Containment Systems Capability

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Abstract
This paper provides a summary of the designs and capabilities of principal containment systems associated with BWR, PWR, and PHWR reactors in operation and under construction in the United States, Canada, West Germany, and Sweden. The many conceptual differences in design and modes of operation following accidents are briefly described, with commentary on their evolution and alternatives considered. Specific examples for each reactor system in operation in the four countries are detailed [1, 2, 3, 4]. The containment design differences and requirements are mainly attributable to the fundamental arrangements of the reactor and secondary side systems, and their demonstrated behaviour during normal operation and following accident conditions. However, two other important considerations that strongly influence design are national regulatory requirements [5, 6, 7, 8] and the number of generating units in a station. These broad issues, together with site conditions and proximity to population, dominate containment performance requirements for economic generation of electricity and public safety. Emphasis is focused on the capability of the various systems to meet design basis accidents. However, the TMI-2 incident has caused plant owners and regulators to examine the ultimate capability of containments, far beyond maximum credible accident bases. Postulated severe degraded core accidents, with a predicted frequency several orders of magnitude lower than other recognized world-wide hazards for which protection is provided, are currently under intense scrutiny [9, 10, 11]. This paper describes the status of some of these studies.

Résumé
Ce document passe en revue les principaux systèmes de confinement associés aux réacteurs à eau bouillante, aux réacteurs à eau sous pression et aux réacteurs à eau lourde sous pression en service ou en construction aux États-Unis, au Canada, en Allemagne de l'Ouest et en Suède. On y décrit brièvement les nombreuses différences existant au niveau de la conception et des modes de fonctionnement à la suite d'accidents tout en commentant l'évolution des systèmes et les solutions de rechange envisagées. S'ajoutent à cela des exemples précis pour chaque filière en service dans les quatre pays [1, 2, 3, 4]. Les différences au niveau de la conception et des exigences sont principalement attribuables au montage fondamental du réacteur et des systèmes du côté secondaire ainsi qu'au comportement démontré en cours de fonctionnement normal et lorsque placés dans des conditions d'accidents. Il est à noter que les exigences nationales de réglementation [5, 6, 7, 8] et le nombre de tranches dans la centrale sont deux autres considérations influant considérablement sur la conception. Ces facteurs importants, joints aux conditions du site et à la proximité de la population, viennent au premier rang des exigences d'efficacité des systèmes de confinement en ce qui concerne la production d'électricité et la sécurité de la population. Le document met l'accent sur l'efficacité des divers systèmes à faire face à des conditions possibles d'accidents. L'incident de TMI-2 a cependant amené les propriétaires de centrales et les organismes de réglementation à pousser leurs recherches au-delà des modèles d'accidents plausibles et à examiner l'efficacité ultime des systèmes de confinement. C'est ainsi qu'on étudie actuellement de façon rigoureuse des hypothèses d'accidents graves impliquant la dégradation du coeur du réacteur [9, 10, 11] et d'une fréquence prévisible de plusieurs fois moindre que celle d'autres dangers reconnus dans le monde entier et pour lesquels il existe un système de protection quelconque. Ce document décrit le stade actuel de certaines de ces études.

Keywords: nuclear safety, containment systems, international experience, design and performance of nuclear safety methods.

Containment Functional and Design Requirements

The universally accepted philosophy for providing assurance of nuclear safety in accidents is the adoption of the principle of 'defense-in-depth,' which prevents or limits the release of radioactive material for a wide range of circumstances. 'Defense-in-depth' embodies a multiplicity of physical and chemical actions attributable to station process systems, but specifically includes three often duplicated and diverse safety systems to effect prompt reactor shut-down, ensure continuing and controlled heat removal, and automatically minimize/prevent radioactive release to the environment.

Containment systems are the ultimate line of defence and safety barrier for preventing the escape of radionuclides to the environment. The functional requirements of containment do not differ in principle for BWR, PWR and PHWR systems, but design requirements are significantly different. The design requirements are not only set by the overall arrangement of the primary reactor and secondary systems, but also (importantly) by national code and regulatory differences. A major additional influence that determines containment system designs is the extent, rate, and duration of accident pressure and temperature transients.

During normal operation the function of all containments is to minimize the release of gaseous, liquid, and solid radioactive materials produced during electricity production, and which are not retained in process systems. The objective is to ensure that emissions are as low as reasonably achievable, economic and social factors taken into account. Following an accident, the objectives are to retain radioactive materials released as a result of process equipment failure.

Thus, containment system designs have evolved from the basis that they should provide radiation shielding and retain all of the steam and water discharged following an internal reactor system piping failure. The primary element of containment systems is a practical engineered and economic 'leak-tight' building that covers and encloses the reactor systems. Piping or ventilation systems that might convey radioactive material outside the containment boundary are isolated immediately after an abnormal condition is detected. Sub-systems to reduce pressure in the building also feature in the designs. In some designs these systems include venting to 'gravel' beds or the atmosphere in a controlled manner, to ensure safe regulatory releases are met.

In addition to the provision for internal containment loads, protection of containment, and hence reactor systems, against external loads (i.e., earthquakes, hurricanes, tornadoes, explosions, aircraft impacts, and plant-induced missiles), are also major design requirements.

Pressurized Heavy Water Reactor Design Requirements

In Canada, PHWR containment designs must adhere to the CSA N290.3 standard, which differs only in detail from similar ASME codes. The CANDU containment design requirements are uniquely influenced by the adoption of multi-unit stations (eight units in the case of Pickering NGS, four units per station in other plants) and on-power refuelling, where a single integrated containment system employing negative pressures, dousing water pressure suppression, and a vacuum building is deployed. Elsewhere in Canada and overseas, AECL-designed 600 MWe single CANDU units include similar dousing water pressure suppression systems and filtered air discharge.

The fundamental difference in design of PHWR's and LWR's, namely the physical separation of the primary coolant and moderator system within the PHW reactors, reduces the probability of core melt in postulated severe accidents by orders of magnitude. In essence, the large heat sink provided by the moderator system gives high assurance of fuel channel integrity and prevents gross fuel melting to the extent that meltdown sequences are not generally considered credible [12].

The design of CANDU containment features are influenced by the structure of the Canadian Atomic Energy Control Board (AECB) regulatory requirements. To provide understanding of this influence, the AECB Siting Guide [6] is briefly described. The logic of this guide is based on a two-tier radiation dose limit applied separately to the most exposed individual and to the population. Process failures are judged against a 'single failure' dose limit (e.g., 3 rem to the thyroid of the most exposed individual). In common with LWR systems, these single failures range up to a guillotine break of the largest-diameter heat transport system piping. The limiting frequency of serious process failures (those requiring intervention by a safety system in order to prevent fuel failures) is one per three years. It must be emphasized that 'single failure' in this context is a different concept than that used in LWR licensing logic. In this case it means total failure of a system with no mitigating action by other process systems; only the safety systems can be credited.

The second part of the AECB Guide requires the analysis of 'dual failures,' involving serious process failures with simultaneous failure of one of the safety systems (either the emergency coolant systems or a major containment subsystem) to perform its function. This particular requirement is unique, but not necessarily more demanding than those of other national regulatory jurisdictions. The thyroid dose limit to the most exposed individual from these 'dual failures' is 250 rem. Containment design is strongly influenced by the requirement to meet this dose limit.

Light Water Reactor Design Requirements

The majority of nuclear power reactors in operation and under construction in the world today are either Pressurized Water Reactors (PWR) or Boiling Water Reactors (BWR).

The design requirements of PWR and BWR Containment Systems must adhere to the national codes and regulatory licensing requirements in the country of plant siting. In the United States principles have been developed for steel and concrete structures by the American Society for Mechanical Engineers (ASME) and American Concrete Institute (ACI). Historically, these code requirements for containments and their subcomponents have developed over the thirty years, culminating in an ASME and a joint ACI-ASME Code that caters to the many different combinations of steel and concrete structures that constitute the containment of operating reactors and those under construction. The ACI-ASME code combines a factored load approach with allowable stress criteria for all internal and external load consequences.

The various national codes, in addition to consideration of the ASME (steel) and ACI-ASME (concrete) codes for containment design, performance, and serviceability, reflect geographical, social, political, and regulatory requirements in their own environment. Thus the German 'Kerntechnischer Ausschuss, the Swedish, and the Canadian codes show differences from those in the United States, as well as alternative requirements for external loads such as historical seismicity, siting conditions, and threats of local explosion and aircraft crashes.

