Nuclear Safety
NEA/CSNI/R(2014)8
June 2014

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Status Report on Hydrogen Management and Related Computer Codes

BETTER POLICIES FOR BETTER LIVES

NUCLEAR ENERGY AGENCY

Unclassified **NEA/CSNI/R(2014)8**

Organisation de Coopération et de Développement Économiques
Organisation for Economic Co-operation and Development

20-Jan-2015

English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Cancels & replaces the same document of 26 June 2014

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JT03369497

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The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 30 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Russian Federation, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Within the OECD framework, the NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, as well as representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety amongst the NEA member countries. The CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and research consensus on technical issues; and to promote the co-ordination of work that serves to maintain competence in nuclear safety matters, including the establishment of joint undertakings.

The clear priority of the committee is on the safety of nuclear installations and the design and construction of new reactors and installations. For advanced reactor designs the committee provides a forum for improving safety related knowledge and a vehicle for joint research.

In implementing its programme, the CSNI establishes co-operative mechanisms with the NEA's Committee on Nuclear Regulatory Activities (CNRA) which is responsible for the programme of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with the other NEA's Standing Committees as well as with key international organizations (e.g., the IAEA) on matters of common interest.

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EXECUTIVE SUMMARY

In the follow-up to the Fukushima Daiichi NPP accident, the Committee on the Safety of Nuclear Installations (CSNI) decided to launch several high priority activities. At the 14th plenary meeting of the Working Group on Analysis and Management of Accidents (WGAMA), a proposal for a status paper on hydrogen generation, transport and mitigation under severe accident conditions was approved. The proposed activity is in line with the WGAMA mandate and it was considered to be needed to revisit the hydrogen issue, especially in the light of the Fukushima accident. This proposal was agreed and approved by CSNI in December 2012.

The writing group consists of participants from Belgium, Canada, Czech Republic, Finland, France, Germany, Italy, Japan, Korea, the Netherlands, Poland, Sweden, Switzerland, Spain, and USA. This report mostly covers the information provided by these countries.

The report is broken down into five Chapters and two appendixes.

- Chapter 1 provides background information for this activity and expected topics defined by the WGAMA members. A general understanding of hydrogen behaviour and control in severe accidents is discussed. A brief literature review is included in this chapter to summarize the progress obtained from the early US NRC sponsored research on hydrogen and recent international OECD or EC sponsored projects on hydrogen related topics (generation, distribution, combustion and mitigation).
- Chapter 2 provides a general overview of various reactor designs: Western PWRs, BWRs, Eastern European VVERs and PHWRs (CANDUs). The purpose is to understand the containment design features in relation to hydrogen management measures.
- Chapter 3 provides a detailed description of national requirements on hydrogen management and hydrogen mitigation measures inside the containment and other places (e.g., annulus space, secondary buildings, and spent fuel pool). Discussions are followed on hydrogen analysis approaches, application of safety systems (e.g., spray, containment ventilation, local air cooler, suppression pool, and latch systems), hydrogen measurement strategies as well as lessons learnt from the Fukushima Daiichi nuclear power accident.
- Chapter 4 provides an overview of various codes that are being used for hydrogen risk assessment, and the code capabilities and validation status for modelling hydrogen related phenomena (e.g., generation, distribution, combustion and mitigation).
- Chapter 5 summarizes the findings identified in Chapters 1 to 4 and discusses the remaining open issues.
- Appendix A presents the detailed descriptions of various containment designs. The main focus is to summarize the containment design features and systems to preserve containment integrity in case of an accident and, in particular, systems foreseen for severe accident management.
- Appendix B includes detailed descriptions of various codes with a focus on code capabilities and validation status for modelling hydrogen generation, distribution, combustion and mitigation.

The report has identified that the hydrogen mitigation strategies vary from country to country and depend primarily on the design of the containments. For NPPs with large dry containment, such as PWR, PHWR, and VVER-1000, a combination of a large free containment volume, the use of PARs, and/or glow plug igniters is commonly used, whereas for NPPs with small containments, such as BWR Mark I, nitrogen inerting in the whole containment is typically applied. The national requirements of the member countries on hydrogen management for SAs vary in details. For instance, some countries define maximum mean and local hydrogen concentrations (typically PWR) for the design of igniter or PAR, and others define maximum oxygen concentration (typically BWR) for N_2 inerting-concept.

In response to the Fukushima accident, hydrogen mitigation systems, particularly PARs, are now required to be installed in most of the countries inside the containment if there was no mitigation concept required before. For the NPPs, where the hydrogen mitigation systems are currently designed for DBAs only, the existing systems are being evaluated and considered to be enhanced under SA conditions.

The Fukushima accident revealed that hydrogen transport from the inerted BWR containment to the surrounding reactor buildings should be investigated. Most countries have not yet adopted specific national requirements for hydrogen mitigation measures outside the containment (e.g., annulus, reactor or secondary building, etc.) or the spent fuel pool areas. Due to the Fukushima accident, many countries have started to study SA conditions within such areas and to consider hydrogen management outside of the primary containment (i.e., reactor building) and at the spent fuel pool area. Question remains regarding the need of hydrogen management outside the containment and decision has to be made whether additional mitigation measures are required.

In most of the member countries, the requirements for operation of the engineering systems (e.g., spray, containment ventilation, local air cooler, suppression pool, latch systems) are defined based on their primary purpose (e.g., heat removal or depressurization). It is expected even though not further substantiated in this report that these requirements may be (or have to be) defined/updated to take into account their effect on hydrogen behaviour. In addition, some NPPs have installed hydrogen measurement systems, and this activity is currently under consideration by many of the member countries, but the implementation details vary (e.g., number and location of samples, actions defined in SAMG).

Regarding the analytical techniques to assess the hydrogen behaviour in a post-accident containment, most of the member countries are using lumped parameter codes (e.g., integral or system codes with mechanistic models) for full plant long term SA analysis combined with 3D-like or 3D codes for detailed short-term and/or local hydrogen analysis (e.g., hydrogen distribution, combustion and mitigation).

Amongst the 11 codes assessed in the report, only the integral or system codes are capable of calculating hydrogen generation in the reactor core and/or from MCCI in the cavity. Most codes have capabilities to model hydrogen distribution, combustion, mitigation systems and engineered safety features. In addition, some member countries have developed more complex models with CFD codes for better assessing hydrogen combustion, recombination and key phenomena such as condensation or evaporation, which can affect hydrogen distribution in the containment.

Code validation performed for hydrogen related phenomena such as generation, distribution, combustion and mitigation vary largely amongst the member countries. Some have performed extensive validations using their own experimental data and/or by attending international benchmarks, but some heavily rely on code developers. None of the codes are fully validated for the entire list of hydrogen generation, distribution, combustion and mitigation phenomena either due to a lack of experimental data to validate desired application range. Engineering judgement and a large degree of experience on code application is therefore needed in order to obtain realistic results.

R&D efforts to date have already significantly enhanced the understanding of the phenomena governing the distribution of hydrogen gas mixtures and their potential combustion. Research will continue to improve the understanding of SA conditions and consequences to equipment and components inside containment. The purpose is to reduce uncertainties and provide insights to refine the SAMGs. Further efforts are still needed to close research gaps, enhance code capability, and reduce code uncertainty.

In conclusion, the present report is an adequate basis for reviewing SAM strategies for hydrogen management. It is recommended that assessment of the SAM strategies or guidelines as well as advantages and drawbacks of the various hydrogen mitigation approaches implemented by different countries be pursued as a follow-up activity.

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ACRONYMS

ABWR	Advanced Boiling Water Reactor
AICC	Adiabatic Isochoric Complete Combustion
ALWR	Advanced Light Water Reactor
AP1000	Advanced Pressurized Reactor (design by Westinghouse)
BDBA	Beyond Design Basis Accident
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
CCVM	Containment Code Validation Matrix
CFD	Computational Fluid Dynamics
CHRS	Containment Heat Removal System
CSA	Complementary Safety Assessment
CSARP	Cooperative Severe Accident Research Program
DBA	Design Basis Accident
DCH	Direct Containment Heating
DDT	Deflagration-to-Detonation Transition
EFADS	Emergency Filtered Air Discharge System
ERCOSAM	Experimental investigations on Reactor COntainment thermal-hydraulics and Severe Accident Management measure
EPR	European Pressurised Reactor (design by AREVA)
FA	Flame Acceleration
FCV	Filtered Containment Venting
HYMERS	Hydrogen Mitigation Experiments for Reactor Safety
IRWST	In-containment Refuelling Water Storage Tank
ISP	International Standard Problem
LAC	Local Air Cooler

LOCA	Loss Of Coolant Accident
LP	Lumped Parameter
LWR	Light Water Reactor
MCCI	Molten Core Concrete Interaction
NPP	Nuclear Power Plant
PAR	Passive Autocatalytic Recombiner
PSR	Periodic Safety Review
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
PHWR	Pressurized Heavy Water Reactor
RCS	Reactor Cooling System
RPV	Reactor Pressure Vessel
SA	Severe Accident
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SARNET	Severe Accident Research NETwork of Excellence
SESAR	Senior Group of Experts on Safety Research
SETH	SESAR Thermal-Hydraulics
SOAR	State-Of-the-Art Report
SFP	Spent Fuel Pool
TMI	Three Mile Island
THAI	Thermal-hydraulic, Hydrogen, Aerosol and Iodine
VVER	Vodo-Vodyanoi Energy Reactor

Organizations:

AECL	Atomic Energy of Canada Limited, Canada
ASN	Autorité de Sûreté Nucléaire, France
Bel V	A subsidiary of the Federal Agency for Nuclear Control, Belgium
CEA	Commissariat à l'Energie Atomique, France
CSN	Consejo De Seguridad Nuclear, Spain
CSNI	Committee on the Safety of Nuclear Installations

CNSC	Canadian Nuclear Safety Commission
EC	European Commission
EDF	Electricity of France
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EPRI	Electric Power Research Institute
FZJ	Forschungs-Zentrum Jülich GmbH, Germany
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit, Germany
IRSN	Institut de Radioprotection et de Sûreté, France
JNES	Japan Nuclear Energy Safety Organization
KAERI	Korea Atomic Energy Research Institute
KIT	Karlsruhe Institute of Technology, Germany
KFD	Nuclear Energy Service, the Netherlands
NEA	Nuclear Energy Agency
NCBJ	National Centre for Nuclear Research, Poland
NRC	Nuclear Regulatory Commission, USA
NRI	Nuclear Research Institute, Czech Republic
NRG	Nuclear Research & Consultancy Group, the Netherlands
OECD	Organization for Economic Co-operation and Development
PSI	Paul Scherrer Institut, Switzerland
RSK	German Reactor Safety Commission
SSM	Swedish Radiation Safety Authority, Sweden
VTT	Technical Research Centre of Finland
WGAMA	Working Group on Analysis and Management of Accidents

1 INTRODUCTION

1.1 Background

During the Fukushima Daiichi nuclear power plant accident, hydrogen explosions occurred in three units, resulting in severe damage of the facilities and primary and secondary containment structures. The hydrogen explosions in units 1 and 3 were caused by the build-up of hydrogen gas within the primary containment due to fuel damage in the reactor and subsequent transport of that hydrogen gas to the secondary containment. The source of hydrogen in unit 4 is believed to be an inverse flow of hydrogen gas mixture through a standby gas treatment system from unit 3.

Following the Fukushima accident, there are lessons being learnt, analyses being conducted, and information being collected to support safety enhancements to cope with events that go beyond the design basis. It has been recognized that significant improvements are needed in national and international communications and information exchange among national regulatory organisations.

The Committee on the Safety of Nuclear Installations (CSNI) has developed a working document, "Considerations and Approaches for Post-Fukushima Daiichi Follow-Up Activities", which identified concepts to be considered in response to the accident. The underlying technical phenomena associated with the Fukushima accident, including such matters as fuel and system performance hydrogen generation and behaviour of the spent fuel pool, was identified as high priority for future research programmes.

At the 14th plenary meeting of the Working Group on Analysis and Management of Accidents (WGAMA) in September 2012, a proposal for a status report on hydrogen generation, transport and mitigation under severe accident (SA) conditions was approved. The proposed activity is in line with the WGAMA mandate and it is considered to be needed to revisit the hydrogen issue, especially in light of the Fukushima accident. This proposal was agreed and approved by CSNI in December 2012 as a Fukushima follow-up activity. A writing group was thereafter formed, consisting of participants from Belgium, Canada, Czech Republic, France, Finland, Germany, Italy, Japan, Korea, the Netherlands, Poland, Sweden, Switzerland, Spain, and USA. The majority of this report only covers the information provided by these countries.

It was defined in the WGAMA proposal that the deliverable of this activity is to write a report and to include the following topics:

- Compile the status on the implementation of hydrogen mitigation means for light water, heavy water and boiling water reactors including systems already installed and contemplated;
- Describe the national requirements on the implementation of hydrogen mitigation means (recombiners, igniters, inert gas injection, etc);
- Briefly describe the different systems available as well as their demonstrated or expected performance for in and ex vessel SA phases;
- Briefly, describe the status of code validation dedicated to hydrogen generation, hydrogen distribution and hydrogen combustion;
- Discuss possible effect of others safety systems on hydrogen risk: effect of water spray system actuation on hydrogen flame acceleration, hydrogen risk in containment venting system, hydrogen management in secondary buildings;
- Identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima accident;
- Identify, from an accident management perspective, if there is room for improvements, both for the hardware and the qualification of the systems;

- Summarize the status of hydrogen risk management strategies as currently implemented, especially the strategies that require interfacing with decision making processes to actuate water spray and containment venting.

Following the above guidelines, this report was prepared and the following items were reviewed within the participating countries:

- Hydrogen management approaches under design basis accidents (DBAs) and SA conditions, including national requirements, mitigation systems and their implementation status inside and outside of the containment (e.g., annulus, reactor or secondary buildings, spent fuel pool area, etc.) as well as hydrogen measurement strategies;
- Consideration of engineering systems (e.g., containment spray, local air cooler, venting system, latch/blow-out panels and doors, etc.) and their potential effect on hydrogen distribution and combustion;
- Lessons learnt from the Fukushima and insights about hydrogen control,
- Hydrogen assessment methodology and its application to probabilistic safety assessment (PSA),
- Codes that are dedicated to hydrogen generation, distribution, combustion, and mitigation, their capabilities and validation status.

