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Status of advanced containment systems for next generation water reactors

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FOREWORD

The nuclear industry has always pursued excellence in enhancing and maintaining a high level of quality in plant design, manufacturing, construction, operation and maintenance. Because of this and because of developments in technology and public concern about the safety of NPPs a number of design improvements will take place for the next generation of nuclear power plants.

Containment is a key component of the mitigation part of the defense in depth philosophy, since it is the last barrier designed to prevent large radioactive releases to the environment. Therefore, while containments of current NPPs are able to meet the objectives they have been designed for in the frame of the defense in depth approach, many improvements are proposed by designers to further reduce the probability of releases to the environment in a broader set of accident situations. In addition, since the containment system is responsible for a significant part of the entire station cost considerable effort is devoted to reduction of the costs associated with construction and maintenance.

The current IAEA programme in advanced nuclear power technology promotes technical information exchange between Member States with major development programmes. It is hoped that this report on the status of advanced containment systems for next generation water reactors will be useful for further dissemination of information and for stimulating international co-operation between the Member States.

EDITORIAL NOTE

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SUMMARY

While current plant containments are able to meet the objectives they have been designed for and provide a substantial contribution to the defense in depth approach, many improvements are proposed for future plants to further reduce releases to the environment for a broader range of accident conditions and also to reduce costs associated with construction and maintenance. The International Working Group on Advanced Technologies for Water Cooled Reactors recommended to the IAEA to prepare and publish the present document to provide a summary of available information on the advanced containments proposed for next generation reactors.

In general, advanced containments have evolved from those of existing reactors. They have been developed on the basis of experience to date, but they also tend to consider carefully:

- extension of the events considered explicitly in the design;
- greater attention to maintenance and operation;
- costs.

The report gives more emphasis to the identification of common trends and to the description of specific challenges and the ways in which these are dealt with, instead of presenting extensive descriptions of all proposed designs. Only a few containments are described as examples.

From a safety standpoint the major innovation is the explicit consideration of severe accidents in the design and the consequent adoption of technical features designed to prevent both early and late containment failures. Future containments are, therefore, expected to remain operational in all situations considered in the design, including severe accidents.

Since economic competitivity of nuclear power has been progressively eroded in a number of countries due to the increased complexity of nuclear plants, to over lengthy licensing procedures, and due to the current low cost of fossil fuels in general, cost reduction of all elements of a nuclear plant is one of the major goals for the next generation of nuclear power plants. This is intended to be achieved by implementing a number of secondary objectives, such as simplification, standardization etc., which will also affect the containment.

The report also identifies, on the basis of published material, the need for additional research and development activities, including deficiencies in existing computer codes for evaluating specific features of some advanced containments such as natural circulation, passive heat removal, efficiency of large passive heat sinks, etc.

The final conclusion is that most of the proposed containment designs appear to be in compliance with the existing and newly emerging safety, technical and economic objectives. Concerns about severe accident protection appear to be well integrated in the design and design criteria appear to be convergent to a large extent. All of the proposed containment designs are intended to provide a step forward in safety and technology.

1. INTRODUCTION

1.1. OBJECTIVES OF THE STATUS REPORT

The present IAEA status report is intended to provide information on the current status and development of containment systems of the next generation reactors for electricity production and, particularly, to highlight features which may be considered advanced, i.e. which present improved performance with evolutionary or innovative design solutions or new design approaches.

The objectives of the present status report are:

- to present, on a concise and consistent basis, selected containment designs currently being developed in the world,
- to review and compare new approaches to the design bases for the containments, in order to identify common trends, that may eventually lead to greater worldwide consensus,
- to identify, list and compare existing design objectives for advanced containments, related to safety, availability, maintainability, plant life, decommissioning, economics, etc.,
- to describe the general approaches adopted in different advanced containments to cope with various identified challenges, both those included in the current design bases and those related to new events considered in the design,
- to briefly identify recent achievements and future needs for new or improved computer codes, standards, experimental research, prototype testing, etc. related to containment systems,
- to describe the outstanding features of some containments or specific solutions proposed by different parties and which are generally interesting to the international scientific community.

The document is intended to be useful for technical experts, providing a concise and updated overview of the states of the art, and to general readers to understand what type of improvements in performance may be expected from future containments.

It is considered beyond the scope of this document to provide a complete, comprehensive description of all available or proposed alternatives. There are cases where the design of the containments proposed for future reactors either do not present significant changes over established practices or has not proceeded beyond a preliminary conceptual stage, leaving open many questions on their development. In these cases it has not been felt appropriate at this time to include a detailed description of the proposed designs. However, many of the general trends indicated in the following sections are also applicable to containments not explicitly covered by specific descriptions. The containment systems described in detail in Section 6 are to be considered examples of the implementation of the general trends, associated with the main reactors of international interest for which a larger amount of information is available. In addition, a few specific features which have been proposed outside the context of a specific reactor, have been included when sufficient material has been found.

This report has been prepared on the basis of available literature, including the presentations given at the Technical Committee Meeting (TCM) on Advanced Containment Technologies held in Aix-en-Provence in June 1992 and organized by the IAEA. The conclusions of the TCM included direct recommendations for the preparation of the present status report, notably:

"The TCM focus was on severe accident mitigation, but attention should be given to other accidents, such as reactivity accidents, earthquakes, external impacts, and containment by-passes. This would also appropriately apply to shutdown conditions.

It should be limited to water reactors and mainly oriented to designs intended to be commercially available around the end of the century. Clear distinction shall be made between licensing requirements and design objectives. Attention shall be oriented to technological evolution more than discussion and interpretation of single phenomena, and general safety objectives."

1.2. EXTENT OF THE CONTAINMENT SYSTEM AND ITS MAIN FUNCTIONS

According to the IAEA Safety Guide No. 50-SG-D12 [1] the containment system consists of the containment structure and associated subsystems required to function under specific accident conditions. These include:

- the containment structure and extensions such as external passive fluid retaining boundaries, which together form an envelope around the reactor coolant system,
- the active features of the containment isolation system, which in general provide closure of openings in the containment envelope on demand,
- energy management features (collectively referring to the functions of pressure suppression, containment atmosphere pressure and temperature reduction and containment heat removal after a postulated accident),
- radionuclide management features, which are provided to reduce the release of radionuclide to the external environment, after their release to the containment volume,
- combustible gas control features, which limit the buildup of combustible gases in the containment envelope and prevent the uncontrollable combustion of these gases.
- other prevention or mitigation features for impulsive loads produced by severe accidents (such as ex-vessel steam explosions).

The main functions of the containment system are:

- to prevent uncontrolled large releases to the environment in all those plant conditions which need to be taken into account, and to minimize the controlled releases,
- to maintain its structural integrity to assure the required leaktightness and the necessary support to systems and components,
- to allow the removal of decay heat and the cooldown of the reactor to safe shutdown conditions,
- to prevent radioactive releases to the environment as a consequence of external events of natural and man induced origin,
- to provide a biological shielding for operating personnel and the public.

It can be debated whether the containment system concepts can be developed separately from the reactors and associated circuits. It could be said that this is in general true for similar reactor types in the same range of power output, if a strict separation, as desirable, is maintained between the containment and other process and safety systems. For example, there are few conceptual difficulties in adopting a suppression pool type containment for a PWR and the choices are generally made on a cost basis. We have to assume that the designers have always optimized their choices with reference to all elements to be considered.

In other cases the solution is not so simple. It appears for example that a Chernobyl type event could not be contained in any containment of a reasonable size, since the energy associated with some conceivable accidents is too large, and the only way to deal with this type of events would be to improve their prevention.

It should be finally mentioned that practically no design presented in this report has been developed to be deployed in a single country. This has required, and is requiring, an extensive effort to provide solutions flexible enough to meet different (and in some cases inconsistent) needs related to licensing requirements, cost objectives and site characteristics.

1.3. STATUS OF NEXT GENERATION WATER REACTORS DEVELOPMENT

A number of reactor designs are currently ready for ordering, representing the state-of-the-art derived from continuous upgrading and improvement based on experience gained from current models. For example the French N4 model (1400 MW(e)), of which the first units are now under construction in France, is derived directly from the standardized P4 series. In Germany the

CONVOY plants are a group of three standard PWRs (1300 MW(e)). The advanced features of the CONVOY plants are mainly in the engineering and project management associated with NPP construction. The Westinghouse-Mitsubishi Advanced Pressurized Water Reactor (APWR-1350 MW(e)), the British 'Sizewell B' PWR (1250 MW(e)), the ABB Combustion Engineering System 80+, the ABB Atom BWR and the General Electric Japanese ABWR are other examples of developments of existing designs.

These reactors, as well as many others currently proposed, may be considered 'advanced' in comparison to the currently operating plants of the previous vintage, but, in general, include containment systems with less significant specific innovations in terms of components or design criteria. However, it is worth mentioning that some designers deviate significantly from their own past practice. For example the containment for ABB's System 80+ reactor is a large volume, spherical steel shell, similar to some existing ones located in Germany.

Table I gives a non-exhaustive list of advanced reactors with some indication of their containment characteristics.

Many reactors have similar containment proposals and few have reached an advanced stage of engineering design. In Section 6 only a few representative examples of engineered complete containment systems are described but even these are unlikely to be available commercially before the late 1990s. Section 5 includes descriptions of some component systems which may be at a less advanced stage of development. The designs included in Section 6 are:

- The Westinghouse Advanced PWR (AP-600);
- The General Electric Simplified Boiling Water Reactor (SBWR);
- The European Pressurized Reactor (EPR) developed by Nuclear Power International (NPI), a joint company established by Framatome and Siemens;
- The ABB Atom PIUS design;
- The advanced CANDU reactors CANDU 396 M 2.

1.4. CLASSIFICATION OF CURRENT CONTAINMENT DESIGN CONCEPTS

Existing containment systems can be classified into a number of different types depending on their principal characteristics. Advanced reactors may use these existing design types or may have additional features designed, for example, to mitigate the consequences of severe accidents. Description of these advanced features and their rationale is the main objective of this report but it will be helpful to list the systems which have been used in reactors existing in 1993. Further detail of these containment systems is given in Appendix A.

1.4.1. Single dry containment

The containment is large enough to withstand the pressure and temperature from the largest pipe break without additional fast acting pressure reduction systems.

Typical applications: PWR, single reactor CANDU.

1.4.2. Dual dry containment

The primary dry containment structure is surrounded by a second containment which provides protection against external missiles and allows for detection and filtration of activity leaking from the first containment.

Typical application: PWR.

TABLE I. CONTAINMENTS FOR ADVANCED REACTOR DESIGNS

REACTOR	SUPPLIER/ COUNTRY	POWER (MW(e))	CONTAINMENT TYPE	FEATURES	STATUS
1. PWR - LOOP TYPE					
System 80+	ABB CE	1300	Dry, dual (steel, concrete)	Active igniters, Corium cooling features	Detailed engineering
Sizewell B	Nuclear Electric, United Kingdom	1250	Dry, dual (steel, concrete)		Pre-op.
APWR	Westinghouse/ Mitsubishi	1050/1300	Dry, steel		Detailed engineering
EPR	NPI	1400	Dry, dual (concrete, concrete)	Corium catcher and cooling	Basic engineering
WWER-88	Russia	1000	Dry, steel, vented	Core catcher, Filtered venting	Basic engineering
WWER-92	Russia	1000/1100	Dry, dual, (concrete, concrete)		Basic engineering
AP-600	Westinghouse	600	Dry, steel	Passive water spray and air cooling	Detailed engineering
WWER 500/600	Russia	500/600	Dry, steel, concrete cover	Core catcher, Cooling to watertank and by air flow	Detailed engineering
B-600	B&W	600	Pressure suppression, concrete cover	Gravel bed pressure suppression, natural circulation air cooling	Concept
AC-300	China	300	Dry, concrete with steel liner	Internal spray	Pre-op.
AC-600	China	600	Dry, steel	Passive water spray and air cooling	Detailed design
MS-300	Mitsubishi Heavy Industries	300	Dry	Internal gravity injection tanks	Conceptual design

MS-600	Mitsubishi Heavy Industries	600	Dry	Passive systems	Basic engineering
NUPACK	Westinghouse	600	Dry, steel, cover	For barge delivery	Concept
LIRA	Italy	1000	Pressure suppression, dual	Igniters, Corium catcher	Concept, Containment only
2. PWR - INTEGRATED TYPE					
PIUS		640	Small, cooled containment	Design minimizes demand on the containment	Basic engineering
SIR	ABB-CE, SWEC, AEA Technology, RRA	320	Dispersed pressure suppression, steel	Inerted, Natural circulation air cooling	Concept
SPWR	JAERI	350/600	Pressure suppression	Water filled containment	Concept design complete
ISER	U of Tokyo	210			Concept
VPBER-600	Russia	600	Steel Guard Vessel and Concrete containment		Detailed engineering
3. BWR					
ABWR	Hitachi Toshiba	1356	Pressure suppression weir wall, concrete	Inerted	Construction
BWR-90	ABB-Atom	830/1170	Pressure suppression, concrete	Inerted	Available
HSBWR	Hitachi	600	Pressure suppression		Concept
TOSBWR	Toshiba	310	Pressure suppression	High level wet well	Concept
SBWR	GE	600	Pressure suppression, concrete	Igniters and inerting, Containment cooling by isolation condenser	Basic engineering
4. HWR					
CANDU 3	AECL	450	Dry	Active igniters	Detailed engineering

1.4.3. Ice condenser containment

This is a pressure suppression system where steam from pipe ruptures is constrained to pass through and condense on racks of ice before reaching the main volume of the containment.

Typical application: PWR.

1.4.4. PWR bubbling condenser containment

This is a pressure suppression system like 1.4.3 above but using water instead of ice.

Typical application: PWRs of Russian design (WWER-440/213).

1.4.5. BWR pressure suppression containment

This uses a water pool for pressure suppression on release of steam from a pipe break. There have been various arrangements but all have a dry well which contains the reactor coolant system and a wet well partially filled with water. The latest version is the Weir-Wall Pressure Suppression Containment System.

Typical application: BWR.

1.4.6. Negative pressure containment

In multiple reactor stations, each individual reactor building can be connected via a duct to a large building maintained at low pressure to relieve pressure resulting from a pipe break.

Typical application: CANDU.

2. GENERAL SAFETY DESIGN CONSIDERATIONS

2.1. GENERAL PLANT SAFETY OBJECTIVES

There is a clear trend today, involving all types of industrial activities, towards a continuous improvement with respect to public safety aspects and environmental protection levels associated with these activities [2].

Therefore, to enable nuclear power to play an expanded role in the future, next generation nuclear power plants should not only be at least as safe as the best plants operating at present, but should be expected to present some significantly enhanced safety, as well as economic characteristics.

Two factors are considered to be of special importance to public acceptability: what technical means are used to limit the off-site consequences of serious accidents (and how large could the most severe consequences be) and how the plants are designed to decrease the sensitivity of nuclear safety to human errors.

A strong incentive has always existed to reduce the off-site consequences of any accident. In the past, the approach has varied from country to country. In some countries the approach has been to design the containment using a design basis accident evaluated on very conservative assumptions. This was expected to be able to contain many severe accidents.

The consequences to the population and environment were calculated assuming an arbitrarily long release to the containment atmosphere and unacceptable doses to the population were avoided by rapid evacuation plans. The difference in most next generation plants is that a combination of design improvements (particularly in the containment system) and a realistic assessment could provide technical elements for a simplification of the emergency plans and, notably, for the elimination of the need for rapid public evacuation (rapid means earlier than 24-36 hours from the initial release) or even any evacuation at any time.

In addition, there has been a recent focus on ensuring, as a separate parallel goal, no health significant contamination of the surrounding land and water bodies, or, at least, a very limited one in space and time, avoiding, as a consequence, the need for long term relocation of large segments of population.

For all reactors which cannot show that a large release from the fuel is practically impossible, despite the highest importance given to preventive measures in the framework of the defense in depth principle, the containment system assumes a key role in achieving the above said goals. Therefore, an advanced containment is characterized by improved performance in terms of leaktightness and structural integrity based on a more direct and realistic consideration of extreme scenarios.

In quantitative terms related to environmental impacts, the goals for the containment system have been expressed in the following way:

Individual doses: taking into account the characteristics of the reference sites, the individual doses at the site boundary should be less than the threshold for the activation of the public protection measures (sheltering and evacuation). These values range from 10 to 250 mSv (1 rem to 25 rem) committed effective dose equivalent, depending on the assumptions and the country.

Land and water bodies contamination: this can be translated into a limit to the total releases of one or more specific isotopes. Some proposed goals have been expressed for example in the range of 10 to 100 TBq of ^{137}Cs, to be calculated using realistic, best estimate assumptions.

It is important to underline that these goals can be compatible only with an essentially leaktight and intact containment, behaving as designed in all considered situations.

Frequently, the above limits have been associated with a probability. A typical value is 1E-6 per reactor year as the probability that release or dose limits are not exceeded. In other cases, the reference accident sequences have been defined on the basis of a combination of probabilistic and deterministic criteria associated with engineering judgement and assessment of uncertainties.

With regard to the human factor the objective for all future plants, including their containment systems, is to reduce the possibility of human errors, in particular reducing the need for and the credit to operator action for a long time after accident initiation.

In addition, plant design shall be 'forgiving' or 'fault tolerant' in the sense that should a human error occur, the plant response would be 'friendly' and ample time, if necessary, will be available to correct the error.

Many studies have shown that with a grace period of about 30 minutes the probability of operator errors becomes very low (for coping with design basis accidents). However, in most of the proposed advanced containment designs the grace periods exceed the above 30 minutes; they range from several hours to several days (with respect to coping with severe accident scenarios).

2.2. GENERAL DESIGN APPROACHES AND RELATED ISSUES

2.2.1. Defense in depth

The defense in depth approach will continue to be the basis for sound design for future plants. Moreover, it will be strengthened by additional margins in the design and, in general, by the explicit consideration of realistically conceivable severe accidents.

The containment system is of course a key element of the strategy and it is designed to provide a last barrier to the release of radioactivity to atmosphere following release internally in severe accidents with a minuscule probability despite all efforts to prevent such accidents. Therefore, in comparison with the present practice, clear improvements may be sought. However, due to different current national licensing practices, rules and regulations, the degree of these improvements may vary from country to country.

A pictorial representation of these improvements, which is intended to cover generally most, but not all, situations is given in Fig. 1 which is derived from an ENEL (Italy) proposal. In this figure also the elements of the defense in depth strategy for each layer of event are intended as examples of the most important ones, but they are not intended to be exhaustive or mutually exclusive. Another example is given in Ref. [3].

2.2.2. Use of passive systems

Passive systems should by definition be able to carry out their mission with minimum or no reliance on external sources of energy and should operate only on the basis of fundamental natural physical laws, such as gravity (see also IAEA definitions in Ref. [4]).

The potential advantages in terms of reliability and independence from other systems and operator actions are important. And their application to containment system appears to be promising in a number of proposed designs.

For example, both the AP-600 and SBWR containments are able to dissipate the decay heat passively for a long time (72 hours and more) without credit to active systems or operator actions. Also EPR 1400 MW(e) is considering not exceeding containment design pressure for a period of about 24 hours even without giving credit to any active system, but realistically considering the effects of all available thermal inertia.

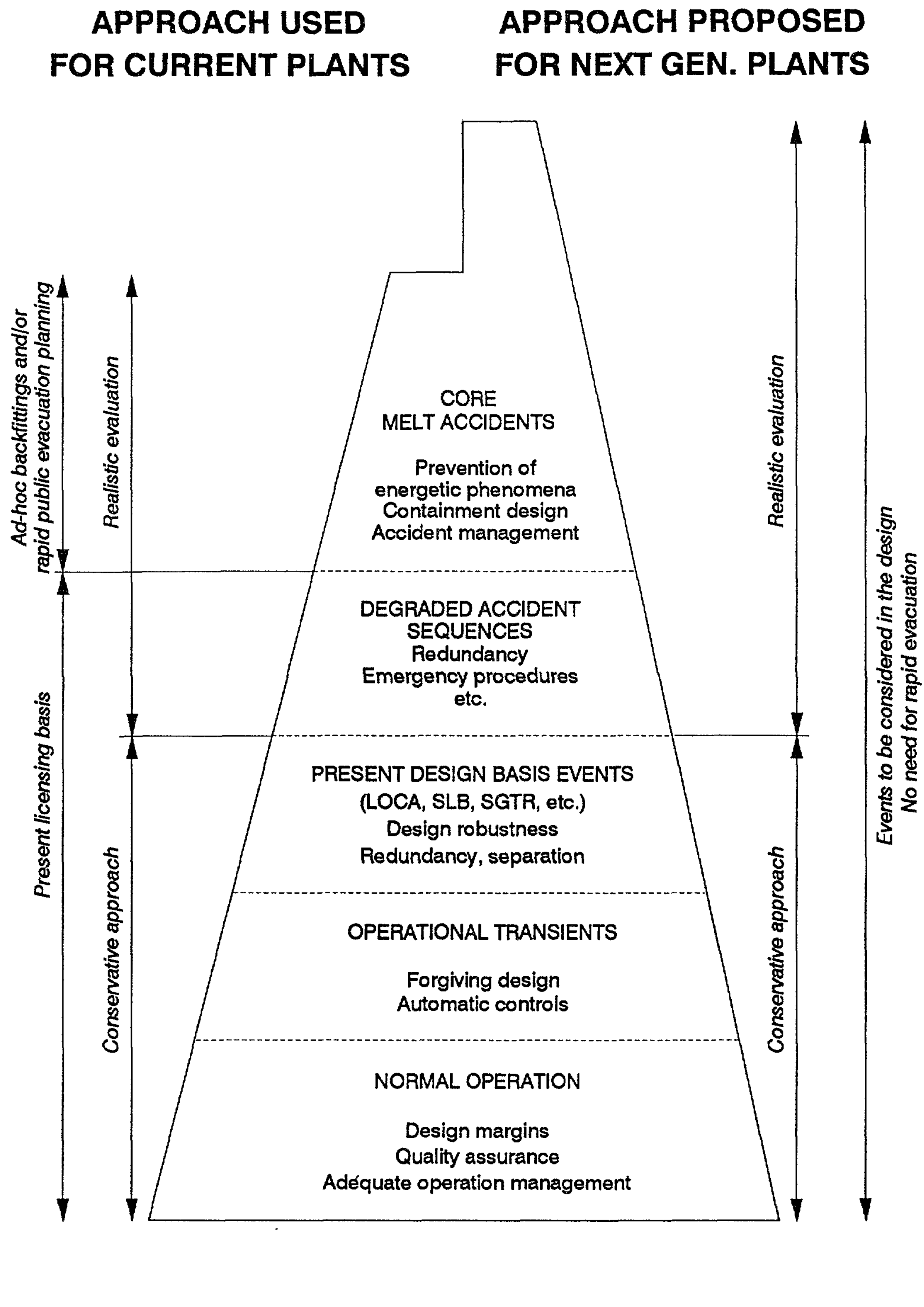


FIG. 1. Defense in depth strategies.

Therefore, use of passive systems at least in the medium term range (in 24 hours) has been encouraged in some advanced containment designs, where they are expected to provide an adequate level of functional performance for the intended purpose and an adequate confidence that this level can be achieved. To have full confidence in these systems proper verification of computer codes and correlations and a review of specific failure modes of passive components/systems is necessary. Testing of the overall containments and of specific components has been required by some safety authorities and is currently underway for some designs.

Other designs adopt a more open approach to the use of passive or active systems, deciding the choice for each single function on a case by case basis.

2.2.3. Credit to non-safety systems/components

All systems and components needed to cope with current design are generally 'safety-grade'. This means that they are designed with margins, they have to comply with certain deterministic rules, and they have to be environmentally qualified according to specific standards.

In future containment designs, consideration of severe accident is given. However, there is still discussion on whether systems and components designed to cope with these very unlikely events must be defined 'safety-grade' in the sense given above. While it is obvious that it must be demonstrated that they are capable of fulfilling their mission, a simplified design and qualification approach may be acceptable.

With reference to systems required only in the long-term after an accident to put the containment in a safe and stable final condition, the same considerations apply. In addition, they are generally required to survive the harsh environmental conditions, and to operate only when these extreme conditions are improved.

2.2.4. Independence, reliability

The proposed containment designs try to separate the containment system from the other plant systems, and in particular from the systems, whose failure may have caused the fission product release. This is generally verified at the mechanical component level as well as at the control and instrumentation level.

Reliability numbers for the containment functions have been proposed in the USA, where a conditional probability of 0.1 for the containment failure in case of a severe accident has been discussed. In general, however, it appears that the uncertainties are so large that a more deterministic approach is preferred.

2.2.5. Single versus double containment

Since the release limits from the plants are very stringent, most of the advanced containments give credit to decontamination factors for the radionuclides that escape the primary containment.

