment pressure to less than one-half its design pressure within 24 hours following a postulated LOCA.

Other major equipment/safety features connected with the containment include the in-containment refueling water storage tank (IRWST), the containment isolation system, the reactor cavity region, the hydrogen control system, and sump pH control.

The IRWST, which is part of the passive core cooling system, is located above the elevation of the reactor pressure vessel nozzles. After ADS is completed water from the IWRST will flow by gravity to the reactor vessel.

The containment isolation system automatically isolates piping that penetrates the containment when an uncontrolled release of radioactive material via this pathway is threatened. The majority of piping and other containment penetrations are located in the middle part of the annulus between the steel containment and the shield building while the various isolation valves are located in compartments in the auxiliary building which is adjacent to the shield building.

The reactor cavity region which is around and under the reactor pressure vessel is about 36ft (11m) high and occupies a large area at the bottom of the containment. The floor and walls (up to a height of 26.5ft (8.1m) are reinforced concrete while above this they are structured modules (a combination of steel plate and concrete). A stairway provides access to the cavity.

The containment hydrogen control system consists of concentration monitors, electric hydrogen recombiners (for hydrogen control during and following a design basis LOCA) and a collection of hydrogen igniters (for hydrogen control during and following a severe accident). Containment structures are arranged to promote
mixing of the atmosphere via natural circulation.

The sump pH control system is a part of the passive core cooling system. It is capable of maintaining the desired post-accident pH conditions in the recirculation sump water after containment floodup.

Severe Accident Challenges Addressed

A variety of potential severe accident challenges to containment integrity can be accommodated in the AP600 design. The chance of a significant direct containment heating (DCH) event is negligible with the presence of the ADS feature since high pressure core melts are essentially avoided. The presence of distributed hydrogen igniters and a geometry favorable to mixing of the atmosphere in the containment protect against the buildup of hydrogen to concentration levels that might support detonation events. The hydrogen control system design is based on preventing containment hydrogen concentrations from exceeding 10 volume percent if 100 percent of the active fuel cladding should react with water. The general design of the containment and its passive cooling system (PCS) is such that the heat liberated from hydrogen burning along with that from other internal energy sources (decay heat, zirconium-water reaction, etc.) can be accommodated. Specifically, it has been stated that with only air cooling via PCS, the containment pressure would not exceed its ultimate pressure during a core melt scenario. This of course requires that no appreciable long-term generation of noncondensable gases should occur.

The AP600 design relies on a large reactor cavity and water flooding by gravity to promote spreading and quenching of molten core debris that might breach the lower head of the reactor pressure vessel in a severe accident. This should avoid significant molten corium-concrete interaction (MCCI) which would be the principal potential source of noncondensable gas production. In addition it should prevent any threat to the integrity of the reinforced concrete basement. Energetic fuel-coolant interactions (FCI) that might possibly accompany molten core debris quenching could generate missiles but the intervening barriers between the reactor cavity and the steel containment shell should protect the latter from missile impact damage. Natural processes such as plateout, settling, and dissolution will operate to remove and immobilize most of the radioactive material that might be discharged to the containment atmosphere in the early stages of a severe accident. Sump pH control should assure that radiiodine will remain in a stable non-volatile state in the water for the duration of the accident. Noble gas fission product activity will remain airborne in the containment throughout the accident but its release to the environment will be low if the various features discussed above function properly to preserve the low leakrate integrity of the containment boundary.
Design/Development Status

The standard safety analysis report for the AP600 was submitted to the USNRC in late June, 1992. Some testing programs are in progress to support the final design and licensing process. This includes evaluation of external cooling of the RPV (after cavity flood-up) as a means to achieve in-vessel termination of core melt. At present the scheduled date for obtaining USNRC final design approval of AP600 is January, 1996 with design certification coming later.

4.2 SBWR

General Characteristics (4b,4c,4d)

The SBWR (Simplified Boiling Water Reactor) is being developed as a cooperative effort involving an international team of BWR vendors, architect-engineers, utilities, universities, and research organizations. The nuclear steam supply is a General Electric 600MWe direct cycle BWR which is housed in a pressure-suppression type of containment. The reactor design uses natural circulation to maintain coolant flow through the core thus eliminating any recirculation pumping or piping. The use of simplified safety systems have resulted in the elimination of all safety-grade pumps and diesel generators. These safety systems use only natural forces or stored energy to cool the core and remove decay heat. The key safety systems include a depressurization system (ADS) and gravity-driven cooling system (GDCS) to provide core cooling. An isolation condenser system (ICS) is used to remove decay heat for all transients. Decay heat is removed from containments using a variation of the isolation condenser that requires the operation of no valves or other moving parts. All systems employ diversity and/or redundancy to enhance reliability.