The accident internal service load is historically associated with any single component failure in the generating plant, having a frequency typically greater than 10^{-7}, events/yr, which causes a maximum energy, pressure, temperature, and radioactive release. In this regard, the design basis accident for which most containment systems are conservatively designed is the largest double-ended primary pipe rupture, (predicted frequency of 10^{-4}/yr), recognizing that while continued operation of the core cooling system is likely, its full credit cannot be assured in all accident conditions.

Certain combinations of extreme internal/external loads are also typically used in design of containments. Perhaps the most famous one is the combination of LOCA with some level of earthquake. On this issue, there is no general worldwide agreement. In the United States, for example, the largest postulated LOCA has been combined with the largest Safe Shutdown Earthquake (SSE). In other countries, while the combination is considered, it is not necessarily assumed that the largest LOCA and the largest earthquake are coincident. The reasoning for this position is that the reactor coolant system is specifically designed to resist earthquakes; therefore earthquakes do not cause LOCA's, but such an independent event cannot be discounted immediately following. The impact of military aircraft, blast waves, and a turbine wheel rupture impact are also considered in many designs.

The following categories of loads are not normally considered in the design process but have received increasing attention in determining containment performance capability.

The first category includes those loads with a negligible frequency ($< 10^{-7}$ per yr). Such loads would typically include meteorites, large commercial aircraft impact, and volcanic eruption.

The second category involves extreme internal accidents. Most countries typically do not combine LOCA with a secondary system failure as a design basis, although analysis of this combination is often undertaken. Rotating equipment and pipe support failure within containment are also not typically considered. Also, major component rupture, including vessel, pump, steam generator, and pressurizer, are not typically a design basis.

The third category involves the question of the degraded core, the so-called 'Class 9' accident. There are three particular types of containment loads that *might* be associated with such a situation. These loads include degraded cores possibly leading to some melting of containment, steam explosion, and hydrogen generation if it results in deflagration. As a result, containment overpressurization at elevated temperatures due to postulated failure of mitigation systems is under study.

Changing Emphasis on Performance Requirements

Table 1 [10] summarizes the evolution of containment performance criteria. The order listed relates to the growing emphasis that each have received over the last forty years, culminating with containment capability for degraded core accidents.

The criteria for radiological releases were the first to be developed. For all but extremely remote sites, this led to the use of containment systems with acceptable leakage related to site-specific characteristics. Most often these pressure retention containments were freestanding steel or steel-lined concrete structures that, for LWR designs, could be demonstrated to leak considerably less than 1 per cent of the containment volume per day during accident conditions. For multi-unit CANDU systems, where accident source terms and

Table 1: Evolution of Performance Criteria

1	Criteria for radiological releases
2	Criteria for direct radiation doses
3	Protection against external missiles
4	Consideration of degraded cores

energy release into containment are lower, and filtered venting to control long-term releases is deployed, leakage rates of less than 1 per cent per hour have been adopted.

These criteria provided protection against leakage, but not from direct radiation due to radioactive material within the containment after accidents. It was initially assumed that people near the site could be evacuated to minimize their exposure from material inside containment if an accident occurred.

The next criterion added was the requirement of shielding from direct radiation at all but the most remote sites. This led to the widespread use of steel-lined, reinforced or prestressed concrete structures for containment that combined low leakage capability with shielding from possible radiation.

The next important criteria to be added were for protection against external phenomena, such as missiles resulting from tornadoes. Similar criteria were developed relating to aircraft crashes at sites, depending on the frequency of air traffic. These additional criteria made the use of reinforced or prestressed concrete containments, or the addition of a special concrete missile shield, essential.

The fourth set of criteria, associated with degraded cores (or more precisely the need for such criteria) have been under intense scrutiny and debate since the Three Mile Island Unit 2 accident on 28 March 1979. There are two major investigations addressing these issues in the United States. One is a series of programs funded by the NRC on containment integrity. In these programs the behaviour of isolation features, structural capacity of containment, leakage characteristics of mechanical and electrical penetrations, and behaviour of the base mat when subjected to a core melt are being investigated [13, 14, 15, 16]. The other major investigation is the extensive U.S. IDCOR Program [9], which is currently under discussion with the USNRC. Also, intensive studies [2], concentrating on the sequences of core meltdowns and the accompanying accident consequences, have been conducted in the Federal Republic of Germany during the past ten years to ascertain the ultimate capability of containment systems for their operating LWR's. In Canada, important fission product distribution studies concentrating on 'Lessons learnt from Three Mile Island' have resulted in containment design modifications. Also, heavy emphasis on dual failure accidents (e.g., a large LOCA resulting in stagnation cooling conditions plus assumed coincidental containment impairment) continues in that country.

Containment Systems and Component Function

The specific details of existing containment systems depend on the project commitment dates, but their generic nature are a function of reactor type, site location, utility preference, economic considerations, number of units per station, and national regulatory influence. Thus, there is considerable design diversity in existing structures, although basic concepts have not radically changed in more than twenty years.

In the 1940s, the control of public exposure following a design basis accident was provided by the use of large exclusion areas, rather than a containment structure. For example, the Clinton pile at Oak Ridge was associated with a 60,000 acre site, the Hanford production reactors with a larger area, and the U.S. National Reactor Testing Station was located in the Idaho desert. The need to locate nuclear power plants nearer the consumer resulted in containment systems. Early containments were static pressure envelopes with few penetrations. These were not practical for commercial electrical generating stations. Subsequently, active containment structures, with a multiplicity of penetrations designed to close on accident signals to form a leak-tight barrier, evolved. Later, systems were introduced to suppress pressure and temperature within containment following accidents, and also mitigate fission product transport to the environment either by chemical means, controlled filtered venting, or returning leakage to containment by the addition of an outer barrier and pumping circuits. Tables 2 and 3, list respectively, the principal containment systems that are in general use, and those that have seen less use, or have just been studied.

Figure 1 illustrates the many variations of PWR containment, either in operation or committed, by 1972, worldwide. The variations on the three basic systems, (i.e., dry pressure retention containment, ice condenser pressure suppression, and subatmospheric pressure suppression) include single versus multiple barriers, the geometry of the steel or concrete structures, and the nature of allowable structural stress. The dominant system is the medium pressure dry containment with a single pre-stressed concrete cylinder. The majority of these containments are in the United States.

Today, the tendency for PWR containments is towards two dry barriers to fission product release, with provision to filter and vent the annular separation space.

All modern BWR containments are of the pressure suppression type (wet well and dry well) in order to reduce containment volume. This is because, in a design basis accident, BWR's would blow down by far the largest volume of high energy fluids of all water reactor systems. There are three variants (General Electric Company, Mark I, II, and III) of this basic system, with specific differences adopted in West Germany and Sweden. The Canadian PHWR System requires the lowest demand for design basis accident energy containment, due to the physical separation of primary, secondary, and moderator systems.

In more recent times there has been a trend towards

Table 2: Principal Containment Systems

Confinement:	Reactor systems enclosed in a low-leakage building, filtered discharge and negative pressure
Low pressure:	Large diameter hemispherical dome, 35 kPa
Medium pressure:	Low-leakage PWR or steel-lined concrete structure (0.2 to 0.5 MPa); variants in France and U.S.
High pressure:	Low leakage, PWR for pressures 0.5 MPa, steel vessel (FRG, U.S. and France)
Pressure suppression:	BWR system within compact low-leakage steel or steel-lined concrete structure, water and drywell energy suppression
Ice condenser:	A PWR energy-suppression system
CANDU pressure suppression:	Reactor and primary systems within steel-lined pre-stressed concrete containment at negative pressure; pressure suppression by dousing
CANDU shared containment:	Large pre-stressed concrete containment at negative pressure surrounding multi-units connected to vacuum building; pressure control via dousing and filtered venting

Table 3: Other Systems in Use or Studied

Multiple containment:	Two pressure-retaining low-leakage barriers
Pressure release:	Controlled filtered venting and scrubbing
Stronger Containment:	Increased wall thickness for 0.85 MPa
Shallow underground:	Standard containment with 10 m overburden
Deep underground:	Containment 30 m underground, turbine at grade
Increased volume:	Double normal volume, 0.42 MPa pressure
Compartment venting:	Vented to high-pressure structure with douse
Thinned base mat:	Permits core melt to inert gravel bed
Evacuated containment:	Operates at 35 kPa or less