In some cases a complete double containment surrounds the primary containment, in other cases only the penetration area is surrounded by structures with a reasonable degree of leaktightness, on the assumption that leaks through the remaining part of the shell should be negligible or, anyway, acceptable for the final goal.

Part of the strategy is also generally to include in the primary containment all systems/components containing significant amounts of fission products after an accident and to avoid recirculation of contaminated fluids outside the containment.

The decision on whether to give credit to a secondary containment for external consequence calculations involves licensability considerations (related to current practice in different countries) and depends upon the adoption of an active or passive strategy, which is important to define the efficiency of the decontamination process.

In the EPR, credit is given for a grace period of several hours to the secondary containment for collection, deposition and hold up of fission products leaked through the primary containment shell.

Both the SBWR and AP-600 adopt the strategy of purely static and passive partial secondary envelopes, reasonably tight, in which only holdup and deposition may be credited. In these cases no filtering is available and releases are considered at ground level.

There are also proposals, such as the SPWR described in Section 6, which use a passive ventilation system capable of maintaining a dynamic leaktightness (i.e. pressure slightly lower than atmosphere) and to provide filtration before exhausting the extracted air.

There have also been proposals to use passive systems to maintain a slight underpressure in the secondary containment taking advantage of the source of thermal energy present in the primary containment in all accidental conditions, in which a secondary containment is needed.

Secondary containment design must also be carefully evaluated from a cost-effectiveness standpoint. Generally a source term reduction factor of at least 5 (in some cases 10 and even more) is required to justify a secondary containment. To reach this goal the secondary structure should be reasonably tight and, if depressurized by active or passive ventilation systems, it should be provided with adequate filtering trains. However, collecting and controlling the releases from primary the containment and releasing them through an elevated point (e.g. a stack) could lead to a dose reduction at short distances higher than one order of magnitude even in the absence of filtration..

Another very important requirement is to minimize any secondary containment bypass, since they are direct releases to the environment jeopardizing the efficiency of the double barrier. Choice between single and double containment may be made also on the basis of the required protection against external events. In particular plants designs for aircraft crash are generally double assuming that at least the primary containment would maintain its performance after the impact and would, therefore, limit any possible release to the environment.

2.2.6. Occupational doses reduction

Plant layout, system design, material selection, etc. are oriented also towards a substantial reduction of occupational exposures for next generation plants. Average occupational doses of 100 man·rem/year unit, and lower have been defined as design goals.

Containment design shall contribute to this goal, particularly in terms of accessibility during normal operation and in shutdown, in terms of easy maintenance or possibility of robotic assistance for maintenance activities, in terms of components separation and shielding, etc.

Post-accident activities are also planned in some cases, giving attention to accessibility and shielding of components which must be accessed in the short or in the long-term after an accident. Many sensitive pieces of equipment may be located in the secondary buildings surrounding the primary containment and may be subject to radioactive contamination after an accident.

2.2.7. Containment filtered venting

In most cases the proposed containments tend to avoid provisions for filtered releases to prevent containment failure in the event of a core meltdown. Containment filtering devices of various designs have been introduced as a backfitting in some current containments as an effective means, or last-resort feature to cope with beyond-design events. Prevention of containment overpressurization can be achieved by early consideration of core melt sequences in the containment design.

It is worth mentioning that containment filtered venting was not designed anyway to cope with very rapid pressurization sequences, and the strategies are different even for current containments.

2.2.8. Accident management

Accident management is one of the important elements of the defense in depth approach. It usually includes appropriate operator actions in the short as well as in the long term after an accident intended both for prevention and for mitigation.

The advanced containment systems are designed to maintain the required performance for a substantial period of time for a broad range of events including severe accidents. This derives both from the fact the core melt initiation is delayed with respect to operating reactors (in addition to being less probable) and to the fact that early containment failures are excluded by specific design provisions. Operator actions are not required in this period. This is a very significant difference with respect to existing containments, where operator actions are required in a rather short term in the case of certain extreme events.

Since the operators are required to act neither in a preventive way nor in a mitigative way for a substantial period of time after accident initiation, they may perform the following supplementary functions:

- identify, if possible, the causes of the accident;
- verify the correct operation of the systems expected to operate and initiate corrective actions, if needed;
- monitor the accident evolution;
- plan medium and long-term actions to reach a final stable and safe condition and to terminate any release to the environment.

These actions could further and substantially reduce the probability of a large release to the environment in respect to PRA estimates.

With reference in particular to the containment systems, i.e. of the mitigative measures, operators will mainly verify containment integrity (e.g. the closure of all isolation valves), act, if necessary, to reduce excessive containment challenges from rapid energetic events, and reduce pressure and temperature inside containment as soon as possible.

Adequate accessibility to all areas where operator actions may be required must be assured as well as adequate radiation protection in these areas. Attention, therefore, will be given to the location of key equipment and to the extension of secondary containment, which has to be assumed to become contaminated.

Of course all these actions must be supported by adequate instrumentation both for monitoring purposes and for control purposes. Containment instrumentation requirements will include pressure and temperature measurements, hydrogen (or oxygen) concentration, radiation levels in the containment atmosphere, etc. Containment gaseous and liquid sampling must be provided. Proper attention must be given both to the instrumentation ranges and to their environmental qualification.

3. OTHER GENERAL OBJECTIVES FOR NEXT GENERATION PLANT CONTAINMENTS

This report gives a major emphasis to improvements introduced in advanced containments from the safety standpoint. It must also take into account, however, that no containment will be ever built unless NPPs will be economic in construction and operational terms as well as safe for their personnel and the public. Therefore appropriate emphasis must also be given to these additional objectives, which have an important influence on the technological choices.

3.1. SIMPLIFICATION

Simplification is generally pursued in all aspects of future plant designs. Simplification may be introduced in several ways, namely:

- **Simplicity in the overall design**: this would reduce construction costs, possible design errors, uncertainties in the assessment of containment behaviour, etc. (examples of this trend are the general reduction in the number of containment penetrations with respect to existing containments, the reduction of containment system interactions with other process or preventive systems).

- **Simplicity in individual component design** (such as penetrations, isolation valves, instrumentation and controls, etc.): this in general will tend to improve the reliability of such components (an example is the careful evaluation of some innovative proposals for isolation valves).

- **Simplicity in operation**: this will reduce the demands on operating staff both in normal and in accident conditions and will reduce the possibility of an operator error; containment status and integrity could be judged more easily and corrective action may be carried out sooner and with greater efficiency; accident management can be facilitated (examples are the greater level of automation of the containment system, a much longer "grace period" after an accident initiation, the use in some cases of passive systems, etc.).

- **Simplicity in maintenance and testing**: this also will improve the reliability of the containment performance, will reduce the possibility of human errors, etc. (an example is again the reduction in the number of isolation valves, the possibility of increasing the interval between the integral tests for leaktightness giving credit to continuous monitoring systems, etc.).

3.2. STANDARDIZATION

A second factor very important for the economics is standardization. Also this term may assume, as simplicity, different meanings, when evaluated from different standpoints.

- **Standardization of systems and components**: a clear trend for next generation plants is the internationalization of the design of entire plants (and their systems and components), since national reactor designs are unlikely to be supported in the future for a number of reasons. On the one hand this type of standardization will reduce engineering costs and will simplify the exchange of operating experience, but on the other hand, it will require a greater convergence in the licensing requirements of different countries. With specific reference to the containment system effective internationalization will require important steps towards convergence in all aspects of licensing, which appears to be difficult to achieve in the short term on a worldwide basis, but may be easier on a regional basis, for example in Europe.

- **Standardization in components among different plants**: a first step is to harmonize component functional specifications and possibly detailed manufacturing specifications, including design codes, materials, etc. The reasons for this are generally the same as above but so are the difficulties. In some cases the difficulties are even bigger because of concerns about proprietary information and the will to keep national industrial codes alive may be stronger.

- **Standardization in components inside the same plant**: another concept is to increase the standardization of the components in the same plant. This concept, while having a number of advantages in term of maintenance, spare parts, simplicity and so on, has to be balanced with the drawback of an increased potential for common mode failures. Its acceptability must be assessed on the basis of PRA evaluations.

3.3. PLANT AVAILABILITY

Even though the basic function of the containment is fulfilled in accident conditions, the containment structures themselves interact with the global availability of the plant in normal operating conditions. Among other topics:

- If a continuous monitoring system is present the detection of a non-acceptable leakflow during operation could, as required by general technical specifications, lead to a plant shutdown; so it is highly desirable to provide adequate:
  · reliability in the procedures for detection of such a leak,
  · ease of determination (and elimination) of the cause of the leak.

- Other containment evaluation tests, performed in outage time (mainly leaktightness of the penetrations, isolation valves and of the global containment), should not constitute the critical path of this outage. This leads to:
  · single access and test procedures,
  · easy single diagnosis of faults and repair procedures.

In reality, periodic evaluation of the overall containment conditions extends the outage. The tests must therefore:

- Be easily preplanned and prepared.

- Require limited time. There are two possible approaches:
  · a periodic test at design pressure provides adequate representation of the behaviour of the containment, but lasts several days (e.g. present French approach),
  · a periodic test at reduced pressure may be quicker, but needs a careful extrapolation to assess the containment stale (e.g. present German approach).

- In outage conditions, the containment function must remain fulfilled as long as accidental conditions remain possible while allowing maintenance operations to be performed. This requirement can lead to delays in access and thus to an increase in outage time. In that case, optimization between the investment cost of better technical features (e.g. adaption of access airlocks) and the cost of an outage increase in outage time must be performed.

- Provisions for LOCA-type or severe accidents in the design of the containment systems must not lead to technical solutions (for the mitigation of the consequences of severe accident situations) which would significantly increase outage times, for example:
  · a shield slab on the primary circuit vessel aimed at retaining eventual upward missiles from this vessel must be designed to be easily and quickly removed for refueling operation;
  · inerting techniques are helpful to reduce the hydrogen combustion risk, but they have a big effect on conditions for access to the Reactor Building; moreover, they necessitate monitoring (and corrective possibilities) on oxygen content during operation;
  · the suppression pool concept may also interfere with operation and maintenance, if contamination spreads to areas where access may be needed;
  · a similar comment may apply to a cavity provided to collect material from a molten core depending on its design (some of them do not provide easy maintenance or inspection).

- The availability of the plant may be linked to the reliability of the containment isolation signal; its logic and technical process should lead to a very low risk of inadvertent operation.

3.4. MAINTENANCE, INSPECTABILITY AND TESTABILITY

The importance of demonstrating (and maintaining at an acceptable level) the reliability of the containment during plant life highlights the role of its maintenance, inspectability and testability.

Containment inspectability procedures exist (e.g. procedures derived from ASME, Addenda to Section XI). They require special technical training for the maintenance personnel to ensure adequate effectiveness (critical judgement is necessary). For advanced containments, they need to be supplemented to address the inspection of components specially devoted to severe accident mitigation (e.g. core melt retention devices and hydrogen recombiners). Moreover, the testability of these components (and the relevance of these tests to severe accident conditions) should be analyzed during the design phase.

A lot of containment leakage test requirements are derived mainly from 10CFR50, Appendix I, "Leakage Rate Testing of Containments of Light-water cooled Nuclear Power Plants" [5]. This document is presently being modified (in a direction that, according to US utilities, "could further reduce contamination, exposure and unnecessary use of resources without any reduction in the health and safety of the public" [5]) in order to clarify, amongst other issues, the test conditions for isolation valves and some of the acceptance criteria. The document allows a choice of approaches; one of the main examples is the regularity and the pressure level of the overall periodic test (mentioned above in Section 3.3).

Conditions under which maintenance must be carried out derive directly from:

- the layout (and resulting accessibility) of the isolation valves and of the associated components;
- the choices, for these elements, made on the basis of the above mentioned simplification and standardization items;
- the complexity of the chosen components, especially those devoted to severe accident mitigation (e.g. the type of H_2 igniter);
- the attention devoted, in the design phase, to the possible, use of innovative solutions (notably robot elements) either initially or at a later date.

The interaction of the containment on the maintenance aspects of the overall plant has been mentioned in Section 3.3.

In addition, the role of the containment as a last barrier in the case of accidents should lead to analysis of a (hypothetical) post-accident phase, in order to maintain its integrity in the mid or long-term. This innovative analysis may be linked to robotics possibilities, (because of their ability for use in a hostile environment) for which some preliminary studies have been made (e.g. the MESSINA project in Germany and the CEA Sherpa Project in France).

3.5. PLANT DESIGN LIFE

Within the last two decades plant life extension[1] has become more and more credible (based on accumulated experience) and realizable. This trend will require

[1] The presently defined goals for plant life vary from project to project but range from 30 years to 100 years (cited in some CANDU presentations); 60 years is often cited (e.g. AP-600).

- demonstration of reliability with respect to safety standards, throughout plant life,

- the constraints, in terms of safety cost, availability and technical feasibility, of possible ideas for the (complete or partial) replacement of containment systems and components during the plant life.

Several factors have led to an increase in proven reliability during the last two decades; advanced containments will take advantage of these factors and in particular:

- better knowledge of ageing and degradation mechanisms of existing containments such as:
  - corrosion of metallic shells,
  - creep and relaxation modes of prestressed concrete,
  - leaktightness of large openings,
  - leaktightness and reliability of isolation valves,
  - concrete attack from aggressive chemicals, reactions with oxygen, etc.
  - corrosion, elevated temperature and effects of radiation on reinforcing steel and prestressing cables,
  - fatigue,
  - coating deterioration;

- improvement of design, construction and installation quality assurance and quality control;

- availability of new materials which may present better performance and resistance to ageing;

- better methods (testing procedures, analytical evaluations) used to assess the ageing of containments and, additionally, to extrapolate its behaviour to the future;

- use of continuous monitoring leakage systems, which improve confidence in containment leaktightness and prevent large openings from going unnoticed.

At the same time containment design should not limit other actions to assure that the plant can be safe and operational for its entire design life, e.g.:

- the inspection necessary for other components to evaluate their ageing,

- the possible replacement of components (see Section 3.6, for example).

3.6. HANDLING AND REMOVAL OF LARGE COMPONENTS

One of the main features of a containment structure is to allow technically the passage of components through it, either in the construction phase or during operation. The recent replacements of steam generators which have occurred in different plants, have recalled this feature.

The initial design must take into account:

- the trend of increasing plant life increases the probability of such an operation;
- if the containment design is not adequate in this respect, special work aimed at enlarging the access opening is needed; today this is plant specific, difficult and costly; moreover, after the reestablishment of its integrity, it must be demonstrated that the reliability of the containment remains acceptable.

3.7. DECOMMISSIONING

In decommissioning operations, the containment type is an important item, because its presence allows work to be performed inside it while retaining within it the radioactivity accumulated in the inner structures. The final decommissioning operation may be postponed for a prolonged period and the containment must retain its confinement function for a duration of several decades; during the initial design a preliminary evaluation of the following items would be useful:

- After shutdown, will the containment shell be able to withstand several supplementary decades with less demanding loading conditions?

- What design features could ease the sealing, at that time, of the penetrations? Can these features be considered economically?

Another alternative, for decommissioning, is total dismantling of the plant. Preparation for such an eventuality is unlikely to be made during the initial design phase since it is not an easy task at present for the initial designer to imagine the possible processes that would be used for such an operation several decades from now.

3.8. ACCEPTABILITY BY PUBLIC OPINION

The present document deals with containment design characteristics related to safety and technical considerations formulated by specialists.

However, the designer must take into account that the containment structure, because it is presented as the last barrier for the retention of fission products in the case of an accident, must also be judged credible by non-technical people. This implies:

- explanations dealing with the overall performance of the containment should be simple and easily understandable,

- information should be given in such a way that non-technical people can easily understand that the design provides clear and straightforward solutions for the accident scenarios which have happened in the past,

- the public should perceive that a consensus, possibly at an international level exist between the designers, the utilities, the public authority and the scientific community on design bases, licensing criteria and performance assessments.

3.9. COST CONSIDERATIONS

Cost reduction is an important design objective for new Nuclear Plants in parallel with goals of improvements to safety. In the case of containment systems, difficulties might be expected, since an advanced containment design has a large potential for being more expensive through its need to:

- withstand a wider range of accidental situations,
- incorporate extra equipment to cope with the additional threats to be taken into account,
- comply with modified (and possibly more complex) licensing and assessment procedures.
- need to develop and test innovative components and systems.

The use of proven technologies, methods and codes, whenever compatible with the new goals is a fundamental means for cost control.

There are both design and management issues which affect the cost of containments. Measures to minimize costs which are part of the design include:

- use of improved materials (concretes, steels, etc.),

- detailed planning of construction schedules to avoid interference between containment construction and that of its contents,

- optimization of construction technique for instance by use of modularization,

- early formulation of design criteria to allow integration of the design of all systems from the beginning taking account of access and maintenance requirements,

- simplification of design, e.g. reduction in the number of penetrations, avoidance of over specified features,

- design for long plant lifetime and including provisions for decommissioning.

Factors which arise from issues of management including the overall management of the industry and its regulatory environment include:

- assuring regulatory stability,
- assuring design stability to share development costs over several stations,
- assuring construction schedules,
- minimizing operation and maintenance costs.

4. DESIGN BASES

4.1. CURRENT DESIGN BASES

Current design bases for the containment are generally based on pipe breaks that are postulated inside the containment and on the consequent releases of mass and energy, causing increase in temperature and pressure. Since these releases are usually accompanied by releases of fission products, it is required that the containment maintains a high level of leaktightness. A double ended guillotine break of the largest primary or secondary pipes, whichever gives the highest peak pressure, is generally taken as the design basis accident for containment design.

This approach has been recognized for a long time as being not the worst accident case; its use, together with conservative assumptions, has been judged as covering the consequences of best estimate assumptions in calculations of more severe accidents. In addition, in the very remote case of an accident with containment failure, protection of the public is assured by the contribution of prompt evacuation of the people in the neighborhood of the plant.

This worldwide approach is described in Sections 1–3. Examples of current containment system design concepts are described in Ref. [1] and Appendix A of this report.

4.2. EXTENSION OF DESIGN BASES

The approach briefly outlined in Section 4.1 has proved to be sound in the specific conditions of the Three Mile Island accident and may continue to be demonstrated as providing satisfactory safety to the public. Today, there is the possibility, based on research results collected in the past few years, to address directly the effects of the most severe accidents, including those involving melting of the core.

Selection of accident sequences which will form the design basis for containment design for the next generation of plants is generally based on a combination of probabilistic and deterministic assessments. Engineering judgement is used to remove the most severe very low probability sequences from the design base. Deterministic calculations for severe accidents are generally carried out with a less conservative approach than is used for e.g. design basis LB-LOCA. This applies both to internal and external initiating events, for example, aircraft impact events.

Special attention is being given to seismic design issues. The extension of the design bases to very low probability events has put into question the need for considering the effects of natural events in the same probability range. This is in general very difficult because it is almost impossible to use historical data for events in the range of 1E-5 or 1E-6 per year. The extrapolation of existing data could lead to unrealistically high acceleration values that would be unacceptable for containment design. Two different approaches can be mentioned as examples of proposed solutions to this issue:

- the first is the assessment of seismic margins,
- the second is a geophysical evaluation of the proposed sites to find any bounding values of the physically possible accelerations.

Containment designs aiming at standardization will use bounding site related events, or, at least, provide the possibility of improving the design to cope with those events at more onerous sites. Since safety analyses are performed giving credit only to the safety systems, additional margin to safety is provided by non-safety systems, which must be adequately tested for operation both during and after the extreme accidents.

4.2.1. Categories of challenges to be considered

A prerequisite for containment design extension into mitigation of severe accidents requires specification of the acceptable limitations on radioactive release to the atmosphere or ground water. These limits will not be the same as those appropriate for the earlier design basis accidents but will be dependent in some way on the probability of the accident sequence. A degree of decoupling between accidents and consequence calculation is often achieved through definition of a source term [2].

To ensure the limitation of source term/dose the fundamental design requirement for the containment is to remain leaktight. Leaktight means that direct leakages to the environment have to be prevented. Leaktightness of the containment requires:

. maintaining the integrity (structural and functional) of the containment structures,
. proper isolation of the penetrations/large passages,
. prevention of containment by-passes (e.g. in the case of uncontrolled interface LOCA or steam generator tube rupture (SGTR)).

Phenomena associated with severe accidents which may lead to additional challenges to the containment, see for example references [2, 6–9], are in principle as follows:

- high pressure RPV failure/direct containment heating,
- steam-explosion (in-vessel/ex-vessel),
- hydrogen deflagration/detonation (local, global),
- melt attack on the containment structure or pressure boundary,
- pressures and temperatures inside the containment caused by gas release and decay heat from the molten core or generation of non-condensable gases by corium/concrete interaction,
- reactivity accidents.

If the containment function is maintained intact in a severe accident, the radiological consequences will be minor. If the containment function does fail, the timing of failure can be very important. The longer the containment remains intact relative to the time of core melt and radionuclide release from the reactor coolant system, the more time is available to remove radioactive material from the containment atmosphere by engineered safety features or natural deposition processes. Delay in containment failure or in containment bypass also provides time for accident management, a very important consideration in the assessment of possible early health effects. Thus in evaluating the performance of a containment, it is convenient to consider

- no failure
- early failure
- late failure
- containment bypass

as separate categories characterizing different degrees of severity. With respect to this characterization, the challenges to the containment can be assigned as follows:

Early containment failure caused by

· high pressure RPV-failure/direct containment heating,
· steam explosions,
· hydrogen combustion,

- isolation failures,
- reactivity accidents.

Late containment failure caused by

- melt attack on the containment structures or pressure boundary,
- pressure and/or temperature increase inside the containment.

Containment bypass caused by

- interfacing system LOCAs,
- SGTR.

4.2.2. Strategies and associated solutions to cope with the challenges

To cope with the different challenges listed in Section 4.2.1, it is necessary to identify the loads on the containment resulting from these phenomena.

It has to be determined whether the containment system can cope with these loads. Realistic assumptions and best estimate calculations are the basis. Gaps in knowledge of specific physical phenomena are filled by technical design solutions with sufficient margins based on the current state of the art.

If the loads are not acceptable to the design (e.g. with respect to steam explosion), two possibilities exist:

- technical measures are taken to reduce the loads to an acceptable limit or to exclude the events,
- or the technical design is improved so that it can withstand the loads,
- or a combination of both.

Dynamic loads are preferably avoided by technical measures, e.g.:

- high pressure core melt sequence converted into a low pressure core melt sequence,
- detonation loads prevented by installation of H_2-reduction devices, or by a large containment volume, or by providing pre-inertisation.

The adaptation of the design to the loads of the different challenges is an iterative process. For each challenge, a description of possible associated general solutions best able to fulfill the above mentioned objective is presented below. Section 6 will deal with more specific individual design projects which are presently underway in the nuclear industry.

Control and reduction of the pressure and temperature

In case of core melt accidents the *pressure and temperature increase* as a result of the decay heat from the core melt (the heat removal systems are postulated to be failed) and by non-condensable gases (Zr-H_2O reaction and molten core–concrete interaction (MCCI)). *The objective* is to limit the pressure increase below an acceptable containment pressure and to reduce the pressure as quickly as possible down to atmospheric pressure in order to limit the release rate from the containment. The dedicated systems should have no active components inside the containment because of the harsh environmental conditions which might persist there for a long time. Adequate shielding and decontamination provisions must be made if containment atmosphere or sump water are circulated outside the containment.

Different technical solutions have been proposed:

- Spray system inside the containment

  A spray system is very effective in reducing the pressure and temperature as long as the partial pressure of the non-condensable gases is not dominant and the sprayed water is cooler than containment atmosphere. The drawback may be that either the active components have to be inside the containment or very highly radioactive fluids have to be circulated and cooled outside the containment. Passive spray systems have also been proposed, but they have limited capacity for pressure reduction and doubtful efficiency for aerosol removal. Consequential damage to these systems has to be taken into account.

- Spray on the steel shell containment from outside

  No active components are inside the containment and no circulation of radioactive fluids outside the containment is necessary. The heat transfer coefficient to the steel shell depends on the partial pressure of the inert gases inside the containment. The pressure can only slowly be reduced so that leakage from the containment persists for a longer period.

- Condenser-system in the atmospheric part of the containment

  This system requires an intermediate circuit to dissipate the heat to the external atmosphere. The intermediate circuit may be passive (natural circulation) or active, and generally makes use of water (single phase or boiling). The heat transfer coefficient to the heat transfer surface depends on the partial pressure of the inert gases inside of the containment. The pressure can again only slowly be reduced and leakage will persist. Consequential damage to this system has to be taken into account. The heat transfer surfaces must be very large.