A diagram which shows the main features of the SBWR design is given in Figure 4-2. The reactor pressure vessel is located in the upper drywell region of the containment. It is approximately 20ft (6m) in diameter, about 82ft (25m) in height, and has a design pressure of 1250psig (8.6MPa). In case of a LOCA emergency core cooling is provided by the GDCS in conjunction with the ADS. When the proper water level signal is received the ADS will fully depressurize the reactor vessel so water can be delivered from the gravity-driven cooling pools to the downcomer annulus region of the vessel. The short-term system supplies flow from three separate GDCS pools (348 cubic meters each) while the long-term system supplies gravity-driven flow from the suppression pool (3225 cubic meters) via a different set of lines.

High pressure decay heat removal and coolant inventory control is
performed with the ICS. It consists of three independent loops each containing a heat exchanger (isolation condenser) that condenses steam on the tube side (which drains back to the vessel) and transfers heat by heating/evaporating water in the isolation condenser and passive containment cooling pool. This pool has an installed capacity that provides at least 72 hours of decay heat removal before replenishment is needed. The heat removal capacity of the ICS (with two of three IC loops in service) is at least 60MWe at normal reactor pressure (1050psig or 7.24MPa) with saturated steam.

The primary containment is a cylindrical reinforced concrete, steel-lined structure. The pressure suppression pool, equipped with a weir wall and horizontal vents, is located outside and surrounding the drywell. The reactor cavity region, which is bounded by the basemat and the reactor pedestal walls, contains the fine motion control rod drive assemblies and the under vessel servicing platform.

The primary containment has a design pressure of 55psig (0.38MPa) and a design leak rate of 0.5 volume percent/day. The total drywell free volume is about 5490 cubic meters and the pressure suppression chamber free volume is about 3790 cubic meters. The reactor is operated with the primary containment inerted, using a purge and nitrogen addition system, to keep the oxygen concentration below 4%. This is backed up by a flammability control system which consists of low power igniter assemblies distributed around the containment to burn hydrogen as required to control excessive buildup of radiolytically generated oxygen.

Heat removal from the containment is accomplished with the passive containment cooling system (PCCS). Like the ICS this system also employs steam condensers located in the large isolation condenser/passive containment cooling (IC/PCC) pool. However unlike the ICS it contains no valves or other moving components. The PCCS consists of three independent loops which are directly open to the containment. The PCC condenser in each loop contains a drain line for delivering condensate to the GDCS pools and a vent line for venting noncondensables that accumulate at the top of the condenser to a submerged outlet in the suppression pool. Each of the three PCC condensers is designed for 10MWe capacity. Together with the pressure suppression containment system the three PCC condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without makeup to the IC/PCC pool.

Other primary containment safety systems include the flooder system and the venting system. The flooder system was incorporated to provide automatic flooding of the lower drywell region in the unlikely event of core debris discharge from the reactor vessel. The system actuates on high lower drywell floor temperature. It consists of three lines that connect each of the GDCS pools to drywell connecting vents. The volume of water in the GDCS pools is capable of flooding the RPV and lower drywell to the top of
active fuel. The venting system allows operators to manually vent the containment from the suppression chamber air space, if required to avoid overpressure failure. This action provides scrubbing of the vented gases by the suppression pool. The set-point for manual venting is 120psig (0.93MPa). After containment pressure has been reduced by venting the operator can re-close the venting path and thus re-establish containment integrity.

The primary containment is surrounded by a multiple compartment reactor building that houses the various auxiliary facilities needed for reactor operation. Offgas from the IC/PCC pool is vented to the atmosphere from the top of the reactor building.

Severe Accident Challenges Addressed

A number of potential severe accident challenges to containment integrity can be accommodated in this design. The ADS feature should reduce the probability of significant DCH events to negligible levels. The use of a nitrogen inerted atmosphere in the primary containment and the igniter system should avoid serious challenges from uncontrolled combustion of hydrogen and the build-up of excessive oxygen during an accident. The passive containment cooling system should provide for efficient rejection of internal energy (stored heat, decay heat, and chemical reaction heat) from the system for the accident duration. The size of the reactor cavity region along with the considerable structural masses located there would tend to promote spreading and partial quenching of entering core debris that might result from a core melt event. Further quenching is expected by flooding with water from above. This should prevent significant basemat attack by MCCl and the accompanying pressurization of the containment from noncondensable gas buildup. Additional protection against overpressure is provided by the manual venting system. The thick walls of the reactor pedestal along with the vessel itself should protect the containment boundary from missiles that might result from FCI events that could accompany core debris quenching in the reactor cavity. Suppression pool scrubbing should be a major passive radioactivity retention process during severe accidents and other natural processes like surface plate-out and aerosol settling will operate to reduce suspended radionuclide concentrations. Releases from the low leakage containment should be further attenuated by natural removal processes in the multi-compartmented reactor building.

Design/Development Status

The SBWR design/development process has been underway for several years. Various test programs have been conducted to support this effort related to performance of both the gravity driven cooling system and the isolation condensers. Some large scale and full
scale tests are still in progress. The standard safety analysis report for the SBWR was issued in August, 1992 and the development team is anticipating receipt of final design approval from the USNRC by May, 1996 with design certification coming at a later date.