It should be noted that this report was also intended to cover the following topics which were proposed in the WGAMA proposal:

- Assess severe accident management (SAM) strategies or guidelines (SAMG) implemented by different countries and determine if improvements are needed;
- Advantages and drawbacks of the various hydrogen mitigation approaches implemented by different countries.

Although it is viewed as being premature to draw conclusions regarding these last two topics, much of the information required to undertake the assessment have been gathered and presented in this document. Assessment of SAM strategies, including examination of advantages and drawbacks of various hydrogen mitigation approaches is being undertaken by individual regulators and it is anticipated that this document will provide part of the technical basis to inform regulatory positions on strategies for severe accident hydrogen management. Follow-up activities are under discussion.

It must be noted that the definition for DBAs or SAs can be slightly different depending on reactors or countries. In this report, the DBAs are referred to events where the extent of the in-core metal–water reaction is limited at low values by the operation of the emergency core cooling systems. The time scale in DBAs involved for the generation of hydrogen allows sufficient time for initiation of measures to control the amount of hydrogen in the containment atmosphere and to prevent any burning. The SAs, involving large scale fuel degradation and possibly even MCCI, raise the possibility of hydrogen release rates greatly exceeding the capacity of conventional DBA hydrogen control measures. The SAs at the Three Mile Island (TMI) and Fukushima illustrated the potential of unmitigated hydrogen accumulation to escalate the potential consequences of a SA.

1.2 Hydrogen Behaviour and Control in Severe Accidents

During the course of SAs in water-cooled nuclear power plants (NPPs), a large amount of hydrogen could be generated and released into the containment. The formation of hydrogen inevitably accompanies the core degradation process or molten core-concrete interaction (MCCI). As the TMI-2 and Fukushima accidents revealed, hydrogen combustion could cause high pressure spikes (mechanical loads) and high temperatures (thermal loads). These loads could damage the equipment (probably partially being used to mitigate the consequences of the accident), and, eventually, cause the failure of the NPP containment, thus breaking the last safety barrier and allowing the release of fission products to the environment. The Fukushima accident showed that hydrogen transport from the inerted BWR containment to the surrounding reactor buildings is another important subject. There the hydrogen accumulation in the reactor building and the hydrogen combustion led to a strong destruction of the reactor building structure.

In order to develop strategies and means to control at least the hydrogen concentrations inside the containment in case of SA scenarios, past and on-going investigations are performed with the aim to quantify the hydrogen produced during a postulated SA scenario and determine the hydrogen behaviour inside the containment, including distribution, combustion and mitigation, as well as its interaction with mitigation measures and safety systems. Investigations on possible hydrogen transport from the containment to the adjacent buildings identified that additional activities are needed to address this issue. A final decision whether additional mitigation measures are required has not been taken yet.

The phenomena associated with hydrogen generation, distribution, combustion and mitigation are discussed in the following sections.

1.2.1 Hydrogen Generation

During a postulated SA in a NPP, hydrogen can be produced during the following processes:

- **Core degradation within the reactor pressure (or calandria) vessel** (RPV), where hydrogen is produced from exothermal oxidation of Zircaloy of the fuel cladding and fuel assembly canisters during the early core degradation process as well as in the later phase through oxidation of other hot metallic components or metals within the molten pool or debris bed in the RPV. More details are given in Chapter 1.3.2.1.
- **Molten core concrete interaction**: After the failure of RPV, the melt can relocate to the reactor pit (or calandria vault) if an in-vessel retention strategy is not considered or fails. Depending on the concrete composition, which is NPP design specific, a large amount of hydrogen and carbon monoxide may be produced during the MCCI. Generation of non-condensable gas from MCCI is a continuous process in contrast to the release of hydrogen from the core oxidation within the RPV. The question whether the MCCI process can be terminated by flooding is still uncertain and under further investigation. More details are given in Chapter 1.3.2.1.
- **Fuel degradation within the spent fuel pools**:

  The irradiated and spent fuel is stored in the spent fuel pool (SFP). In case of loss of cooling for a SFP, the SFP pool water can be heated-up by the decay heat of the fuel assemblies. As a consequence, the water level in the SFP may decrease due to boil-off of the SFP water, or leak in the SFP, or rupture of a pipe connected to the pool, especially if it is connected to the pool at a low level. With no or without sufficient water supply, the fuel assemblies may become uncovered and the fuel can be overheated. Oxidation of the cladding by steam may lead to a large amount of hydrogen production, as shown in experiments of the Sandia Fuel Program [1.1].

  Following the Fukushima accident, concerns on SFP cooling have been increasingly raised in the international nuclear community. It is still under discussion whether hydrogen mitigation measures are needed or what measures can be implemented, although no specific requirements have been implemented in most countries. Nevertheless, the primary focus is on preventive measures, e.g., implementation of additional injection and heat removal systems from the SFP.

1.2.2 Hydrogen Distribution

The hydrogen can be released into the containment or reactor building through engineered pathways and breaks of the reactor cooling system (RCS), or during MCCI, or from fuel degradation within the spent fuel pools. After the initial blow down, transport of hydrogen is mostly driven by convection loops due to the release of hot steam/gas mixture or steam condensation on cold walls and structures, if no other source of forced flow exists. Depending on the level of mixing in the containment atmosphere, hydrogen can be well-mixed or stratified.

If considerable hydrogen stratification exists, local concentration of hydrogen could become a concern because pockets of high hydrogen concentrations may lead to Flame Acceleration (FA) or Deflagration-to-Detonation Transition (DDT) if the combustible mixture is ignited.

The hydrogen distribution and local concentration may also be affected by the operation of other safety systems. For instance, spray systems or local air coolers are used in some reactors to limit the containment pressure and to provide heat removal by steam condensation on water droplets or cooler surfaces. On the one hand, these measures may homogenise the hydrogen distribution in the containment due to enhanced mixing, but on the other hand, they can significantly reduce the steam concentration, which may lead to more sensitive gas mixture compositions by removing steam inerting.

Moreover, systems installed for SAs, such as passive autocatalytic recombiners (PARs) or igniters, can start hydrogen recombination or combustion at low hydrogen concentrations before the hydrogen accumulates to sensitive mixtures. The related processes may also enhance the gas mixing. The use of thermal recombiners (electrically powered) implemented for DBA conditions is recommended in SAM strategies, in case other systems fail or are not implemented. The main purpose of thermal recombiners is to initiate the recombination process below the low flammability limit. If local conditions exceed this value, the recombiner can act as a reliable ignition source to initiate lean hydrogen deflagrations.

1.2.3 Hydrogen Combustion

Once released into the containment, hydrogen is mixed with the atmosphere and can lead to flammable mixtures. The flammability of a gas mixture depends on its temperature and pressure (see Figure 1.2.3-1), composition, and the availability of an ignition source. In practice, the point representing the mixture composition (hydrogen, air, steam) on the Shapiro diagram (see Figure 1.2.3-2) is generally used to determine whether the mixture is flammable. In this diagram, the combustible regions (yellow – non-complete combustion, light orange – complete combustion) and the detonable region (dark orange) are delimited by the exterior and interior curves, respectively. The detonation limit is not an intrinsic property; it is geometry dependent. The red lines are simplified combustion limits which are often used in codes.

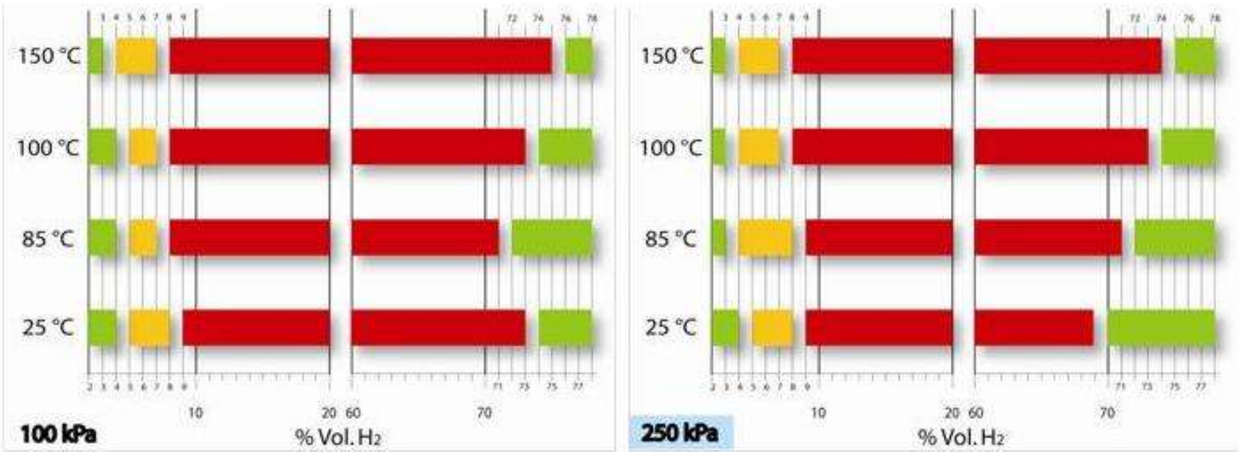

Figure 1.2.3-1 Flammability limits diagrams of H_2/air mixtures between 25°C and 150°C at an initial pressure of 100 kPa and 250 kPa [1.2] (Green: no flame propagation; Yellow: upward propagation; Red: full propagation)

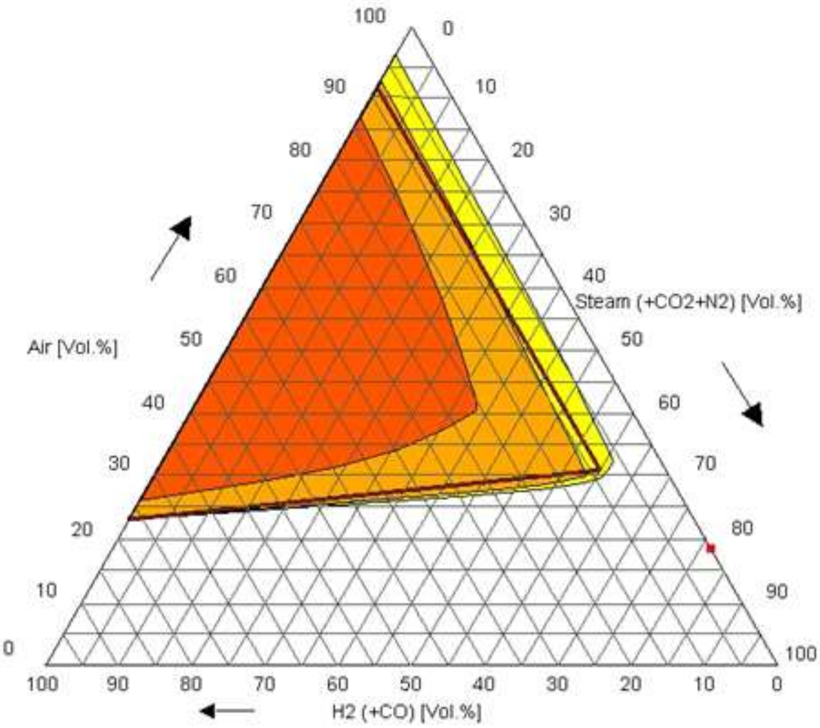

Figure 1.2.3-2 Shapiro diagram for hydrogen-air-steam mixtures at 1 bar and ambient temperature used in COCOSYS (yellow area – limited/upward combustion, light orange area - complete combustion with flame propagation in all directions, dark orange area: detonation) [1.3]

For a flammable hydrogen-air-steam mixture, deflagration may be initiated by an energy source of a few mJ (i.e., actuator switching, hot points developing, hot gases released from the reactor coolant system or use of igniters). In contrast, a much more powerful energy source (>100 kJ) is required to trigger a stable detonation. Direct initiation of detonation is highly unlikely inside the containment or reactor building; the only mechanism likely to provoke a detonation is through FA and DDT. In fact, due to the effect of hydrodynamic instabilities and turbulence (caused primarily by obstacles in the flame path), an initially slow flame (with a flame velocity around 1 m/s) may accelerate. Fast combustion regimes may also develop, involving FA (a few hundred m/s), DDT and detonation (over 1000 m/s). These fast combustion modes may pose a threat to the integrity of the containment structure and reactor safety equipment.

1.2.4 Hydrogen Mitigation

Comprehensive R&D programmes were developed in many different countries since the TMI-2 accident to develop SAM strategies to prevent fast hydrogen combustion in case of a SA in the reactor core. The main focus was to prevent or mitigate challenges posed on containment integrity. The major SAM strategies were extensively discussed at the first OECD/NEA workshop held by AECL and the CANDU Owners Group in Winnipeg, Canada in May 1996, focusing on the *Implementation of Hydrogen Mitigation Techniques* [1.4] [1.5]. In the same year, the CSNI principle working group on the confinement of accidental radioactive releases published a technical opinion paper on *Implementation of Hydrogen Mitigation Techniques during Severe Accidents in Nuclear Power Plants* [1.6]. The common SAM strategies are as follows:

- the deliberate ignition of the mixture as soon as the lower flammability limit is reached,
- the consumption/recombination of hydrogen (e.g., by PARs),
- a combination of ignition and recombination measures,

- the replacement of oxygen by an inert gas (N_2 typically) already during plant operation,
- the dilution of the atmosphere to prevent the formation of flammable mixtures either by increasing the volume of the containment, or by injecting an inert gas, or
- the release of hydrogen by containment venting.

The experts concluded that the choice of a hydrogen mitigation strategy for SAs depends primarily on the design of the containment (see more details in Chapters 2 and 3) and the overall hydrogen concentration expected to be released into the containment during a SA and its release rate, respectively. Geometrical sub-compartmentalization is also very important because significant amounts of hydrogen could accumulate in compartments to create high local concentrations of hydrogen that could be well within the detonability limits [1.6].

It is pointed out in Reference [1.6] that once accident management measures aimed at preventing SAs from occurring have failed and hydrogen is being generated and released to the containment atmosphere in large amounts, three steps can be taken:

- Step 1: Reduce the possibility of hydrogen accumulating to flammable concentrations.
- Step 2: Minimize the volume of gas at flammable concentrations if flammable concentrations cannot be precluded.
- Step 3: Prevent further increasing hydrogen levels from the flammable to detonable mixture concentrations.