- Cooler in the sump water

  A sump water cooler can reduce the pressure down to atmospheric. Consequential damage to this system has to be taken into account. A sufficient level of natural circulation in the sump water is needed and therefore a good mixing of the sump water is a prerequisite for this system.

It may be that one system is not able to meet all the objectives so that an adequate combination of the different systems has to be optimized especially for the different requirement for the short-term and long-term.

Corium-concrete interaction

If the corium cannot be maintained inside the reactor vessel, it will reach the reactor cavity and eventually the containment basemat.

In the case of severe accidents the decay heat may cause a *basemat penetration* by the core melt. *The objective* is to prevent basemat penetration in the long-term phase. The dedicated features and systems should cool the corium so that migration of the corium can be stopped in time and it can be solidified. The technical solution should not require the function of active components inside the containment.

Different technical solutions have been proposed.

- Spreading of the core melt

  A sufficiently large area could be provided to allow the corium to spread into a coolable geometry. The corium is then flooded from the top. Understanding of the spreading behaviour of the corium is a very important prerequisite.

. Retention device

A special retention device could catch the corium. The corium may be then flooded from the top and/or cooled from the bottom by natural circulation or active means. The retention device must withstand all the loads resulting from the RPV failure (e.g. steam explosion, missiles).

Airborne fission product (FP) reduction system

The objective is to reduce the aerosol and iodine content in the containment atmosphere released from fuel damage or core degradation fast enough to minimize radioactivity release by containment leakages.

Systems which support this objective are:

- *Containment structures as a FP reduction system*
  The containment structure and its internals provide a large surface area for aerosol deposition.

- *Ventilation systems*
  Ventilation systems can be used for cleaning exhaust air for mitigation of accident consequences, especially for sorption of iodine. Ventilation systems can be used in the annulus of dual-containment systems or from secondary confinement which may become contaminated with airborne fission products during accident conditions, as a result of leakage from the containment.

- *Containment spray systems*
  The aim is to dissolve or to entrain airborne FP and to retain them in the water of the containment sump. Steam and air may also be passed through suppression pools or other bubbling mechanisms with the effect of reducing the content of iodine and aerosols. Chemicals such as sodium hydroxide, sodium thiosulfate or hydroxine are added to the water spray or to pools to enhance the removal of FP (especially iodine and caesium). As mentioned before, passive spray systems do not appear to be effective due to the limited capacity and plan rate obtainable with them.

- *pH control systems*
  Water pools where iodine and iodine components may be trapped after an accident have generally a controlled pH ≥ 7 throughout the accident to prevent reevolution of activity in the long-term.

Hydrogen management

In the case of severe accidents there is a possibility of a large generation of H_2 by the Zr-H_2O reaction and by metal (Cr, Fe) H_2O reaction particularly in the case of MCCI and there is an additional contribution from radiolysis. Proposed design basis metal-water reaction percentages vary between 75% and 100% of active cladding. In the long-term the radiolysis contribution should be taken into account. The objective is to avoid a strong H_2 deflagration or even detonation which could lead to a damage of the containment.

Different solutions have been proposed. For all of them it is very helpful if MCCI is limited by appropriate measures.

· Preinertization

In the case of relatively small containments, preinertization is the most effective measure. During power generation, the containment is not accessible and at other times only with some difficulty.

- Large containment volume

  A large containment volume is provided to keep the global and local H_2 concentration below the detonation limit. Maximum proposed acceptable concentrations vary between 10% and 13% of H_2 in dry air. A good mixture of the containment atmosphere by natural circulation has to be ensured, and, therefore, compartmentalization should be limited.

- H_2 reduction devices

  Two categories of devices can be considered inside the containment. Igniters, catalytic or dedicated battery driven burn the hydrogen at a low concentration avoiding unacceptable pressure peaks. Catalytic recombiners or other types are generally needed to reduce the H_2 concentration in the long-term phase in particular when the H_2 concentration is below the ignition limit.

In the case of preinertised containments, proper attention should be given to periods (such as refueling) when the containment is not inerted. At the same time oxygen build-up caused by water radiolysis should be monitored and controlled.

Features to improve leaktightness

The main function of the containment during a severe accident and, during any accident with the potential for releasing significant quantities of radioactivity to the environment, is to maintain a high level of leaktightness. This function, of course, must be assured under a harsh environment (involving pressure, temperature, radiation, aggressive chemicals, etc.) and various types of loads on the containment structures.

Leaks may occur from:

- containment structures,
- penetrations,
- hatches and locks (including the bolted head of the SBWR drywell),
- isolation valves,
- electrical penetrations,
- systems open in the containment after an accident and crossing its boundaries.

In addition, all types of the so-called containment bypasses must be taken into account; in particular the steam generator tube rupture event in the case of PWRs and the interfacing system LOCAs. Leaks through all these pathways must be avoided or, at least minimized. The first attempt to improve the leaktightness is to reduce the number of penetrations; this is being pursued by all designs and it goes along the general line of plant simplification. In addition some specific provisions have been proposed, which are of high importance particularly for single containments and for direct pathways to atmosphere in double containments (some of these are not completely new and are implemented in some existing containments):

- pressurization systems designed to keep the penetrations at pressures higher than the peak containment pressure,
- suction systems designed to collect any leak from the first barrier (e.g. the inner containment isolation valve), to treat it and to release it through a controlled point,
- special components such as bellow's type valves,
- seal welds on large equipment hatches, etc.

Leakages from containment structures are generally very low both for steel shell containments and for concrete containments with liner. In the case of concrete containments without a metallic liner these leaks directly via the concrete structure may be a significant percentage of the overall leakrate. Special leaktightness problems may arise, when considering potential accidents initiated while the plant is in a shutdown or refueling conditions, i.e. when the containment configuration may differ from that at power and when some openings may be left for operational reasons. Since PRAs for current plants have shown that the risk of a core melt is not trivial in these conditions, attention must be given to the means to reach the desired leaktightness before the core releases a significant amount of radioactivity.

The importance of secondary containment or, in general, secondary buildings is underlined elsewhere in the report.

A very important issue is the demonstration during plant life that the containment will actually achieve these challenging design goals.

While existing plants are generally leak tested at peak design pressure (e.g. LOCA pressure) or lower, advanced containment must also be proven to provide the required leaktightness at pressures caused by the severe accidents considered. In some cases these pressures may exceed the design pressures (depending on design criteria) and periodic testing may, therefore, not be possible. Adequate rationale must be developed and a proper testing programme must be implemented to support the assumptions used in the safety case.

A second issue is to improve the detectability of the existence of large openings in the containment boundary, which may go unnoticed for a long time, increasing the probability of a 'preexisting opening' should an accident occur. In addition to strict administrative procedures, several systems have been proposed for a continuous or semi-continuous check on containment integrity during normal operation. One of these systems has been developed in France by EdF and is currently used in many operating plants; the system is called SEXTEN and gives an indication of the leaktightness of the containment by the rate of increase of containment pressure caused by the usual leaks in the instrument air system. This system is also likely to be adopted in some advanced containments.

4.2.3. Selection criteria

With respect to the challenges to the containment system design resulting from severe accidents, selection criteria have to be established. These must address:

- which challenges have to be considered explicitly in the design,
- use of margins,
- how are dynamic loads to be handled.

These items lead to definition of the preconditions for deciding on a strategy. To achieve the most effective overall solution, a fourth item must be added to the list:

- how to combine the variety of possible solutions for each individual function into an overall design.

The selection process must take into account:

- impact on layout,
- impact on plant operation during DBAs,
- use of proven materials and technologies,
- efficiency,
- reliability,
- passivity level,
- simplicity,
- maintenance and testability,
- impact of failure and spurious operation.

Reference [9] describes a survey on how to proceed with this process.

5. EXAMPLES OF CONTAINMENT SYSTEMS IMPLEMENTATIONS

5.1. MEASURES TO PREVENT FUEL-COOLANT INTERACTION (FCI)

(a) In-vessel FCI

Under some postulated accident conditions an in-vessel FCI could occur. The resulting steam spike or steam explosion is generally not considered to be sufficiently energetic to cause failure of the containment. However, combining several conservative assumptions, some authors believe that there is potential for rupture of the upper or lower head of the RPV, possibly causing damage to the containment internal structures or essential safety equipment within the containment.

In-vessel steam explosion defense strategies are

- to exclude the possibility by design or by other physical considerations,
- to demonstrate that vessel would not fail [10],
- or to require, if the vessel has to be assumed to fail, a reinforced reactor pit design able to withstand the consequential loads, possibly including an upper shield slab to protect the containment from induced missiles [11].

Specific measures to prevent in-vessel FCI have not been developed.

It should be pointed out that some FCI models imply a greater probability of in-vessel steam explosions if the vessel is at low pressure at the time of the event. This situation is much more likely in many next generation reactors since primary system depressurization is part of the prevention strategy. It should be pointed out, however, that in this case static pressure load should not be added in the evaluation of vessel integrity, and, on the other hand, this event is not likely to cause containment failure.

(b) Ex-vessel FCI

If a substantial portion of the molten core comes into contact with an equivalent mass of water in the reactor cavity, an FCI and consequently a steam explosion could occur. Particularly in small containments, local failures of the containment structures may occur. Several preventive features are possible.

The absence of water in the reactor cavity at the time of RPV melt-through reduces the possibility of an ex-vessel FCI. The various core-catcher and ex-vessel corium cooling systems described in the next section may be relevant to ex-vessel FCI problems.

5.1.1. In-vessel corium cooling

New interest has recently been focussed on the possibility of preventing reactor vessel penetration by the corium, if the vessel itself is externally flooded to an adequate level, and the primary circuit is depressurized. In particular, the AP-600 PRA seems to give credit to this feature, basing this assumption on a number of analytical and experimental results.

If these results are confirmed, the same conclusions could apply to other reactors with similar power density/vessel bottom surface ratio. A need for additional confirmatory tests is claimed by some organizations to give a better understanding of the heat transfer phenomena inside and from the molten corium mass, and the mechanical effects on the vessel caused by high temperatures and temperature gradients.

Finally, it should be pointed out that if there are sequences which would not flood the reactor cavity, then the problem of ex-vessel corium cooling remains unchanged.

5.1.2. Ex-vessel corium cooling

Various means of cooling the corium within the reactor cavity subsequent to RPV failure have been proposed. Most of these measures involve allowing the corium to spread out either in the vertical or horizontal direction, and to cool the corium by flooding with water or allowing heat transfer to water without flooding. Active and passive systems have been proposed for removing heat from the corium.

It should be pointed out that two levels of concepts may be differentiated:

- the concepts which are based on isolated physical phenomena considerations and which need more theoretical studies to be part of an integrated containment design,
- the devices which have been envisaged, at the early phase of design, to be consistent with the overall containment features in respect to physical knowledge of the phenomena.

5.1.3. Systems which allow flooding of the cavity

Figure 2 [10] shows such a concept where the corium dropping into a dry cavity is allowed to spread horizontally over a sacrificial layer. As this layer is eroded hollow plugs become uncovered, allowing sump water to contact the corium and eventually to cover the corium material; cooling of the corium will be maintained by evaporation of the water. Experiments are being performed to investigate the performance of the concept.

Figure 3 [10] shows another solution where a staggered pan configuration is embedded in an oxide ceramic particle bed; the upper part of which remains dry to avoid vapour explosions, and the lower part is flooded with sump water. Natural convection of the sump water is indicated in the figure.

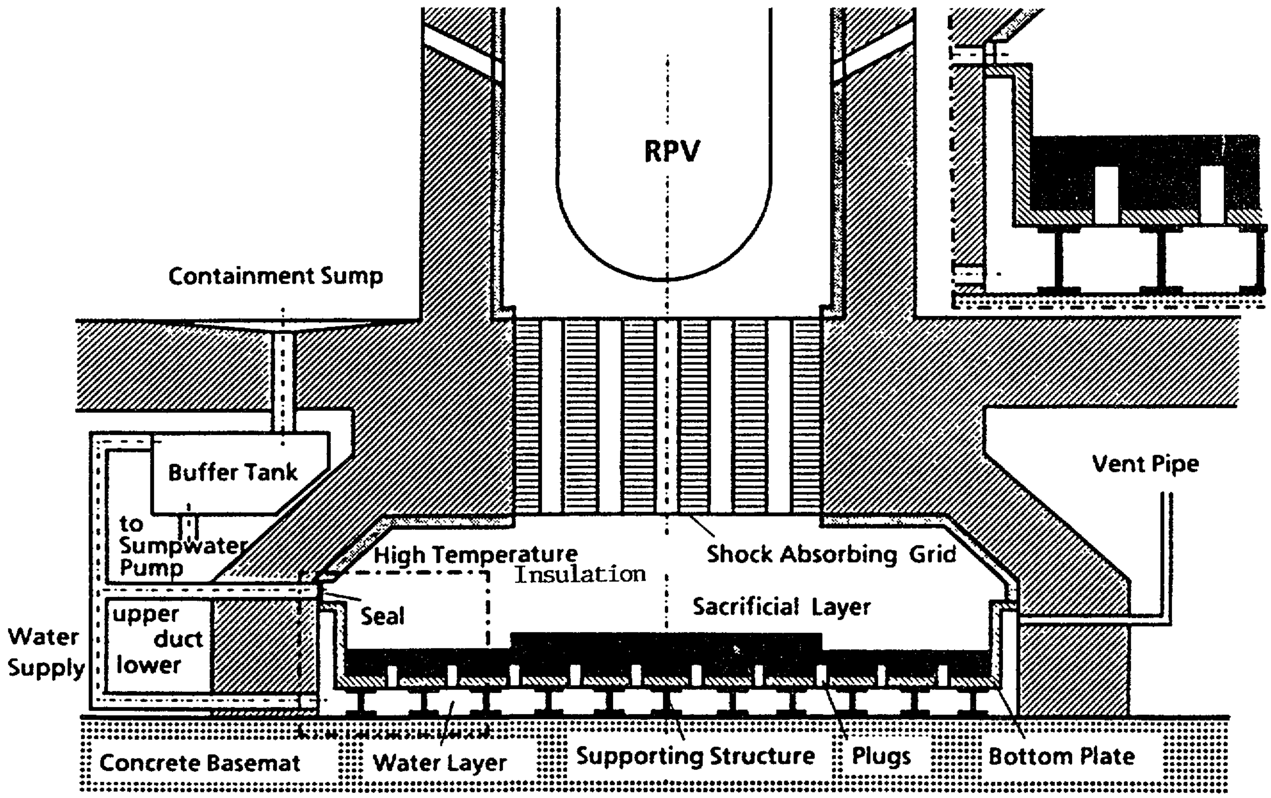


FIG. 2. Core retention device for a modified PWR concept.

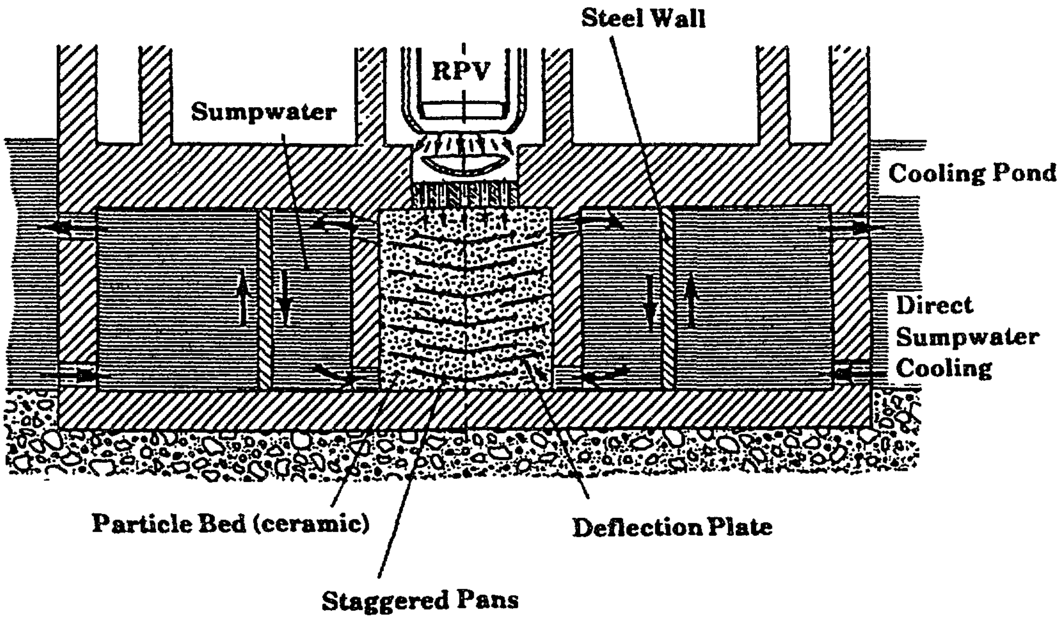


FIG. 3. Core catcher with integrated sump water cooling.

5.1.4. Dry core catcher concepts and devices

A device proposed for the Franco-German European Pressurized Reactor (EPR) [12] allows the corium, after RPV melt-through to leave the dry reactor cavity and to spread horizontally into a dry area of about 150 m^2. Relatively low melting point plugs would then be exposed to the corium and which when melted would allow water from the IRWST to flood the corium (see Figs 4 and 5). Cooling of the corium is maintained in the short-term (between 0.5 and 1 day) by evaporation of water. After that period a dedicated containment heat removal system is put into operation, which ensures the cooling of the containment atmosphere by spraying and for the cooling of the water of the spreading compartment and of the IRWST. The system has the following characteristics:

- no active components inside containment,
- cooling of corium ensured by an intermediate active cooling system with the above mentioned grace period for initiation,
- cooling and electric power supply independent of the safety grade cooling train and emergency power supply.

Special provisions are taken to deal with the recirculation of the water outside the containment.

Experimental research is being performed in the USA (MACE tests) and in Germany, notably, to investigate the performance of core melt retention and cooling by spreading.

Figure 6 shows a dry vertical multi-crucible core catcher concept. The corium is collected in the crucibles which are cooled externally by natural circulation of water [13].

Figure 7 shows a dry core-catcher concept with promotion of thermal radiation [13–15].

5.2. CONTROL AND REDUCTION OF PRESSURE AND TEMPERATURE

Following a severe accident, the pressure and temperature within the containment need to be limited and reduced as quickly as possible. Internal spray systems and various types of heat exchangers are used in current LWR plants. Listed below are some design concepts advocated for advanced reactor containments.

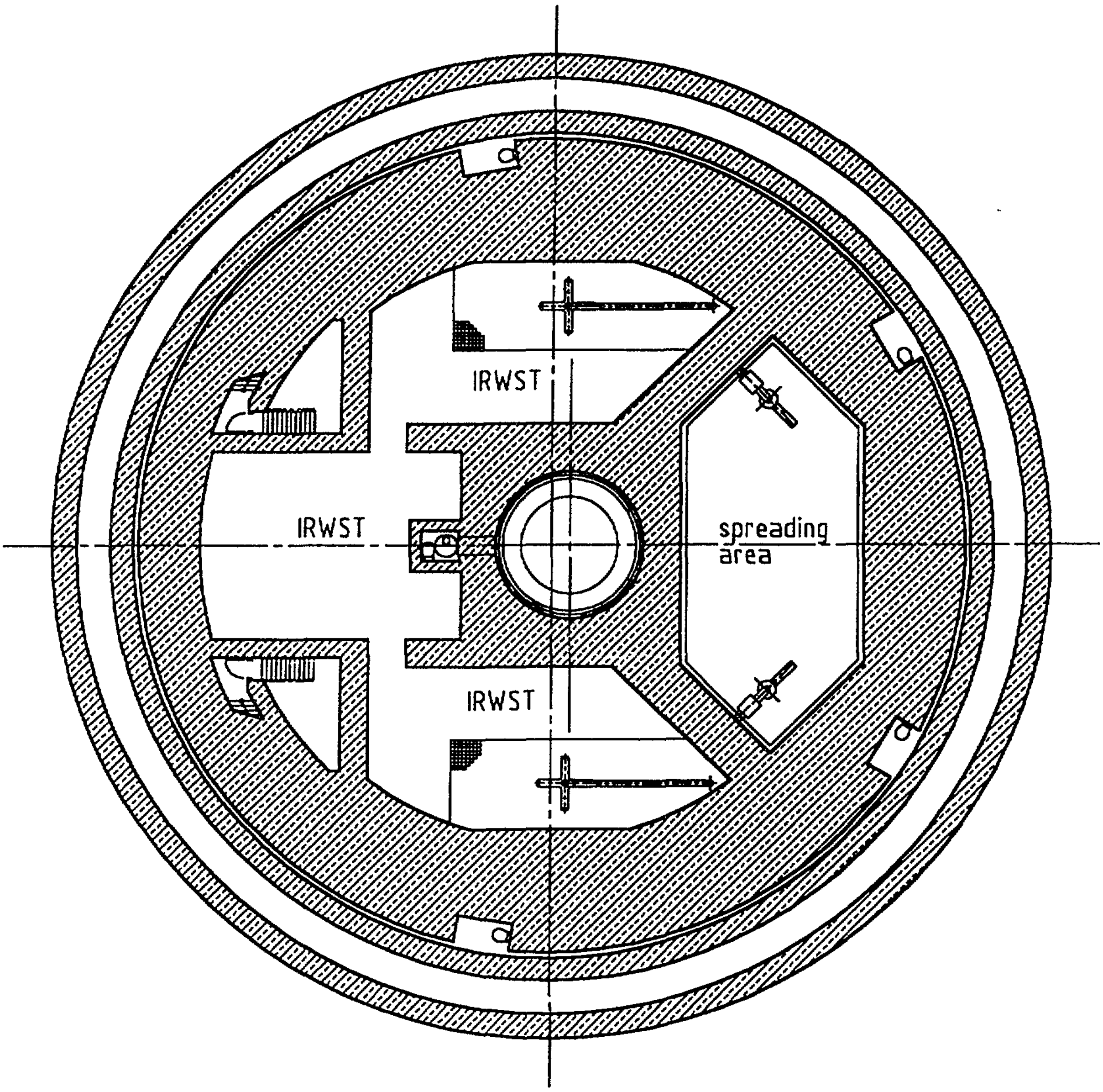


FIG. 4. EPR spreading concept — Plan view.

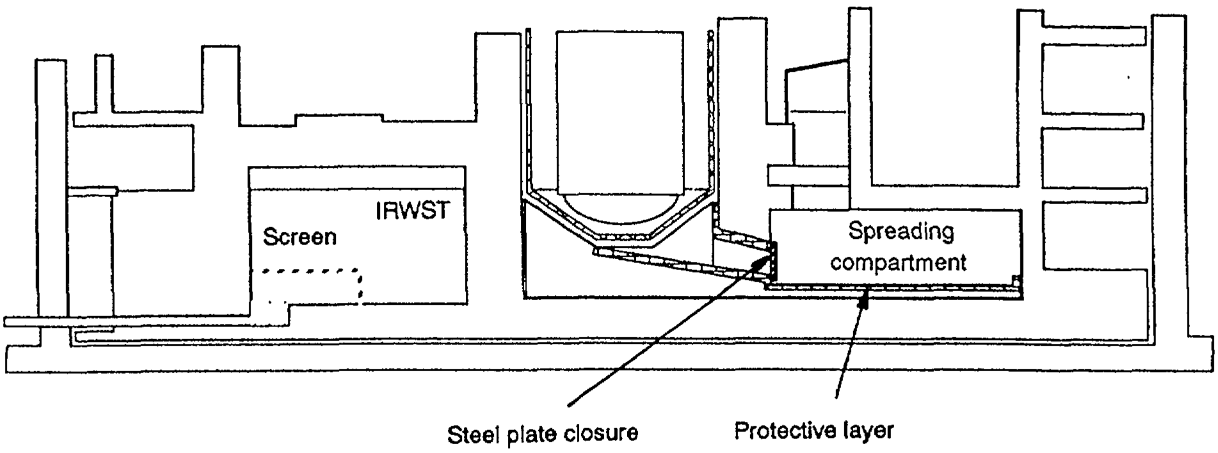


FIG. 5. EPR spreading concept — Containment section.