4.3 MS600

General Characteristics (4e,4f)

The MS600 is a simplified PWR developed by Mitsubishi Heavy Industries which utilizes the concept of hybrid safety systems (combinations of active and passive safety systems). The reactor has a power output of 600MWe and is a 2-loop PWR having top mounted in-core instrumentation, horizontal steam generators, and special reactor coolant pumps. The active safety systems are used to terminate higher probability DBAs while the passive safety systems are in place to terminate low probability DBAs like large LOCA. A diagram of the hybrid safety system configuration is given in Figure 4-3. The circle in this figure represents the spherical primary containment shell. The passive safety systems consist of an ADS, advanced accumulators, gravity injection tanks and the horizontal steam generators. For the MS-600 steam generator cooling is used as part of the depressurization process. The system therefore consists of the primary and secondary depressurization valves and the horizontal steam generators. The primary valves blow down from the pressurizer steam volume into the gravity injection tanks. The secondary valves discharge to the atmosphere through relief valves on the main steam lines. The advanced accumulators employ a fluidic flow control device which provides a high initial flow rate followed by a prolonged injection at a much lower rate. This matches the flow rates required in a large LOCA.

Gravity injection tanks have been provided which inject water when the RCS pressure has been reduced to close to containment pressure. Water spilling from the break will completely submerge the RCS.

Horizontal steam generators have been adopted for the MS-600 to provide natural circulation cooling under accident conditions. The horizontal arrangement prevents gas bubbles forming in the U-tubes, which could obstruct natural circulation.

The secondary side of the steam generators is supplied by gravity from a condensate storage tank. No operator action is required for three days. After three days, operators refill the condensate storage tank for continuous decay heat removal and available active systems are put in service. The MS-600 uses a double containment (see Figure 4-4) consisting of a steel primary containment and a concrete-filled steel secondary containment. The pri-
Figure 4-3 Configuration of Hybrid Safety Systems for MS-600

Figure 4-4 Diagram of the Containment System for the MS-600

- Double containment vessel (steel + concrete filled steel)
- Passived annulus system
  (Conventional) - Make annulus portion negative pressure by fan and ventilate through charcoal filter
  (Passive) - Make annulus portion leaktight
  - Leak gas is ventilated through charcoal filter without fan
Primary containment is a 51m steel sphere which gives a large operating floor space. The spent fuel pit and its cooling system are located inside containment, together with the gravity injection tanks, which are also used as refuelling water storage tanks. No containment spray system is needed nor provided for the primary containment. The secondary containment, or shield building, is a concrete-filled steel plate structure which can be built quickly and is leaktight. The annulus inside this leaktight structure is vented through a charcoal filter to reduce any radioactive release to atmosphere. The annulus vent system consists entirely of passive components. This type of secondary containment also provides good protection against external missiles.

Severe Accident Challenges Addressed

The available documentation on this plant indicate a relatively large double containment system that is well equipped with passive core cooling and containment cooling features. These should be effective in terminating large LOCA's and similar design basis level accidents. However, it is not clear how effective they would be in core melt situations. The ADS capability would address the issue of DCH but insufficient information is available concerning protection against potential hydrogen detonations, noncondensable gas production, basemat attack, missiles from energetic FCIs, and possible containment bypass events such as large scale steam generator tube ruptures.

Design/Development Status

The conceptual designs have been completed and the basic design work for the MS-600 is now in progress. Analytical and test programs are underway to confirm the suitability of the design. Preparation of a safety analysis report is scheduled to begin soon. A detailed design and design certification test program is scheduled for completion by 1997.
5.0 Conceptual Design Phase Designs

5.1 L.I.R.A. Containment

General Characteristics (5a,5b,5c)

A containment system known as L.I.R.A. has been under study at ENEL (Rome, Italy) with the general design objective of limiting the release of radioactivity to the environment from any accident to less than one-millionth of the core inventory. The design uses substantially passive features to absorb energy and maintain a low leakage system. This discussion pertains to a containment system for a PWR type reactor but with some modifications it could be adapted to BWRs and HWRs as well.

A diagram of the proposed containment system is shown in Figure 5-1. The volume and design pressure of the primary containment are similar to that of present large dry containments (e.g. 70,000-80,000 cubic meters and 0.3-0.4MPa for a 1000MWe plant). However L.I.R.A. also adopts the principle of vapor suppression with the primary circuit located inside a drywell that is surrounded by a wetwell. The primary containment is a steel-lined reinforced or prestressed concrete structure having a leak rate in the range of 0.2%-0.5% per day at design pressure. It is surrounded by a reinforced concrete secondary containment structure which shares a common basemat with the primary containment.