Deliberate ignition and recombination by PARs are the two major hydrogen mitigation measures implemented in the containment atmosphere for containments with larger volume, whereas replacement of oxygen by an inert gas (N_2) is typically employed for smaller BWR containments and containments using ice condenser units (see more details in Chapter 2) [1.4] [1.5].

1.3 Overview of International Research on Hydrogen Behaviour

1.3.1 Early Research (1980 – 1998)

Since the TMI accident in 1979, there has been a great deal of interest regarding the concerns on hydrogen generation and combustion in water-cooled reactors. Comprehensive R&D programmes were developed to address issues on hydrogen by the international communities. In particular, a series of hydrogen combustion tests were performed by the US national laboratories between 1980 and 1998. These studies have established a foundation for understanding of hydrogen behaviour and development of analytical tools for hydrogen management. A brief overview of the hydrogen research programmes sponsored by US NRC is introduced in the following. Generally, testing was dominated by slow combustion in the 1980s and the 1990s testing was directed towards fast combustion.

In 1983, US NRC published a *Light-Water Reactor Hydrogen Manual* [1.7] to address concerns on hydrogen behaviour in Light-Water Reactors (LWRs) for both normal operations and accident situations. The topics included hydrogen generation, transport and mixing, detection, and combustion, and mitigation. Basic physical and chemical phenomena were described and plant-specific examples were provided. Another report on *Hydrogen Combustion Characteristics Related to Reactor Accidents* [1.8] provided knowledge of combustion phenomena and their characteristics necessary for accident analyses and perspective on combustion processes which could not be derived easily from the enormously diverse combustion literature.

In 1982, the report on *Final Results of the Hydrogen Igniter Experimental Program* [1.9] evaluated capability of glow plugs to function as ignition sources in hydrogen-air-steam environments to support the proposal of intentional ignition of hydrogen by glow plug as an active component for hydrogen mitigation system in nuclear reactor containments.

In 1984, the report on *Hydrogen-Burn Survival Experiments at Fully Instrumented Test Site (FITS)* [1.10] examined equipment survivability in a hydrogen-burn thermal environment and provided data for a better understanding of thermal response of components and verification of computational models.

In 1985, the report on *Hydrogen-Steam Jet-Flame Facility and Experiments* [1.11] and a later report on *Analysis of Diffusion Flame Tests* [1.12] investigated physical characteristics of hydrogen diffusion flames and their controlling parameters, including auto-ignition, flame stability, flame-length dependence on jet mixture, and heat transfer to various objects. The report on *Data Analyses for Nevada Test Site (NTS) Premixed Combustion Tests* [1.13] provided an in-depth analysis of premixed large-scale combustion experiments conducted at the Nevada Test Site (NTS), covering hydrogen concentrations of 5 to 13 vol.% and steam concentrations of 4 to 40 vol.% as well as some tests incorporated with spray systems and/or fans, which enhanced the combustion rate and significantly altered the post-combustion gas cooling.

In 1986, the report on *Hydrogen Air Steam Flammability Limits and Combustion Characteristics in the FITS Vessel* [1.14] presented data on flammability limits of hydrogen-air-steam mixtures in both turbulent and quiescent environments and a correlation that described the three-component flammability limit.

In 1989, the report *Experimental Results Pertaining to the Performance of Thermal Igniters* [1.15] summarized the findings relevant to the performance of thermal igniters from numerous research programmes conducted since 1980. These included studies of the combustion characteristics of hydrogen burns initiated by igniters, flammability limits of hydrogen-air and hydrogen-air-steam mixtures, and hydrogen mixing processes. Several programmes investigated the effect of water sprays on igniter performance. Others studied the effect of steam condensation on igniter effectiveness. Most research programmes reported heat transfer characteristics of hydrogen combustion and resultant flame speed data.

Between 1987 and 1998, the research was directed towards detonation studies, including reports published on:

- *Detonability of H2-Air-Diluent Mixtures* [1.16] that presented detonation cell width and velocity results of H_2-air mixtures, undiluted and diluted with CO_2 and H_2O for a range of H_2 concentration, initial temperature and pressure.
- *FLAME Facility: The Effect of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation for Hydrogen-Air Mixtures at Large Scale* [1.17] that studied FA and DDT of hydrogen-air mixtures and its relevance to nuclear reactor safety.
- *Hydrogen-Air-Diluent Detonation Study for Nuclear Reactor Safety Analyses* [1.18] that investigated detonability of hydrogen-air-diluent mixtures for the effects of variations in hydrogen and diluent concentration, initial pressure and temperature.
- *The Effect of Lateral Venting on Deflagration-to-Detonation Transition in Hydrogen-Air-Steam Mixtures at Various Initial Temperatures* [1.19] and *The Effect of Initial Temperature on Flame Acceleration and Deflagration-to-Detonation Transition Phenomenon* [1.20] that investigated the influence of gas venting and initial gas temperature on FA and DDT in an obstacle-laden tube with hydrogen-air-steam mixtures.
- *Deliberate Ignition of Hydrogen-Air-Steam Mixtures in Condensing Steam Environments* [1.21] that presented large scale experiments to determine if a detonation or an accelerated flame could occur in a hydrogen-air-steam mixture which was initially non-flammable due to steam dilution but was subsequently rendered flammable by rapid condensation of steam due to water sprays.

At the same time, many other countries (i.e., Canada, France, Germany, and Russia) also performed extensive research to examine various aspects of hydrogen phenomena, such as hydrogen combustion studied in References [1.22] through [1.29], which provided the basis for the FA and DDT state-of-the art report to be discussed in Chapter 1.3.2.

1.3.2 Recent Research (1999 – Present)

Over the past 30 years, significant advances in the understanding of hydrogen behaviour and codes development have been gained through various experimental programmes. In the recent 15 years, extensive studies are still continuing by the international nuclear communities to assess the threat of hydrogen in SAs. For instance, the US NRC sponsored Cooperative Severe Accident Research Program (CSARP), the OECD/NEA projects, including SESAR Thermal-Hydraulics (SETH /SETH2), Thermal-hydraulic Hydrogen Aerosol and Iodine (THAI/THAI2) and Hydrogen Mitigation Experiments for Reactor Safety (HYMERES) projects, which have been (are being) carried out under the auspices of OECD/NEA with financial support of large number of organizations around the world, as well as European Commission (EC) sponsored Severe Accident Research NETwork of Excellence (SARNET)/SARNET2 and Experimental investigations on Reactor Containment Thermal-hydraulics and Severe Accident Management Measure (ERCOSAM) projects. The ERCOSAM project has been coordinated with the Russian ROSATOM SAMARA project. The main purpose of these projects was to address the remaining research needs on hydrogen and to provide data for model development and validation. Many State-Of-the-Art Reports (SOARs) have been published on various aspects of hydrogen issues, e.g., generation, distribution, combustion and mitigation. A brief overview of these researches carried out after 1998 is discussed in the following.

1.3.2.1 Hydrogen Generation

In 2000, OECD/NEA published a report [1.30], *Perspective on Ex-Vessel Hydrogen Sources*, which identified the potential ex-vessel hydrogen sources and addressed the question whether further investigations were required considering the uncertainties associated to these sources. It was concluded that the hydrogen sources must be considered as a whole and cannot be separated into in-vessel and ex-vessel issues. For significant sources that may not be accommodated by mitigation means associated to DBAs, the uncertainty was largely dominated by the unknown extent of Zr oxidation during the in-vessel phase. Hydrogen production during corium quenching by water was not adequately understood and additional work was required.

In 2001, OECD/NEA published another report [1.31], *In-Vessel and Ex-Vessel Hydrogen Sources*, which identified potential in-vessel hydrogen sources and reviewed the state of knowledge on underlying physical and chemical processes. The availability of models and codes for accident analysis was addressed, including the remaining uncertain areas. It was recognized that the hydrogen source term can be predicted within the measurement errors during the cladding oxidation of intact fuel rods by steam. Major uncertainties existed for oxidation during reflood processes, which were under study in ongoing experimental programmes. A very limited database was available for the late phase of SA scenarios, where the rod-like geometry had degraded to a debris bed or a molten pool. In these configurations, key uncertainties in prediction of the hydrogen source term were related to oxidation behaviour of Zr-rich mixtures including both the kinetics of oxidation and the steam-debris interaction surface.

In 2011, IAEA published a report [1.32], *Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants*, which discussed potential hydrogen sources during a SA from in-core degradation to MCCI. It was recognized that the in-vessel hydrogen generation processes are the main factors influencing the hydrogen risk, which is generally well simulated when the core geometry is still intact, but some knowledge is needed for the late phase of the core degradation (i.e., U-Zr-O oxidation and core reflooding), for which some studies have been done in the frame of the EC sponsored COLOSS project and some integral experiments (CORA, LOFT). The current German QUENCH experiments at KIT will extend the knowledge database on hydrogen production associated with core flooding, but uncertainties are expected because prototypic materials are not used. Ex-vessel hydrogen production is not considered to be an issue because all the released vessel metallic components will be fully oxidized within about an hour after the vessel failure.

The first OECD/NEA MCCI project was completed in 2005 ([1.33], [1.34]) and dedicated to provide experimental data to resolve two important accident management issues:

- Verify that molten debris that has spread on the base of the containment can be stabilised and cooled by water flooding from the top;
- Assess the two-dimensional, long-term interaction of the molten mass with the concrete structure of the containment, as the kinetics of such interaction is essential for assessing the consequences of a SA.

After successful completion of the first MCCI Project at the Argonne National Laboratory (ANL), a second phase OECD/NEA MCCI-2 project was initiated using the same ANL facilities from 2006 to early 2010 [1.35]. The purpose was to help fill gaps that were not fully covered in the first phase. The testing fell into four categories:

- Combined effect tests to investigate the interplay of different cooling mechanisms and to provide data for model development and code assessment.
- Tests to investigate the effectiveness of new design features that enhance debris coolability.
- Tests to generate additional 2D core-concrete interaction data for model development and code validation.
- Integral test at larger scale to confirm synergistic effect of different cooling mechanisms and to provide data for validation of SA codes.

In parallel to these tests, a supporting analysis task was carried out to further develop/validate debris coolability models that form the basis for extrapolating the experiment findings to plant conditions. In total, 10 tests were conducted in this programme (all successful).

At KIT, experiments are also performed at the MOCKA facility to gain a detailed understanding of the interaction of the corium with the concrete. The MOCKA facility is a new facility which is designed to investigate the corium/concrete interaction in an anticipated core melt accident in LWRs, after the metal melt is layered beneath the oxide melt. The experimental focus is on the cavity formation in the basemat and the risk of a long-term basemat penetration by the metallic part of the melt [1.36].

1.3.2.2 Hydrogen Distribution

In 1999, OECD/NEA published a SOAR on *Containment Thermal Hydraulics and Hydrogen Distribution* [1.37]. The objective was to assess the code capabilities in predictions of pressure, temperature and gas concentration-distribution inside the containment under SA conditions; and to address strengths and limitations of analytical methods to predict the effectiveness of a chosen mitigation technique. The report described hydrogen mitigation techniques and phenomena relevant to containment thermal-hydraulics and hydrogen distribution under SA conditions. It also assessed experimental activities performed in 1980 and 1990’s investigating phenomena related to containment thermal hydraulics and hydrogen distribution. It also discussed code development and application activities, particularly the International Standard Problem (ISP)-23, 29, 35 and 39 benchmarks against the HDR and NUPEC hydrogen mixing and distribution tests. It also evaluated remaining uncertainties derived from experimental and analytical experience with a focus on code application for containments of NPPs. One of the outcomes of this SOAR is that lumped parameter codes are satisfactory to make predictions of pressure history and average steam content, and that the use of field codes to analytically predict gas concentration distributions will require “considerable validation and accumulation of code application experience” before they are used for plant analysis.

In the OECD/NEA SETH-PANDA project started in 2002 [1.38], basic phenomena, such as jets and/or plumes, inducing gas mixing, transport and stratification in a containment volume, were examined in the PANDA facility at integral scale, using “CFD-grade” instrumentation.

In May 2003, OECD/NEA published two reports on *ISP-42 (PANDA Tests) Blind and Open Phase Comparison* ([1.39], [1.40]). The ISP-42 PANDA test scenario was established to cover many typical LWR and Advanced LWR (ALWR) containment and primary system phenomena. It was designed to answer questions with respect to how far the systems or containment codes can go in assessing the classes of phenomena investigated and what could be expected either from available commercial CFD codes or codes with 3-D capabilities for containment analysis. One of the outcomes was to clarify model development needs in relation to calculation uncertainty and safety relevance to certain particular phenomena.

In 2004, the EC established a 4-year network (SARNET) to resolve the most important remaining issues for SAs [1.41]. One of its work package (WP12) was devoted to examine the hydrogen distribution in the different parts of the containment. The influence of containment sprays on atmosphere behaviour was investigated through benchmark exercises based on experiments performed in TOSQAN (IRSN, France) and MISTRA (CEA, France) facilities [1.42]. It was concluded that the level of validation obtained was encouraging for the use of spray modelling for risk analysis. However, some more detailed investigations are needed to improve model parameters and decrease the uncertainty for containment applications as well as to increase the predictability of the phenomena within the containment analyses. In 2009, the follow-up project, SARNET2, started. Hydrogen mixing was again one of the topics [1.43], including modelling of containment sprays and condensation modelling with CFD codes.

In September 2007, OECD/NEA published a report on *International Standard problem ISP-47 on Containment Thermal-hydraulics* [1.44]. The main objective of the ISP-47 was to assess the capabilities of LP and CFD codes in the area of containment thermal-hydraulics. Data generated in three experimental facilities TOSQAN, MISTRA and THAI (Becker Technology, Germany) were used for benchmarking in steady-state and transient conditions. Except the pressure transients and gas temperature field as used in the former benchmark exercises, detailed gas velocity and gas concentration (air, steam and helium) fields were obtained. It has been realized that further work is needed in code development, establishment of nodalisation rules, and code user training to achieve more accurate predictive capabilities for containment thermal-hydraulics and atmospheric gas/steam distribution. Uncertainties in modelling of condensation, gas stratification and jet injection appeared to be the major issues.