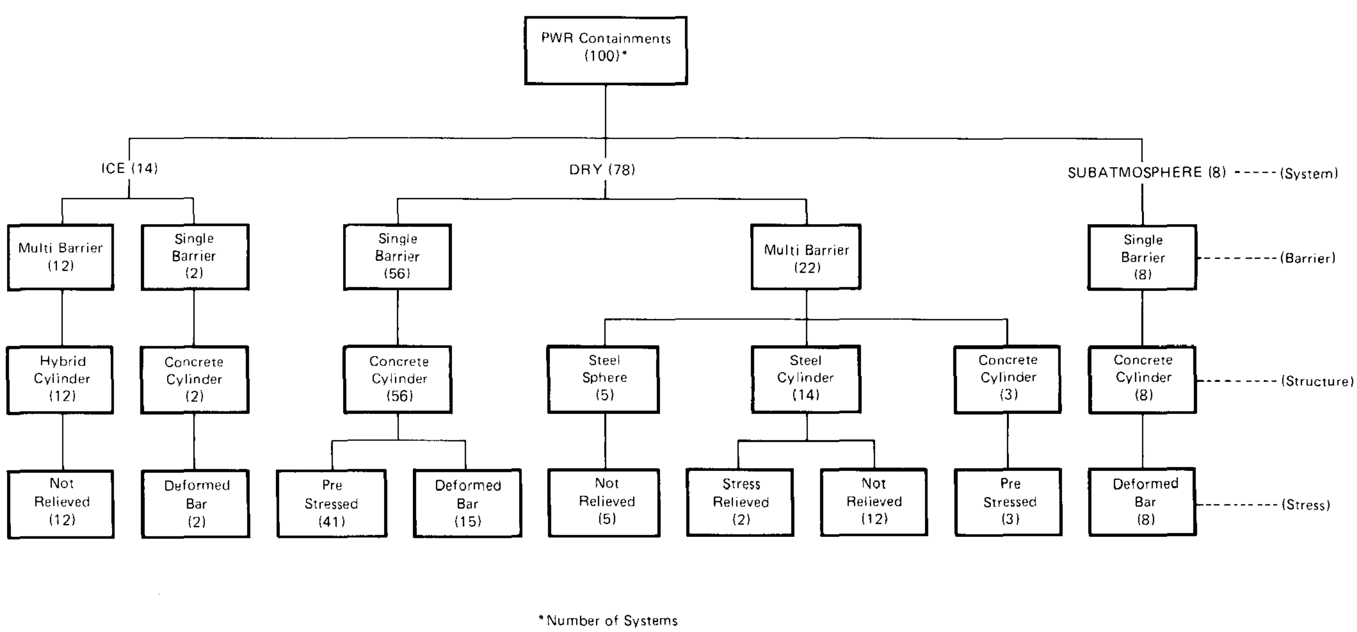

Figure 1 PWR containment designs (committed by 1972).

standardization for PWR, BWR, and PHWR containments, with differences in detail only dependent on the country of siting. Selected designs for the United States, West Germany, Sweden, and Canada follow.

Pressurized Water Reactor Containments

Figure 2 shows a low-leakage, pressure retention design, consisting of a pre-stressed concrete cylinder with a steel liner. A vertical buttress system together with a horizontal ring at the spring line, is used to anchor pre-stressing tendons. The dome and cylinder are separately pre-stressed. This design is widely used in the United States. More recent modifications to the design eliminates the dome ring, introduces partial buttresses in a hemispherical dome, and anchors the wall and some dome tendons at the base mat. As noted in Figure 1, this type of single-barrier containment is the most widely used in PWR stations operating today. Another version of this type of containment is the deformed bar-reinforced concrete cylinder and dome.

Steel containments, either cylindrical or spherical, are widely used in U.S., West Germany and Japan. In these double-barrier designs a concrete biological shield, which also serves to protect against external loads, surrounds the steel containment. The cylindrical design shown schematically in Figure 3 has wide application in the United States and Japan.

A common form of double-barrier containment in the future is expected to be the steel sphere surrounded by a concrete shield building, as developed in West Germany, and also applied to some plants designed in

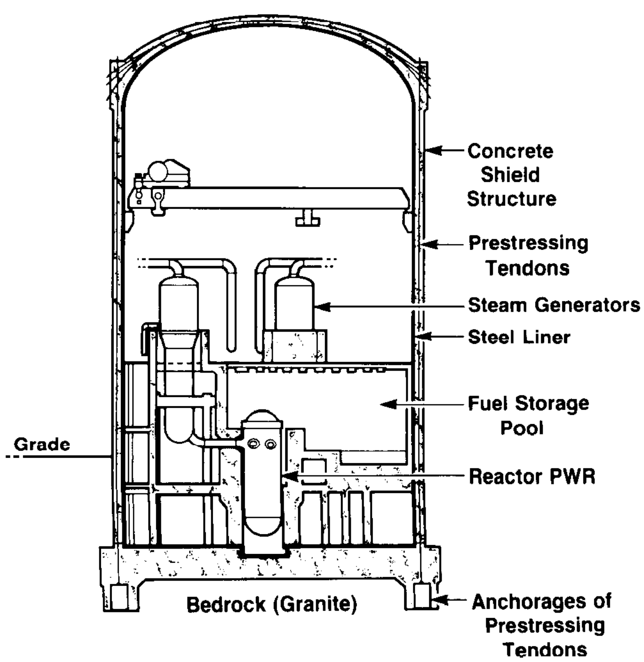

Figure 2 Dry containment: Steel-lined pre-stressed structure.

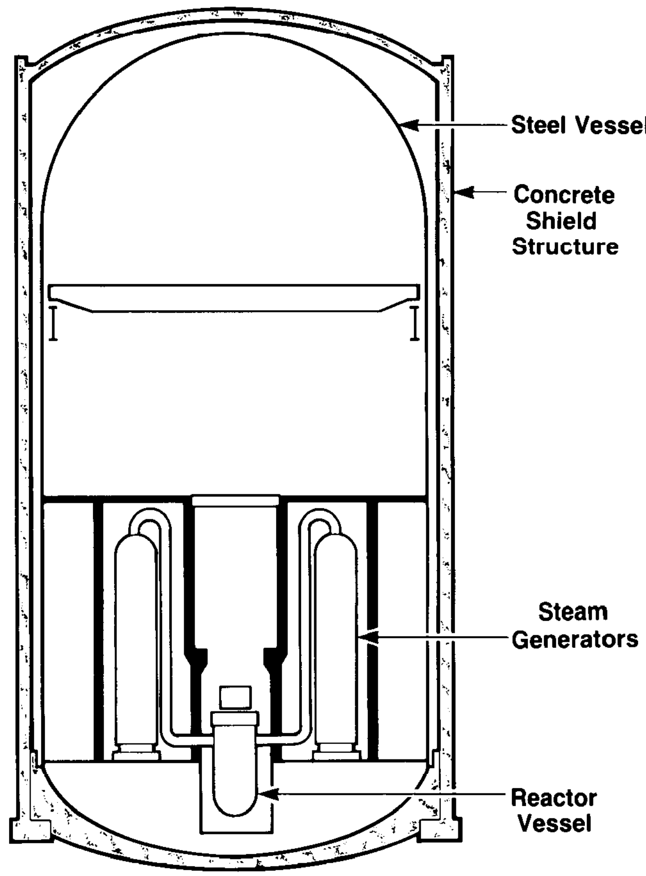

Figure 3 Dry containment: Steel vessel within concrete shield structure.

the U.S. Figure 4 shows a sectional view of the German, 1300 MWe Biblis B plant [2]. The inner detached steel shell of the containment (wall thickness 29 mm) constitutes a passive pressure-tight barrier. The containment sphere has a free volume of 70,000 m^3. The concrete structures within the steel containment (about

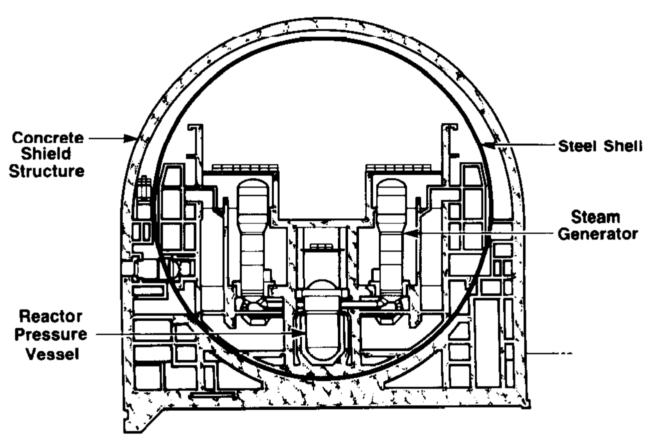

Figure 4 Dry containment: Spherical steel vessel within concrete shield structure.