The pressure suppression pool contains about 8000 cubic meters of borated water (for a 1000MWe reactor) which should absorb the internal energy of the core and primary coolant, the energy released by fuel cladding oxidation, and the decay power for 24 hours without reaching the boiling point. Vertical discharge pipes from the drywell to the pool are shown in Figure 5-1 but horizontal pipes within a weir wall could also be used. Conceptually, both pressurizer relief discharge and steam generator pressure relief discharge would be piped to the suppression pool. The suppression pool would also serve as a reliable radioactivity scrubber during design basis and severe accidents.

The reactor would be equipped with an ADS feature which could discharge to the drywell and/or directly to the suppression pool. The primary containment is large enough to accommodate an RHR system and thus reduce the number of active containment penetrations during an accident. The proposed containment is also equipped with both hydrogen igniters (DC powered) and a collection of catalytic foils for hydrogen recombination during and following an accident. In the design shown here the large reactor cavity (about 250 square meters) is fitted with a slab of graphite (made up of adjacent blocks 1m high)(5c). The purpose of this design is that of spreading the molten debris into a thin layer, and of solidifying and cooling it down by means of the
Figure 5-1 Schematic Arrangement of the L.I.R.A. Containment
heat capacity and high conductivity of the graphite slab. After this initial solidification the cavity is flooded by connecting it to the lower part of the suppression pool through the actuation of either active (valves) or passive isolation devices (fusible plugs or fusible links). In this manner molten fuel-coolant interactions should be avoided along with thermal attack of the concrete containment basement. As an alternative to the flat slab of graphite, a solution based on a stack of staggered graphite beams has been proposed (5b) for cases where, for civil engineering reasons, a more standard cavity size (e.g., 50 square meters) should be preferred.

The large water inventory of the suppression pool allows deferral of actuation of the normal (active) pool cooling system for two or three days. If it were desirable to rely on passive systems even after this time, the L.I.R.A. design could accommodate a filtered venting system, which should have a filtration efficiency of 99.9% for both aerosols and iodine.

Severe Accident Challenges Addressed

As suggested in the above survey this containment concept was developed with particular attention to meeting severe accident challenges. The ADS capability of course addresses the DCH issue quite effectively. The presence of hydrogen igniters (above the suppression pool) and backup catalytic foils (in the containment dome region) should protect against the buildup of appreciable hydrogen concentrations in the main volume. Within the drywell volume steam inerting should minimize the probability of energetic hydrogen burns although igniters or catalytic foils could be added if necessary. The proposed containment has sufficient heat capacity to absorb the expected energy sources for a period of at least 48 hours before overpressurization would become a concern. This provides time to establish a positive heat rejection capability, but if that were unsuccessful filtered venting use would permit controlled venting operations. The issue of MCCI and its attendant problems of basement attack and noncondensable gas generation is in principle addressed by the slab of graphite (or stack of graphite beams) and subsequent bottom flooding in the reactor cavity region of the L.I.R.A. conceptual design. The bottom flooding plan should also avoid high-energy FCI events. Natural processes (pool scrubbing, plateout, and settling) would be expected to remove and immobilize most of the radioactivity that might be released to the primary containment during a severe accident. The secondary containment structure would provide an additional barrier for escape of radioactivity. Placing the steam generator pressure relief system and the RHR system within primary containment would clearly reduce the probability of containment bypass events.
Design/Development Status

The L.I.R.A. Containment design is currently only in the conceptual stage although most of the features utilize presently available technology. Supporting studies are in progress involving the overall system configuration, the core distribution/cooling system for the reactor cavity, and the development of projected cost estimates.

5.2 SPWR

General Characteristics (5d,5e)

The SPWR (System-integrated Pressurized Water Reactor) has been in the design process at JAERI (Japan Atomic Energy Research Institute) since 1986 as a next generation power reactor. The basic SPWR is a 1100MWt unit with all the primary coolant system components including the steam generator, the main circulation pump, the pressurizer, and the core located within a single pressure vessel. Recently (5f) some information has appeared on an 1800MWt version of the SPWR. The SPWR has no primary loop piping, the water inventory is relatively large, and all other pipes connected to the vessel are located above the elevation of the core. The reactor has no control rods and reactivity control during operation is done by adjusting the boron concentration in the primary coolant. Emergency shutdown uses a passive shutdown system in which highly borated water flows by natural convection to the core region from an in-vessel poison tank.

The reactor is housed in a cylindrically-shaped, pressure-suppression type containment. Figure 5-2 illustrates the general layout as well as important safety systems. These are a combination of active and passive safety systems. For LOCAs the active high/low pressure injection systems (HLIS) would be used first if possible, followed by the passive pressure balanced water injection systems (PBIS) if needed. The HLIS introduces borated water from the condensate storage tank (located outside containment) or from the suppression pool inside containment. The PBIS injects borated water by gravity flow from a tank located at the top of the containment when hydraulic pressure valves open at a designated water level. High capacity ECC systems are not required because only small LOCAs are considered possible in this reactor design. The grace period for restoring core cooling capability is about 3 days after the PBIS is put into use.