In 2007, the OECD/NEA sponsored THAI project was initiated and lasted 3 years to address open questions concerning the behaviour of hydrogen, iodine and aerosols in the containment of water cooled reactors during SAs [1.45]. The experiments were conducted in the intermediate scale THAI facility. One of the test series, helium/hydrogen material scaling tests, demonstrated that helium can be used as a substitute for hydrogen for experiments investigating containment atmosphere flow dynamics, thus the transferability of the various available helium test data to hydrogen distribution problems was confirmed. In parallel with the experimental programme, benchmark simulations performed with various LP and CFD-codes exhibited a major progress in analytical modelling of stratification and mixing processes.

In 2007, the OECD/NEA sponsored SETH-2 Project [1.46] was established and lasted 4 years to investigate the break-up of hydrogen stratification in a LWR containment caused by heat and mass sources and sinks, as well as by the effect of the activation of a safety system (e.g., spray, heat source simulating PAR operation, containment cooler). The experimental investigations were carried out in the multi-compartment, large-scale PANDA (PSI, Switzerland) and MISTRA facilities, which are equipped with CFD-grade instrumentation. The effect on the hydrogen concentration of the sudden opening of a connection separating the two containment compartments was also investigated. The accompanying analytical activities performed with various computational tools during the project have already proven the relevance of the experimental database for the assessment of advanced LP and CFD code capabilities.

In 2010, the European Union and Russian State Atomic Energy Corporation co-sponsored the 4-year *ERCOSAM-SAMARA* [1.47] projects to investigate the hydrogen concentration build-up and stratification under typical postulated accident scenarios and the effectiveness of accident management systems (i.e., local air cooler, spray, and PAR) in destabilizing a stratified hydrogen layer. The experimental investigations were carried out in the PANDA, MISTRA, TOSQAN and SPOT (JSC, Russia) facilities. The use of different facilities allows insight on scaling effects and to draw general conclusions, not specific to a particular containment design. In parallel to the experimental programme, planning, pre- and post-test calculations are being performed by the member countries to assess the experimental conditions, evaluate the test results, and verify codes/models currently used in reactor safety analysis.

In 2013, OECD/NEA initiated another 4 year project on *Hydrogen Mitigation Experiments for Reactor Safety* (HYMERES) [1.48] to improve the understanding of the hydrogen risk phenomenology inside the containment in order to enhance its modelling in support of safety assessment. It was aimed to address:

- realistic flow conditions (single-phase jets impinging on containment structures, two-phase jets in containment) to provide crucial information in the evaluation of the basic computational and modelling requirements (mesh size, turbulent models, etc.) needed to analyse a real nuclear plant.
- interaction of safety components to study different combinations of “safety elements”, e.g., the thermal effects created by two PARs, spray and cooler, spray, and operating simultaneously to open connections between compartments.
- system behaviour (e.g., opening of connections between compartments) to examine hydrogen concentration build-up inside the containment.

The experiments will be performed in the PANDA and MISTRA facilities. Analytical activities are also expected to accompany the experimental programmes.

1.3.2.3 Hydrogen Combustion

In March 2000, OECD/NEA published a report on *Carbon Monoxide - Hydrogen Combustion Characteristics in Severe Accident Containment Conditions* [1.49]. It reviewed the knowledge base on CO/H_2 combustion from the perspective of assessing the potential combustion threat inside the containment during a SA. Extensive studies have been carried out to address concerns on hydrogen in a SA, but the contribution of carbon monoxide to the combustion threat has received less attention. Assessment of scenarios involving ex-vessel interactions require additional attention to the potential contribution of carbon monoxide to combustion loads on the containment, as well as the effectiveness of mitigation measures designed for hydrogen to effectively deal with particular aspects of carbon monoxide.

In August 2000, OECD/NEA published a SOAR on *Flame Acceleration and Deflagration-to-Detonation Transition* [1.50]. The report addressed the key issue of hydrogen combustion that may occur as a result of SA in NPPs, namely to predict the combustion modes, i.e., a slow flame, a fast turbulent flame, or a detonation. It was pointed out that despite the significant advances in understanding of FA and DDT that had been obtained between 1990 and1999, the knowledge of FA and DDT is largely empirical and numerical simulation is unable to provide a truly predictive capability for DDT. Situations sometimes arise in which the possibility of FA and DDT cannot be ruled out, but there are no means to predict whether these will actually occur.

In the frame of the SARNET project [1.41], hydrogen combustion and the associated risk mitigation were studied, concentrating on the formation of combustible gas mixtures, local gas composition and potential combustion modes: slow deflagrations, fast accelerating flames, DDT and detonation. The influence of hydrogen gradients on FA and deceleration was studied in the ENACCEF facility (IRSN, France) [1.51]. The experimental results had been used in a benchmark using different CFD codes. In the SARNET2 project [1.41], hydrogen combustion was also one of the topics, including a benchmark for FA and DDT as well as hydrogen combustion in the course of Direct Containment Heating (DCH).

In the frame of the OECD/NEA THAI project [1.45], hydrogen deflagration tests were performed in the THAI facility to quantify the influence of initial pressure, initial temperature, steam content, burn direction and spatial gas distribution on pressure build-up, temperature development, flame front propagation and completeness of combustion. In the follow-up THAI2 project started in 2010 [1.52], the hydrogen combustion behaviour during spray operation is being investigated. The data of selected tests have been used for open and blind post-test calculations for the validation of combustion models.

In 2010, OECD/NEA initiated a benchmark activity, ISP-49, on hydrogen combustion [1.53]. It was aimed to identify the contemporary level of the numerical tools in the area of hydrogen combustion under the conditions typical for safety considerations for NPPs. It was intended to rank the existing models, identify knowledge gaps of the current expertise, and to lead to significant improvements of the quality and adequacy of the numerical prediction, thus facilitating accident management procedures. The experiments selected for the simulation were performed in the THAI and ENACCEF facilities, and characterised by completely different geometrical and initial mixture composition conditions and hence completely different deflagration features.

1.3.2.4 Hydrogen Mitigation

In 2001, IAEA published a report on *Mitigation of Hydrogen Hazards in Water Cooled Power Plants* [1.54]. The report summarized concepts for hydrogen mitigation in containments, concentrating primarily on measures that were already being implemented or those that show promise in the near future for hydrogen mitigation in SAs. It was concluded that the nature of the hydrogen threat to containment and the choice of measures to mitigate the hydrogen threat depend strongly on the containment design. There is no single strategy or technique that is universally appropriate for all designs and accident scenarios, or even, for all phases of an accident in a particular design. Different measures may be more appropriate at different locations and at different times during an accident. A completed safety assessment for the particular plant is the only valid context for judging the adequacy of safety systems and accident management measures, including hydrogen countermeasures.

In 2002, OECD/NEA issued a report on *Implementation of Severe accident management measures – Summary and conclusions* [1.55]. It showed that SAM had been implemented in various ways in many plants, but not yet in all plants. The approaches followed in the different countries did not fit into one single pattern. Harmonisation did not seem feasible at that stage. It appeared that this difference in approach did not necessarily result in differences in safety levels, but reflected differences in regulatory requirements in different countries, and/or differences in the options adopted by utilities. The observations regarding hydrogen are summarized as follows:

- The extent of implementation of hydrogen mitigation measures, mainly PARs, had significantly increased in many countries, in particular in some Western European countries.
- Ignition under conditions of PAR overload continued to be an area for research. There was consensus that PARs are only one of the possible sources of ignition. However, PARs do not add a new threat of initiating combustion. Rather, the reduction of hydrogen concentration by PAR action is safety oriented with respect to possible combustion threats.
- Damage to equipment and power cables under hydrogen burn conditions was raised as an issue. The magnitude and duration of thermal loads from combustion are expected to be within existing qualification for most equipment.

- Carbon monoxide from ex-vessel chemical reactions can be a significant addition to energy available for release by combustion. It was generally agreed that PARs can effectively remove CO at the same time as H_2.

In the frame of the OECD/NEA THAI project [1.45], as well as the ongoing OECD/NEA THAI-2 project, significant progress has been achieved in the knowledge on the behaviour of different PAR designs under SA typical conditions (onset of recombination, recombination rate under different conditions and ignition potential). Oxygen starvation significantly reduces the recombination rate. A highly important result is that PAR ignition potential is limited to a relatively small area of mixture compositions in the air-hydrogen steam ternary diagram capable to provide the high catalyst temperatures required for ignition. In the follow-up THAI-2 project [1.52], PAR behaviour under extremely low oxygen content is being examined.

In the frame of the SARNET project [1.41], the studies on hydrogen also included the reaction kinetics inside PARs. The small-scale REKO-3test facility (FZJ, Germany), allows the investigation of catalyst samples inside a vertical flow channel under well- defined conditions such as gas mixture, flow rate and inlet temperature [1.56]. It was important for model development, as separate effects could be studied. The results have been used for the validation of numerical codes. Recently, the parallel recombination of hydrogen and carbon monoxide has been investigated [1.57].

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2 CONTAINMENT DESIGN FEATURES

This chapter provides an overview of various containment/reactor building designs, including Western Pressurized-Water Reactors (PWRs), Eastern European Vodo-Vodyanoi Energy Reactors (VVERs), Boiling-Water Reactors (BWRs) and Pressurized Heavy-Water Reactors (PHWRs) that are currently under operation or construction, or new designs. It is known that hydrogen management cannot be separated from the reactor containment strength and hydrogen distribution can be strongly affected by containment layouts [1.32]. Therefore, it is necessary to realize the difference in various containment designs, thus having a better understanding of hydrogen mitigation measures and hydrogen management strategies.

It must be noted that the description may not cover all aspects of a reactor and the report does not contain all NPP designs worldwide. The main focus is to present the following characteristics that have a close link to the hydrogen generation, transport, and management during an accident with severe core damage:

- Reactor characteristics (e.g., type, power, number of loops)
- Containment design features (e.g., containment type, volume, design pressure, reactor building, and spent fuel pool location)
- Containment and/or mitigation systems (e.g., spray, fan cooler, blow-out panels, latches, hydrogen-mitigation systems, and venting system).

In this report, **containment** is referred to as the structure that provides a confinement of radioactive material in a NPP, including the control of discharges and the minimization of releases. It is an engineered barrier supporting fundamental safety function of confinement of radioactive material to be ensured during normal operation, anticipated operational occurrences, DBAs, and, to the extent practicable, in selected Beyond Design Basis Accidents (BDBAs) [2.1]. The containment provides radiation shielding under normal operation and accident conditions, and in some NPP design, it provides in addition the protection of the reactor against external events. The containment structure and its associated systems with the functions of isolation, energy management, and control of radionuclides and combustible gases are referred to as the containment systems. In accordance with the concept of defence-in-depth, this fundamental safety function is achieved by means of several barriers and levels of defence [2.2].

The **reactor building**, which fully surrounds the containment and is often called secondary confinement, is to capture leak(s) from the containment. The reactor building usually consists of a reinforced concrete structure. The concrete of the reactor building surrounding the containment also serves as protection against external events. The reactor building is held typically at a slightly negative gauge pressure in both operational states and accident conditions. In the event of an accident, leaks from the containment into the reactor building can be extracted and filtered by an air removal system to permit the use of controlled emission from the plant stack [2.2].

The general containment/reactor building design features of the Western PWRs, Eastern European VVERs, BWRs and PHWRs (CANDUs) are summarized in the following sections. Detailed descriptions of various NPPs are provided in Appendix A. The details in national requirements and status of hydrogen mitigation measures, SAM considerations in using the engineering systems (e.g., spray, local air cooler, containment venting system, etc.) will be presented in Chapter 3.

2.1 PWRs

Thirteen types of Western PWRs are presented in Appendix A.1. The descriptions are provided by the participants from Belgium, Finland, France, Germany, Japan, Korea, the Netherlands, Spain, Sweden, Switzerland and USA.

The size of Western PWRs covers ranges from two-loop 365 MWe to large four-loop 1650 MWe units. The key design characteristics and systems that are related to hydrogen management are summarized in Table 2-1. Despite the differences in their fission-product and core-material inventories; there are no substantial differences in the basic nuclear and thermal-hydraulic parameters of the reactor circuit, but the design of the containment is significantly different. Three types of PWR containments exist:

- steel containment within a surrounding concrete reactor building,
- pre-stressed concrete containment with an inner steel liner, where the upper part is not surrounded by a reactor building, and
- concrete double containment.

For the Framatome and Westinghouse PWRs, spray systems are typically installed to limit the containment pressure in DBAs, but no spray systems are needed nor installed in the German PWR containments because they are full pressure containments. The long term heat removal from the PWR containment after DBAs is provided by different systems, sometimes including air ventilation systems. Measures to prevent hydrogen combustion during DBAs inside the containment are installed often with additional systems to be used under SA conditions, such as thermal recombiners for DBA conditions and PARs or glow plug igniters for SA conditions. To prevent a long term containment over-pressure failure under SA conditions, many plants are back fitted with filtered containment venting (FCV) systems as part of their accident management concepts.

The current PWRs with large dry containments in the US do not use PARs nor igniters as part of their hydrogen control measures, whereas the PWRs with ice condenser pressure suppression inside containment do utilize hydrogen igniters.

2.2 VVERs

Six types of VVERs are presented in Appendix A.2, including three types of operating NPPs and three types of next generation ones. The descriptions of operating VVER-440 and VVER-1000 are extracted from References [2.3] and [2.4].

For VVER-reactors, a distinction has to be made between the old VVER-440/W230 units on one side and the modern VVER-440/V-213 and VVER-1000/V-320 reactors as well as the next generation of VVER-1000 or VVER-1200 on the other side. Data provided in Table 2-2 are only for the most popular operating VVER units, with a size ranging from six loop plants with ~500 MWe to large four loop plants with more than 1000 MWe.