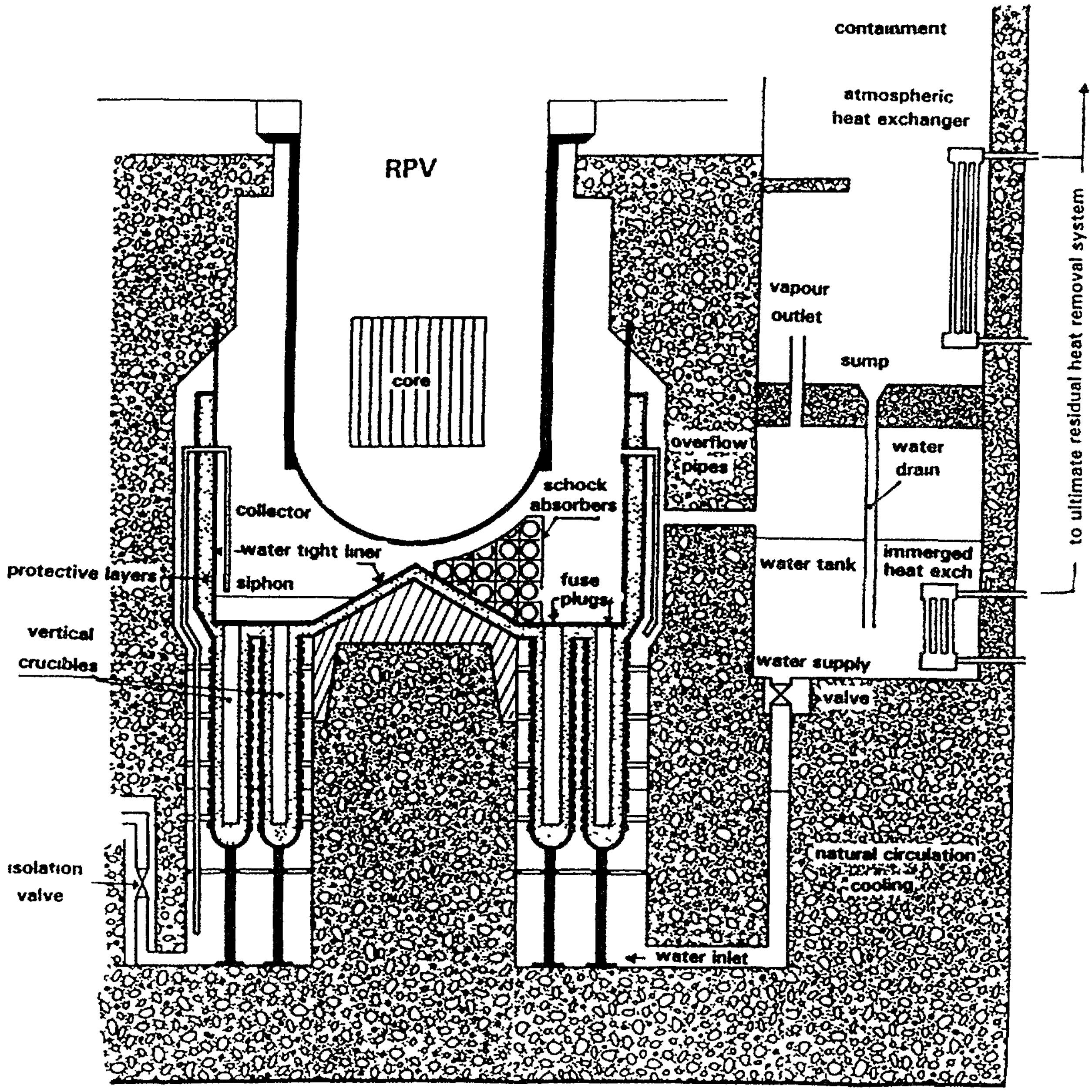


FIG. 6 Dry vertical multi-crucible core catcher concept [13]

As in the previous section these concepts can be divided between those systems which have been fully integrated into an overall containment design and those which are more speculative isolated systems still under development.

5.2.1. Decay heat removal systems

Passive containment cooling systems (PCCS) provide the safety-grade ultimate heat sink that prevents the containment from exceeding its design pressure in a large variety of advanced reactor types:

In the case of AP-600, the steel containment vessel is cooled externally by a combination of downward water flow and subsequent evaporation and an upward flow of naturally convected air as shown in Fig. 8 [14, 15]. (For a review of experiments and tests designed to confirm the performance of the arrangement, see Section 7) The same principle applies to the SPWR

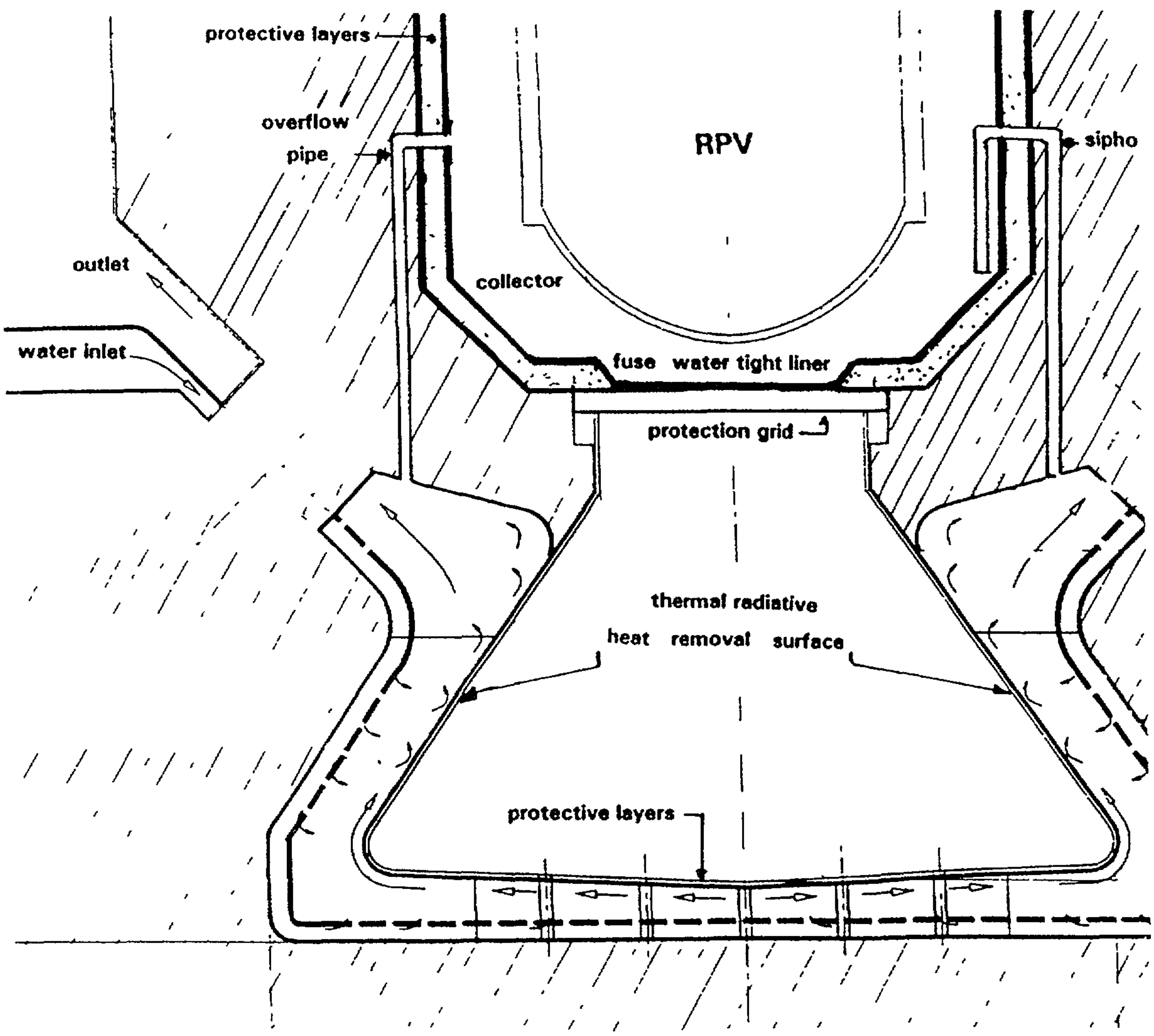


FIG. 7. Dry core-catcher concept with promotion of thermal radiation [13–15].

In the case of the reinforced concrete cylindrical containment structure of SBWR [19, 23, 36] with an internal steel liner, PCCS is employed [8] to remove decay heat from the containment after a LOCA, in order to maintain the containment pressure within its design limits for 72 hours. The system consists (see Fig. 9) of a passive containment cooling condenser located outside the containment in a large pool of water vented to the atmosphere. The condenser receives a steam/gas mixture from the containment drywell; the steam is condensed inside the vertical tubes of the condenser and drained to the gravity driven cooling system (GDCS) pool located inside the containment drywell (Figure 10) [19]. Non-condensable gases are routed to the suppression pool in the wetwell through a vent line. The PCCS is thus both passive in operation and initiation which is an important consideration.

Figure 11 shows a system for passively removing heat from a suppression pool contained within the containment structure [16]. The systems involve natural circulation of the suppression pool water, conduction of heat through a steel primary containment shell and natural circulation/evaporation of water in an outer pool. (Water wall type passive containment system.)

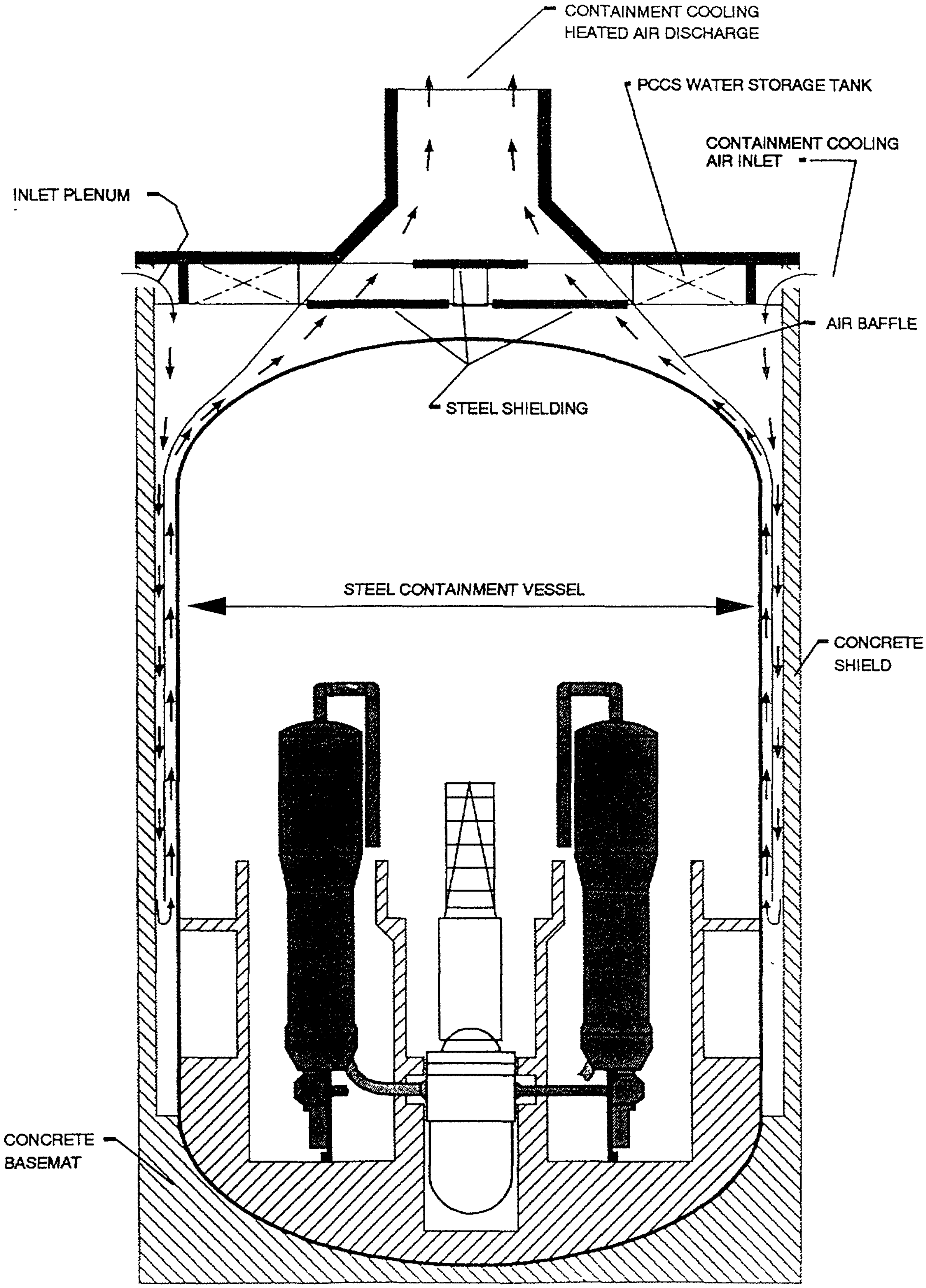


FIG. 8. AP600 containment cooling [14, 15].

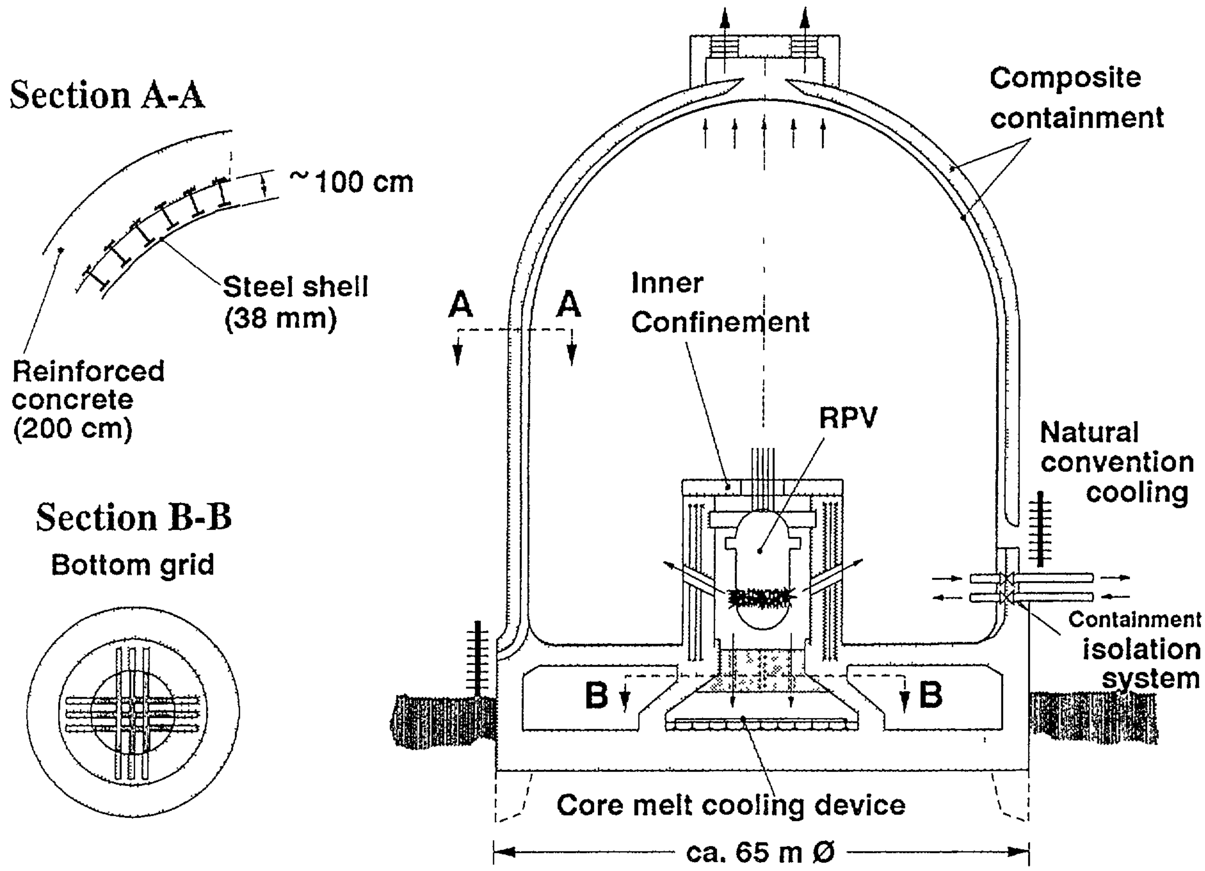


FIG. 9. Conceptual design of a composite containment (PWR) [10, 11].

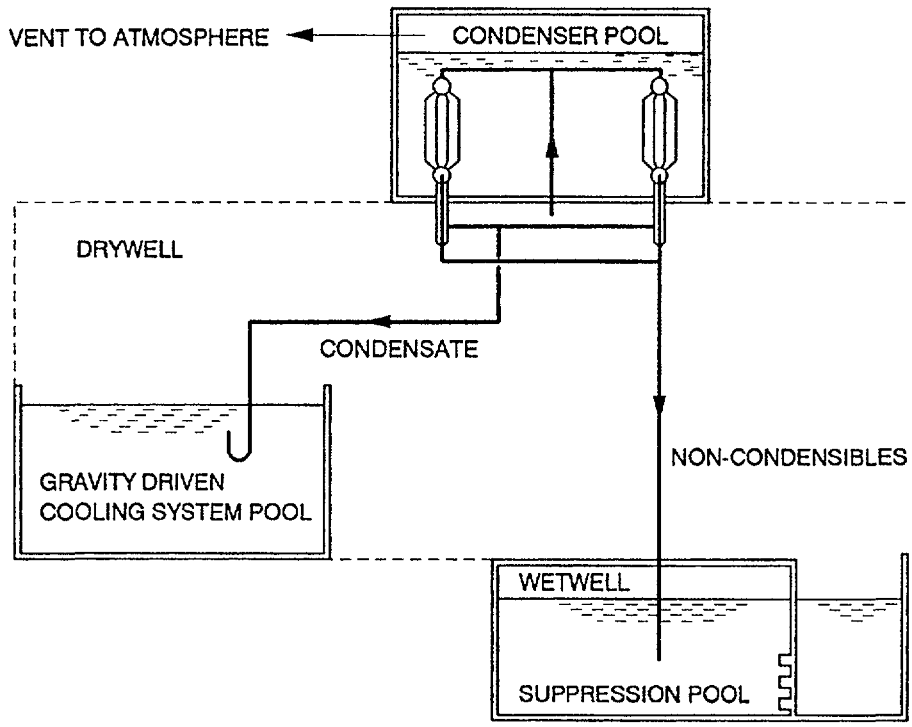


FIG. 10. Passive containment cooling system [17, 19].

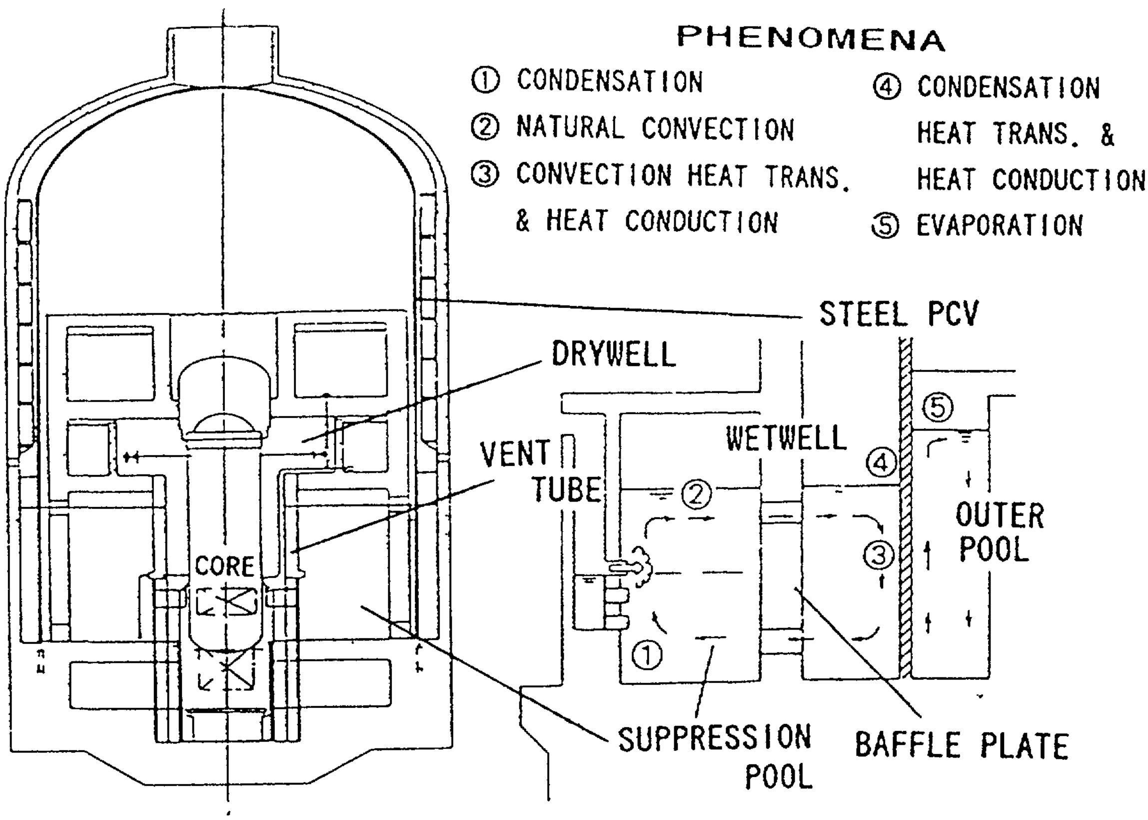


FIG. 11. Concept of water wall type PCCS [16].

5.2.2. Conceptual designs

5.2.2.1. Containment proposed by Kernforschungszentrum Karlsruhe (KfK)

A conceptual design of a composite containment has been proposed by Kernforschungszentrum Karlsruhe (KfK), Fig. 9 [10, 11].

With respect to pressure build-up and control of pressure and temperature the following enveloping extreme "worst-case" conditions have been taken into account:

- Static internal overpressure mainly due to hydrogen deflagration (2.0 MPa),
- Dynamic internal overpressure due to hydrogen detonation (10 MPa/12 ms),
- Failure of the reactor pressure vessel due to high system pressure or steam explosion (Fig. 12),
- A core catcher system (described in the previous section),
- Passive removal of decay heat,
- Leak tight closure of all penetrations,
- Optimized disassembly and removal of the power plant.

Of the various alternatives [22, 24] proposed only the first one is described briefly, inviting the readers to refer to references for a full description.

A composite wall structure is proposed (see Fig. 9) consisting of an inner steel shell 40 mm thick surrounded by a strong reinforced outer concrete shell, about 2 m thick. The containment loads are shared by the two containment shells. Natural convection air flow within the annulus formed between the two shells removes decay heat from the containment. Preliminary assessments suggest that about 10 MW of heat could be removed by this system.

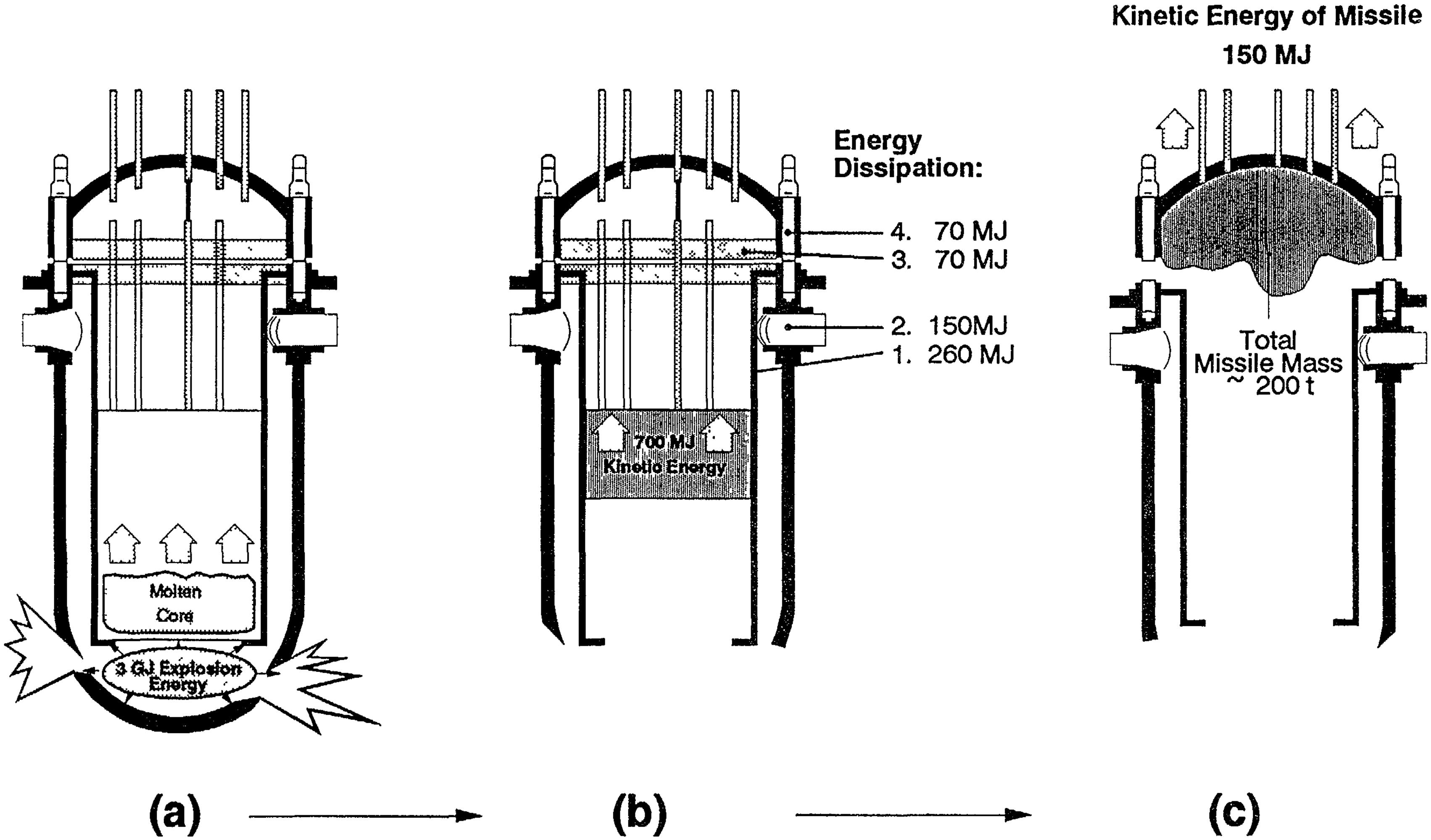


FIG. 12. Energy partition associated with an energetic steam explosion [10].