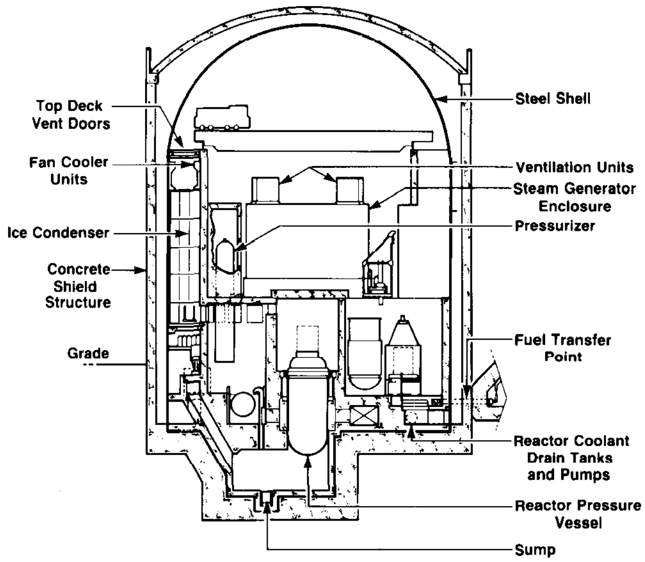

Figure 5 Ice condenser: Steel shell within concrete shield structure.

15,000 m^3) also reduce long-term pressurization by their heat storage capacity, and physically separate safety systems and the irradiated fuel storage pool.

The annulus between the steel containment and the outer concrete shielding (1.8 m thick), which is exhausted through a qualified filter system and stack, provides for additional deposition of radioactive products in the event of containment impairment. A subatmospheric pressure system is designed to direct flows from compartments having lower activity to those with higher activity following any accident.

Another double-barrier annulus concept, developed in France, includes a cylindrical concrete containment lined with steel and an outer concrete shield. Recently, France has developed a design for 1300 MWe plants which does not require the steel liner.

Two types of pressure control containments have been developed for PWR's, the subatmosphere containment (−5.0 psig operating pressure), and the ice condenser. A typical ice condenser containment is shown in Figure 5. Steam and air resulting from an accident is forced by the pressure from the lower compartment through the ice beds where the steam is condensed. The design pressure for this containment is one bar, whereas a PWR dry containment for the same rating would range from three to five bar. However, current economic considerations have limited this design to 1000 MWe units and larger.

Boiling Water Reactor Containments

All modern BWR containments are of the pressure-suppression type, incorporating drywells and wetwells as pressure-suppression chambers. Following a LOCA, the steam/water flow causes a rapid increase of pressure and temperature in the drywell. The pressure difference between the dry and wet wells forces the contained water out of the blowdown pipes, and high pressure steam then flows to the wetwell pool. Steam condensation occurs and non-condensible gases collect in the wetwell airspace or compression chamber. Given the relatively small containment volume of BWR's compared with other reactor systems, this condensation process is the key element in limiting maximum pressures to 3 bar or less.

During the last thirty years there have been progressive changes to the shape, geometry, size, and location of the various suppression chambers relative to the reactor core within containment. The latest Mark III General Electric design is shown in Figure 6. The quenching pool has been moved to the side, whereas in the previous Mark II design it was underneath the reactor vessel. This made it possible to reduce the elevation of the reactor vessel, and created the best compromise with regard to the height of the vessel, its accessibility, and construction of containment. The design shown in Figure 6 uses a steel containment within a concrete shield. However, because of localized dynamic loading from the wetwell during LOCA and Safety Relief Valve discharge, the steel containment was replaced by a hybrid shell in later designs. This hybrid employs a concrete base mat, a concrete shell in the pool region with a steel containment shell above the pool. In future applications of U.S.-built BWR's, it is anticipated that a full-height reinforced concrete shell would be the preferred arrangement, which is also the practice in other countries.

Modern large BWR's typified by the U.S. General Electric Mark III design, the Gundremmingen KRB-2 1300 MWe units in West Germany, and the Swedish BWR-75 1000 MWe units, have steel liners and cylindrical pre-stressed concrete containment structures. The

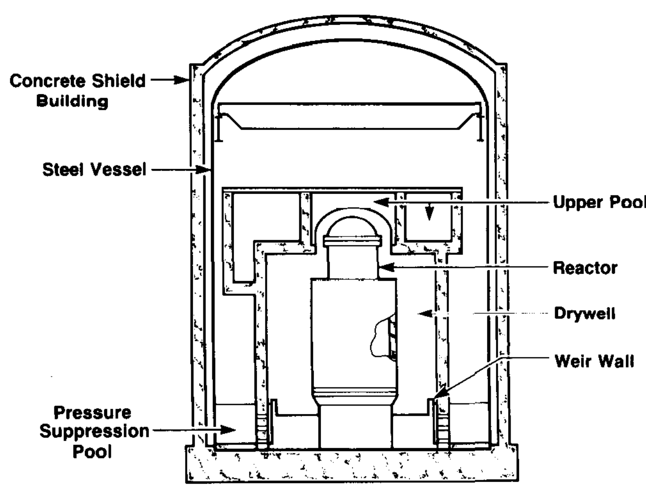

Figure 6 Pressure-suppression containment: Steel vessel within concrete shield building (BWR Mark III containment).

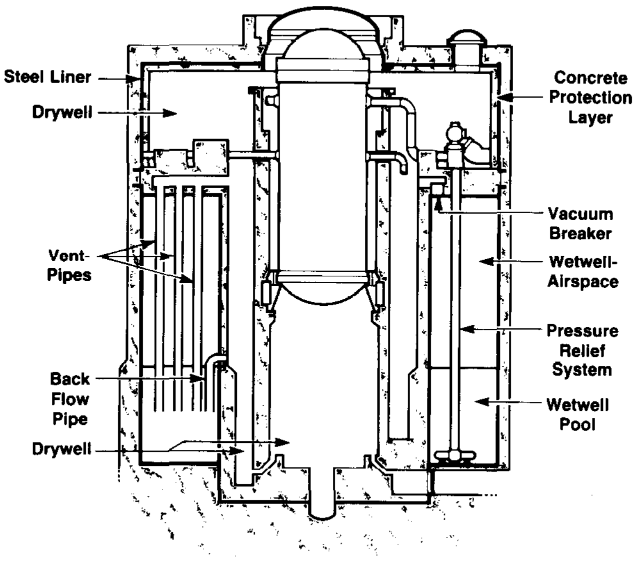

Figure 7 KRB-2 pressure-suppression containment (BWR concept).

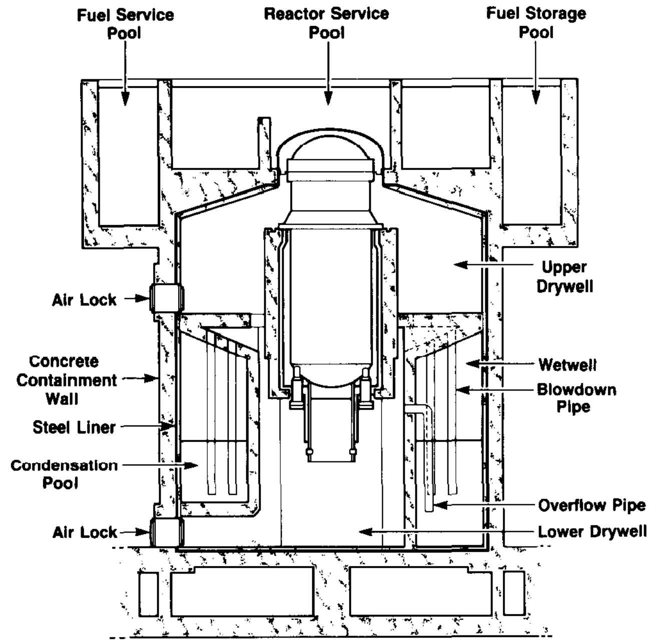

Figure 8 Swedish BWR 75 containment: Steel-lined concrete shield structure.

German and Swedish designs, however, have retained features of the Mark II concept, where the drywell is both under and over the reactor vessel. The overall objective of these systems is to maintain low design pressure with relatively small containment volumes, and to provide for an emergency condenser during plant transients and accidents.

In West Germany, earlier BWR containments were a spherical steel shell. The current KRB-2 design is changed as shown in Figure 7. It consists of a cylindrical pre-stressed concrete structure with an embedded steel liner that is protected by additional concrete. The drywell space surrounds the reactor vessel and heat transport piping extending to the second isolation valve. Many large-diameter vent pipes from the drywell extending into the pool provide the path to condense LOCA-induced steam/water mixtures. A separate pressure-relief system provides for coolant pressure control. The containment is protected from large wetwell overpressures relative to that in the drywell during LOCA by vacuum breaker swing check valves that allow pressure equalization in the two chambers. The suppression system design pressure is typically 4 bar compared with maximum expected LOCA pressures of less than 3 bar. The wall of the reactor building serves as a secondary containment, and the annular space between it and containment is sub-atmospheric, to prevent leakage to the environment. German regulatory authorities require the reactor building walls to withstand an external blast wave of 0.45 bar, a site-dependent earthquake, and the crash impact of military aircraft. To provide further assurance of containment integrity from external events, the reactor buildings are not rigidly joined, apart from the common foundation.