There are substantial differences in the basic nuclear and thermal-hydraulic design of the reactor circuit and the containment of these two general plant types. The VVER-440/W-213 standard containment design consists of a reinforced concrete accident localization structure (hermetic rooms) connected to a pressure suppression system (bubble condenser tower with air traps) and has a spray system, providing the active pressure suppression function and the radioactivity removal function. The modified VVER-440 operated in Finland has a steel containment with external and internal spray systems (note the external spray system was back-fitted as a SAM system in 1990) and two ice-condensers in the containment connecting the lower and upper compartments. Many VVER-440 units use the RPV external cooling as an accident management measure to prevent its failure in a SA. As well PARs are installed to prevent hydrogen combustions during SAs, while glow-plug igniters are installed in addition in the Finnish VVER NPP containments in the lower compartment of the containment (steam generator room) to mitigate rapid hydrogen generation. The older VVER-440/W-230 reactors (not described in the report in detail) had no pressure suppression system, while the reactor system was confined by a reinforced concrete accident localization structure (hermetic rooms) with flaps opening into the environment during DBA. Most of the units are already permanently shut down or different upgrades for the confinement have been made. During the design phases of the Russian VVER-1000 NPPs, great scientific efforts were made to meet the optimal core configuration based on the small series of V-187, V-302 and V-338 models, and the final large series of V-320. The containment design of all these series was similar and consists of a pre-stressed reinforced concrete containment construction with an internal steel liner. A spray system is used to limit the pressure build-up during DBA. PARs are as well under discussion to be implemented into the containment as SA measure to prevent hydrogen combustions and FCV systems are implemented also in some plants. The future developments comprise of VVER-1000 (AES-91 and AES-92 project) design with a larger number of passive systems, a core catcher and hydrogen mitigation features. The later development of VVER-1200 (AES-2006) has a larger reactor power and additional safety features. Details of these new designs are not provided in the overview tables.

2.3 BWRs

Ten types of BWRs are presented in Appendix A.3. The descriptions are provided by the participants from Finland, Germany, Japan, Spain, Switzerland, and US.

The size of BWRs and the construction of the containments vary significantly. Typical NPP designs are summarized in Table 2-3. The earlier designs had less reactor power, in the order of several 100 MWe, while the latest ones have similar power output as large PWRs. The containment design consists of steel containments within a concrete reactor building or a pre-stressed concrete containment with inner steel liner often surrounded by a reactor building. The containment is always separated into a drywell containing the reactor and a wetwell (passive heat sink) with a water volume of ~2000 – 3500 m³.

The well-known BWR containments are of the Mark I, II and III types, which are very different especially related to the wetwell design. In Germany, only plants of the so-called BWR type 72 (pre-stressed concrete containment with inner steel liner surrounded by a concrete reactor building) are still in operation. The older BWR type 69 plants with a steel containment were already shut-down in August 2011 based on a German government decision after the Fukushima accident.

In most BWR containments, spray systems are installed at least in the drywell, but not all of them are safety related. Most of the BWR containments – drywell and wetwell – are inerted by nitrogen during normal operation to prevent hydrogen combustion during DBAs and SAs. If not, like typically for Mark III containments, igniters are installed. Another exception is the German BWR type 72 units, where only the wetwell is inerted by nitrogen and PARs are installed in both the drywell and wetwell to prevent hydrogen combustion. To prevent a long term containment over-pressure failure in case of a BDBA or SA, filtered or unfiltered systems are backfitted often as part of accident management concepts. The systems are connected typically to the gas phase of the wetwell and also to the drywell of the containment in many plants and have a separate off-gas pipe towards the stack or the environment. Flooding the cavity below the RPV is another measure, foreseen to prevent concrete erosion after reactor pressure vessel failure in a SA.

2.4 PHWRs – CANDUs

Three types of PHWRs are presented in Appendix A.4, including two types (single and multi-units) of operating CANada Deuterium Uranium (CANDU) NPPs and one type of an enhanced CANDU model (Generation III). The descriptions are provided by the participants from Canada and Korea.

All CANDU reactors (see Table 2-4) follow the same basic design (i.e., natural uranium with heavy water as coolant), although variations can be found in most units. The CANDU pressure tubes, holding fuel bundles immersed in high pressure and high temperature heavy water coolant, act like the reactor pressure vessel for LWR. They are located inside a calandria vessel filled with low pressure (near atmospheric) and low temperature heavy water moderator. The calandria vessel is located inside a reactor vault built of concrete and filled with light water, which functions as a biological shield under normal operating conditions and a passive heat sink under certain SA scenarios. Ex-vessel cooling, considered to be an important accident management programme in PWRs, is inherently in the CANDU design.

There are basically two types of CANDU reactors; single and multi-unit stations. The multi-unit stations share a single containment system consisting typically of four units at each station with power output ranging from 540 to 940 MWe per unit, while the single units have a stand-alone containment with a power output in the 700 MWe range. Most of the single units are of CANDU 6 design. For CANDU6, the containment and reactor building are a single combined structure and the two terms are interchangeable.

For the multi-unit stations, each unit has a rectangular or cylindrical reinforced reactor building, which serves as a support and an enclosure for the reactor and some of its associated equipment. The portion of the reactor building, forming part of the containment envelope, is called the reactor vault. The reactor vaults of all the units are connected with a common vacuum building through a fuelling duct and pressure relief duct. The vacuum building is equipped with an emergency filtered atmospheric discharge system (EFADS) to vent the containment atmosphere following an accident (maintaining the vacuum building sub-atmospheric).

The spray system and air coolers exist in the containment of the CANDU 6 single units. The multi-unit stations have the spray system in the vacuum building and air coolers in the reactor vault. PARs and/or igniters have been installed in most of the CANDU reactors.

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Table 2-1 Summary of NPP Design Information – PWRs

NPP type	German PWR KONVOY	Spanish PWR 1000 (Trillo)	Swiss PWR 1000 (Gösgen)	Netherlands PWR 500 (Borssele)	French PWR 900	French PWR 1300
Power [MWe]	1345 – 1485	1066	1035	512	900	1300
No. of loops	4	3	3	2	3	4
Containment						
Type	Steel containment	Steel containment	Steel containment	Steel containment	Single pre-stressed concrete wall with inner steel liner	Double concrete wall; inner wall pre-stressed
Volume [m³]	~70000	~61100	~55200	~36000	~48000	~70000
Design Pre. [bar (a)]	~6.3	6.38	5.89	4.8	5.0	4.8 – 5.2
Reactor building (*)	Reinforced concrete	Reinforced concrete	Reinforced concrete	Reinforced concrete	no	no (double concrete containment)
SFP location	Inside containment	Inside containment	Inside containment and in external spent fuel storage building	Inside containment	BK building outside containment	BK building outside containment
Containment systems	H_2 mixing system with thermal recombiners, air coolers, burst elements between rooms	H_2 mixing system incl. thermal recombiners, burst elements between rooms	H_2 mixing system incl. thermal recombiners, air coolers, burst elements between rooms	Spray system, air coolers, filtered recirculation system, burst elements between rooms	Spray	Spray
SA Mitigation Systems						
Hydrogen mitigation	PARs	PARs	Ignition by thermal recombiners (SAMG)	PARs	PARs	PARs
Filtered venting system	yes	No, but required after Fukushima	yes	yes	yes	yes

(*) Reactor building is referred to as the building that fully surrounds the containment with the reactor and provides protection against external events.

Table 2-1 Summary of NPP Design Information – PWRs (cont.)

NPP type	French PWR 1450	French EPR 1650	Finnish EPR 1600	USA AP1000	Swiss Westinghouse PWR 365	Spanish Westinghouse PWR 1000
Power [MWe]	1450	1650	1600	~1000	365	~1000
No. of loops	4	4	4	2	2	3
Containment						
Type	Double concrete wall; inner wall pre-stressed	Double concrete wall with inner steel liner	Double concrete wall with inner steel liner	Steel containment	Reinforced concrete with inner steel liner	Reinforced concrete with inner steel liner
Volume [m³]	~71000	~80000	~80000	58333	36380	56900 to 62100
Design pressure [bar (a)]	5.3	5.3	5.3	5.08	4.1	3.78 to 4.7
Reactor building *)	no (double concrete containment)	no (double concrete containment)	no (double concrete containment)	steel/concrete composite, reinforced concrete	Reinforced concrete	Reinforced concrete
SFP location	BK building outside containment	Fuel building outside containment	Fuel building outside containment	Auxiliary building	Separate building outside containment	Separate building outside containment
Containment systems	Spray	Spray, melt spreading, IRWST	Spray, melt spreading, IRWST	Passive containment cooling	Spray and fan cooler	Spray and fan cooler
SA mitigation systems						
Hydrogen mitigation	PARs and convect#) system which allows atmosphere homogenization	PARs and convect system which allows atmosphere homogenization	PARs and convect system which allows atmosphere homogenization	Igniters and PARs	PARs	No, but PARs required after Fukushima
Filtered venting system	yes	no	yes	no	yes	No, but required after Fukushima

*) Reactor building is referred to as the building that fully surrounds the containment with the reactor and provides protection against external events.

#) The convect system opens panels in some areas to enhance mixing.

Table 2-1 Summary of NPP Design Information – PWRs (cont.)

NPP type	**APR1400 in Korea**	**OPR1000 in Korea**	**Japanese Mitsubishi PWR**	**Japanese Mitsubishi PWR**	**Japanese Mitsubishi PWR**
Power [MWe]	1400	1000	570	900	~1100
No. of loops	2	2	2	3	4
Containment					
Type	Pre-stressed concrete with inner steel liner	Pre-stressed concrete with inner steel liner	Steel containment	Steel containment	Pre-stressed concrete with inner steel liner
Volume [m³]	90000	77220	42400	67900	73700
Design pressure [bar (a)]	5.1	4.8	3.6	3.6	5.0
Reactor building *)	no	no	Reinforced concrete (outer shielding building) Annulus formed to wrap the entire containment		Annulus formed to wrap the lower part of containment
SFP location	SFP building	SFP building	Separate building outside containment	Separate building outside containment	Separate building outside containment
Containment systems	IRWST, spray,	Spray, fan cooler	Spray, annulus venting	Spray, annulus venting	Spray, annulus venting
SA Mitigation Systems					
Hydrogen mitigation	PARs and glow plug igniters	PARs and glow plug igniters	Prescribed as the basic requirement after Fukushima; PARs, igniters and hydrogen monitoring referred as mitigation measures		
Filtered venting system	no	no	no	no	no

*) Reactor building is referred to as the building that fully surrounds the containment with the reactor and provides protection against external events

Table 2-1 Summary of NPP Design Information – PWRs (cont.)

NPP type	USA PWR "Large Dry"	USA "Ice Condenser"	Belgium PWR 440	Belgium PWR 1000	Swedish PWR Westinghouse
Power [MWe]	~500 - 1300	~1100	440	960-1040	804 (R2), 1050 (R3), 935 (R4)
No. of loops	2 - 4	4	2	3	3
Containment					
Type	Steel containment/ Reinforced concrete with inner steel liner	Steel containment/ Reinforced concrete with inner steel liner	Steel containment	Pre-stressed concrete containment with inner steel liner	Single pre-stressed concrete wall with inner steel liner
Volume [m³]	~68000	~34000	42830	~60500 or ~70000	(R2) ~51000, (R3) ~58000, (R4) ~59000
Design pressure [bar (a)]	~4 - 5	~2	3.86	4.1 – 4.7	5.14
Reactor building *)	Reinforced concrete	Reinforced concrete	Reinforced concrete	Reinforced concrete	no
SFP location	Separate building outside containment	Separate building outside containment	Separate building outside containment	Separate building outside containment	Separate building outside containment
Containment systems	Spray/fan coolers	Spray/air return fan	Spray and fan cooler	Spray and fan cooler	Spray and fan cooler
SA Mitigation Systems					
Hydrogen mitigation	Thermal Recombiners	Igniters	PARs	PARs	PARs
Filtered venting system	no	no	no	No, but required after Fukushima	yes

*) Reactor building is referred to as the building that fully surrounds the containment with the reactor and provides protection against external events

Table 2-2 Summary of NPP Design Information – VVERs

NPP type	VVER-440/W-213 Standard	VVER-440 Loviisa	VVER-1000/V-320
Power [MWe]	500	496	~1000
No. of loops	6	6	4
Containment			
Type	Concrete containment with bubble condenser tower	Steel containment with ice condensers	Pre-stressed concrete with steel liner
Volume [m³]	55000	58000	~61000
Design pressure [bar (a)]	~2.5	~1.7	5.1
Reactor building *)	Reinforced concrete (partly)	Reinforced concrete	no
SFP location	Reactor building	Inside containment	Inside containment
Containment systems	Spray	Spray (inner and outer**), ice condenser	Spray
SA Mitigation Systems			
Hydrogen mitigation	PARs	PARs and glow plug igniters	PARs under discussion
Filtered venting system	no	no	Yes in some NPPs

*) Reactor building is defined as the building the fully surrounds the containment with the reactor and provides protection against external events

**) The external spray system of the VVER-440 in Finland was back-fitted as a SAM system in 1990.

Table 2-3 Summary of NPP Design Information – BWRs

<table>
<tr><th>BWR type</th><th>German BWR of Type 72</th><th>Spanish BWR Mark I</th><th>Japan BWR Mark I</th><th>Japan BWR Mark I</th><th>Swiss BWR Mark I</th><th>Japan BWR Mark II</th></tr>
<tr><td>Power [MWe]</td><td>1344</td><td>460</td><td>460</td><td>784</td><td>373</td><td>1100</td></tr>
<tr><td>Reactor</td><td>RPV with internal pumps</td><td>BWR-3</td><td>BWR-3</td><td>BWR-4</td><td>BWR-4</td><td>BWR-5</td></tr>
<tr><td colspan="7">Containment</td></tr>
<tr><td>Type</td><td>Pre-stressed concrete with inner steel liner</td><td>Steel containment Mark I</td><td>Steel containment Mark I</td><td>Steel containment Mark I</td><td>Steel containment Mark I</td><td>Steel containment Mark II</td></tr>
<tr><td>Total vol. of containment [m³]</td><td>17800</td><td>7056</td><td>7780</td><td>10380</td><td>7100</td><td>13200</td></tr>
<tr><td>Water vol. in wetwell [m³]</td><td>3000</td><td>1856</td><td>1750</td><td>2980</td><td>2100</td><td>3400</td></tr>
<tr><td>Design P [bar (a)]</td><td>4.3</td><td>5.3</td><td>5.4</td><td>4.9</td><td>5.8</td><td>4.2</td></tr>
<tr><td>Reactor building *)</td><td>Reinforced concrete</td><td>Reinforced concrete (partly)</td><td>Reinforced concrete (partly)</td><td>Reinforced concrete (partly)</td><td>Reinforced concrete with outer torus</td><td>Reinforced concrete (partly)</td></tr>
<tr><td>SFP location</td><td>Reactor building</td><td>Reactor building</td><td>Reactor building</td><td>Reactor building</td><td>Reactor building</td><td>Reactor building</td></tr>
<tr><td>Containment systems</td><td>Spray (not safety relevant)</td><td>Spray</td><td>Spray</td><td>Spray</td><td>Spray/Flood system</td><td>Spray</td></tr>
<tr><td colspan="7">SA mitigation systems</td></tr>
<tr><td>Hydrogen mitigation</td><td>N_2-inerting of WW; PARs in DW and WW</td><td>N_2-inerting of DW and WW</td><td colspan="2">N_2-inerting of DW & WW; thermal recombiners for DBA</td><td>N_2-inerting of DW & WW, thermal recombiners (DBA)</td><td>N_2-inerting of DW & WW & thermal recombiners (DBA), PARs planned for reactor building</td></tr>
<tr><td>Venting system</td><td>FCV connected to WW</td><td>Hardened vent system connected to DW and WW, FCV required after Fukushima</td><td colspan="2">Hardened vent system connected to DW and WW FCV planned to be installed</td><td>FCV connected to DW & WW through outer torus water pool</td><td>Hardened vent system connected to DW and WW; FCV to be installed</td></tr>
</table>

*) Reactor building is defined as the building the fully surrounds the containment with the reactor and provides protection against external events; DW-drywell, WW- wetwell

Table 2-3 Summary of NPP Design Information – BWRs (cont.)