5.2.2.2. *LIRA*

A conceptual containment design for advanced PWRs has been proposed with the general objective of limiting the accidental radioactivity release to the environment to less than one millionth of core inventory by using substantially passive features.

The design objectives are claimed to be achieved by adopting the vapour suppression principle, with a drywell that is concentric with the wetwell, but still retaining a large and strong containment, (see Fig. 13).

A distinguishable feature [23] is the protection of the integrity of containment basemat realized with a stack of graphite beams in the reactor cavity promoting a 3-D corium spreading and solidification (see Fig. 14).

5.2.3. Direct containment heating (DCH)

Overpressures due to direct containment heating (DCH) should be avoided. Approaches that have been advocated are to provide a reliable primary circuit depressurization system (interaction with FCI phenomenon: see Section 5.1), or construct a system of ledges and walls to deflect core debris and provide an indirect path from the reactor cavity to the upper containment volume.

5.3. HYDROGEN CONTROL SYSTEMS

Hydrogen within the containment can arise from oxidation of Zirconium, steel, concrete, etc., and the objective of hydrogen control systems is to avoid strong deflagration or detonation of the gas. Current reactors usually have hydrogen recombiners and/or igniters within the containment volume.

One solution is to have the containment inerted during plant operation, although this may imply operational disadvantages.

5.4. SOURCE TERM REDUCTION SYSTEM

Minimization of fission product inventory in the containment atmosphere for all accidents, especially for these severe accidents where corium has penetrated outside the pressure vessel, relies upon various features or natural processes, including:

- quenching the molten material and cooling it in a stable way at, or shortly after, its ejection from the reactor vessel,
- avoiding or minimizing molten core-concrete interaction (MCCI),
- fission product scrubbing in large pools of water,
- natural process for the removal of suspended aerosols (e.g. agglomeration, sedimentation, hygroscopicity, diffusioporesis); for this activity large surface areas for deposition are advocated,
- aerosol deposition and removal by the passive containment cooling system (SBWR only),
- internal containment spray, if proved feasible and effective,
- scrubbing effect of water flooding the core debris,
- adequate pH conditions in water pools in order to avoid elemental iodine resuspension.

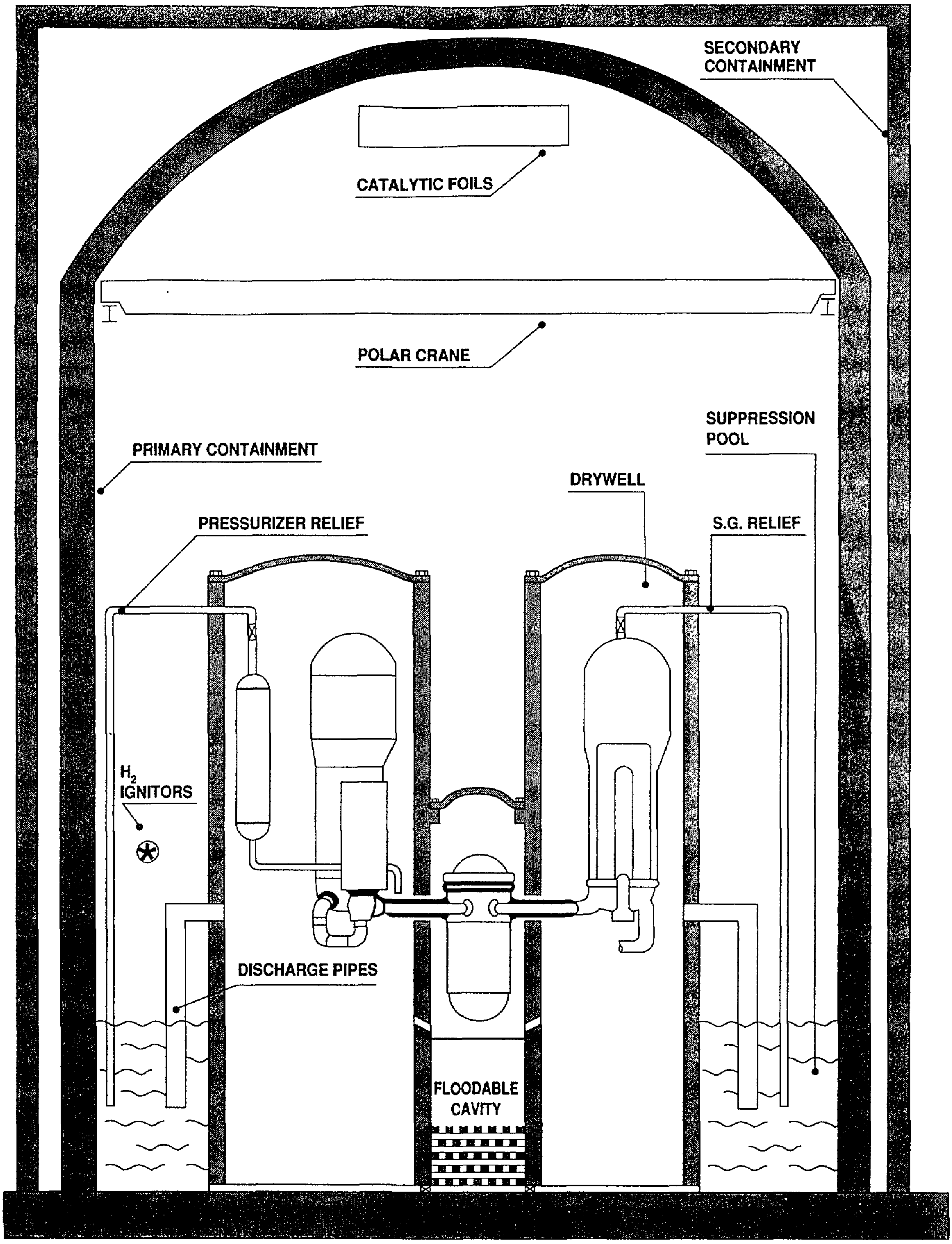


FIG. 13. Schematic arrangement of the LIRA containment [23].

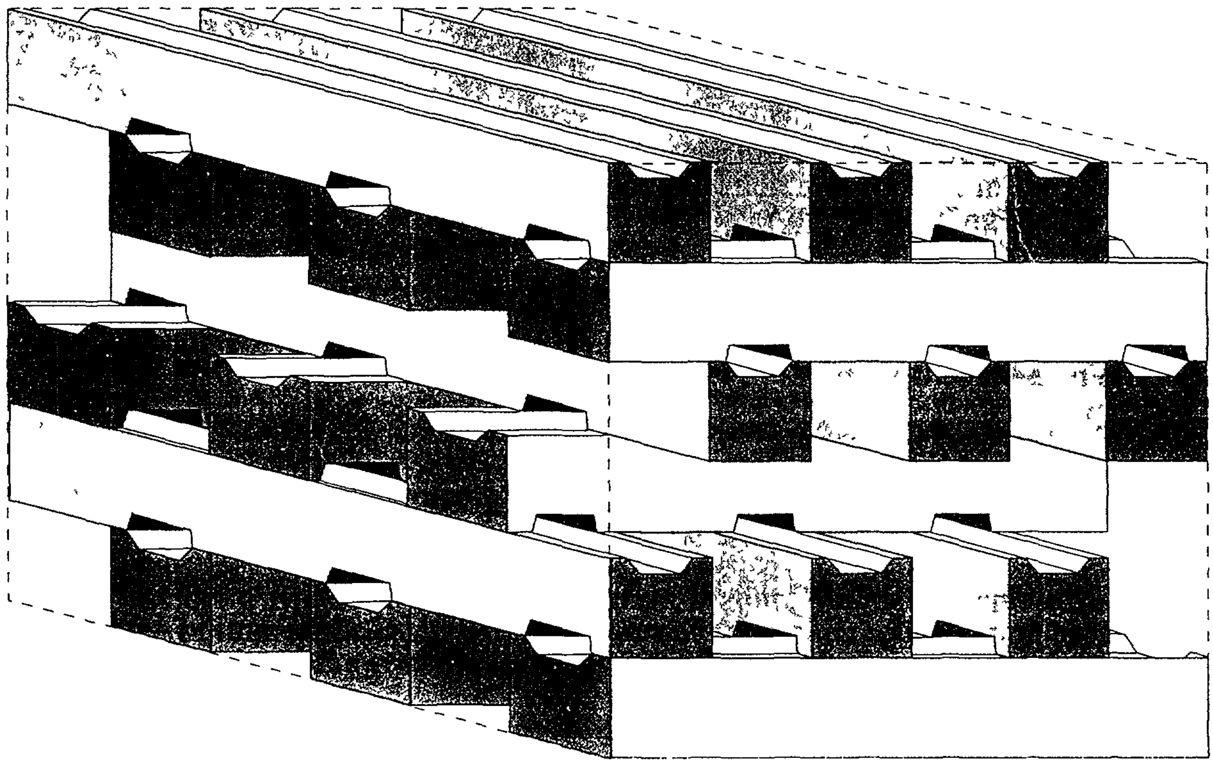


FIG. 14. Axonometric view of the stack of graphite beams to be located inside the reactor cavity.

Some containment designs have additional containment systems/buildings surrounding the primary containment in which a slight subatmospheric pressure is maintained, in order to collect any fission products leaking from the primary containment. Controlled release via filters and elevated stacks is then possible.

An important aspect in this context is foreseen for the SPWR [18], which is a design currently developed by Westinghouse and Mitsubishi on the basis of AP-600 (with participation of EdF) design but with an electrical output of about 900–1200 MW.

The SPWR containment is cooled as in AP-600 by a PCCS during the post-accident period. Therefore no secondary containment is possible in the upper part of the containment.

However, in the penetration area, where in Japan 97% of the total leaks are assumed to occur, an emergency passive air filtration system (EPAFS) has been proposed and tested. This system will exhaust the air in this area by ejectors supplied by air bottles. The air will be filtered before being mixed with the cooling air of the PCCS. A specially designed turning diffuser on the top of the shield building would provide direction and credit for elevated releases of the radioactivity (see Fig. 15).

5.5. SYSTEMS AND COMPONENTS TO IMPROVE LEAKTIGHTNESS

A primary requirement for all proposed containment is obviously focussed on primary containment leaktightness in all considered design conditions. In recent years, two different trends in this respect may be noted. The first relates to a reduction of the assumed leak rate used for current reactors (in the range of 0.25%–0.5% per day for most PWR containments and 0.5%–15% per day for BWR containments) believing that this is achievable with current technology based on operating

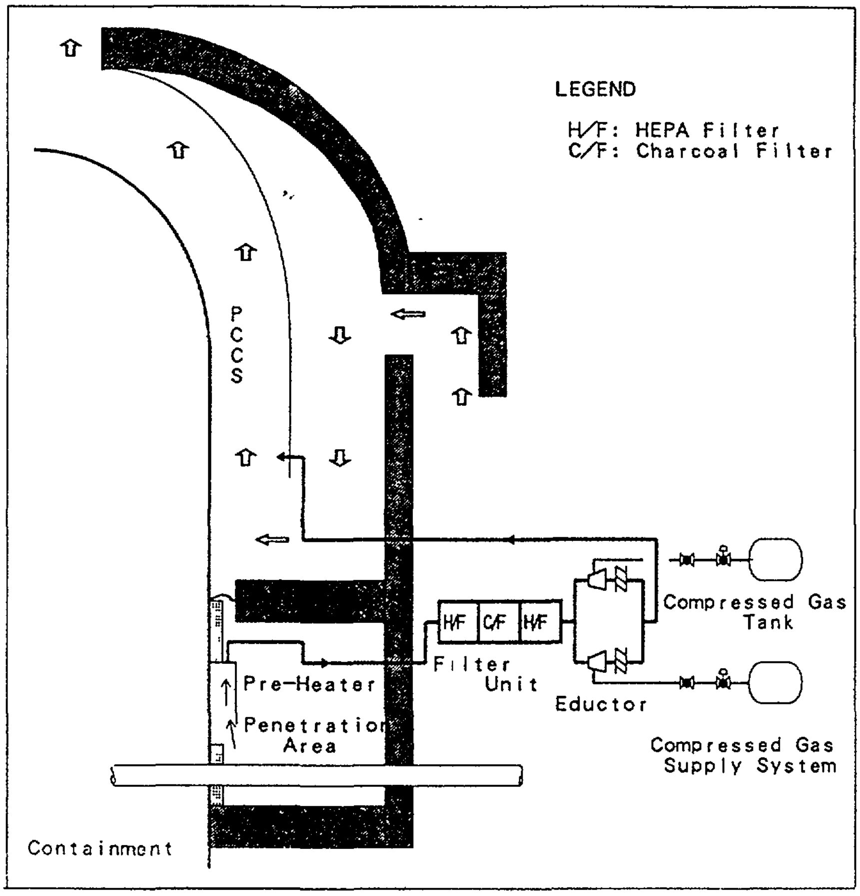


FIG. 15. Concept of emergency passive annulus filter system [18].

experience (e.g. AP-600 proposes 0.12% per day). The second refers to some cases when the requirements have been relaxed on the basis that part of the required source term reduction task has been assigned to the secondary containment.

Designers have drawn from the current experiences in containment integrated leak testing to identify weak points to be improved; mainly, the isolation valves and the large containment penetrations.

Specific attention has been given to the path allowing a direct release path to the environment bypassing any secondary structure. In some cases, pressurizing systems have been proposed in the designs to prevent any release from these paths.

Attention has been given also to design all large penetrations as pressure sealing. Remaining specific issues concern how to assure the required performance in the case of severe accidents, with reference to the harsh environmental conditions in term of temperature, pressure and radiation and the possibility of local effects due to aerosol accumulation or corium splashing. Deformation of the containment shell, when subjected to high pressures combined with high temperature, should be considered as potentially increasing the outflow area. Dislocation of mating sealing surfaces due to differential thermal expansion (such as a metal hatch mating on a concrete wall) have been taken into account in some cases. Some designs proposed pressurized electrical penetrations to eliminate any contributions by this potential leak path.

In addition, containment bypass may occur for interfacing systems LOCA and steam generator tube rupture (SGTR). As containment leaktightness and performance improve, these items progressively become more important. Some designs propose improved integrity at full reactor pressure of all pipes connected to the primary circuit in order to decrease the possibility of an interfacing system LOCA. Another proposal is to incorporate more interfacing systems within the containment. For coping with SGTR, isolation valves on the feed water and steam lines are proposed. Designs are also discussed where secondary side heat removal is performed by a safety condenser, no longer providing an extra barrier to fission product release to the environment.

6. EXAMPLES OF PROPOSED CONTAINMENTS

6.1. OVERALL CONSIDERATIONS

Before presenting some examples of the next generation of advanced containment systems, it is of interest to identify a few considerations related to the overall integrated concepts not yet mentioned in the previous sections.

While in the past both spherical and cylindrical containments were proposed, essentially all advanced containments are cylindrical. This may have been induced by layout consideration, but it is likely that the main cause was the tendency, mentioned below, to adopt reinforced and prestressed concrete primary containments.

In fact, among the examples presented in this section only AP-600 has steel containment, which is a key element to assure a completely passive heat removal by external natural air draft in these designs.

In other cases, the primary containment is either reinforced concrete, such as the SBWR (which is capable anyway of passive heat removal) or prestressed concrete, such as the EPR.

Some designers have indicated a reason for preferring concrete structures, based on more stable and predictable behaviour at loads exceeding the design values and the elastic range.

Finally, it should be noted that, after some interest expressed in the 1970s and 1980s in underground siting, no new proposal has been advanced in this area.

6.2. AP-600

The AP-600 containment (see Fig. 8) includes a primary steel containment shell and a shield concrete building.

The steel containment vessel is an integral part of the containment system and serves both to limit releases in the event of an accident and to provide the safety grade ultimate heat sink. The portion of the vessel above the operating floor is surrounded by the shield building, but is exposed to ambient weather conditions as part of the passive cooling flow path. A flexible watertight and airtight seal is provided between the containment vessel and the shield building. The portion of the vessel below the operating floor is fully enclosed within the shield building, with the bottom head embedded in concrete.

The containment vessel is a freestanding cylindrical steel vessel, 39.62 m in diameter, with ellipsoidal upper and lower heads with a total height of 57.9 m. The wall thickness is 41.27 mm.

The passive containment cooling system (PCCS) provides the safety-grade ultimate heat sink that prevents the containment shell from exceeding its design pressure of 310230 Pa. The PCCS uses natural air/circulation between the steel containment shell and the shield building. In accident situations, air cooling is enhanced by distributing water onto the steel containment shell. The water is gravity fed from a 1325 m^3 annular tank designed into the roof of the shield building. This tank has sufficient water to provide three days of cooling. After three days, additional cooling water would be provided by operator action to maintain low containment pressure and temperature. But even if no additional water were provided at this time, air cooling alone would be sufficient for continued public safety.

6.3. SBWR

The SBWR containment system [19, 23] is of the pressure suppression type, as are all the recent BWR containments proposed by General Electric.

The containment structure (as shown in Fig. 16) is a reinforced concrete cylindrical structure which encloses the reactor pressure vessel (RPV) and its related systems and components. The containment structure has an internal steel liner providing the leaktight containment boundary. The containment is divided into a drywell region and a suppression chamber region with an interconnecting vent system. The functions of these regions are as follows:

The drywell region is a leaktight gas space surrounding the RPV and the reactor coolant pressure boundary which provides containment of radioactive fission products, steam, and water released by a LOCA prior to directing them to the suppression pool.

The suppression chamber region consists of the suppression pool and the gas space above it. The suppression pool is a large body of water to absorb energy by condensing steam from safety relief valve (SRV) discharges and pipe break accidents. The pool is an additional source of reactor water makeup and serves as a reactor heat sink. The gas space above the suppression pool is leaktight and sized to collect and retain the drywell gases following a pipe break in the drywell, without exceeding the containment design pressure.

A safety related passive containment cooling system (PCCS) is incorporated into the design of the containment to remove decay heat from the drywell following a LOCA. The PCCS uses three elevated heat exchangers (condensers) located outside the containment in large pools of water at atmospheric pressure to condense steam that has been released to the drywell following a LOCA. This steam is channeled to each of the condenser tube-side heat transfer surfaces where it condenses and the condensate returns by gravity flow to the gravity driven cooling system (GDCS) pools. Non-condensable gases are purged to the suppression pool via vent lines. No forced circulation equipment is required for operation of the PCCS. There is sufficient inventory in these pools to handle at least 72 hours of decay heat removal without tube uncovery.

6.4. EUROPEAN PRESSURIZED REACTOR (EPR)

For the EPR [20–22] (Figs 17 and 18) a double wall containment is selected to ensure the containment function. The primary containment consists of a prestressed concrete cylinder with elliptical head. To provide the dual-wall containment function, the containment is surrounded by a secondary wall in reinforced concrete. All possible leaks are collected in the intermediate space, the annulus, which is maintained at subpressure and filtered before release via the stack, so that no direct leaks to the environment need to be considered.

The containment is surrounded by the safeguard and fuel building which contain the safety systems. The redundant safety grade systems are strictly segregated within the safeguard bildings into four divisions. This divisional segregation is provided for both the mechanical and electrical part of the systems. Both, reactor and adjacent buildings will be erected on one raft. The fluid systems will be installed in the lower part of the adjacent building, the electrical and I&C equipment including the main control room in the upper part. Water storage for safety injection is provided in the in-containment refuelling water storage tank (IRWST) inside the containment. This water inside the containment also mitigates core melt situations. Adjacent to the reactor pressure vessel a spreading area for molten core is provided which is designed to prevent molten core–concrete interaction. The residual heat removal system is located inside the containment avoiding the risk of leaks outside containment and of containment bypass.

The spent fuel pool is located outside the containment in the fuel building which also houses the associated cooling system and the pumps and borated water storage for the chemical and volume control system.

The active components of these two systems are duplicated with provision of emergency power. They are geographically segregated. An additional intermediate pool is located inside the containment, which is only used during refuelling, and from where the spent fuel is transferred to the pool outside the containment. This arrangement helps to shorten the refuelling outages.

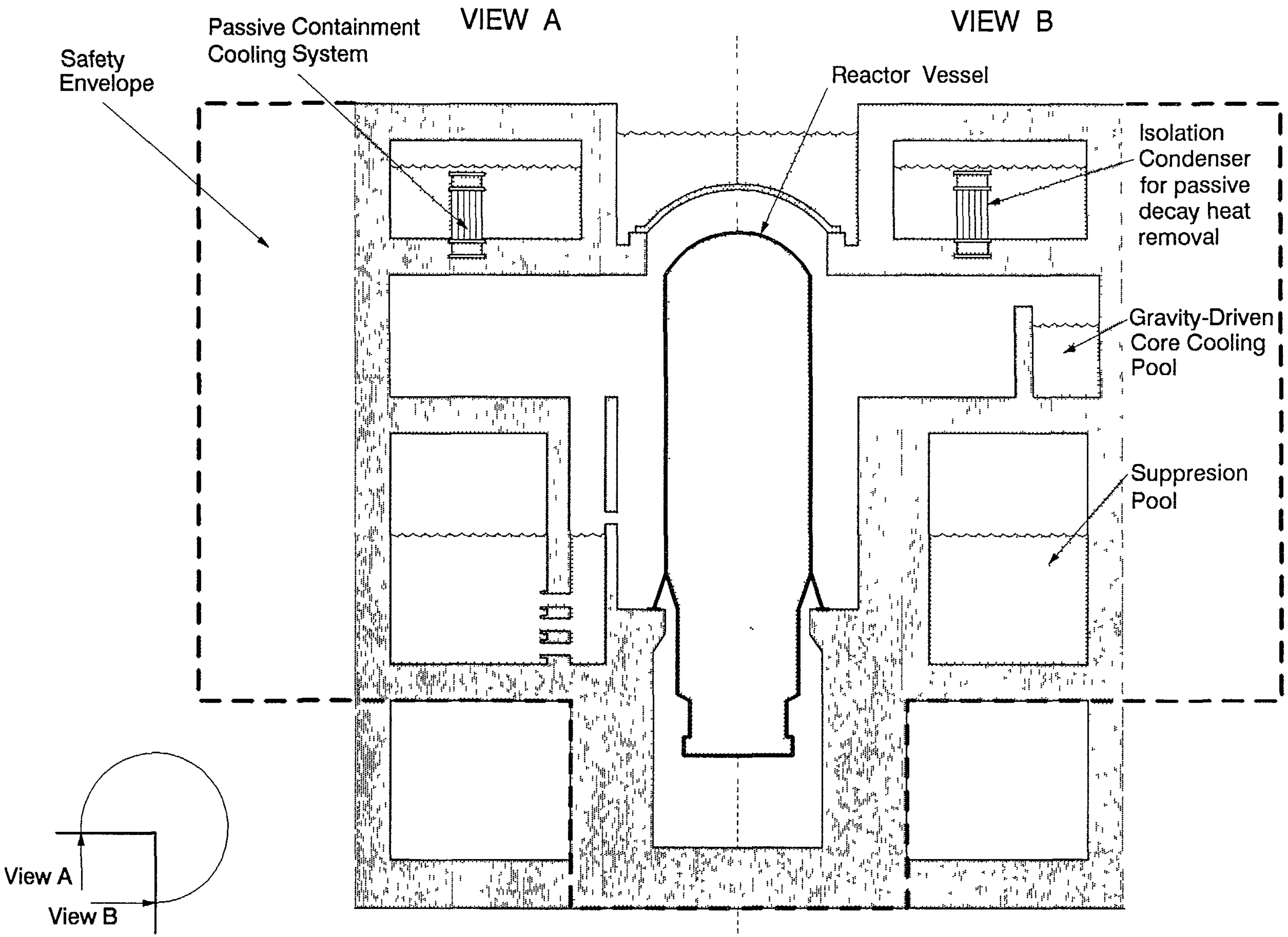


FIG. 16. SBWR containment system [19].