Figure 8 shows a sectional view of the Swedish BWR 75 containment [4], which is a reinforced, partly pre-stressed concrete cylinder provided with an embedded liner of carbon steel. The drywell, wetwell, and blowdown pipes are similarly arranged to the German KRB design, and the entire containment is totally steel lined. A different labyrinth arrangement exists between the upper drywell and wetwell than in the West German design. The containment and reactor building basement structure is different, but each design has no structural tie (other than expansion joints) between containment and adjacent buildings. The steel liner embedment of between 20 to 30 cms within the concrete is deeper than the KRB containment. The upper drywell contains primary and secondary reactor process systems, including main steam, feed water, and containment cooling systems. The lower drywell contains systems such as the control rod drives and recirculation pump motors. The wetwell is an annular enclosure. Blow-out panels in the lower part of the reactor concrete shield provide a path to the lower drywell in the event of a LOCA within the reactor compartment.

Pressurized Heavy Water Reactor Containment

This section concentrates on the CANDU containment system associated with the multi-unit stations in Canada [3]. The single 600 MWe units designed by AECL use similar negative pressure containment (NPC systems, with the omission of a vacuum building.

The NPC1 design concept, (where reactor units are

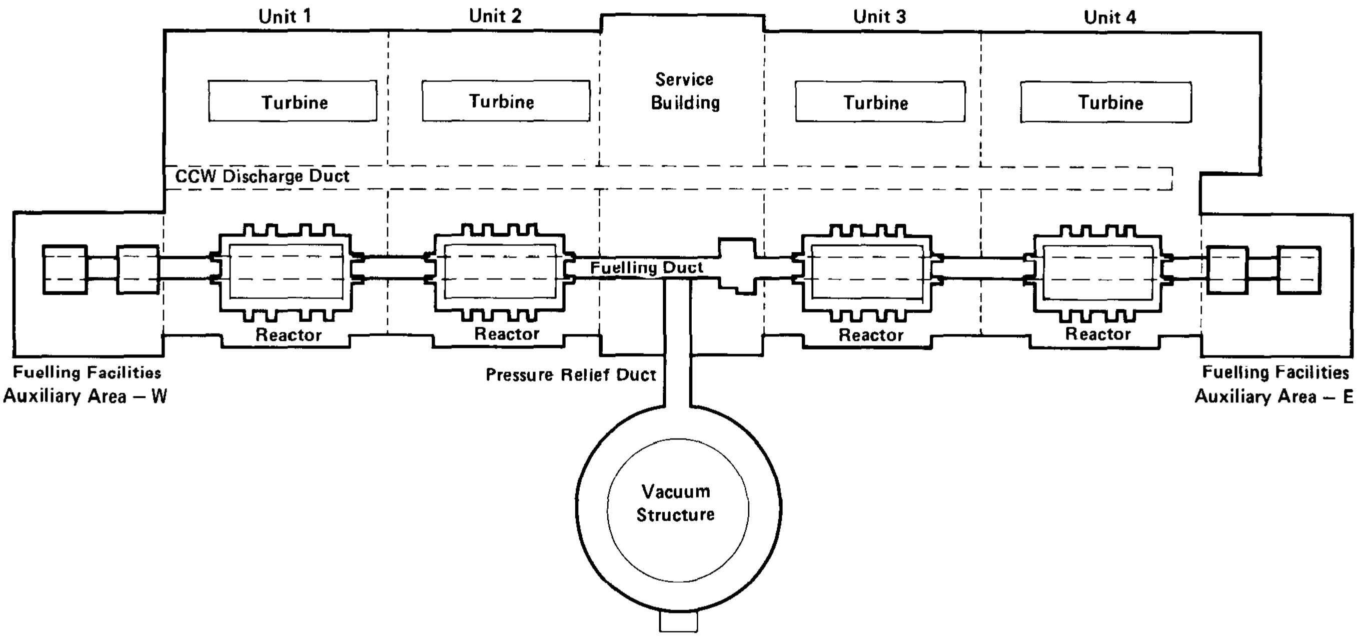

Figure 9 Multi-unit CANDU containment.

isolated from one another), is used in the eight-unit Pickering NGS, which came into service in the period 1971–1985. The second major type (NPC2), was used in the four-unit Bruce NGS A, which came into service in 1977–1979, and in all subsequent four-unit stations.

The prime difference between the NPC1 and NPC2 concepts is that the latter locates most of the supporting process equipment outside the primary containment envelope, although it follows that some equipment must be in secondary confinement areas. Another feature of NPC2 is that the four reactor vaults are interconnected during normal operation due to the choice of common on-power fuelling systems for all units.

The main reason for adoption of the NPC containment concept was increased effectiveness required to satisfy concerns for relative close population siting that existed at the time of the Pickering NGS A project commitment. The NPC2 design was developed primarily to improve maintenance access to process equipment during operation.

The basic operating principle of negative pressure containment is to maintain a negative pressure such that air leakage through the structure is inward. Any discharge required to maintain this negative pressure differential is along defined pathways that can be filtered, treated, and monitored to control releases to the environment.

Figure 9 shows the NPC2 containment envelope, which is normally at sub-atmospheric pressure. In the event of a LOCA, various systems act to provide for short- and long-term pressure and effluent control. The short term period extends from the LOCA, when very fast pressure transients are experienced with possible 'puff' releases of radioactivity, to the re-establishment of sub-atmospheric pressure within containment. The long term period is associated with the initial activation of the Emergency Filtered Air Discharge System (EFADS) until cleanup operations are complete. EFADS is manually activated when containment pressure approaches atmospheric several days after the event. Figure 10 lists the systems that collectively perform the containment function in the two time frames.

The principles of pressure control used in the CANDU NPC2 containment in the short term are 'pressure relief' followed by 'steam-suppression' as depicted in Figure 11. Following LOCA, the reactor vaults and fuelling duct connecting the multi-unit station are pressurized by the resulting high-enthalpy fluid flashing to steam. The extent of pressure rise is limited by the very large volume of the containment envelope. The increase in pressure, acting across the Pressure Relief Valve (PRV) pistons, automatically opens the valves and releases the air-steam mixture into the vacuum building (VB).

The steam suppression function is carried out by a dousing system located in the vacuum building. When the PRV's open and VB pressure rises, water is forced over a weir structure and into spray headers located under the dousing tank. The spray water falls through the steam-air mixture, reduces pressure, and provides for soluble fission-product retention.

The principle of effluent control used in the short

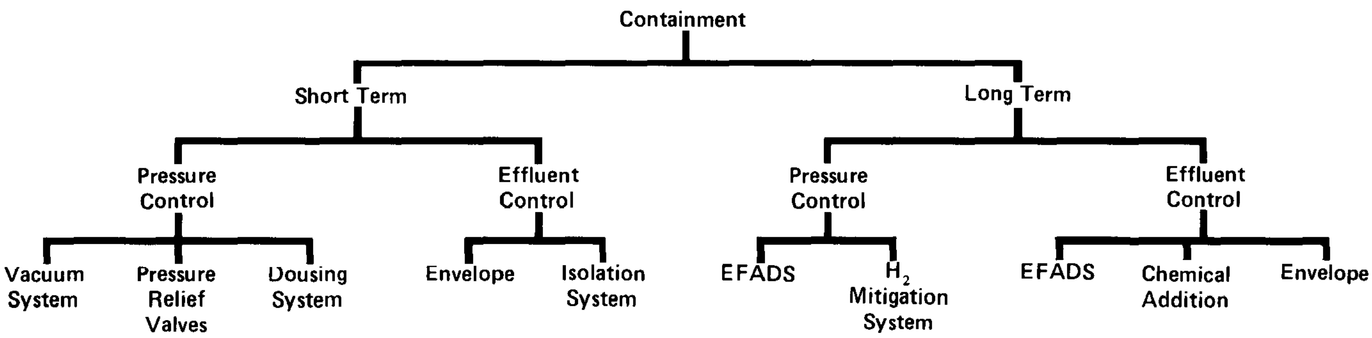

Figure 10 CANDU containment systems.