Containment pressurization in a LOCA is of course minimized by the presence of the pressure suppression pool which utilizes a weir wall and horizontal vents to connect to the containment (drywell) gas space. The suppression pool is also apparently used to quench overpressure discharges from the reactor vessel, the PBIS tank, and the main steam lines as required. Heat re-
Figure 5-2 Diagram of SPWR and Associated Systems
moval from the suppression-pool (and hence the containment) is by either an active or passive system which is still under development.

Severe Accident Challenges Addressed

The design of the SPWR, like that of the other new/advanced concepts, makes the occurrence of a core melt accident a very rare event. However, if such an accident should happen as a result of multiple safety system failures the SPWR would be exposed to the range of severe accident phenomena discussed earlier. Unfortunately the available documentation provides insufficient information to determine the capability of the containment design to survive the challenges that would arise from most of these phenomena. Areas of specific concern in this case are DCH events, MCCII effects, FCI effects, and radioactivity control measures. The SPWR containment employs nitrogen inerting which should avoid challenges from hydrogen combustion.

Design/Development Status

The SPWR is apparently still in the developmental stage. Various experimental and analytical studies are being done in support of the design effort and some demonstration scale tests are also under consideration.

5.3 HSBWR-600

General Characteristics (5g,5h)

The HSBWR (Hitachi Small Boiling Water Reactor) has a rated capacity of 1800MWe (600MWe). The reactor employs natural circulation in the core and no steam separators. An important safety design basis for the plant was to satisfy current safety criteria regarding DBAs while providing enough margin to allow an adequate grace period for mitigation of severe accidents.

A diagram showing the configuration of the HSBWR-600 reactor and containment system is given in Figure 5-3. The emergency core cooling system is composed of a steam-driven reactor core isolation cooling (RCIC) system and a low pressure accumulator for short-term cooling. The reactor is equipped with ADS to allow effective use of the accumulator whose capacity is sufficient to supply emergency coolant for one day after reactor scram. Several options exist for longer term core cooling. The normal mode
would be use of the residual heat removal (RHR) system (see Figure 5-3). Alternately the accumulator could be manually re-filled by using attachable fire engine pumps. Finally a reflooding line, equipped with a check valve, runs between the suppression pool and the RPV. On opening the check valve the RPV would be flooded to above the top of the core.

The primary containment vessel (PCV) is a steel cylinder and dome structure having a total volume of 6190 cubic meters. As shown in Figure 5-3 it contains a pressure suppression pool equipped with horizontal, submerged vents. The PCV is surrounded by a reactor building with volume of 11,300 cubic meters. The basemat and the lower walls of this structure are constructed of reinforced concrete. The annulus between the cylindrical PCV and the reactor building concrete wall is filled with water making an outer pool. Long term heat removal from the PCV can be achieved by natural circulation in the suppression pool and heat conduction through the steel PCV wall to the outer pool which undergoes gradual evaporation loss. This heat removal from the PCV to the outer pool uses no powered components and can continue for three days without adding water to the outer pool. Means are provided for supplying additional water if required at that time. Experimental and analytical modeling programs have been conducted to demonstrate the successful operation of this heat rejection concept. (5i,5j)

Severe Accident Challenges Addressed

The available documentation on this plant emphasizes the passive heat removal aspects of the design. The redundant and passive core cooling system options combined with the passive PCV cooling design appears quite adequate for DBA type energy releases. However, insufficient information is available to determine the capability of the containment design to withstand the various challenges that can be presented by severe accidents. Issues of concern include large-scale hydrogen production/combustion, MCCl effects, FCI effects, and possible containment bypass. The DCH issue probably has been resolved with the ADS capability, and the suppression pool would provide passive radioactivity removal.

Design/Development Status

The HSBWR-600 is apparently still in the developmental stage. Considerable R&D effort has been applied to study and demonstrate the satisfactory operation of the PCV passive cooling system design.
5.4 EPR

General Characteristics (5k,51)

Design of the EPR (European Pressurized Water Reactor) which is to be a next generation PWR is a joint effort of Framatome and the Power Generation Group of Siemens, coordinated through a common subsidiary known as Nuclear Power International (NPI). Electric utility groups in both France and Germany will have some involvement in the overall effort. At present the power rating of the EPR is expected to be about 4200 MWe (1400 MWe). The general safety objectives involve minimizing the probability of severe (core-melt) accidents while incorporating measures to maintain containment integrity even if a core-melt should occur. The former objective is to be met by improving the quality of primary system components such as the RPV ductility, by increasing the volume of the primary coolant system (including the secondary side of the steam generators) to lengthen LOCA response times, and by improving the reliability of the safety injection and residual heat removal systems. This includes special efforts to assure physical separation of redundant safety systems as well as to minimize effects of external events (i.e., seismic, aircraft crashes, etc.) on plant operations and equipment. Meeting the latter objective involves design of a robust containment equipped with a series of systems to stabilize/cool molten core debris, accommodate noncondensible/combustible gas generation, and provide heat removal to keep the containment leaktight over the long term. Also under consideration is a passive safety condenser system, connected to the secondary side of the steam generators, which would greatly reduce radiological releases from severe steam generator tube rupture accidents.