BWR type	**US BWR Mark III**	**Spanish BWR Mark III**	**Swiss BWR Mark III**	**Japan ABWR**	**Finnish Asea Atom BWR**	**Swedish BWR 75**
Power [MWe]	~1300	1092	1275	1356	880	1180
Reactor	BWR-6	BWR-6	BWR-6	ABWR	RPV with internal pumps	RPV with internal pumps
Containment						
Type	Mark III steel or concrete with a steel liner containment surrounding WW and DW	Mark III steel containment surrounding WW and DW	Mark III steel containment surrounding WW and DW	Reinforced concrete containment with steel liner	Pre-stressed concrete with a steel liner embedded	Pre-stressed concrete with a steel liner embedded
Total vol. of containment [m³]	~35000	~39000	~36000	16000	10000	11700
Water vol. in WW [m³]	~3700	3650	3769	3600	2700	3166
Design Pre. [bar (a)]	2.0	2.05	2.034	4.2	4.7	6.0
Reactor building *)	Reinforced concrete	Reinforced concrete	Reinforced concrete	Reinforced concrete (partly)	Reinforced concrete	Reinforced concrete
SFP location	Separate Building	Separate Building	Separate Building	Reactor building	Reactor building	Reactor building
Containment systems	Spray/fan cooler	Spray	Fan coolers	Spray	Spray	Spray
SA mitigation systems						
Hydrogen mitigation	Glow plug igniters	Glow plug igniters	Glow plug igniters	N_2-inerting of DW and WW, thermal recombiners (DBA), PARs planned for reactor building	N_2-inerting of DW and WW	N_2-inerting of DW and WW
Venting system	Limited capability	Hardened vent system connected to WW, FCV required after Fukushima	FCV	Hardened vent system connected to DW and WW; FCV to be installed	FCV from WW (preferred) or DW	FCV from DW

*) Reactor building is defined as the building the fully surrounds the containment with the reactor and provides protection against external events

Table 2-4 Summary of NPP Design Information – PHWR CANDUs

PHWR type	CANDU 6 single unit (Point Lepreau, Canada)	CANDU multiple unit (Darlington, Canada)	CANDU 6 single unit (Korea)
Power [MWe]	700	935 per unit	700
No. of units	1	4	1
No. of loops	2	2 per unit	2
Containment			
Type	Pre-stressed concrete with polymer coating	Pre-stressed concrete with inner steel liner	Pre-stressed concrete with inner epoxy liner
Free volume of reactor building*) [m³]	~48000	13000 reactor vault 80000 vacuum building 35000 fuelling duct	~48000
Design pressure [bar (g)]	1.24**	Positive: 0.965 Negative: 0.531	1.24
SFP location	Outside containment	Outside containment	Outside containment
Containment systems	Dousing, air coolers	Dousing, air coolers, vacuum building	Dousing, air coolers
SA mitigation systems			
Hydrogen mitigation	PARs	PARs and igniters	PARs for Wolsung unit 1, PARs and igniters for Wolsung units 2, 3 and 4
Filtered venting system	Yes	Yes (EFADS, designed for DBAs, FCV to be installed during refurbishment outage),	Yes for Wolsung unit 1; to be installed for Wolsung units 2, 3 and 4

*) For CANDU single unit NPPs, the containment and reactor building are a single combined structure. For multi-unit CANDUs, the reactor building includes the reactor vault of a specific unit, the common vacuum building and the fuelling duct. These three structures form the reactor containment.

**) The design pressure of the CANDU 6 containment is 1.24 bar (g), but the actual failure pressure can be much higher, e.g., failure of airlock seals occurred until ~235 kPa (g) according to a previous station test.

3 HYDROGEN MANAGEMENT

Hydrogen mitigation measures, as part of the SAM in NPPs, have been implemented in many of the countries in the world. The goal is to prevent mechanical and thermal loads, resulting from hydrogen combustion, which could threaten containment integrity. This chapter will cover the following topics:

- Description of the national requirements on hydrogen management for SAs inside and outside the containment and the hydrogen mitigation measures implemented (or to be considered) in the operating NPPs of the participating countries.
- Considerations of the use of engineering systems (i.e., spray, containment venting, local air cooler, suppression pool, latch systems) in SAM or of other events (i.e., gas transport within containment compartments and containment leakages) that can affect the hydrogen distribution in the containment buildings surrounding the SFP and the build-up of combustible mixtures.
- Hydrogen measurement strategies for SA conditions adopted or to be considered in the participating countries.
- Lessons learned from the Fukushima accident in terms of hydrogen management in SAs in general.
- Hydrogen assessment methodology and the PSA level 2 assessments of hydrogen mitigation measures.

3.1 Hydrogen Mitigation Measures

For the containment, and the reactor and auxiliary buildings, implementation of hydrogen mitigation measures is aimed in general to prevent and limit hydrogen explosion consequences. Therefore, depending on the NPP type, hydrogen mitigation measures are designed to meet specific safety criteria and requirements. In addition to mitigation measures, gas composition monitoring system is often used to check if the requirements are satisfied and to provide relevant information to NPP operators during accident and SA conditions.

While the deliberate ignition concept was explored especially in the USA after the TMI-2 event, leading to installation of glow plug igniters (active systems) especially into Mark III BWR containments (see Table 2-3) or the US PWR (see Table 2-1) and Finnish VVER-440 with ice condenser containments (see Table 2-2), effort was undertaken in Germany, France and Canada to find an alternative passive device. This led to the design of PARs by different manufacturers, especially Siemens/AREVA, NIS (Germany) and AECL (Canada).

As discussed in Reference [1.6], igniter technology is established as a method of preventing damaging burns by ensuring ignition near the limits of flammability. Igniters can deal with higher hydrogen flow rates than PARs but they have to be appropriately located, and especially a glow plug system needs external power. Therefore, reliable power supply and igniter placement are important for the effective reduction of hydrogen concentration. Implementation of igniters requires study of gas flow patterns in representative scenarios to optimize location selection. Igniters and PARs are not mutually exclusive solutions; they can be used in some combinations for a better hydrogen control.

Catalytic recombiners use platinum- and/or palladium-based catalysts to oxidize (recombine) hydrogen, which usually accomplish the chemical reaction at lower temperatures, in a wider hydrogen/oxygen concentration range, and even under steam-inerted conditions and over longer times than it would be with gas combustion. PARs are available in two general different designs. They vary in the shape of the reacting surfaces (flat plate or pellets contained in cartridges) and the layout of the channel box enhancing the flow natural flow conditions. The working principle of these PARs is almost identical, but they vary in size and shape, thus recombination rate, but the appropriate PAR size can be determine based on the containment volume and size.

There are two broad categories of catalytic recombiners:

- Conventional catalytic recombiners designed for DBAs, functioning in essentially the same way as thermal recombiners, operating mostly external to containment, delivering the containment atmosphere to heated catalyst with the use of powered gas pumps,
- Passive autocatalytic recombiners, which are situated inside the containment and use the heat of the oxidation reaction to produce flow through the unit by natural convection. As a consequence of their passive self-start and self-generated flows, they do not require external power or operation actions.

PARs are in line with the general trend toward passive safety devices as key elements in NPPs; hence PARs are nowadays the first choice of mitigation measure for hydrogen management in most countries, but still not applicable for all containment designs. Several criteria must be fulfilled (see Chapter 3.1.1.1 for national requirements). It should be noted that PARs are ultimately subject to mass transfer limitations and may not be capable of removing hydrogen at rates required under fast-developing conditions, for example, in the vicinity of the hydrogen release.

PARs are commercially available from vendors in Canada (AECL designed and provided by Candu Energy Inc.) and Europe (AREVA and NIS). Additional systems are under development in Russia and Korea.

The three main vendors undertook extensive testing of their PARs in different experimental facilities in the 80’s and 90’s. The latest experimental tests have been performed in the frame of OECD/NEA THAI project [1.45], where three smaller PAR units provided by AECL, AREVA, and NIS were tested under comparable experimental conditions. The PAR performance tests provided new data (onset of recombination threshold, recombination rate under various conditions, and PAR induced ignition) used for code model development and improvement. The PAR behaviour varies slightly, but the general trend was similar, except the question of PAR induced ignition, where a difference was found between plate type and pellet type PARs. In addition to extensive qualification testing performed by the PAR vendors in the 80’s and 90’s, research on PAR performance under SA conditions, such as iodine poisoning and catalyst plates covered by solid aerosols or droplets [1.45], extremely low oxygen [1.52], H_2-CO recombination [1.57], has been carried by several R&D organizations recently.

3.1.1 Inside the Containment

As discussed in Chapter 2, the choice of mitigation means is primarily dependent on containment designs. For instance, the hydrogen mitigation strategy is mostly a combination of large free volume with the use of PARs and/or igniters for NPPs with large dry containments (i.e., PWRs and PHWRs). For small containments (i.e., BWRs), gas inerting is typically used, but PARs are also used if the dry-well cannot be inerted. For larger BWR containment designs, igniters are used.

3.1.1.1 National Requirements

To prevent the occurrence of an accident or to limit its consequences, nuclear safety authorities define rules and criteria for the implementation of prevention or mitigation means and elaboration of accident management guidelines. Regarding hydrogen mitigation, the choice of strategy is strongly dependent on reactor type, containment design and requirements adopted in each country. The national requirements on hydrogen management inside the containments are described in the following and summarized in Table 3-1.

In **Belgium**, the design requirements for the existing NPPs are documented in the Safety Analysis Report (SAR). For DBA, the hydrogen control requirements are based on the USNRC requirements in [3.1], however without taking into consideration the rule revised in 2003 (i.e., without eliminating the requirements for hydrogen thermal recombiners and purge systems and without relaxing requirements for hydrogen monitoring equipment). For SA, after performing the first PSA Level 2 and analysing the different hydrogen mitigation strategies in the framework of the Periodic Safety Review (PSR) in the early nineties, it was decided in 1993 to implement PARs in all Belgian PWRs. Since 1996-1998, all Belgian units are equipped with PARs.

SAM strategies and SAMG have been developed and implemented for each unit. The SAMG are either based on the generic SAMGs developed by the Westinghouse Owners Group (Tihange), or based on an upgrade and extension of existing SA procedures (Doel).

The Belgian Royal Decree of 30 November 2011 stipulates high-level safety requirements for nuclear power plants, taking into account the WENRA Reference Levels published in 2007. This Royal Decree requires a hydrogen risk management for both DBA and SA conditions.

In **Canada**, the design requirements for a new water-cooler reactor are established in a regulatory document RD-337 [3.2]. In general, the design must provide systems to control the release of hydrogen to prevent deflagration or detonation that could jeopardize the integrity or leak tightness of the containment. It should be noted that the containment and reactor building are a single combined structure for CANDUs with a single unit, and for multi-unit CANDUs, the reactor building includes the reactor vault of a specific unit, the common vacuum building and the fuelling duct, forming the reactor containment. The Canadian Nuclear Safety Commission (CNSC) action items and expectations with respect to the hydrogen issue are not exactly the same for new, refurbished and existing NPPs [3.3]. The expectations for resolution of hydrogen issues for new NPPs are as follows:

- Adopt the safety approach of RD-337 and containment design and analysis expectations, including those applicable to hydrogen assessment and mitigation.
- For DBA, it is expected to consider and assess the potential adverse effects of the bounding hydrogen releases and burns. The designer needs to preclude destructive global/local hydrogen combustion modes, demonstrate the effectiveness of any introduced hydrogen mitigating measures, such as the PARS and establish compliance of the containment post-accident performance with the applicable RD-337 deterministic expectations via appropriate design and analysis rules. Appropriately validated codes and models (e.g., GOTHIC, DDTINDEX, etc.) need to be used.
- For BDBA with limited core damage (such as LOCA + ECCS failures in CANDU), the designer needs to show that there are no issues with the short and long term hydrogen releases and demonstrate the effectiveness of the hydrogen mitigating measures, in precluding destructive potential hydrogen combustion modes. The designer also needs to establish compliance of the containment post-accident performance with the RD-337 containment performance expectations applicable to BDBA, albeit with different design and analysis rules more relaxed than those from DBA. Mechanistic models (e.g., GOTHIC, DDTINDEX, etc.) are expected to be used to assess the hydrogen challenges.
- For SAs, the designer is expected to choose a set of representative scenarios and assess the corresponding flammable gas releases including the hydrogen and CO releases from MCCI. The designer then needs to demonstrate that the introduced hydrogen complementary features such as the PARS provide effective mitigation and that destructive potential combustion modes are avoided. The designer also needs to establish compliance of the containment post-accident performance with applicable RD-337 containment performance expectations for BDBA/SA using appropriate design and analysis rules. Use of integrated models, such as MAAP to assess the hydrogen challenges, is acceptable.
- The designer is expected to show that the containment probabilistic safety goals, such as the large release frequency, are met**,** with the adverse effects of hydrogen combustion considered.
- For defence in depth of the containment design for BDBA/SA, the designer (and the licence applicant) is required to develop a SAM programme in addition to complementary design provisions for hydrogen mitigation. RD-337 requires the installation of a hydrogen monitoring

and sampling system to manage the potential hydrogen challenges to safety and to the containment via SAM [3.4].