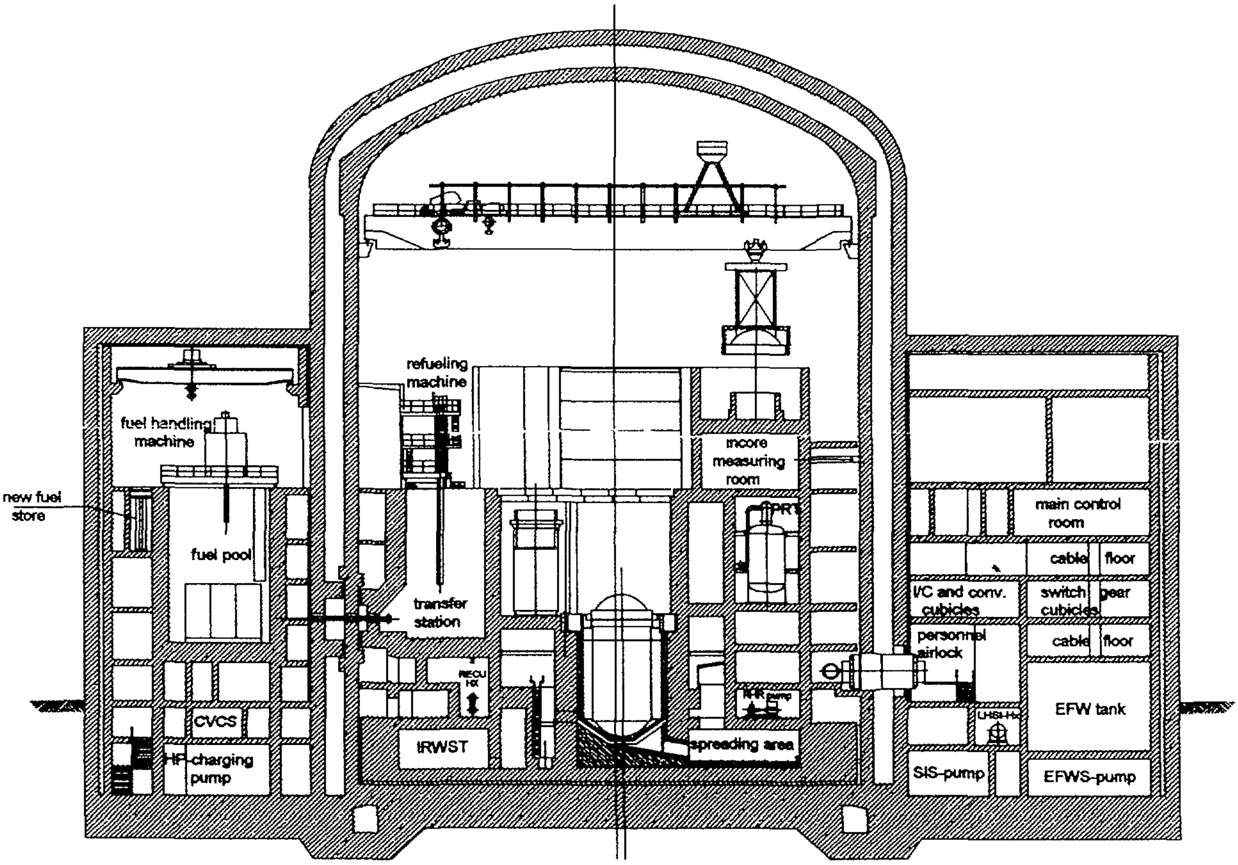


FIG. 17. EPR reactor building arrangement — Containment section AA [20].

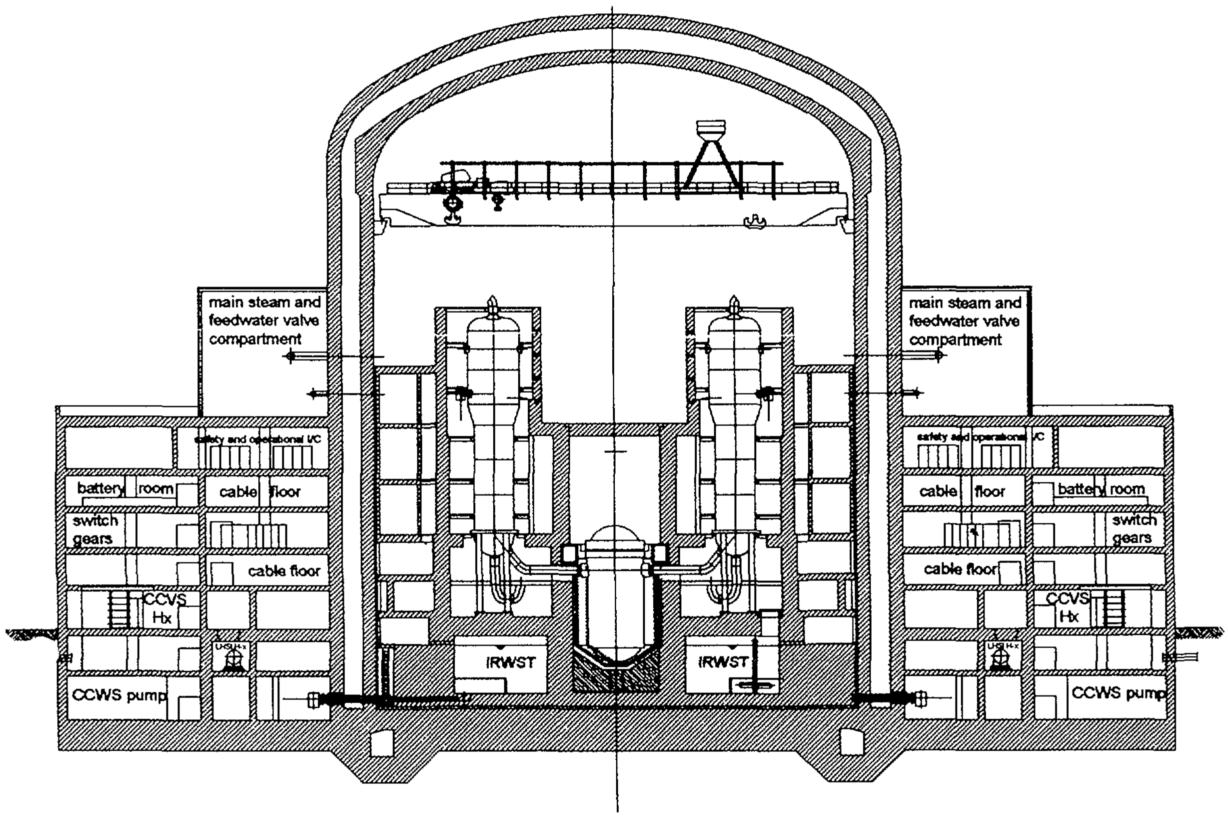


FIG. 18. EPR building arrangement — Containment section BB [21].

The arrangement means that all equipment that is important for safety is protected. The plant is designed to withstand external hazards including aircraft crash.

The nuclear auxiliary building adjoins the safeguard buildings and houses e.g part of the chemical and volume control system, boron and water make up and storage systems. The centerline of the turbine building runs through the center of the reactor building to reduce the possibility to damage of reactor building due to turbine missiles. Both emergency power generator buildings, each equipped with two emergency diesel generators, are located in separate buildings, on either side of the reactor building

6.5. PIUS

The PIUS containment concept (see Fig. 19) proposed by ABB [19] is conditioned by the extremely low probability of a core melt accident, given the innovative characteristic of its primary circuit

The containment is essentially a small volume pressure suppression pool containment preliminary selected as the optimum solution taken the large water type masses of the PIUS primary loop and the surrounding reactor pool.

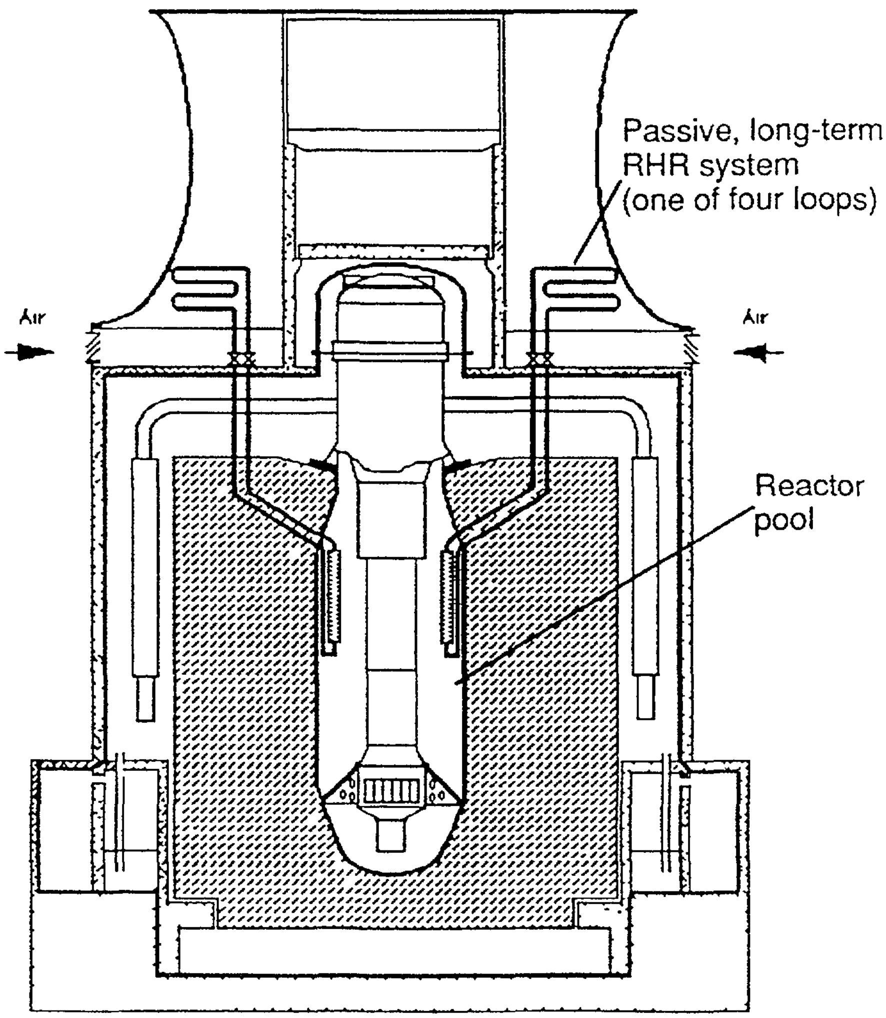


FIG. 19 PIUS safety-grade structures [19]

The distinctive elements for the design are:

- The largest conceivable LOCA, i.e. a double-ended cold leg break, combined with a loss of an AC power, is the design reference accident.
- Following the initial blowdown of the primary system, the decay heat removal is ensured by a fully passive system, removing the heat from the core to the reactor pool in the concrete vessel cavity, in natural circulation, and from the pool, via submerged coolers and natural circulation cooling loops, the heat is dissipated in dry, natural draft, air cooling towers on top of the containment.
- The initial pressure peak is, in the same way as in other pressure suppression containments, determined by the heat capacity of the suppression pool and the size of the compression chamber.
- Following the initial blowdown, the steam (and hot water) outflow to the containment stops, and the pressure in the containment decreases due to condensation on containment walls and structures.
- The passive residual heat removal will keep the reactor pool water from boiling, and the iodine spike, occurring after shutdown, will remain in the water; the only fission products and radioactive material that are released to the containment, are those that were present, prior to the accident.
- The core integrity is protected by passive, self-protective means, and no core damage sequence has been identified; therefore, a mechanistic source term will be applied in safety analyses.
- No containment spray is considered necessary.

7. ISSUES RELATED TO RULES AND REGULATIONS, COMPUTER CODES AND RESEARCH

Despite enormous progress in the solution of problems associated with the design of ALWRs and their containment systems there exist many issues that must be resolved in full depth in order to meet the developing licensing requirements of regulatory authorities. Examples of such issues are:

- harmonization between the requirements set by utilities and vendors and those of the licensing authorities,
- better understanding of phenomena associated with severe accidents,
- the confirmation of design margins for specific solutions,
- improvement of analytical and experimental tools for verification of proposed engineering solutions.

7.1. HARMONIZATION OF REQUIREMENTS SET BY UTILITIES AND VENDORS WITH LICENSING REQUIREMENTS

At present, current reactor designs do not explicitly cover severe accidents. However, in the case of ALWRs direct consideration of severe accident prevention, and mitigation of their consequences, is required in the design by the licensing process. Utilities in cooperation with vendors have set requirements addressing most important issues associated with severe accidents (EPRI Requirements in the USA, EUR — European Utilities Requirements in Europe). Discussions between utilities and appropriate regulatory authorities are proceeding at present with the aim of harmonizing their positions in such areas as:

- source term,
- fire protection,
- intersystem loss of coolant accident issues,
- hydrogen control,
- core debris coolability,
- high-pressure core ejection prevention and mitigation measures,
- early depressurization of primary circuit,
- leak tightness of the containment under severe accident conditions,
- equipment survivability,
- exclusion of potential containment bypasses,
- containment leak rate testing,
- post-accident monitoring and sampling,
- long-term residual heat removal.

Changes in the design bases for ALWRs will generally involve the need for revising both currently established licensing criteria and industrial design standards. The Fifth Workshop on Containment Integrity addressed this issue with two presentations [7, 25]. Whereas in Ref. [7] it is proposed to incorporate new criteria with respect to severe accidents in the General Design Code, Ref. [25] proposes formulation of a code modification for direct consideration of severe accident loads in containment structural design codes. Suggestions have been made for changes of ASME Section III, Division 2, Subsection CC, and ASME Section III, Division 1, Subsection NE and ACI 349.

Along with harmonization of licensing requirements at a local level it is also desirable to reach greater consensus within the international nuclear community on these issues.

7.2. AVAILABLE COMPUTER CODES

Despite extensive R&D activities in many countries (e.g. USA, France, Germany, Italy, United Kingdom, Switzerland, Spain and Japan) it is highly desirable to reach better understanding of such processes as hydrogen generation and combustion, core debris coolability and in-vessel and ex-vessel steam explosions. Moreover, verification of the new safety provisions is necessary in order to demonstrate that their design requirements are met. Problems associated with corium retention systems (mainly material problems) and leaktightness of containment penetrations (personnel airlocks, equipment hatches) under severe accident conditions can be mentioned as examples. From this it follows that efficient analytical and experimental tools must be available. The development of computer codes must continue in order to treat all the relevant physical phenomena in a realistic manner. In particular the development of best estimate codes is of great importance.

7.2.1. Thermal-hydraulic and fission product transport codes

A variety of codes in this area have been developed in order to predict thermal-hydraulic behaviour and fission product transport within the containment. The codes can be used for the analysis of design basis accidents as well as for beyond design basis accidents. These codes can be divided into two main groups:

- integral codes,
- detailed mechanistic codes.

The integral codes are designed to be fast running and to cover the complete course of an accident. They tend to be plant specific and are difficult to apply to new systems or experiments. The detailed mechanistic codes tend to apply to a specific area of severe accident scenarios and model the relevant phenomena in detail. They can be applied to new systems and experiments. They also take longer to run in comparison with integral codes. The integral codes are used to make many calculations of different scenarios which would be prohibitively expensive for the detailed mechanistic analyses. The range of applicability of a given code depends on what it was designed for and what it is validated against.

Many integral codes are based on specific models valid for specific plants. Therefore, modifications are required for advanced containment designs. For example, modifications were needed to the MAAP 3.0 PWR code to enable its use in evaluation of AP-600 safety problems.

Condensation pools are a common feature in many designs. The system codes do not treat condenser pools adequately. Modelling of them in the system codes could be improved in relation to thermal mixing, condensation including direct contact condensation in the presence of non-condensables and 3D effects. System codes need to be benchmarked against validated CFD codes.

Primary circuit/containment coupling (thermohydraulic coupling) is important. Integrated system codes should be improved or new ones developed. For example, RELAP5/CONTAIN is now under development by the US NRC.

A short list of codes that can model specific phenomena associated with accident conditions in containments is given below.

1. The oxidation of large amount of Zircaloy in the core: SCDAP/RELAP5, ATHLET-SA, CATHARE/ICARE, MAAP, MELCOR, STCP.
2. Boron-carbide reactions: SCAP/RELAP5, KESS, MAAP, MELCOR, STCP.
3. Inerted Containment atmosphere: MAAP, MELCOR, THALES, STCP, FUMO, JERICO, WAVCO, CONTAIN.
4. Aqueous fission products: MAAP, MELCOR, THALES, STCP.
   The modelling of fission products and aerosols in the presence of liquid water is very difficult. It is believed that above listed codes can carry out only simplified calculations.
5. Steam explosions: A steam explosion evaluation presents still some uncertainties and appropriate deterministic and probabilistic codes should be improved.
6. Oxygen ingress to inerted containment: MAAP, MELCOR, THALES, STCP, CONTAIN.
7. Effects of hydrogen on natural circulation cooling: MAAP, MELCOR.

8. Heat exchange from and to the containment shell: AP-600-MAAP, CONTAIN.
9. Direct containment heating: CONTAIN, MAAP.

More detailed information on the above listed codes is presented in Appendix B. It is, however, necessary to note that this list serves only as illustration of codes applied mainly in the USA and in Europe. Therefore, it does not represent an exhaustive list of all codes used in this area on the broader international level.

7.2.2. Codes for structural assessment

The rapid progress in computer capabilities has enabled improvement to be carried out on classical codes based both on mechanical force equilibrium and on the more complex finite element approach. The aims of such improved codes are:

- the evaluation of stresses, strains and displacements of containment components in defined accident situations (the quality of the results obtained depends to a large extent on the adequacy of input data),
- the determination of the ultimate failure mode of the whole containment under increasing pressure and/or temperature loadings using inelastic analyses methods.

It is generally agreed that methods and input data with respect to assumed loads are highly conservative in the evaluation of the performance of present day containments. As stated in [27] the application of 'best estimate' approaches is desirable in analyses of future advanced containment. On the regulatory level the applicability of these approaches in the licensing process should be clarified.

Up to now only a few computer codes are able to assess the behaviour of containment structures under dynamic loads acting in some severe accident scenarios, namely these dynamic loads arising from hydrogen detonation. However, this fact is not caused by deficiencies in the applied methodology. But the main difficulty is in the determination of appropriate boundary conditions.

Another important topic is the thermo-mechanical assessment of corium retention devices used in some projects and investigated in various laboratories (e.g. France–CEA TRIREM, Italy–ENEA activities) [30, 31, 33].

7.3. EXPERIMENTAL RESEARCH

Experimental efforts in the area of ALWRs can be divided into three groups:

- behaviour of containment components and structures under severe accidents conditions,
- phenomenological research,
- system and components testing.

7.3.1. Behaviour of containment components and structures under severe accident conditions

This group comprises:

- Experiments for establishing input data to be used later in computer code analyses (e.g. material behaviour laws, determination of loads arising as a result of severe accident conditions),

- Mock-ups loaded to ultimate loading conditions (e.g. Sandia 1/6th reinforced concrete, Sandia 1/8th metallic, Sizewell B 1/10th prestressed concrete [32]). The goal of these experiments is not only determine the ultimate failure loads but also to evaluate the effect of such factors as:

  · effect of reinforcement and prestressing,
  · effect of random defects,

· effect of the size and location of containment penetrations.

In this field some further activities are under way [29]:

- prestressed concrete mock-ups (MITI-NRC),
- steel mock-ups (MITI).

It is necessary to note, that the main issue that should be solved to obtain representative results is the scaling problem.

7.3.2. Phenomenological research

In order to gain more detailed information on phenomena arising in severe accident conditions and thus to determine reasonable design margins for applied engineering solutions further R&D efforts are necessary. An overview of existing experimental results along with the plan for further activities in the USA can be found in [35]. In the area of containment behaviour under severe accident conditions the following main issues are investigated:

- failure of concrete structures due to thermal stresses, decomposition, etc.,
- leak-rate assessment in the course of pressure increase,
- investigation of refractory materials with respect to protection of concrete against molten core attack,
- source term assessment.

The US NRC experimental and analytical research has included the testing of large models of containment structures as well as large and full scale testing of penetration components. Results are being used to predict failure thresholds and failure modes along with associated leak-rates.

In the area of severe accident phenomena and containment challenges the US NRC has established research agreements with 16 countries in the framework of the Cooperative Severe Accident Research Programme. Several further research programmes are proceeding in the world, such as ACE/MACE, FARO (JRC-ISPRA), etc.

As for experimental research concerning source term evaluation the following can be mentioned:

- the Sophie Programme (France): Deposition and resuspension of fission products and aerosols like CsI, CsOH, I_2, Te;
- Piteas and Tuba (France): Analyses of aerosol behaviour (soluble, insoluble) in the containment filled by steam with time-varying saturation values;
- Vanam tests (Germany): Performed in the Battelle Model Containment (BMC) in which the effects of multicomportment geometry are taken into account along with stratification and natural circulation;
- Falcon Test Facility (United Kingdom): This facility which is located in the Winfrith Technology Centre is suitable for performing small scale physical-chemical experiments with the goal of studying fission product and aerosol behaviour taking into account the complexity of the path to the containment;
- other international research programmes as LACE, ACE and PHEBUS.

It is also appropriate to mention two further programmes:

- The PSI ALPHA programme (Switzerland), which includes the following major items:

  · the large-scale, integral system behaviour test facility PANDA, which is used to examine multi-dimensional effects in the SBWR heat removal system,

· an investigation of the thermal-hydraulics of natural convection and mixing in pools and large volumes (LINX),
· separate effects study of aerosol transport and deposition in plena and in tubes (AIDA).

The results will be valuable for providing code validation data.

- The RASPLAV project, recently launched by OECD, which plans to study in the Russian Kurchatov Institute Laboratories several phenomena related to the possibility of cooling the corium inside the pressure vessel by externally flooding it.

7.3.3. System and component testing

For most advanced reactor concepts it is necessary to demonstrate that engineering solutions can meet all their functional requirements. As an example, the AP-600 passive safety systems test programme is briefly described.

The AP-600 passive safety systems test programme:

A major portion of this test programme [34] is devoted to the study of the passive containment cooling system (PCCS). The aim of the tests that have been performed, or are in progress, is to demonstrate the heat removal capability of the containment design and to get more detailed information about:

- water film behaviour and wetting of steel containment outer surface,
- heat transfer by evaporation of water from the steel containment surface and heat transfer by air cooling,
- hydraulic pressure drop in the flow paths of the air for external containment cooling,
- condensation heat transfer on inclined cooled surfaces inside the containment in the presence of non-condensable gases.

Moreover, the following additional tests are currently being performed:

- PCCS air flow path resistance test (a 1/16-scale replica of a 14 degree section of the PCCS air flow path was constructed with the aim of determining the flow path resistance, to identify the possibilities for its aerodynamic improvements and to demonstrate the effectiveness of such improvements),

- PCCS water distribution tests (a large-scale demonstration of the effectiveness of water distribution capability to distribute water on the steel containment shell surface on the top and sidewall for the full range of expected water flow rates),

- PCCS wind tunnel tests (a detailed scale model for simulation of conditions existing in the inlet and exhaust portions of the outer concrete containment structure),

- PCCS small-scale integrated test (an experimental apparatus for simulation of heat transmission processes on both inside and outside steel containment surfaces with the aim of determining the effects of air flow velocity, water flow rate, air humidity and air temperature in the annulus).

- PCCS large-scale integrated test (the main goal of this test was simulation of the involved phenomena, in a geometry with a realistic height to diameter ratio, such as non-condensable gas mixing, steam condensation and flow pattern inside the containment).

Finally, it is appropriate to note that the above examples of experimental research demonstrate the intensity of activity in this area. To give an exhaustive review of all relevant international experimental research is outside of the scope of this document.

8. PHENOMENOLOGICAL RESEARCH AND SYSTEM/COMPONENTS NEEDS

8.1. PHENOMENOLOGICAL RESEARCH

There is a wide and comprehensive literature available on the topic of severe accident phenomena (Section 4.2). In view of the different approaches to address the phenomena in design and in view of still lacking "harmonize guidance on the associated methodology" it is clear that further R&D efforts are necessary to confirm design margins for specific solutions eventually to be selected. An overview of insights gained by experiments and definition of further research activity is outlined in Ref. [35] for the USA.