A simplified diagram of a containment configuration is shown in Figure 5-4. It consists of a large prestressed concrete containment, which can be fitted with a steel liner, and an outer non-prestressed concrete shield building on a common basement. The annular region between the two structures could be equipped with filtration devices to treat leakage that might enter from the inner structure. Thus no direct leaks from the primary containment to the atmosphere would have to be considered. As indicated the containment houses a large in-containment refueling water storage tank (IRWST) which would serve as the water supply for the safety injection system during LOCA events.

Severe Accident Challenges Addressed

The strategy for limiting the effects of hypothetical core melt accidents and the corresponding measures provided have the essential objective to preserve the short and long-term function of
the containment. Efforts have been governed by the following principles:

Core melt phenomena must be controlled in such a way that their side effects, in terms of forces, temperatures and pressure are kept as low as possible.

Corresponding safety measures must be independent from special accident sequences, and their effectiveness must be as easy as possible to prove.

On the basis of the main phenomena to be controlled during accidents the strategy and safety measures can be outlined as follows:

High-pressure core melt accidents shall be eliminated by safely bringing the HP core melt scenario to an LP core melt scenario. This can be achieved by the safety relief valves already installed as part of the accident management
measures, together with additional measures if necessary, to provide additional relief capacity before penetration through the reactor pressure vessel, thus preventing loss of integrity due to high pressure.

In order to maintain a sufficient distance from the hydrogen detonation limit, the formation of hydrogen must be kept as low as possible. This can be achieved by preventing interaction between the corium and the concrete. Accumulation of hydrogen should be eliminated by burning and/or catalytic recombination.

The stabilization of the corium within the containment has been the object of extensive research for quite a long time. After thorough examination, two main solutions for new PWR concept have been arrived at as follows:

a) distribution of the corium within the reactor pit and its stabilization by cooling from above and beneath, or

b) distribution of the corium beyond the reactor pit, with cooling from above only.

To prevent loss of containment integrity as a result of internal pressure use is first made of all the existing heat storage capacities. However, the design will also provide for cooling measures in the long term. The time required to start up these cooling systems should be at least one day. For heat removal from the containment there are several types of solution, e.g. atmospheric or sump cooling. The effectiveness of such systems in meeting the various requirements is still being looked into in detail.

The open literature should be examined periodically to obtain the results of further design work in these areas as it becomes available.

Design/Development Status

Conceptual design work on the EPR is apparently completed. Work is now entering a basic design phase which should last about two years, whereupon a detailed design phase will get underway. Siting applications to French and German licensing authorities is scheduled for 1995 with the intent of beginning construction of actual plants before the end of the decade.
5.5 KfK New Containment Concept

General Characteristics (5m)

R & D studies under the direction of the Kernforschungszentrum Karlsruhe (KfK) have been in progress to investigate the feasibility of containment structures designed to withstand the various static and dynamic loads that might develop during the very unlikely occurrence of a severe accident in a large PWR (1400 MWe). The parameters addressed in the studies include: static overpressure from hydrogen deflagration, dynamic loading due to hydrogen detonation, RPV rupture by either overpressurization or large-scale in-vessel FCI events, the retention of molten core material, passive decay heat removal from the system, and maintaining leak tightness of all containment penetrations. Bounding mechanical and/or thermal loadings or design requirements were estimated for these parameters by analyzing the effects of the mixture of relevant phenomena.

Figure 5-5 illustrates a conceptual containment design based on the results of these investigations. It consists of a steel shell and prestressed concrete structures and is currently designed to withstand an internal static pressure of 2 MPa and a dynamic impulse of 0.2 MPa-s (upper-bound figures stemming potentially, as discussed in the previous section, from hydrogen detonation). The rocket-like behavior of the vessel in case of a high pressure melt-through event must be taken into account when designing the RPV support structure. Impact loading of inner containment structures surrounding the RPV is caused by moving parts of the vessel, while the actuating pressure is caused by the expansion of the water-steam mixture. It can be shown that the kinetic energy of the upward moving vessel head can be absorbed by unbonded prestressed cables that are anchored into the very stiff, hollow-box-type structure formed by the integrated core catcher and the concrete structure on the ground floor. The concrete structures surrounding the RPV could also withstand a catastrophic rupture (longside crack) of the RPV.