It is generally expected that refurbished plants would approach the safety level similar to that of a new NPP. For the operating NPPs that are not planned to be refurbished in the near future (or at all), the industry is expected to conduct a gap assessment against the modern expectations applicable to the containment and hydrogen control systems. The identified gaps then would need to be addressed. It is important to acknowledge that the extent of justified modifications could be different than those for the refurbished plants. Practicability may significantly limit the options available; nevertheless, the effort that went into advancement of the hydrogen behaviour knowledge base and development of PAR does provide acceptable ways to improve safety.

In the **Czech Republic**, there is no specific requirement, regulatory guide or other document issued by Czech Republic State Office for Nuclear Safety. Conditions for a design of the hydrogen removal system under SA conditions were prepared within the projects performed by the UJV Rez a.s. for the NPPs. Generally, definitions of the hydrogen sources into the containment were based on the realistic scenarios selected from the PSA studies (level 1 and 2) and some other assumptions – operation of spray system**,** location of sources inside of the containment, potential absolute mass of hydrogen and melt behaviour in the containment (possibility of corium spreading from cavity to neighbouring room, which influence location of hydrogen source during MCCI phase). The aim is to cover all important variants of scenarios. Criteria for a design of hydrogen removal system are based on evolution of hydrogen concentrations, criteria for FA and DDT, and AICC pressure. PARs are preferred components and igniters could be taken into account too.

In **Finland**, new safety requirements came into force in 2013, but the requirements on hydrogen management have not changed. In fact, the requirements on severe accidents were already in place long before the Fukushima accident. The following quotes are taken from a draft translation of the regulation YVL B.6 [3.5].

- “The containment shall be dimensioned so as to ensure that it retains its leak tightness in a SA even if 100% of the easily oxidising reactor core materials react with water.”
- “The leak tightness of the containment in SAs shall be demonstrated using the containment temperature and pressure obtained from the SA analyses performed in compliance with Guide YVL B.3 by increasing the maximum pressure (gauge pressure) by a 50% safety margin and by pressure increase due to hydrogen burn calculated according to the AICC principle. The 50% safety margin compensates for the uncertainties associated with the calculation methods and selection of calculation cases in SAs.”
- “The containment structure and systems used for managing accidents shall prevent such gas burns, gas explosions or other energetic phenomena that may jeopardise containment leak tightness or the operability of the components needed for accident management.”
- “Combustible gases shall be primarily managed by systems and components that are located inside the containment and do not require an external power supply.”

In **France**, the safety standard requires implementation of a system of “defence in depth” that consists of a series of redundant and diversified measures (automatic mechanisms, systems or procedures) to prevent the occurrence of an accident or to mitigate its consequences. These measures are checked at each stage in the life of the nuclear installations (examination of the safety options, creation authorization, commissioning authorization, etc.) and re-examined systematically during the 10-year safety reviews. This PSR provides the opportunity for an in-depth examination of the condition of the NPPs, to check that they comply with all the safety requirements. An additional aim of the review is to improve the safety of the installations, particularly by comparing the applicable requirements with those applied by the licensee to more recent NPPs. The PSRs therefore constitute one of the keystones of safety in France, by obliging the licensee not only to maintain the level of safety of its NPP but also to improve it.

In the framework of this process, the decision of implementing PARs in French PWRs has been taken in 2004. Due to the standardization of the French NPPs (see Section A.1.5), the adopted hydrogen mitigation has been designed to fulfil the same following requirements:

- For all SA conditions, in case of ignition, the pressure induced by a complete adiabatic and isochoric combustion remains under the design pressure (7 bar for PWR900 , 6 bar for PWR1300 and PWR 1450),
- During the SA transient, the mean hydrogen concentration remains under 8 vol.%. This limit ensures the non-completeness of hydrogen combustion in case of ignition,
- During the SA transient, the local hydrogen concentration must be below the target value of 10 vol.%, which permits to avoid FA phenomena and the possibility of high dynamic pressure loads.

In **Germany**, the legal framework for the peaceful use of nuclear energy in Germany is based on the Atomic Energy Act (Atomgesetz - AtG), which initially entered into force in 1960. With the amendments which took effect on 22 April 2002, however, essential elements of the German nuclear energy law have been reformulated. Now, the purpose of the Act is to regulate the orderly procedure for ending the use of nuclear energy for commercial generation of electricity. The existing nuclear power plants are to be operated at a high level of safety for their remaining service lives.

According to the German Constitution (Grundgesetz – GG), the Länder (regulatory bodies at the level of the federal states) are responsible for the implementation of the Atomic Energy Act. To ensure a uniform implementation of the Atomic Energy Act, the Länder are subject to federal supervision. The provisions of the Atomic Energy Act are supplemented or specified by further laws and ordinances.

It should be noted that the efforts undertaken by the Licensees in the BDBA and SA area related to the implementation of SAM programmes since the late 1980s has been initially on a voluntary basis. The licensees agreed to implement the respective German Reactor Safety Commission (RSK) recommendations. In the context of the now legally required PSR, every ten years the defence in depth and the fundamental safety functions have to be reassessed using current site conditions and impacts conceivable at the plant site (http://www.ensreg.eu/EU-Stress-Tests).

First requirements for a SAM programme regarding BDBA events during power operation were published in autumn 1988 after intense discussions within the RSK [3.6]. The concept was part of the NPP emergency protection and was called "Anlageninterner Notfallschutz" (on-site emergency protection). The primary intention was the prevention of SAs during power operation. Some selected mitigation measures, e.g., FCV and hydrogen counter measures for dominating SA phenomena were proposed as well. For both, necessary hardware modifications have been considered, e.g., the replacement of pressurizer safety valves or the installation of a FCV system. The final RSK recommendation regarding a SAM programme was published in 1992 [3.7] and provided all details for SAM concepts to be developed and implemented by the licensees to deal with SAs starting from full power operation.

For German BWR containments the inerting by nitrogen during normal plant operation was chosen as preferred option, due to the smaller containment volume compared to PWR. This was already recommended by the RSK in 1986 [3.8].

Main requirements have been determined in [3.8] as follows:

- The inerting of the containment must be performed in parallel to the start-up procedures of the NPP.
- The de-inerting of the containment is not allowed to be started earlier than 24 h before the shut-down starts.

- The remaining oxygen content in the containment should prevent combustions under accident conditions. A content of 4 vol.% of O_2 is reasonable.
- The control rod driving room may get de-inerted for inspection if needed.
- The oxygen and hydrogen concentration inside the containment should be continuously monitored.

The BWR type 69 NPPs (all are shut down after the Fukushima accident) have been fully inerted by nitrogen (wetwell and drywell).

The containments of the BWR type 72 NPPs differ considerably from those of the BWR type 69. The licensee of BWR type 72 developed an inerting/recombination concept which took into account the differences of the plant design and considered the RSK recommendations. The concept was separately discussed and approved by the RSK [3.6] and thereafter realized. It consists of an inerted wetwell (as access to the drywell may be needed during plant operation) and in addition a large number of PARs in both the drywell and the wetwell.

In all German PWR containments, PARs are installed to remove the hydrogen generated in a SA. This was recommended by the RSK already in 1997 [3.9] after intense discussion of the boundary conditions to be assumed for the concept development.

Requirements have been determined in [3.9] as follows:

- To further reduce the risk of an early or late loss of integrity of the containment vessel of PWR plants as a result of hydrogen combustion processes associated with events going beyond DBA, the RSK recommends installation of PARs as a plant-internal accident management measure.
- The number of PARs to be installed in a containment vessel, and their locations, must be determined taking the hydrogen release rates and characteristic gas transport times within the containment into account.
- On the basis of present knowledge, it is possible to sufficiently accurately determine by numerical analysis with lumped parameter codes and engineering estimates the distribution of hydrogen determining the number and the locations of the required PARs. The RSK assumes that the analysis results are further supported by CFD code analysis.

The current nuclear rules and regulations in the scope of the “Safety Criteria for Nuclear Power Plants”, the “RSK Guidelines for Pressurized Water Reactors” and the “Incident Guidelines” date back to the late 1970s and early 1980s. Therefore, in September 2003, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) launched a comprehensive programme for the revision of the nuclear rules and regulations. It was completed End of 2012. The new nuclear rules and regulations modernised on this basis comply with the current recommendations of the IAEA and the Western European Nuclear Regulators Association (WENRA) in the field of requirements for ensuring nuclear safety. The results of the work are documented in http://regelwerk.grs.de/. Severe accidents and accident management programmes are now included in the regulatory framework. The requirements are in principle the same related to the hydrogen issue.

In **Japan**, the new regulatory requirements are enforced on July 8th 2013 after intensive analyses and discussions on a SA at TEPCO’s Fukushima Dai-Ichi NPS. Requirements on the SA measures are mentioned to prevent and mitigate both core degradation and containment failure. For hydrogen explosion, it is interpreted to prevent detonation that can cause the containment failure. These national requirements are substantiated by the guidelines that show how to prove adequacy of measures against hydrogen explosion:

- A representative scenario is chosen from the containment failure sequences that are established based on PSA.

- An amount of hydrogen generated before failure of the reactor vessel corresponds to reaction of 75% of Zircaloy located in the core.
- After failure of the reactor vessel, generation of combustible and non-combustible gases due to MCCI is taken into account.
- Generation of hydrogen and oxygen due to radiolysis is taken into account.
- Validated computer codes are applied to predict the hydrogen distribution inside the containment.
- Other factors that give significant influences are taken into account.
- Criteria for preventing the destructive detonation are interpreted as maintaining the mean and local hydrogen concentration at 13 vol.% or less without steam condition or the mean and local oxygen concentration at 5 vol.% or less.

The followings are exemplified as commendable measures.

- Glow-plug igniters
- PARs
- Inerting the containment (e.g., with nitrogen).

In **Korea**, the regulatory requirements for combustible gas control during a SA are prescribed in the section 16.1, "Capabilities to Respond Severe Accidents" of the Regulatory Guidelines, based on the "Additional TMI-related requirements" of the U.S. NRC's regulations, 10 CFR 50.34(f)(2)(ix) and (f)(3)(v) [3.10]. Until now SA requirements are not legally treated, but they have been applied under the auspices of the Severe Accident Policy issued by the Nuclear Safety Commission in August 2001. Currently, activities to enact these requirements are under discussion. The Regulatory Guidelines specifies the combustible gas control requirements as follows:

- Hydrogen concentration shall not exceed 10 vol.% during and after a SA where hydrogen generated from reaction of 100% cladding metal and water under the assumption of uniform distribution of hydrogen inside containment building. The concentration of combustible gas in each compartment of containment building shall be low enough for preventing wide-scale FA or DDT. Facility shall be installed for protection of containment building from damage due to combustion of combustible gas in containment building.
- Structural integrity of containment building shall be kept intact during and after a SA where hydrogen is generated and released by reaction of 100% cladding metal and water, and pressure is increased due to deactivation after hydrogen combustion or accident (assuming carbon dioxide as deactivation agent). As for steel containment building, the requirements for Class C acceptable operating limit of Korea Electric Power Industry Code (KEPIC) MNE 3220 shall be satisfied. As for concrete containment building, the acceptable factored load category defined in Korea Electric Power Industry Code SNB 3720 shall be satisfied.

All the above should be included in the design or pertinent countermeasures.

In **the Netherlands**, the safety requirements for the Borssele two loops PWR NPP are set by the Duch government in the 'Nucleaire Veiligheidsregels' (NVR). The nuclear rules and regulations in the NVR are based on the IAEA Safety Standards, including the Western European Nuclear Regulators Association (WENRA) Reference Levels (www.wenra.org). Recently the license requirement of the Borssele NPP has been updated, including the WENRA RHWG reference levels of 2008.

The features for the control of combustible gases should be designed to eliminate or reduce the concentration of hydrogen, which can be generated by water radiolysis, by metal–water reactions in the reactor core or, in SA conditions, by in-vessel metal-water reactions (mainly Zr-oxidation) at elevated temperatures and by MCCI. Features used in various designs include hydrogen recombiners (i.e. passive recombiners or active igniters), large containment volumes for diluting hydrogen and limiting the hydrogen concentration, features for mixing the containment atmosphere, features for inerting and devices for ensuring that any burning of hydrogen is controlled

SAMGs have been in operation at the Borssele NPP since 2000 as an outcome from the PSR at the plant in 1993. The SAMGs are based on the generic SAMGs produced by the Westinghouse Owners Group (WOG). They are intended to address scenarios deriving from severe external hazards, such as earthquakes and floods, where there is the imminent potential for core melt. The SAMGs include guidance for using the pressure relief valves and various pressurizer spray options to control the RPV pressure. The SAM guidelines SAG-6 and SCG-3 give guidance on how to measure the hydrogen concentration within the containment and how to manage its flammability or prevent deflagration or detonation.

For an ex-vessel event the containment has filtered venting, a spray system, air coolers, a filtered recirculation system and PARs. The containment is designed for overpressures of 3.8 bar.

In **Spain**, the requirements for hydrogen mitigation are:

- For DBA conditions, all licensees must have the necessary systems to control hydrogen concentration to ensure containment integrity [3.11].
- For DBA and BDBA conditions, BWR with Mark I type containment must have an inerted atmosphere and BWR with Mark III type containment must have the capability for controlling combustible gas (igniters) generated from a metal-water reaction involving 75% of the fuel cladding so that there is no loss of containment structural integrity [3.1].
- For DBA and BDBA, the equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere [3.12].
- For DBA at PWR-KWU type containment, if the calculations reveal that the hydrogen concentration can reach values above the ignition limit in certain areas of the containment vessel, active features shall be provided which will assure sufficient forced flow mixing of the containment atmosphere [3.13].
- For SA at PWR-KWU type containment, the hydrogen mitigation system (PAR) is able to reduce overall hydrogen concentration under various conditions including steam inert (non-flammable mixtures) , thus limits the probability of fast deflagration in large volumes and assures the safe enclosure of radioactivity in the containment, in accordance with [3.9].
- Implementation of PAR system has been required to the rest of Spanish NPPs after the “stress test” by the end of 2016. The PAR system (effectiveness recombination per unit, number and location) should be able to limit hydrogen concentration during different phases of SA to eliminate the possibility of deflagration or detonation that threaten the containment integrity. The considered sceneries should maximize hydrogen generated from metal-water reaction and other hydrogen sources [3.14].