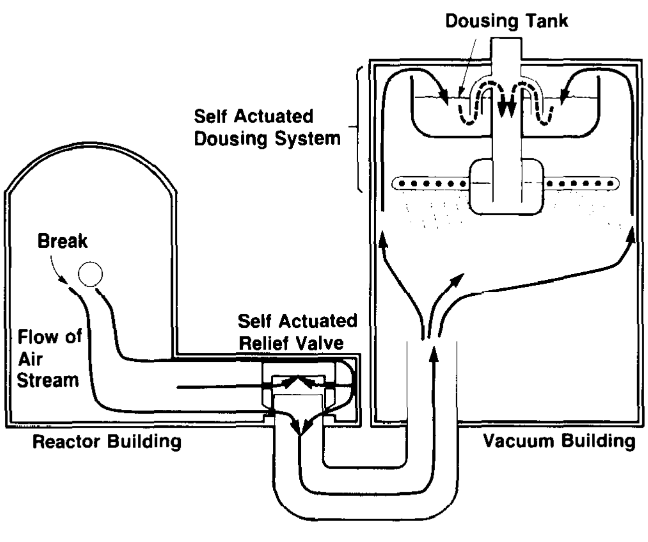

Figure 11 Operation–Negative-pressure containment.

term is isolation by physical barriers. Containment operates at 98 kPa (− .5 psig) and the vacuum building at 7 kPa (− 13.7 psig). Typical design pressures for containment are 170–200 kPa (10–14 psig) and 50 kPa (− 7 psig) for the VB.

Containment Capability Studies

Over many years, there have been numerous containment studies [2, 3, 11, 13] performed with the aim of establishing design parameters, proving that regulatory limits for design basis events are met, and identifying ultimate capability to withstand severe postulated accidents. Given that current research to provide 'best estimate' source terms and fission-product transport is also important to demonstrating containment capability, there is no doubt that high emphasis of nuclear reactor safety R&D today, is on containment systems.

This section summarizes this containment R&D, and provides a few examples of the many studies performed in West Germany, Sweden, the U.S.A., and Canada to demonstrate containment capability for LWR and PHWR nuclear stations. Present studies are largely associated with very low-frequency ($\leq 10^{-7}$ events per year), high-consequence events, since it is generally recognized that all containment systems are adequately designed for likely accidents.

Containment Research

Tables 4 and 5 provide a synopsis of typical integral containment tests for PWR's and BWR's to verify containment analysis codes and assure adequate designs. Many experiments have been performed elsewhere, notably in Japan.

In addition to those integral tests, there have also been numerous separate-effects tests performed in all countries (often involving international collaboration, as at Marviken) to understand jet impingement loads, vent flows, and condensation heat transfer. Experiments [17, 18] to determine the effects of external missiles (including large steel piping and segments of a turbine rotor) impacting on containment have been performed in the U.S. and elsewhere.

In Canada, as elsewhere, there were a number of on-site containment tests during the period 1970–1983, conducted by AECL and Ontario Hydro, to determine leakage rates and the thermal utilization of dousing flow in the Vacuum Buildings, and/or containment. In addition separate-effects tests of all containments were performed over the period 1960–1984, to understand transient compressible flow in interconnected volumes, jet loading, tee-junction losses, vessel-pipe fluid mixing, and liquid-steam phase separation at tee junctions.

The majority of current containment research is centred on the ultimate capability of LWR systems when subjected to severe accidents in the Class 9 category, as typified by the *idcor* program. The Industry Degraded Core Rulemaking (IDCOR) Program in the U.S. is supported by 62 nuclear utilities, architect-engineers, LWR vendors in the United States, and by Japan and Sweden. The IDCOR mission was to develop

Table 4: Integral PWR Containment Experiments

Year	Facility	Measurement purpose	Specific information
1965	CSE, U.S.A.	Vessel blowdown	Fission product transport and removal
1970	CVTR, INEL, U.S.A.	Peak pressure and temperature effects	Axial wall temperature distribution, heat transfer coefficients
1975	Battelle, Frankfurt	Pressure and temperature measurements during blowdown	Pressure waves, wall temperatures, H/T coefficients, jet impingement and hydrogen distribution
1981	Lucal Heights	Pressure/temperature response, small steel containment	Compartment pressure/temperature and heat transfer
1982	HDR, Karlstein	Blowdown for different break sizes/locations	Wall temperatures, steam-air concentrations, jet impingement, strains accelerations
1983–6	Sandia Nat. Labs., NM	Failure conditions modes beyond DBA	Structural failure mode, leakage paths, penetrations behaviour, base mat melt, bypass, margins

Table 5: BWR Containment Experiments

Year	Facility	Measurement, purpose
1960s	Humboldt Bay Bodega Bay	Drywell, wetwell pressure transients
1972/73	Marviken, Sweden	Full scale containment tests
1972,75	GKM 1, KKB	Vent pipe loads, full-scale
1975,77	Karlstein large tank and concrete cells	Multivent pipe tests
1976/77	GKM 2S	Vent pipe and pool wall loads, condensation, transient and static tests
1978/80	Studsvik, Sweden	Pool swell in different geometries
1984	GKSS	Vent clearing, pool swell and fall back
1983,86	Sandia Nat. Labs, NM	Large scale, Mk I, II, III overpressure tests, failure mode/timing, and design margins

a comprehensive, technically sound position on the issues related to potential severe accidents in light water power reactors.

IDCOR resulted from the USNRC's evaluation of the TMI-2 degraded core condition, which was more severe than that previously assumed in a design-basis accident. In October 1980, the NRC initiated a 'long-term rulemaking to consider to what extent, if any, nuclear power plants should be designed to deal effectively with degraded core and core melt accidents.' The NRC's rulemaking proposed to address the objectives and content of a degraded core-related regulation, the related design and operational improvements under consideration, their effects on other safety considerations, and the costs and benefits of design and operational improvements.

Subsequently, the NRC issued a proposed Commission Policy Statement [19], to implement the 2 October 1980, 'Advance Notice of Rulemaking,' and identify the severe accident decision process on specific stan-

Table 6: Severe Accident Phenomena Addressed by IDCOR to Establish Ultimate Containment Capability

1	Steam explosions causing pressure pulses; liquid slugs or missiles
2	Overpressure due to rapid steam generation
3	Overpressure due to hydrogen generation combustion
4	Containment by-pass via interface systems to environment
5	Overpressure due to noncondensable gases
6	Melt-through of containment base mat
7	Overpressure due to loss of containment heat removal
8	Containment failure modes
9	Radionuclide release and transport

dard plant designs, and on other classes of existing plants, which may or may not include rulemaking.

IDCOR identified key issues and phenomena, developed analytical methods, analyzed the severe accident behaviour of four representative plants, and extended the results as generically as possible. The methods used in the study were 'best-estimate,' rather than the conservative engineering approaches in technical analysis usually characteristic of licensing submissions. Existing methods and experimental data were thoroughly reviewed and new programs were undertaken where confident support of prior positions was uncertain. In general, IDCOR has demonstrated that consequences of dominant severe-accident sequences are significantly less than previously anticipated. Most accident sequences require long times to progress, allowing time to achieve safe stable states. Table 6 lists the reactor safety phenomena considered in reaching these conclusions.

While studies continue, the most important results to date are: containment overpressure capability is several times the pressure associated with the design basis accident; limited impairments of the containment envelope would likely occur on failure, thus stabilizing or gradually reducing pressures, which would limit the rate of radioactive release; hydrogen-related concerns can be mitigated or do not exist; and early failures of containment due to all causes are most unlikely, thus permitting sufficient time for interdictory actions.

Corium and fissium experiments [20, 21] are also on-going in the U.S., West Germany, and Sweden. Concrete-corium interaction tests to determine the extent of base-mat erosion are continuing in West Germany and U.S.A. Current experiments at Sandia National Laboratories [13, 14, 15, 16] sponsored by USNRC are addressing the issue of 'when, where and how' various steel and concrete containments will fail, and the resultant extent of radioactive release. Large-scale models have been or are being constructed to identify containment safety margins, and the integrity of containment pipe and electrical penetration assemblies when subject to overpressure loads. This large program is scheduled for completion by the end of 1986. In West Germany [2], studies suggest that the weak point of their PWR containment is associated with the sealing box which is part of the main airlock, in the event of overpressure accompanying a Class 9 accident. It is considered that the failure mode will be 'leak instead of break,' which will either result in a maximum stabilized containment pressure below ultimate capability, or reducing pressure. In other words, containment-pressure relief will occur rather than gross containment failure. Experiments to prove this engineering assessment are now being planned [2].

Table 7: IAEA Review of Hydrogen Studies

1	Hydrogen distribution in containment
2	Lower flammability limits
3	Combustion limits of H_2-air-steam-CO_2 mixtures
4	Available hydrogen and oxygen detectors
5	Pre-inerting as a mitigation scheme
6	Effectiveness of various ignition sources
7	Controlled burning and extinguishing systems
8	Fog/spray suppression
9	Minimum equipment to survive degraded-core accident.