The objectives are in general to identify and to obtain a better understanding of the problems related to the integrity of the containment system, modelling, material characterization and failure modes in order to evaluate the safety margins of the structures. In particular the response of the containment under severe accident conditions has to be researched:

- Assessment of structural failure of concrete due to thermal stress, decomposition, etc.
- Leak rate assessment, during pressure increase,
- Investigations on refractory materials to protect concrete against molten core attack,
- Source term assessment.

The NRC's experimental and analytical research has included the testing of large models of containment structures as well as large and full-scale testing of penetration assemblies. Their research results are being used to predict failure thresholds and modes, and their related leak rates. In the area of severe accident phenomena and containment challenges, the NRC and 16 countries have research agreements under the Cooperative Severe Accident Research Program.

Several research programmes are under way in the world, such as ACI/MACE, FARO (IRC-ISPRA), etc.

Some of the experimental facilities concerning source term evaluation are:

- the Sophie Programme (France), regarding the deposition and resuspension of fission products and aerosols like CsI, CsOH, I2, Te;
- Piteas and Tuba (France), to analyze aerosol behaviour (soluble, insoluble) in the containment with steam characterized by time-barying saturation rates;
- the Vanam tests (Germany), performed in the Battelle Model Containment (BMC) in which the effects of a multi-compartment geometry are taken into account and stratification and natural circulation are also considered;
- the Falcon test facility (United Kingdom), located in Winfrith Technology Center, useful to perform small scale physical-chemical experiments in order to study the fission product and aerosol behaviour taking account of complex paths up to containment.
- Other international research programmes include LACE, ACE and PHEBUS experiments.

8.2. SYSTEM/COMPONENTS TESTS NEEDS

For the advanced reactor concepts it is necessary to demonstrate by adequate testing of systems and components that design features fulfill in reality their dedicated function. In the following, for AP-600 reactor, the specific test programme with respect to containment system function is briefly reviewed.

AP-600 passive safety systems test

A major portion of the AP-600 tests [34] are devoted to the study of the passive containment cooling system (PCCS). Tests have been performed or are in progress, to characterize the heat removal capabilities of the AP-600 containment design. Several tests have been performed during the AP-600 conceptual design progress to provide the data base for models. These include the following:

- Study of water filter behaviour and wetting of a steel plate simulating the containment exterior surface.

- Heated plate tests to examine the evaporating heat transfer of water from the steel surface of the containment and heat transfer with only air cooling.

- Containment external cooling air flow path pressure drop tests to characterize the hydraulic losses.

- Steam condensation heat transfer experiments on a flat cool surface at different angles of inclination to simulate the condensation on the inside of the containment in the presence of non-condensable gases.

  Additional tests which are currently being performed, or have just finished, are as follows:

- PCCS air flow path resistance test
  A 1/16-scale replica of a 14 degree section of the entire PCCS air flow path was constructed to qualify the cut flow path resistance to determine if aerodynamic improvement were needed, and demonstrate the effectiveness of these improvements.

- PCCS water distribution tests
  The PCCS water distribution test provided a large-scale demonstration of the capability to distribute water on the steel shell containment chrome outer surface and top of the sidewall for the entire range of expected water flow rates. The tests also confirmed the detailed design of the water delivery and distribution system.

- PCCS wind tunnel tests
  Detailed scale model tests were used in the test to simulate the structural details of the shield building air inlet and exhaust and surrounding buildings.

- Small scale integrated PCCS tests
  The experimental apparatus was built to simulate the entire PCCS heat transfer processes occurring both on the inside and outside containment surfaces. Special items of interest were:

  · overall condensation heat transfer rates inside and outside the containment surface,
  . effect of air flow velocity, water filter flow rate, air humidity and air temperature in the annuals.

- Large scale integrated PCCS test
  The large scale test used to simulate the steel containment shell with a height-to-diameter ratio more typical of the actual containment shell than was available for small scale tests. The larger vessel made it possible to study in-vessel phenomena such as non-condensable mixing, steam release jetting and condensation as well as flow pattern inside of containment.

9. CONCLUSIONS

A first conclusion is that containment designs of the next generation reactors are essentially not drastically different from the existing ones. Only a few institutions outside the large engineering organizations have proposed very innovative solutions, whose feasibility, advantages and economy have still to be proved.

Industry is aiming at profiting from the huge accumulated operating experience and is, therefore, proposing technical evolutions of existing designs. However, in specific areas, such as protection against severe accidents, important improvements have been introduced in order to keep the containment intact and also operational for all realistically conceivable severe events.

Important improvements are specifically addressed at preventing early containment failure to eliminate the need for prompt public evacuation on the one hand, and on the other hand, to leave ample time for operators to initiate additional mitigation procedures and to reach a satisfactory long-term status.

Other interesting proposals, which have the potential for further improving safety, are those involving the use of passive systems to remove decay heat from the core (both intact or melted) and to transfer it to the environment.

Since safety objectives are becoming more and more demanding, other sequences, which in the past were not dominating the assessment of public risk may come back into play. This is the case for containment bypasses, even without extensive core damage.

At the same time a more realistic approach to the evaluation of the former design basis events is needed in order to avoid possible inconsistencies such as larger consequences, calculated with very conservative assumptions and methodology, in the case of events of much higher probability involving less or no core damage.

In general, however, proposed containment designs appear to provide a substantially higher degree of protection against a larger set of events, including those which have emerged in the past PRAs as those dominating the public risk profile. Therefore they give support to the defense in depth approach and the challenging goal of a step forward in safety, which has been set for next generation reactors.

Appendix A

CLASSIFICATION OF CURRENT CONTAINMENT SYSTEM DESIGN CONCEPTS

In the following a brief description of the most common design solution for the existing containment systems is presented (derived from the IAEA Safety Guide [1]). Conceptually advanced containments can be classified in the same categories; however, some specific features are different as discussed in the main body of the report.

Single dry containment (typical application PWRs)

In this concept (see Fig. A-1), the primary containment envelope is a steel shell or concrete building (cylindrical or spherical) with a steel liner which surrounds the nuclear steam supply system. Energy management in the building can be accomplished by an air cooler system, or by a water spray system. In addition, the free volume of the containment and the structural heat sinks (the containment envelope and the structures within it) are used to limit peak pressures and temperatures for postulated pipe rupture accident conditions. The initial supply of water for the spray system and for the

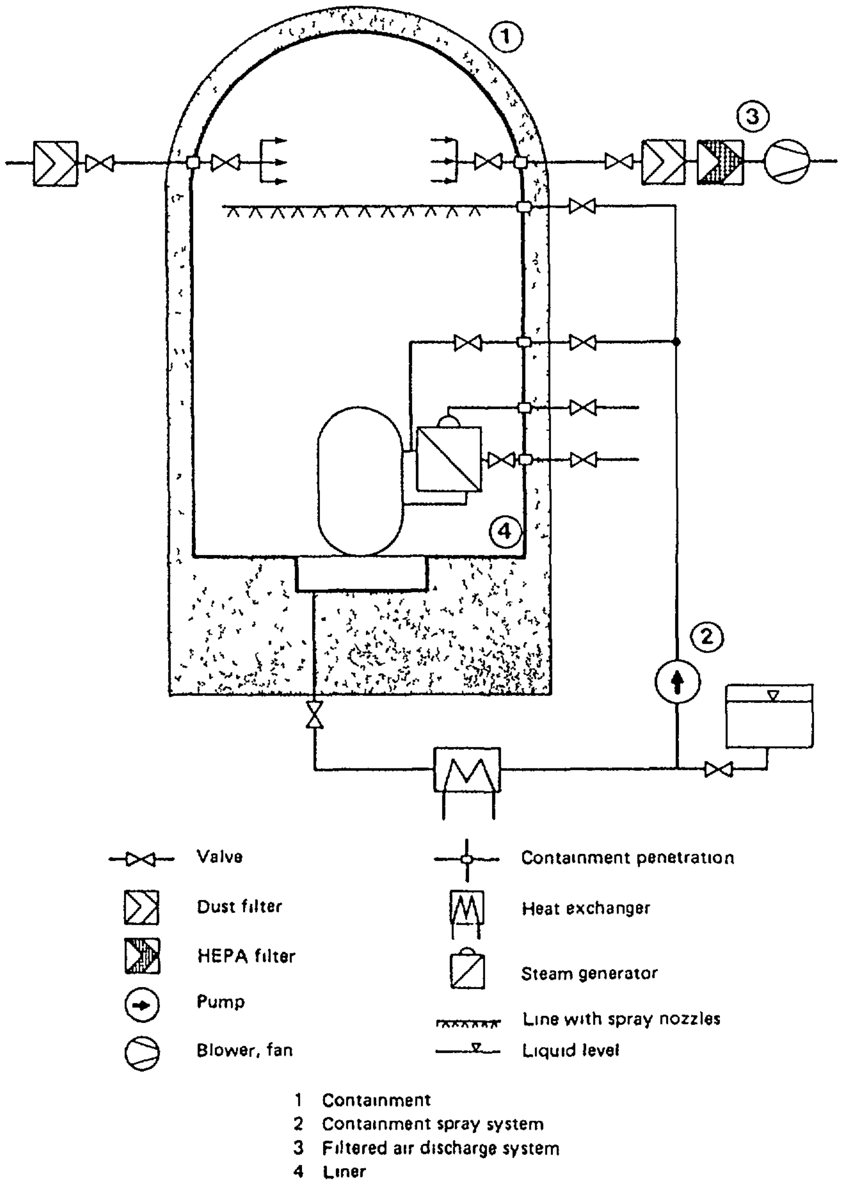


FIG. A-1. Single dry containment [1].

emergency core cooling system (ECCS) is in a large tank. When this water is used up, suction for both the spray system and the ECCS is switched to the containment building sump. Water that is recirculated to the reactor vessel is sometimes cooled by heat exchangers. In most designs the recirculation water for the spray headers — which is also used to limit the containment atmosphere contamination — is cooled by heat exchangers. When pipes rupture in systems other than the reactor coolant system, only the spray system, in the recirculation mode, (is operated).

Dual dry containment (typical application PWRs)

A typical full-pressure dual containment (see Fig. A-2) consists of:

- a steel or concrete shell, basically cylindrical or spherical in shape (containment);
- a concrete shell surrounding this containment (secondary confinement);
- an air extraction system for the annulus (the space between the containment and the secondary confinement).

The containment embraces all components of the reactor coolant system under primary pressure. It constitutes the pressure resistant and gastight containment. It is designed as a full-pressure containment, i.e. it is able to withstand the increases in pressure and temperature that would occur in the event of any design basis accident, especially a loss-of-coolant accident. Full-pressure containments do not utilize any kind of special pressure suppression system. In the medium and long term, energy removal and pressure reduction is accomplished by using an ECCS recirculation heat exchanger. Concrete containments may be reinforced or prestressed and, in the second case they may be without a metallic liner.

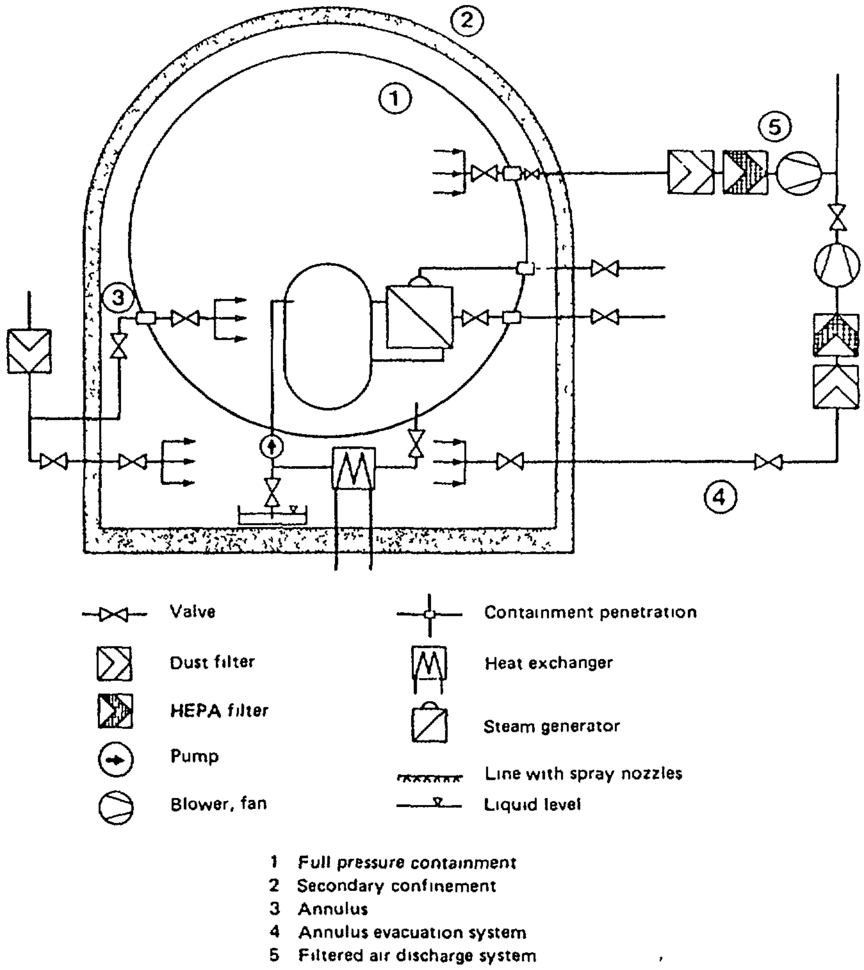


FIG. A-2. Dual dry containment [1].

The secondary confinement is a concrete shell which surrounds the containment. It has three functions:

- to provide radiation shielding for the environment and plant personnel in combination with the containment, during normal operation and under accident conditions,
- to protect the systems and components inside the secondary confinement against external events,
- to capture leakage from the containment in the annulus between the two shells.

Leakage from the containment into the annulus may be extracted and filtered during accident conditions by an air removal system, and then released through the plant stack in a controlled manner.

Ice condenser containment (typical application PWRs)

The PWR ice condenser containment (Fig. A-3) utilizes a pressure suppression system concept in which the high-pressure steam/air mixture resulting from pipe rupture accident conditions is

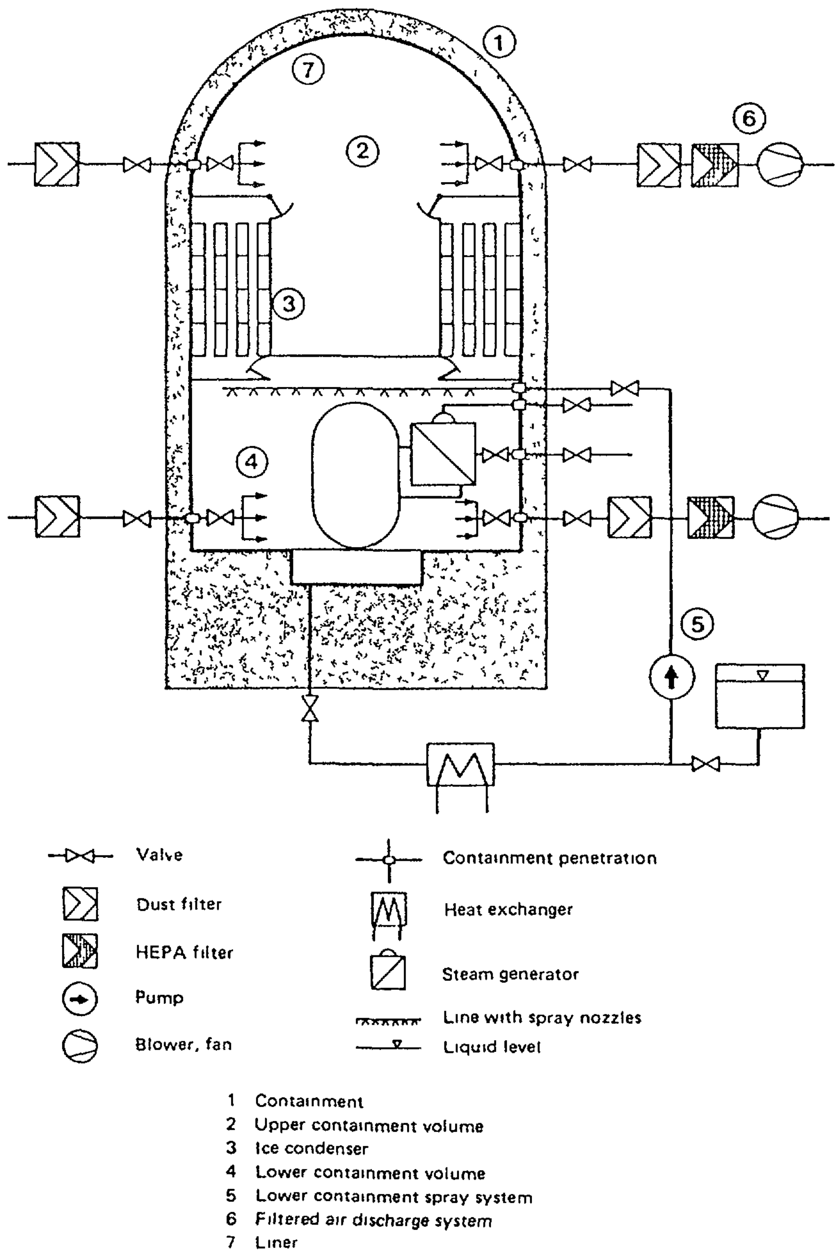


FIG. A-3. Ice condenser containment [1].

directed through vent doors into chambers containing baskets with ice. The steam is condensed on the surface of the ice in the baskets.

The proposed containment is a cylindrical structure divided into three isolated compartments — the lower area which contains all major reactor coolant system components, the ice condenser chambers and the main upper containment volume. Non-condensable gases (including noble gas fission products), which are forced into the ice condenser chambers, are vented through doors into the main upper containment volume.

An active spray system is utilized in the lower containment volume to reduce pressures and temperatures and to remove airborne radioiodine from the containment volume. The initial source of water for this system is a water-storage tank. After exhaustion of this supply, a recirculation mode is initiated wherein water is pumped from the building sump through a heat exchanger and then returned to the spray headers.

PWR bubbling condenser containment (typical application, early WWER)

The PWR bubbling condenser containment system (Fig. A-4) utilizes a suppression pool concept in which the high-pressure steam/air mixture resulting from loss-of-coolant accident conditions is directed into pools of water through submerged tubes. The steam is condensed in the bubbling condenser pools.

The proposed containment is a cylindrical concrete structure divided into three isolated volumes — the lower one, which contains all major primary reactor coolant system components, the bubbling condensers (suppression pools), and the main upper containment volume. Non-condensable gases (including noble gas fission products) which are driven into the bubbling condenser chambers are vented through openings into the main upper containment volume. Radioiodine and other soluble or particulate fission products are trapped in the bubbling condenser water pools.

Open tanks located in the upper containment volume are connected through U-tubes to water spray nozzles in the lower containment volume. During fast pressure transients in the containment system the passive sprinkler system is activated by the pressure differences between the water inlet of the U-tubes submerged in the tanks and the nozzle outlet. An active spray system, with an independent stored water supply, is used to provide both energy and radionuclide management functions. When the water supply in the spray tanks is exhausted, a recirculation mode is initiated and water from the building sump is pumped through a heat exchanger and sprayed into the lower containment volume. After a few minutes, the pressure in the lower volume falls below atmospheric pressure, and an inverse pressure differential is created between the upper and lower volume. Air is prevented from returning from the upper to the lower volume by hydroseals formed in the bubble tubes. Once a negative pressure has formed in the lower volume the leakage of radionuclides from it will cease.

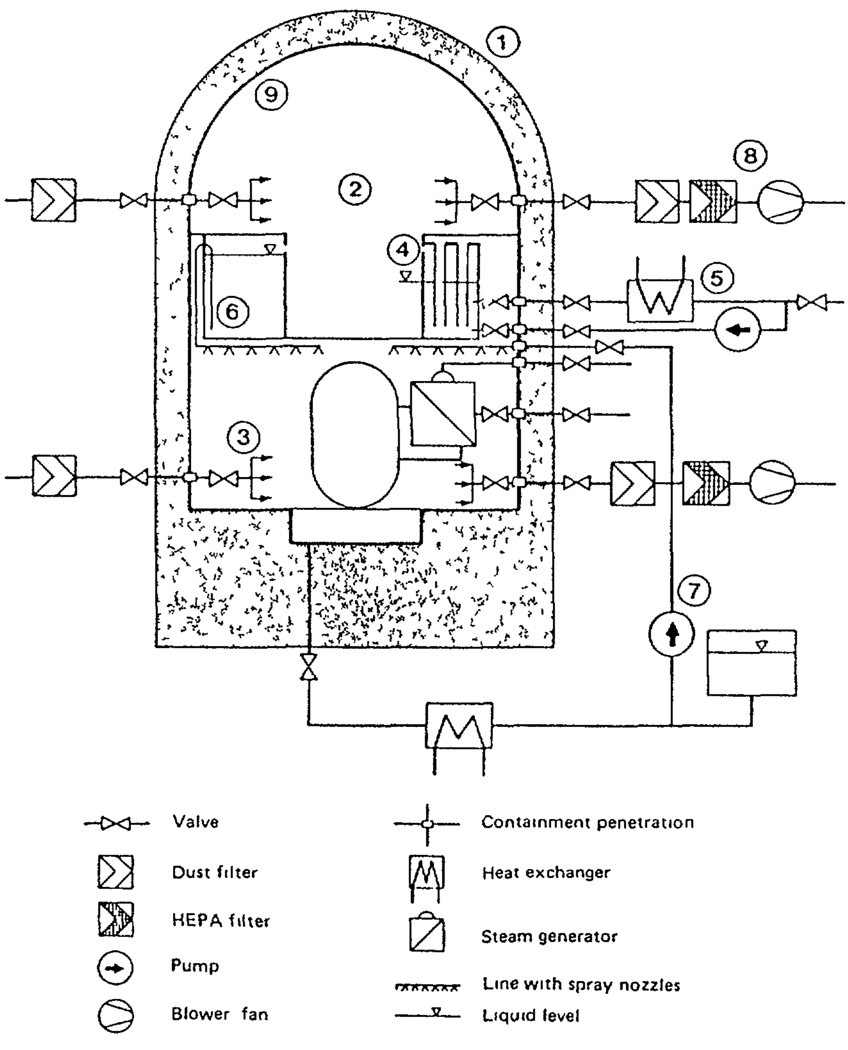
BWR pressure suppression containment

A BWR containment with a pressure suppression system (Fig. A-5) is a structure divided into two main compartments: a dry well, housing the reactor coolant system and a wet well, partly filled with water whose function is to condense the steam in case of a loss-of-coolant accident. The remainder of the wet well serves as a compression chamber for non-condesables. The two compartments are connected by pipes that are submerged in the water of the wet well. A spray system is often installed in the dry well and in the wet well. The reactor building surrounding the containment forms a secondary confinement which captures leakage from the containment. The containment envelope can consist of a concrete structure with a steel liner for leaktightness or a steel shell.

The purpose of the pressure suppression system is to reduce pressure when a pipe in the reactor coolant systems ruptures. The steam from a leak in these pipes enters the dry well and is passed through pipes into the water of the wet well, where it is condensed, and the pressure in the dry well is reduced. The pressure suppression system helps in reducing the airborne radioiodine by wash-out into the condensation pool.

The wet well is also used as a heat sink for the automatic pressure relief system. This serves to limit the pressure rise in the reactor coolant system when the reactor cannot discharge steam to the turbine condenser system. The steam still produced by residual heat after reactor shutdown is passed into the water in the wet well via safety relief valves connected to the steam pipes within the dry well.

The concrete or steel structure of the reactor building surrounding the containment serves as protection against external events.



1 Containment
2 Upper containment volume (wet well)
3 Lower containment volume (dry well)
4 Bubbling condenser system (suppression pool)
5 Suppression pool cooling system (not required if the heat capacity of the condenser system (4) is sufficiently large)
6 Lower containment volume passive spray system
7 Lower containment volume active spray system
8 Filtered air discharge system
9 Liner

FIG. A-4. PWR bubbling condenser containment [1].