Figures 5-5 and 5-6 depict the conceptual design of a core catcher system that can withstand high pressure and basemat erosion. A heavily latticed concrete structure in the upper part of this core catcher cellar is designed to absorb the kinetic energy of the downward moving end-cap missile from the pressure vessel, thus preventing the early destruction of the core retention device located in the lower part of the containment. The molten core is cooled by water evaporation. For this purpose, a water/vapor circulation system is induced by design to operate completely inside the outer containment. With this design, the heat from the molten core is transferred to the steel shell of the outer containment and then dissipated by natural air convection through the chimney formed between the steel shell and the outer concrete shell of the containment. The composite concrete-steel
Figure 5-5 Conceptual KfK Containment

Figure 5-6 Conceptual Molten Core Retention/Cooling Device
wall system shown by the Section A-A drawing in Figure 5-5 reacts to close the gap between the steel shell (about 40 mm thick) and the outer concrete walls (about 2 m thick) when the increasing pressure inside the containment causes the steel to approach the yield limit. However, the support beams prevent full closure thus maintaining a flow path for natural air convection and passive heat removal.

Other conceptual designs have been published to accommodate the several severe accident challenges discussed above (5n,5o,5p) but the design illustrated here reflects the general principles and objectives of the KfK program.

Severe Accident Challenges Addressed

The documentation shows that serious efforts have been made to develop design concepts for integral and very robust containment systems that can accommodate all potential high energy challenges that might occur during severe accidents. The concepts have merit particularly when combined with comprehensive probabilistic safety assessments to assure that events such as failure to isolate or containment bypass are reduced to negligible levels.

Design/Development Status

The KfK work is obviously in the conceptual stage. Supporting analytical and experimental studies have been done on some aspects of the work and more is planned. The commercial development of these concepts will depend upon a number of factors such as siting requirements, compatibility with normal operational needs, compatibility with recovery from lesser accidents, and cost comparisons.

5.6 ICS CONCEPT

General Characteristics (5q)

The Integrated Containment System (ICS) is a product of ongoing research that has been conducted by the Department of Mechanical & Nuclear Engineering at the University of Pisa, Italy. It consists of a primary containment at low pressure and a condensing pool inside a vacuum building which acts as a secondary containment. A diagram of the arrangement is shown in Figure 5-7. The primary containment houses a standard PWR equipped with the full complement of emergency safety features. In principle several primary containments at a site could share one vacuum building,
in a manner that is similar to the arrangement presently used at various power stations in Ontario, Canada.

Pressure buildup in the primary containment during LOCAs would be controlled by venting through the condensing pool in the secondary containment whenever the primary containment pressure would increase above the hydraulic head of the pool vents. The primary containment is also equipped with a spray system to provide post accident heat and radioactivity removal. The atmosphere of the secondary volume is vented through a standby gas treatment system which is downstream of the condensing pool. The system operates as required to keep the secondary volume at subatmospheric pressure. The condensing pool is quite large which provides a passive heat sink and also provides water for the various emergency safety features. The ICS is also equipped with an automatic post-accident containment inverting system using carbon dioxide accumulators. The reactor cavity is fitted with a combination crucible and flooding system to promote controlled dispersal and quenching of molten core debris that might reach this location in a severe accident.
Severe Accident Challenges Addressed

The post-accident containment inerting system should eliminate potential challenges to containment integrity from hydrogen combustion phenomena. Long-term overpressure challenges should be effectively neutralized by the natural venting to the vacuum building and use of the standby gas treatment system as needed to exhaust excess noncondensables. The cavity crucible and flooding arrangement is intended to prevent energetic FCI events and minimize basement attack and noncondensable gas generation from MCCI. The large condensing pool will serve as a passive radioactivity retention site that should be effective in station blackout accidents as well as in other severe accidents. The available documentation contained no mention of ADS capability, which would address the issue of DCH challenges, but it could be easily incorporated with the discharge delivered to submerged spargers in the condensing pool. Another uncertain issue with respect to the ICS concept is the level of protection against containment bypass challenges.

Design/Development Status

Apparently work on the ICS design is mostly conceptual. Some supporting analyses have been performed and more may be planned but no significant effort to further develop or provide strong engineering support for the concept appears in place as yet.
6.0 Summary and Conclusions

In order to obtain a quick reference to the key features of the various containment designs discussed in previous sections the summary given in Table 6-1 may be consulted. The definitions of the several symbols and alphabet characters appearing in the table are given below. The entries in the table were made on the basis of the available information. In a few instances some subjective judgement was needed but an attempt was made to fairly represent the important aspects of each design.