Another area of research of importance to all nuclear power systems, and prompted by TMI-2, is that of ensuring control of hydrogen generation in severe accidents. This subject is the focus of attention of a current IAEA working group who are reviewing the issues identified in Table 7, using information from the major investigations already carried out by EPRI, Sandia Labs, and WNRE.

These studies are confirming that hydrogen recombiners or igniters for controlled burning, will prevent large containment overpressures. In many containments, the predicted volumetric concentration of hydrogen is far too low for combustion to occur.

Containment Response Analyses

The objectives of containment analysis are to establish design parameters and to verify that regulatory dose limits are not exceeded following any process system failure that leads to a release of radioactive material within the containment envelope. Design and regulatory processes require that containment response be analysed for a large number and variety of postulated system pipe failures ranging from a small leak up to a guillotine failure of the largest piping in the heat transport system.

PHWR Analyses

For CANDU reactors, accidents are characterized according to the postulated LOCA break-discharge rate, since this parameter has the dominant effect on subsequent containment response. A coolant channel end-fitting failure is used to bound the radiological consequences of small breaks in the heat transport system piping. The accident sequence postulated is an instantaneous maximum opening break, with the resultant ejection of all 13 fuel bundles from the channel.

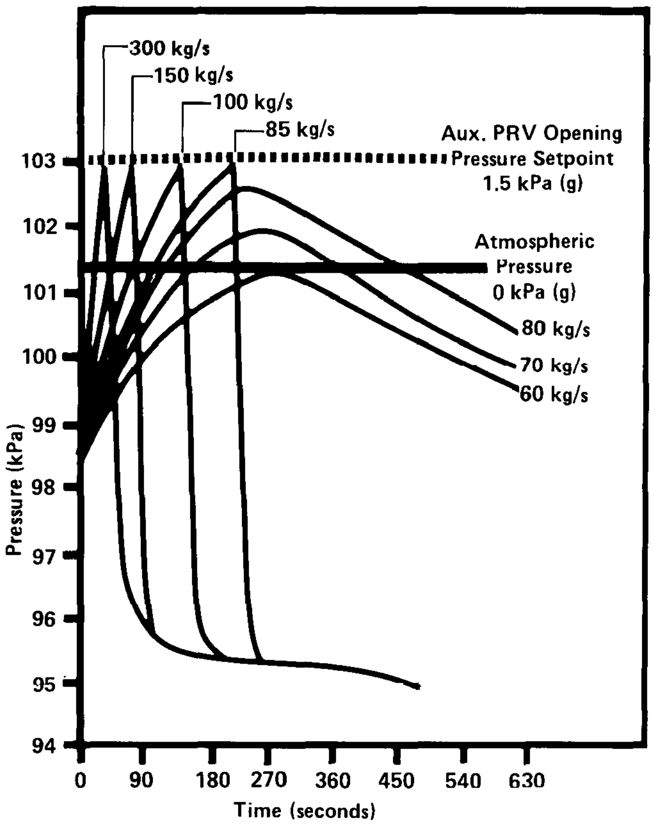

Figure 12 Negative-pressure containment response (small break).

Severance of an end fitting results in an initial coolant discharge rate up to 200 kg / s. The ejected fuel bundles will likely be damaged on impact with the reactor vault, and will release fission products into containment at a rate dependent on the extent of fuel cooling.

The containment pressure due to small breaks is strongly affected by containment heat sinks and, in particular, by the number of vault air-cooling units assumed operational at the time of the break. Figure 12 shows containment-pressure transients for various initial-break discharge rates. For small breaks above 80 kg / s, the duration of the overpressure period is determined by the time for the pressure relief manifold to pressurize to the setpoint of the pressure relief valves (PRV's). Below this discharge rate, energy removal due to air coolers and condensation on cold surfaces is sufficient to offset the energy addition from the break, with the result that the containment pressure can remain slightly above atmospheric without initiating PRV opening. The containment overpressure period will then last until either the break-energy discharge rate decreases sufficiently that the heat sinks are able to reduce the pressure to sub-atmospheric by steam condensation, or the operator manually intervenes by switching the PRV's to control mode. In spite of the potentially extended containment overpressure period for certain small breaks, releases into the environment are very small, since fuel damage is limited to a fraction of the core inventory.

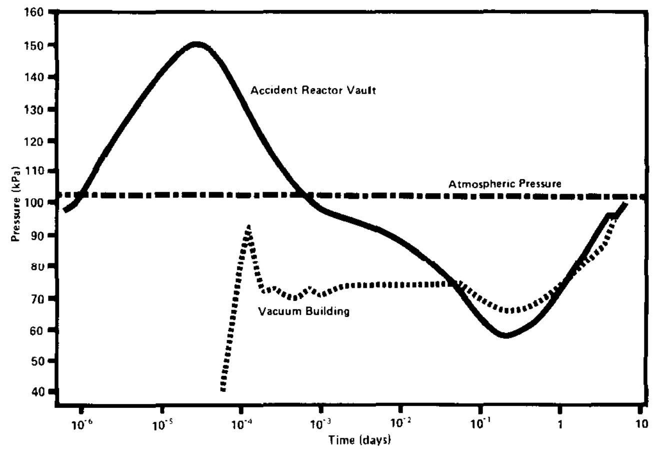

Figure 13 Negative-pressure containment response (large break).

Certain large breaks in the heat transport system, which could result in coolant stagnation within fuel channels, are capable of producing extensive fuel failures throughout the core. In addition, the initial pressure excursion presents a challenge to containment integrity. Figure 13 shows the estimated pressure transients in the accident vault and vacuum building following a postulated guillotine break in a pump suction line.

In this event, a peak pressure of 150 kPa occurs in the accident vault at less than 3 seconds. This is substantially below the containment design pressure. One minute after the break, the accident reactor vault becomes sub-atmospheric. From this time onward, the heat-removal rate exceeds the steaming rate at the break. The containment atmosphere continues to cool down and depressurize, until in the long term it becomes repressurized by air in-leakage, instrument air, and any gas evolution within the containment envelope.

Even with the fuel cladding damaged, the fission product release from the fuel is initially limited to a gradual escape of the 'free' inventory of volatiles. Only when the fuel heats up to high temperatures (well in excess of 1000°C) can a significant amount of volatiles start escaping from the 'bound' inventory. Thus, the concentration of activity in containment takes some time to build up to appreciable levels. With the effective pressure suppression provided by the NPC system, the amount of activity escaping to the environment by pressure-driven leakage is correspondingly small.

The bulk of activity enters the containment during the sub-atmospheric holdup period. These fission products then experience decay and undergo numerous interactions before a small portion is gradually

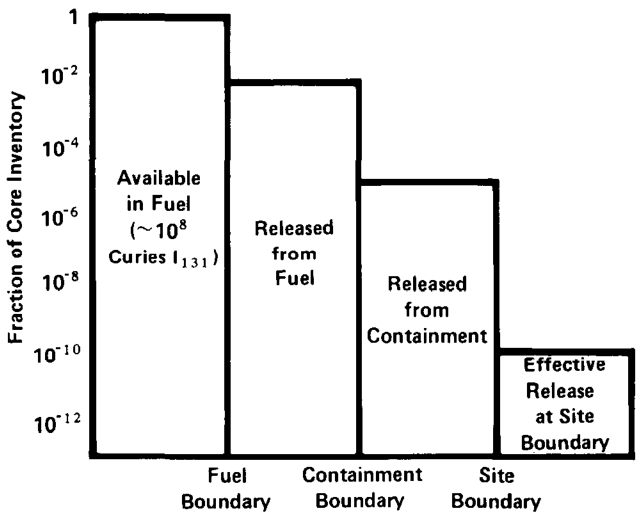

Figure 14 Typical attenuation CANDU (large break).

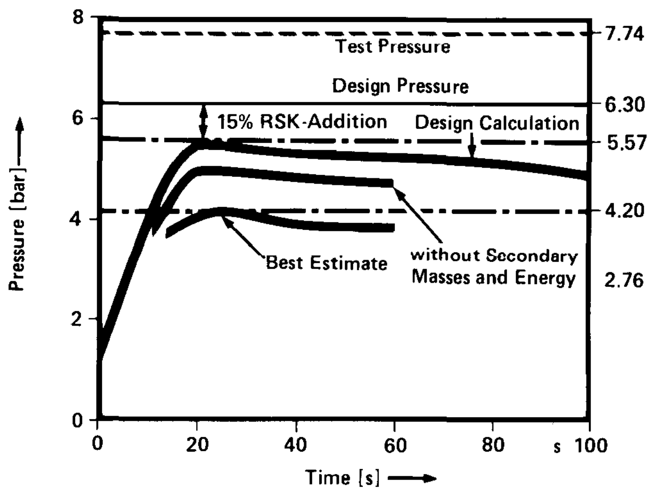

Figure 15 DBA pressure transients for standard 1300 MWE German PWR.

released by (EFAD) filtered venting. With the exception of noble gases, the fission products become trapped in water within containment, either by dissolution in liquid droplets or by becoming nucleation centres for liquid aerosols. Eventually, the airborne activity consists of only the noble gases and a small amount of volatile chemical compounds (e.g. organic iodides) in equilibrium with the solution on the floor.