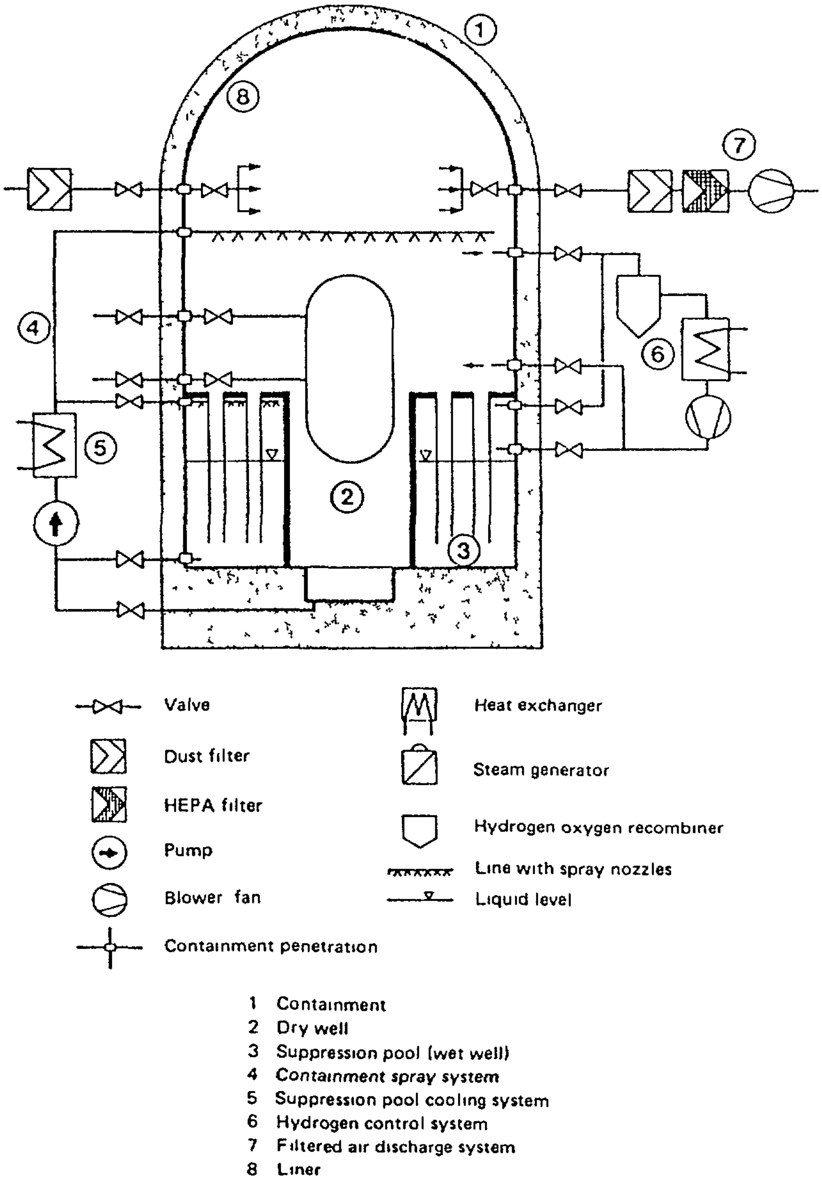


FIG. A-5. BWR pressure suppression containment (the reactor building with its confinement function is not shown) [1].

The reactor building is held at a slightly negative gauge pressure during both operational states and accident conditions In the case of an accident, leakage from the dry well into the reactor building is extracted and filtered by an air removal system, to permit controlled emission from the plant stack.

BWR weir-wall pressure suppression containment

The BWR weir-wall pressure suppression containment system (Fig. A-6) consists of three different structures, the dry well structure, the containment envelope, and the reactor building.

The function of the dry well structure is to completely enclose the reactor pressure vessel (RPV), and to create a pressure boundary which separates the RPV and its recirculation system from the containment vessel and the main body of the suppression pool The dry well structure vents the steam/air mixture to the suppression pool. It also provides radiation shielding from the reactor and

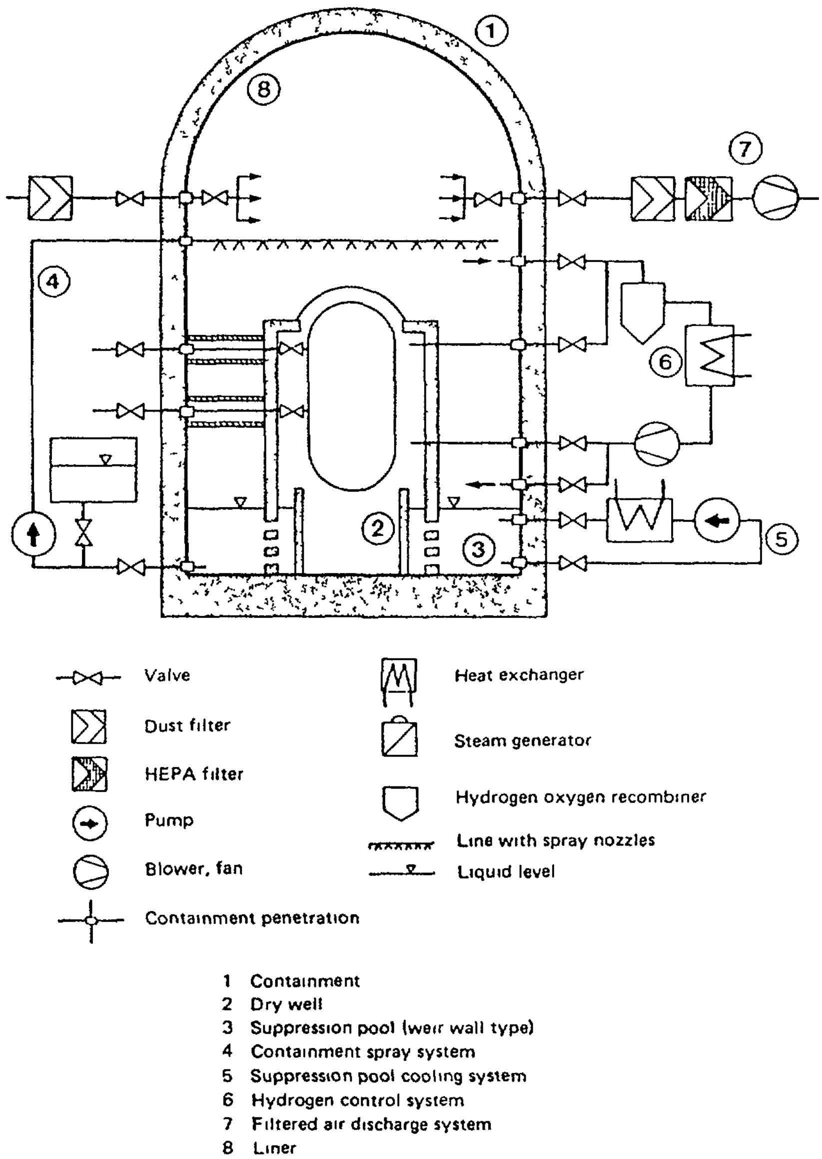


FIG. A-6. BWR weir-wall pressure suppression containment (the reactor building with its confinement function is not shown) [1]

the nuclear steam supply system piping. The weir-wall portion of the dry well structure functions as the inner wall of the suppression pool and serves to channel the steam released by a postulated loss of coolant accident (LOCA) through horizontal submerged vents into the suppression pool for condensation.

One of the functions of the reactor building is to provide protection against external missiles for the containment envelope, personnel and equipment. It also provides shielding against the fission products in the secondary confinement envelope, functions as a secondary containment barrier and provides a means for the collection and filtration of fission product leakage from the steel or concrete containment vessel following a LOCA.

During postulated LOCA conditions, the pressure rise in the dry well drives the water level between the weir wall and the dry well structure wall down, uncovering the vents in the dry well structure wall, and forces the dry well structure steam/air mixture through the vents and into the suppression pool. The steam is condensed in the suppression pool water. Fission product noble gases and other non-condensables from the dry well structure escape from the surface of the pressure suppression pool into the containment envelope.

In the long term, an active spray system is utilized for the containment envelope. This system takes water by suction from the suppression pool, through a heat exchanger, and the water is pumped to spray headers located in the dome of the containment envelope.

PTR negative pressure containment

The term 'negative pressure containment' is used to describe a containment system that typically consists of the following subsystems (Fig. A-7):

(a) A containment envelope that comprises the reactor buildings, the connecting pressure relief duct, vacuum ducts, the vacuum building and all the containment extensions.

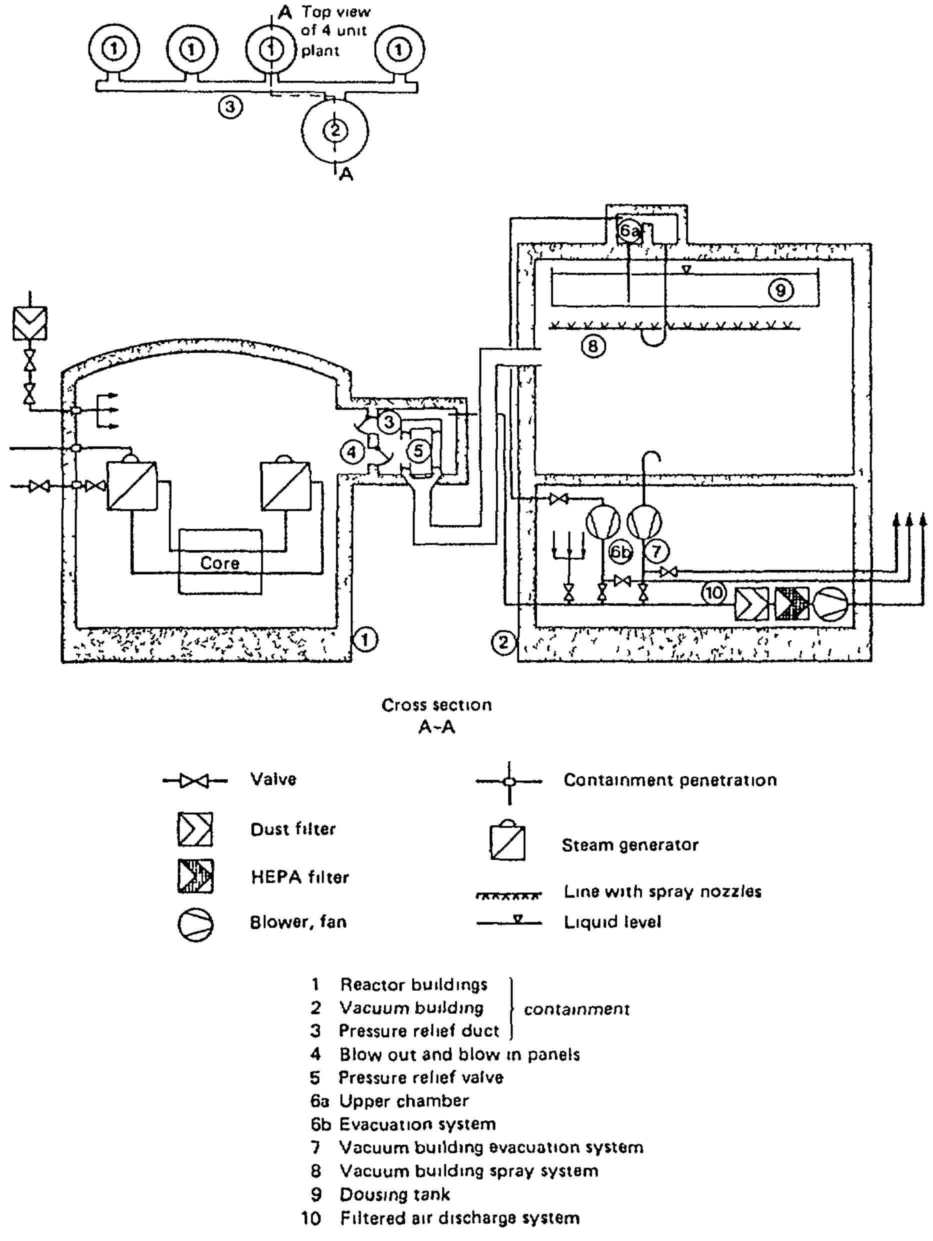


FIG. A-7. PTR negative pressure containment (typical application CANDU reactors) [1].

(b) A containment isolation system that closes all out-leakage paths automatically upon detection of radioactivity or pressure within the envelope.
(c) A pressure relief system that comprises the pressure relief panels which isolate the reactor buildings from the connecting pressure relief duct and the pressure relief valves which isolate this relief duct from the vacuum building.
(d) A vacuum system that maintains subatmospheric pressure inside the vacuum building so that when this building is connected to the containment, the atmosphere from the containment passes into the vacuum building.
(e) An energy suppression system, comprising a dousing tank, upper chamber vacuum system and spray header, that is housed inside the vacuum building and can absorb all the energy released to the vacuum building.
(f) An atmospheric control system that controls the atmosphere within the reactor buildings.
(g) A filtered air discharge system that would help to maintain subatmospheric conditions within the containment envelope if the need were to arise in the long term after an accident. The reactor buildings are maintained at slightly negative gauge pressures during both operational states and post-accident conditions.

Energy management is achieved by relieving the reactor building peak pressure to the vacuum building via the pressure relief system, which is actuated by a small increase in the reactor building pressure. Additional energy suppression takes place when the steam drawn into the vacuum building is condensed by the spray system, which is automatically actuated by a change in pressure within the vacuum building. Long-term heat removal from the containment is achieved by the atmospheric control system that cools the building air, and by the heat exchangers in the ECCS recirculation system. Radionuclide management is accomplished by plate-out on the internal surfaces of the containment envelope, by washout afforded by the spray and by the leaktightness of the containment envelope.

Single dry containment (typical application CANDU reactors)

The pressurized containment system (Fig. A-8) used in some pressure tube reactors single-unit station designs typically consists of the following subsystems:

(a) A containment envelope comprising a prestressed, post-tensioned concrete reactor building and its extensions.
(b) A containment isolation system that closes all out-leakage paths automatically upon detection of radioactivity or pressure within the envelope.
(c) An energy suppression system that consists of a dousing tank and a spray system that suppresses the initial peak pressure.
(d) An atmospheric control system that controls the atmosphere within the reactor building.
(e) A filtered air discharge system that helps to maintain subatmospheric pressure within the envelope in the long term and aids containment cleanup operations.

Upon detection of radioactivity or high pressure in the reactor building, the isolation system closes the appropriate penetrations of the containment envelope. When high pressure is detected in the reactor building, the dousing system is also activated. The initial peak pressure following a LOCA is suppressed by condensation of steam through the dousing spray system. Long-term energy management is provided by the atmosphere control system (building air coolers) and by the heat exchangers in the ECCS recirculation system. Radionuclide management is accomplished by plate-out on the internal surfaces of the containment envelope, by washout afforded by the dousing spray system and by the leaktightness of the containment envelope.

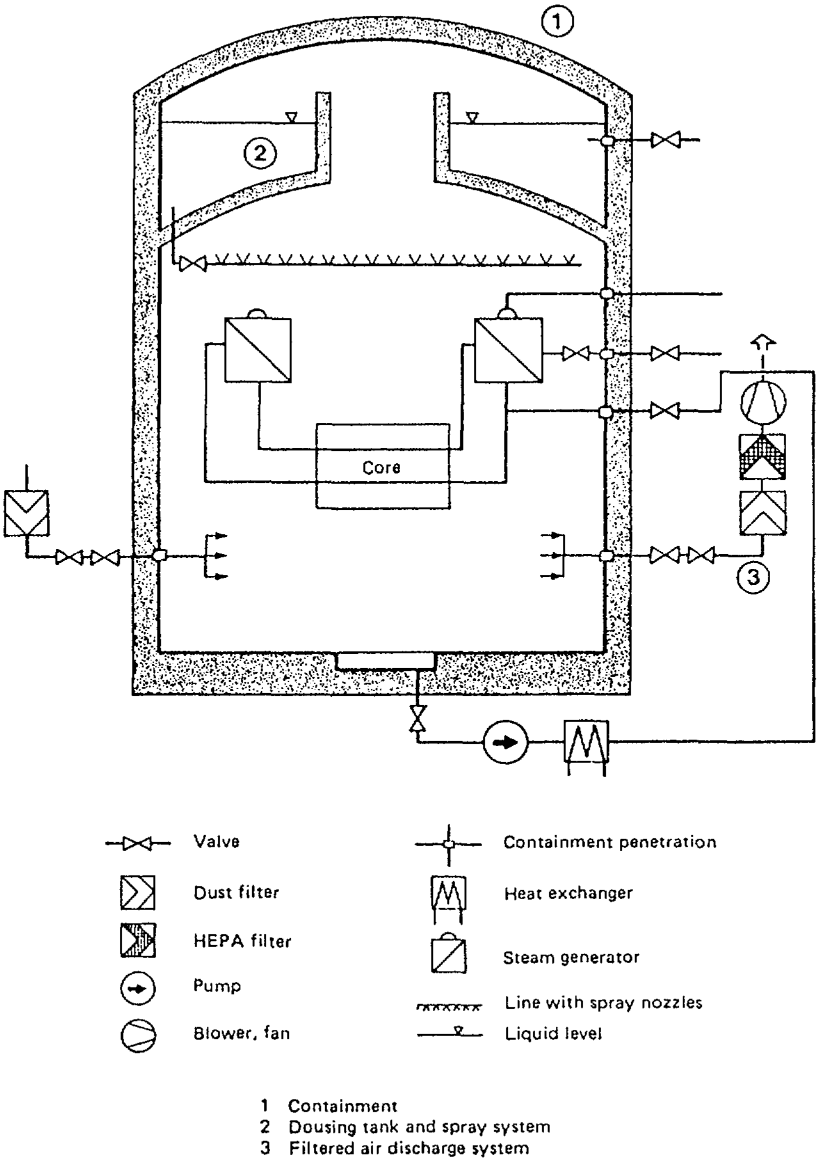


FIG. A-8. Single dry containment (typical application CANDU reactors) [1].

Appendix B

MAJOR CONTAINMENT CODE DESCRIPTIONS

Code: ATHLET-SA **Function:** Severe accident up to fission product transport in RCS.

Owner: GRS, Germany

Description: ATHLET-SA is a detailed mechanistic code and is modular in construction. It includes modules fur such things as

1. Thermal hydraulics.
2. Heat transfer and conduction.
3. Core degradation.
4. Neutron kinetics.
5. General control simulation.

ATHLET-SA is an extended version of the thermal hydraulics code ATHLET with the above additional modules to cover severe accident scenarios. The severe accident modules come from the KESS code and work is underway to include fission product release and transport models from other codes as well as KESS.

Limitations: One of the major limitations of the ATHLET-SA code is its inability to model Stainless steel Oxidation by steam and B_4C oxidation. There is also some doubt whether the current versions can model eutectic interactions between UO_2 and Zircaloy. Some of these phenomena can already be modelled in KESS so they will probably be incorporated in future versions. There are no modules for debris bed formation, fuel coolant interactions and vessel rupture and attack. Fission product release is based on empirical rate constants. An improved model based on diffusion is under development. Fission product transport will be included by linking with the TRAP-MELT code.

Code: CATHARE/ICARE **Function:** Severe accidents up to fission product transport in RCS.

Owner: CEA, France.

Description: This code is the result of merging the Thermal Hydraulics from CATHARE and core degradation from ICARE. Aerosol and fission product behaviour is modelled using the TRAPF programme a French derivative of TRAP-MELT. The ICARE2 code is a stand alone code and derives boundary conditions from CATHARE. Current developments are concentrating on late phase degradation e.g. debris bed formation, pool melts fuel coolant interactions up to vessel rupture. The ICARE code is available in the ESTER package and so can be used by the AEA.

Limitations: The CATHARE code is not recommended for the analysis of BWR type reactors. Grid spacers are not represented in the core models. Models are included for stainless steel oxidation but there are no models for B_4C oxidation. There are no models for fission product decay. Late phase melt progression modelling is under development.

Code: **MAAP** **Function:** Severe accidents up to release from containment.

Owner: EPRI, USA

Description: The modular accident analysis programme MAAP is an integrated code and has specific models for specific plants and phenomena. There are BWR and PWR versions.

The main differences are the choice of the containment and whether Ag-In-Cd (in the PWR version) is included in the fission product calculation. The code will model a complete accident up to containment failure. Steam explosions cannot be calculated. Fission product transport is included but few chemical reactions are considered. MAAP version 4.0 will consider chemical interactions during the melt phase, and add a generalized containment building model. MAAP 4.0 is under development. A version specific to the AP600 is also being developed.

Limitations: MAAP is limited to the designs hard wired into the code. This makes it difficult to perform calculations on new plant or examine the implications of new features. During the melt phase no consideration is given to chemical interaction of core materials or to heat generated by the oxidation of anything other than Zircaloy clad.

Code: **MELCOR** **Function:** Severe accident up to release from containment.

Owner: USNRC, USA

Description: MELCOR is an integrated code that can model a severe accident sequence from initiation through to release from the containment. The code is designed to be used in parametric studies. Detailed mechanistic models are included where this does not impact on the codes objectives.

Limitations: Many calculations have been carried out on BWR sequences with less attention to PWR problems. Steam explosions cannot be modelled but the mixing prior to a steam explosion can be modelled in the containment. It may not be possible to model B_4C in clad different from current BWR designs. No chemical interactions are considered between core materials. Chemical interactions between steam, stainless steel, zirconium and B_4C are considered. This code is currently subject to USNRC peer review, recommendations have been made for future developments. The code at present has not had extensive use on PWRs: this will change in a few years time.

Code: FUMO **Function:** Containment thermal hydraulics and non-condensable flow.	
Owner: University of Pisa, Italy	
Description: The FUMO code is aimed at the study of the thermal hydraulics in the containment following a LOCA or severe accident. It is a lumped parameter code. Mass and energy is conserved at nodes and momentum balance is conserved at junctions. Six non-condensable gases can be modelled Ar, CO, CO_2, H_2, N_2 and O_2. Homogenous gas/water mixtures which may or may not be in thermal equilibrium with a pool can be modelled.	
Limitations: It is the intention to model fission product and aerosol transport and their interaction with containment thermal hydraulics. Fission products and Aerosols and their interaction with themselves and containment materials. It is not clear whether hydrogen burning can be modelled.	
Code: JERICO **Function:** Containment Thermal Hydraulics and non-condensable flow	
Owner: CEA	
Description: The code is aimed at calculating containment thermal hydraulics. It is a lumped parameter code with nodes connected by flow paths and heat structures. Each node is split into two sub-nodes a liquid system and a gaseous system. There may be thermal non-equilibrium between the sub nodes. The gaseous system can contain 5 non-condensable gases N_2, O_2, H_2, CO, and CO_2. Hydrogen and carbon monoxide burning can be modelled.	
Limitations: Fission product and aerosol transport and chemistry are not modelled although decay heat can be included. The code has problems handling natural convection due to ignoring gravity.	

Code: CONTAIN **Function:** The modelling of containment during severe accidents.
Owner: USNRC, USA
Description: CONTAIN is a code to model the thermal hydraulics and fission product transport in containments. Models are included for such things as hydrogen burns, carbon monoxide burns, direct containment heating and core concrete interactions. Allowance is made for the radioactive decay and transmutation of elements.
Limitations: Rate constants are modelled in chemical interactions as a function of temperature alone. In the advanced plants where longer accident scenarios may occur other effects could be important. These could include pH dependent effects and radiolysis. Calculations should be carried out to see the significance of the phenomena.
Code: GOTHIC **Functions:** This is a general purpose thermal hydraulic code for the design, licensing safety and operation of nuclear containments
Owner: EPRI, USA
Description: This code solves mass, momentum and energy balances for three separate phases, vapour, liquid and drops. The vapour phase can include both steam and non-condensable gases. Containment compartments can be modelled in 1, 2 or 3 dimensions. Lumped parameter modelling is also possible and 3D and lumped parameter models in different compartments can be connected. Many models are available for such things as valves, pumps, pools, pressure suppression systems, and metal water reactions.
Limitations: Gas burns and aerosol transport are no modelled neither are fuel/coolant/ concrete interactions.

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ABBREVIATIONS

AP-600	Advanced Passive (Westinghouse reactor)
CANDU	Canada deuterium-uranium (reactor)
DBA	design basis accident
DCH	direct containment heating
ECCS	emergency core cooling system
EdF	Electricité de France
EPAFS	emergency passive air filtration system
EPR	European Pressurized Reactor
FCI	fuel-coolant interaction
FP	fission product
GDCS	gravity driven cooling system
IRWST	in-containment refuelling water storage tank
KfK	Kernforschungszentrum Karlsruhe
LIRA	Limitazione Intrinseca dei Rilasci Accidentali (Intrinsic Limitation of Accidental Releases)
LOCA	loss of coolant accident
LB-LOCA	large break LOCA
MCCI	molten core-concrete interaction
NPI	Nuclear Power International
PCCS	passive containment cooling system
PIE	postulated initiating event
PIUS	Process Inherent Ultimate Safety (reactor)
PRA	probabilistic risk assessment
PTR	pressurized tube reactor
PWR	pressurized water reactor
RPV	reactor pressure vessel
SBWR	simplified boiling water reactor
SGTR	steam generator tube rupture
SPWR	simplified pressurized water reactor
SRV	safety relief valve

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