Design Status

A = commercially available
B = detailed engineering phase
C = basic engineering phase
D = conceptual phase
* = research project effort

Containment Type

a = pressure suppression
b = steel-lined reinforced concrete
c = large steel vessel with concrete shield building
d = steel-lined reinforced concrete with concrete secondary building
e = includes an adjacent vacuum building

Severe Accident Challenges/ Phenomena

AFR = accident frequency reduction
DCH = direct containment heating
EFCI = energetic fuel-coolant interactions
HYD = hydrogen combustion events
OP = overpressure protection
HR = heat rejection (short and long term)
DC/BA = debris cooling/basemat attack
FPC = fission product control
CI/CB = containment isolation loss/ containment or suppression pool bypass events

Strategies/Techniques Used for Protection

AFR a) = improvements made to the NSSS
b) = improvements made in active ECC systems
c) = presence of passive cooling method(s)

DCH a) = installed ADS
b) = other protective means (see text)

EFCI a) = presence of robust internal structures
b) = addition of special shields
HYD
a) = installed inerting system
b) = active igniters used
c) = passive igniters used
d) = other protective means (see text)

OP
a) = use of large free volume & heat capacity
b) = use of pressure suppression system
c) = provision for controlled venting

HR
a) = use of multiple active cooling systems
b) = presence of engineered passive cooling system(s)

DC/BA
a) = promote debris spreading and water quenching
b) = installed core catcher device with cooling

FPC
a) = pool scrubbing action
b) = natural deposition processes
c) = installed active spray system
d) = filtration of primary containment leakage and/or vent flows

CI/CB
a) = reduce number of penetrations/ improve isolation systems
b) = enclose interfacing systems in containment

Inspection of the entries in Table 6-1 reveals substantial variations in plant sizes and containment parameters among the thirteen designs that are represented. Also, diverse means are being employed to cope with the various challenges presented by severe accident phenomena, and the full spectrum of these potential challenges is receiving consideration. This shows that in most cases significant resources have been directed towards maintaining control of all accident situations - even those that are well beyond the conventional DBA level.

It is emphasized that the entries in Table 6-1 and the discussion in the report apply to designs and/or concepts as described by open documentation that was available up to approximately June 1, 1993. Changes that may have occurred since then are not necessarily addressed here and the reader will have to consult more current literature for such updates. This is particularly relevant for any design that has not yet reached commercially available status because engineering work is continuing and various changes may take place before the design becomes finalized.
<table>
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<tr>
<th>Plant/Concept</th>
<th>Size, MWe</th>
<th>Design Status</th>
<th>Containment Parameters</th>
<th>Type</th>
<th>Des. Press. &quot;MPa&quot;</th>
<th>Free Vol. cu.m.</th>
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</table>

"MPa" = pressures given are absolute pressures  
ng = not given in available documentation
Table 6-1  Summary of New/ Advanced (cont'd) Containment Design Features

<table>
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<tr>
<th>Plant/Concept</th>
<th>AFR</th>
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<th>EFCI</th>
<th>HYD</th>
<th>OP</th>
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<td>-</td>
</tr>
</tbody>
</table>

* These were mainly containment design efforts but it should be assumed that in commercial service each would be equipped with the best available reactor power system.
7.0 References

(1a) Letter from J. Royen (OECD) to R. L. Ritzman dated April 14, 1992.


(2b) "The New Reactors" Nuclear News, September, 1992, p. 70


(3b) "The New Reactors," Nuclear News, September, 1992, p. 68

(3c) "CANDU-3 personal communication from K.N. Tennankore, AECL-Chalk River Laboratories, February 8, 1993


(5f) "Passive Safe Reactor for the Next Generation" personal communication from J. Sugimoto, JAERI, February 1, 1993.


(5m) H. Hennies & G. Kessler, "Improved Containment and Core Catcher for Future LWRs & FBRs" International Conference on Emerging Nuclear Energy Systems, ICENES '91, June 16-21, 1991, Monterey, California, USA.


APPENDIX

List of Members of
the CSNI-PWG4 Task Group on
Severe Accident Phenomena in the Containment (SAC)
(March 1993)

Austria
Dr. Gert Schniz (FZS)

Belgium
Mr. Baudouin Arien (CEN/SCK)
Dr. Jean Snoeck (Tractebel)

Canada
Dr. Kannan N. Tennankore (AECL)

Finland
Mr. Risto Sairanen (VTT)

France
Dr. Jean-Pierre L'Héritéau (IPSN)

Germany
Dr. Hans Alsmeyer (KfK)
Prof. Dr. Helmut Karwat (TUM c/o GRS)
Mr. Bernd Schwinges (GRS)

Italy
Dr. Leonardo Arnu (ENEA/DISP)
Mr. Arnaldo Turricchia (ENEL)

Japan
Mr. Kikuo Akagane (NUPEC)
Dr. Jun Sugimoto (JAERI)

The Netherlands
Mr. Simon Spoelstra (ECN)

Sweden
Dr. Veine Gustavsson (Vattenfall AB)

Switzerland
Mr. Georg Megaritis (PSI)
Dr. Günter Prohaska (HSK)

United Kingdom
Dr. Brian Turland (AEA Technology - Chairman)

United States
Dr. Sudhamay Basu (USNRC)

Commission of the
European Communities
Dr. Elena Bonanni (JRC Ispra)

OECD (NEA)
Mr. Jacques Royen - Secretary