Figure 14 illustrates the mitigating processes of CANDU containment systems in terms of I^{131}- attenuation resulting from the largest LOCA. Assuming the most adverse weather conditions, the 'effective release' is an equivalent amount of I^{131} that an individual could receive if present at the exclusion boundary for several months. The total attenuation for this severe accident is at least ten orders of magnitude, and the resultant dose, if individuals remained indefinitely at the site boundary, is within regulatory requirements.

In Canada, the current emphasis is on studies to delay or reduce the extent of containment venting even though regulatory limits are met. Regulatory requirements demand that dual-failure dose limits not be exceeded for LOCA's coincident with various containment impairments, including failure of isolation dampers, simultaneous deflation of four airlock seals on a double-door system, failure of pressure-relief valves, and loss of reactor vault air-cooling units.

PWR Analyses

As an example, the containment-response analysis performed in West Germany [2] to establish DBA parameters, and capability in severely degraded core accidents for the standard 1300 MWe PWR, is next described. While assumptions required by the German Advisory Committee on Reactor Safeguards (RSK) are not the same as those in the United States, the analysis results are generally typical of most PWR's. Also, while the extent and timing of BWR severe accidents is different from PWR's, the questions on ultimate containment capability are not dissimilar. The German containment design-basis accident (DBA) is a double-ended break in a main coolant pipe. For containment design purposes, RSK also requires a number of conservative assumptions which include:

(a) decay heat according to ANS Standard plus 20 per cent;
(b) maximum LOCA pressure assuming a 2 per cent decrease in containment volume, and a 2 per cent increase in primary and secondary circuit volumes (blowdown mass and energy include one secondary steam generator content);
(c) a 15 per cent safety margin applied to calculated maximum LOCA pressure;
(d) the steel containment shell to be designed for maximum containment temperature (145°C), rather than its expected temperature (60°C).

Figure 15 shows that the 'best estimate' of the maximum LOCA pressure will be 4.2 bar. Also shown, are the design calculation results, assuming (a), (b), and (d) above, for the pressure transients when the additional energy from the assumed secondary break is either excluded or included. The containment design pressure of 6.3 bar includes the additional assumption (c). The containment test pressure of 7.74 bar, prior to reactor criticality, is set by the difference in yield at the testing and LOCA temperatures. Thus, there is a substantial margin (up to 84 per cent) between the expected LOCA pressure and the demonstrated test capability.

In the event of a core melt-down, there would be a substantial release of fission products and steam/water to containment. The extent of release to the environment is highly dependent on the containment isolation

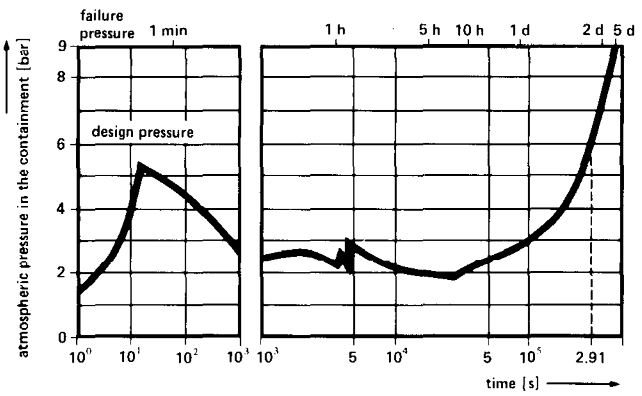

Figure 16 German PWR containment in core melt accompanied by failure of the sump cooling system.

time, the extent and nature of any containment leakage, and the transport and driving force paths from the annular space between containment and the reactor building. A very important mitigating process in this regard is the finding [22] that all radioactive substances, with the exception of the noble gases and airborne gaseous iodine, are bound to aerosol particles and subject to highly effective removal mechanisms. These removal mechanisms, involving plate-out and absorption on containment structures, reduce aerosol mass concentrations by five to six orders of magnitude within five days. Containment overpressure failure at the weakest point is not expected during this period.

Figure 16 shows the predicted pressure variation in containment for a core melt-down sequence. The maximum transient pressure during blowdown of 5.3 bar is reached at 17 seconds after LOCA. The transient LOCA pressure, which is relieved by the containment volume and condensation, reduces to 2.5 bar during the next 10^3 seconds. The core melt-down process due to the onset of evaporation of the moderator water and assumed complete absence of emergency forced cooling, commences at about 20 minutes.

Core degradation then proceeds, accompanied by hydrogen production due to steam/zirconium reactions. Shortly after one hour, the core structure is predicted to fail, allowing significant amounts of core material to drop into the water contained in the lower plenum of the reactor vessel, with the resultant violent evolution of steam.

Subsequently, at about 1.9 hours after blowdown, reactor vessel failure is predicted to occur, and core melt interaction with the concrete basemat begins. Given that 80 tonnes of metallic melt and 130 tonnes of oxide melt at a temperature of 2400°C are assumed available, it is predicted that the concrete shielding surrounding the reactor vessel will be eroded within 7 hours, causing the containment sump water to contact the melt. Violent evaporation of the sump water in the isolated containment subsequently results in its pressurization to design pressure after 3 days, and to 9 bar after 5 days, as shown in Figure 16. This sequence of events raises the question of ultimate containment overpressure capability, the mode and extent of containment failure, and subsequent extent of radioactive release from the annulus between containment and the reactor building, via filtration to the environment. These questions also highlight the 'defense-in-depth' provided in LWR stations for public protection, and the extended time available for any necessary emergency evacuation.

Recent studies in West Germany [2], and those of IDCOR [9] in the United States, are showing that containment overpressure capability before failure is up to 2–3 times design pressure for the undisturbed steel shell. As an example, West German experts predict overpressure failure of the 1300 MWe standard PWR containment to be above 14 bar in this case, and that the mode of failure will be 'leak instead of break.'

It is considered that containment overpressure will result in a leak at weak points, such as in main airlock components or at electrical/pipe penetrations, and will either permit a stabilization or reduction of containment pressures. The net result is the maintenance of major containment integrity, and only gradual activity release (likely after filtration), to the environment. Also, recent West German studies and those in the United States have demonstrated that previously assessed source terms are too high by several orders of magnitude.

Summary

This paper has discussed the functional requirements, the evolution of designs, and the influence of national regulatory requirements on containments for PWR, BWR, and PHWR reactors. Particular containment designs are not only a function of national siting requirements in the United States, West Germany, Sweden, and Canada, but also relate to specific reactor system performance in perceived accident conditions, and the number of reactors constituting the generating station. In all cases, it is evident that the various containment systems easily meet their design-basis accidents. Since the TMI-2 accident, all jurisdictions have examined the need for design changes to meet post-accident scenarios.

In Canada, increasing attention has been paid to large stagnation LOCA's with assumed coincidental containment impairments. In this regard, methods to delay or reduce the extent of atmospheric venting of containment to relieve pressure are under active study, even though regulatory limits are met.

In West Germany, the United States, and Sweden, emphasis is on the ultimate capability of PWR and BWR containments to withstand overpressures and evaluate environmental releases for class 9 core melt accidents, which are beyond, or bordering on the range of credible frequency. Current information from the U.S. based IDCOR study and the independent West German and Swedish research work, indicates that LWR containment designs are capable of withstanding overpressures up to three, and possibly four times their design pressure. In the event of containment failure, it is predicted that radioactive release will likely result from gradual leakage from weak points (rather than from a gross containment break) to the reactor building, resulting in a slow and delayed discharge to the environment. Experiments and scaled tests of containments have been performed, or are currently underway in many countries to determine ultimate containment ability and failure modes. These tests, together with allied fission-product source-term and transport tests, and comprehensive experiments on hydrogen generation and mitigation, are an important area of reactor safety research today. In addition, research in the U.S. is directed at decoupling LOCA plus SSE as a design basis, and reducing postulated high-energy system pipe breaks and loading phenomena.

If containment failure were to occur, in either the CANDU, PWR, or BWR reactor systems, it is predicted to do so many days following the most severe postulated accident, permitting adequate time for assurance of public safety.

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