In the **Swedish** regulatory framework; there is no specific requirement explicitly addressing hydrogen management or hydrogen related issues. However, the hydrogen issue is covered by the following regulatory requirement, “The reactor containment shall be designed taking into account phenomena and loads that can occur in connection with events in the event class highly improbable events, to the extent needed in order to limit the release of radioactive substances to the environment”.

In the SSM’s general advice on the application of the regulations, it is stated that to meet the above requirement, a safety evaluation should be performed of events and phenomena which may be of importance for containment integrity in highly improbable events. Hydrogen combustion is listed as one of the phenomena which can result in the need to take measures.

In the document HSK-R-103/d [3.15], the **Swiss** regulator indicated the internal plant measures against the Consequences of major accidents. With respect to the hydrogen control, appropriate measures should prevent a serious accident such that hydrogen in the containment - global or local – accumulates in such a concentration, that the containment would be endangered by an ignition.

Independent of the measures, which were implemented for the detection and monitoring of the hydrogen concentration in the containment, different requirements for the design of the FCVS has to be taken into account (Swiss Guideline HSK-R-40/d) [3.16]. Already in the early 90, all Swiss NPPs had a refit with a FCVS in accordance with the Swiss Guideline HRS-R-40.

In the **United States**, the US NRC revised the hydrogen control requirements in [3.1] in 2003, "Combustible Gas Control for Nuclear Power Reactors." This rule eliminated the requirements for hydrogen thermal recombiners and hydrogen purge systems in currently licensed LWRs, and relaxed the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance. However, the rule retained existing requirements for ensuring a mixed atmosphere, inerting BWR Mark I and II containments, and providing a hydrogen control system capable of controlling an amount of hydrogen generated from a metal-water involving 75 percent of the fuel cladding surrounding the active fuel region in BWR Mark III and PWR ice condenser containments. The technical bases for the regulations were established from experience at TMI along with bounding estimates for the amount of hydrogen likely to be generated by a severe core damage accident.

This rule also specifies requirements for combustible gas control *in future* water-cooled reactors which are similar to the requirements specified for existing plants. However, a key difference is the need to accommodate an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction. Particularly, if a containment does not have an inerted atmosphere, it must limit hydrogen concentrations in containment during and following an accident that releases hydrogen (equivalent to 100 percent fuel-coolant reaction) when uniformly distributed to less than 10 percent (by volume); and maintain containment structural integrity and appropriate accident mitigating features. Consequently, the PWR large dry containments are now required to install a hydrogen control system to adhere to the above specified limits.

3.1.1.2 Mitigation Measures and Expected Performance

To fulfil the requirements mentioned above, mitigation measures have been (or are being) implemented in various NPPs to remove hydrogen or to control the hydrogen concentration inside the containment. The design of the appropriate mitigation measures is then performed based on the results of numerical simulation and dedicated experiments. The mitigation measures and their implementation status are detailed in the following. The major mitigation methods are summarized in Table 3-2 for these countries.

For the **Belgian** PWRs, the hydrogen mitigation strategy combines the existence of a large free volume and the installation of PARs in the primary containment. Since 1998, all NPPs are equipped with a large number of AREVA PARs in order to remove hydrogen during a SA and to prevent global combustions threatening the containment integrity.

The design procedure applied at that time was based on the sizing of the total catalytic surface for a reference plant according to the design criterion "mean hydrogen concentration < 5 vol.%". In addition, the total catalytic surface was increased by 20 %, with respect to the minimum required surface derived from a set of representative SA sequences, in order to provide a design margin covering uncertainties. Since the installation of the PARs, calculations with the MELCOR code using a multi-compartment containment model have confirmed the sizing of the PARs.

For **Canadian** NPPs, several engineered design features, such as a large dilution volume and the installation of vault hydrogen mitigation systems (i.e., igniter and/or PARs), are utilized to provide effective hydrogen management for DBA and/or BDBA/SA.

The primary short-term hydrogen management strategy is by dilution due to large containment volume. For CANDU single unit reactors, this can be achieved by two means. Firstly, the internal configuration of the containment has been designed to promote natural convection. The fuelling machine vaults have openings at the top as well as at the bottom. Hydrogen released into the fuelling machine vault can migrate to the rest of the containment building (e.g., dome region) very quickly by natural convection. Secondly, local air coolers installed in the fuelling machine vault and in the dome region can quickly mix the hydrogen with the surrounding air preventing local stratification of hydrogen.

For the multi-unit stations, the mixing lengths between the release region (fuelling machine vault) and the balance of the containment volume are much longer. Glow plugs have been employed within each reactor vault and they can be energized either automatically on receipt of containment button-up signal or manually by hand-switch immediately following a LOCA. The basic principle of this mitigation is to initiate a burn as soon as the hydrogen-air-steam mixture permits, thus minimizing potential pressure excursions, but this method needs a large number of igniters to ensure that there is no localized stratified region. The characteristic of TAYCO type igniter has been extensively tested and verified by experiment programme at AECL. Igniters are credited for short term DBAs for the multi-unit stations.

For the CANDU 6 plants in Canada, hydrogen igniters are not provided because it was determined that the average hydrogen concentrations were low based on DBA conditions due to enhanced mixing. In the refurbished Point Lepreau NPP, PARs have been installed to mitigate hydrogen generation under BDBA conditions and to assist with SA conditions. In addition, the FCV system has been installed to protect the containment integrity in case of the need to depressurize the containment.

In response to the Fukushima accident, the multi-unit stations have committed to install PARs to improve hydrogen control for BDBA and SA accidents. Installation of PARs is ongoing and on schedule. There are other plans to enhance defence in depth with other SAM strategies (i.e., delayed venting, turning off air coolers, etc.) for hydrogen control.

In **Czech Republic**, both Czech NPPs have installed a system for hydrogen removal, based on PARs, but they cover only DBA hydrogen generation and are insufficient for SA conditions. A project on installation of extended system for hydrogen removal for SA conditions was launched in early 2013 and installation of PARs has to be finished by the end of 2015. The PAR layout and the number of PARs will be determined by numerical analysis. Nevertheless, only PARs will be used for hydrogen mitigation (no igniter is considered), and it is expected to use AREVA PARs at the Dukovany NPP and NIS at the Temelin site. In both NPPs, the existing PARs designed for DBA conditions will continue to be in service.

In **Finland**, there are three different reactor types and each of them has different hydrogen risk mitigation measures:

- Loviisa 1 & 2 (VVER-440): 154 PARs, made by AECL, and glow plug igniters.
- Olkiluoto 1 & 2 (Asea Atom BWR): containment is inerted with nitrogen.
- Olkiluoto 3 (EPR): about 50 PARs, made by AREVA.

For the **French** PWRs, the adopted hydrogen mitigation strategy combines the existence of large free volume and the use of PARs to consume hydrogen.

Before implementing PAR systems, the experimental programmes based on two facilities were led by IPSN (former name of IRSN) and CEA in Cadarache in the 1990s, in cooperation with Electricity of France (EDF), to assess the efficiency of PARs in representative SA conditions:

- The H2PAR programme [3.17], conducted by the IRSN with support from EDF, aimed mainly to verify that catalytic hydrogen recombiners continue to function in an atmosphere representative of SAs and containing several chemical compounds in aerosol form (risk of catalyst poisoning). This programme also investigated the risk of PAR-initiated combustion and determined thresholds for the specific PAR studied. The impact of various parameters on recombination performance was also analysed, including geometric parameters (number of catalytic plates, height of stack location), physical parameters (molar fraction of hydrogen) and chemical parameters (several catalytic plates replaced by chemically neutral plates). Concerning aerosols poisoning, H2PAR results didn't show any significant effect on PAR efficiency.
- The KALIH2 test programme, conducted by the CEA with support from EDF, had similar objectives. It evaluated the effects of the following phenomena on recombiner performance: humidity, smoke from cable fires, and carbon monoxide. Unlike H2PAR, KALIH2 enabled to study the impact of overpressure on the PAR efficiency.

The decision of PARs installation inside the containment was made in 2004 and has been based on the analysis of representative and bounding in-vessel accidents scenarios. The number and the location of PARs have been adjusted to fulfil the requirements presented in Table 3-2.

Since 2007, PWRs 900 have been equipped with AREVA PARs and PWR1300 and PWR 1450 were equipped with AECL PARs. The EPR under construction in Flammanville will be equipped with AREVA PARs.

In **Germany**, the BWR type 69 containments (all NPPs are shut down after the Fukushima accident) have been fully inerted by nitrogen (wetwell and drywell) during normal operation.

The BWR type 72 NPPs uses a combined inerting and recombination concept. It consists of an inerted wetwell (as access to the drywell may be needed during plant operation) and in addition a large number of PARs (NIS type) in both the drywell and the wetwell to prevent large combustions. Deterministic analyses have been performed by the utility using the WAVCO containment code. The scenarios analysed have been "station black-out" and a "main steam-line break LOCA". During the review process, GRS made detailed analyses using the RALOC code to support the conceptual design. The scenarios analysed have been "station black-out" and a "LOCA at the RPV head", the latter was a similar case to the steam-line break scenario. Both resulted in an upgrade of the system towards a larger number of PARs to be installed.

In all German PWR containments a large number of PARs are installed to remove the hydrogen generated in a SA and to prevent global combustions. In most PWR plants, AREVA type PARs have been installed, while in three other PWRs, NIS type PARs are used.

The PAR concepts have been systematically investigated [3.18]. At first five representative SA sequences for the design and valuation of a PAR system were selected. These have been cases with fast, intermediate and slow core degradation as well as partial core degradation with reflooding. For the case selection, criteria considering the release conditions of H_2 and CO into the containment atmosphere and the pre-conditioning of the containment at the time of the release of these gases were applied. Due to the preventive accident management action of primary bleed and feed for German PWR plants, high pressure core melt sequences were not taken into account for the design of a PAR system. For these sequences, detailed investigations by GRS with the containment code RALOC (now part of COCOSYS), based on MELCOR calculations for core degradation and MCCI after vessel failure, were performed for a reference plant. The following specific topics have been investigated:

- Positioning of PARs in a multi-compartment containment configuration (development of generic criteria).
- Determination of the local and overall capacity of a PAR system, needed to prevent high hydrogen accumulation.
- Influence of the PAR system on the gas distribution in the containment under accidental conditions (extent of gas mixing).
- Consequences of a failure of local catalytic devices due to blow-down forces.

The findings are summarized as follows [3.18]:

- The chosen concept for the use of PARs in a large dry containment demonstrates a high safety profit.
- An active PAR-system results in a lower containment pressure in the long-term due to the mole reduction and steam condensation. The general temperature level is increased due to the exothermic H_2-O_2 reaction (about 20 – 30 K).
- Inside the inner containment missile shield of German containment design, combustible gas mixture compositions could be developed locally for short times, also with concentrations, exceeding 10 Vol.% H_2 (mainly in the neighbourhood of the H_2-release location). Such conditions are in general no threat to the containment integrity in case of an ignition (negative H_2-concentration gradients towards the outer containment shell).
- The integral recombination rate depends on the local and global convection pattern inside the containment (e.g. influence of door leakages between different areas of the containment and stair cases).
- An unequal hydrogen supply (short term high peak) to a PAR located in the neighbourhood of the release location could result in an overheating and by this to an ignition of a combustible gas mixture.
- After approximate 1 day, the containment atmosphere becomes inert due to the continuous O_{2}-consumption by the catalytic reaction. This could lead to a long-term increase of the H_2 content in the containment (important for venting device!).
- The continuous recombination of H_2 and O_2 during the first day mitigates always potential combustion threats to the containment.
- It should be pointed out, that due to the preventive accident management action of primary bleed and feed for German PWR plants, high pressure core melt sequences were not taken into account for the design of a PAR-system.

Regarding for PAR locations within complex German PWR containments, generic recommendations have been derived based on the work done at GRS for the PAR concept development. PARs in German PWR containments are required to be placed [3.19]:

- Near expected main convection flow paths between inner containment rooms (main component rooms) and the dome area as well as the containment periphery considering the various possibilities of steam and gas releases from the reactor circuit into the containment.
- Within stair cases near doors or other connections which will be opened in case of an accident; related to German PWR containments these are the lower middle and upper part of the stair cases connected to the missile protection cylinder within the containment.
- The selection of PAR positions within stair cases should consider escape paths for people from the containment in accident situations; rescue operations are not blocked.
- In so called dead-end-zones (the room above the RPV or the special room containing numerous pipelines in German PWR containments).
- At different heights in large rooms to support local convections and to enhance the efficiency.

- Not near the containment steel shell because of the hot off-gas of the PAR during operation.
- In a way that the hot off-gas of the PAR during operation will not challenge other systems, components, cables, penetrations, etc.

Besides this analytical work a wide experimental programme was issued by the PAR vendors for experimentally qualification of the components.

In **Japan**, applications have been submitted for permission for the establishment amendment for the existent PWRs and two BWRs in order to examine conformity to the new regulatory requirements.

The Japanese PWRs adopted a few AREVA type PARs suppressing global combustion as a mandatory measure as well as more than ten igniters suppressing a peak hydrogen concentration as voluntarily measure. Rationality of their numbers and locations are now under discussion. In case of PARs, experimental conditions should be identified regarding flow fields where the hydrogen processing performance was measured. Their locations in the actual containment should be designed so that an equivalent or higher level of performance will be ensured.

On the other hand, BWRs adopted hydrogen mitigation strategy that combines inerting with nitrogen and a filtered venting.

Furthermore, an additional filtered venting system is required for both PWRs and BWRs as "specialized safety facility" to cope with extremely harsh external events such as the aircraft crashes, etc.

For the **Korean** PWRs and PHWRs, the adopted hydrogen mitigation strategy combines the existence of large free volume and the use of PARs to consume hydrogen. PARs installation inside the containment had been based on the analysis of representative scenarios. The number and the location of recombiners had been adjusted to fulfil the requirements presented previously.

In **the Netherlands**, the Borssele PWR NPP has a hydrogen control system consisting of PARs located in the containment. Three types of Siemens PARS with different capacity are installed in the Borssele NPP. The PAR system is sized and designed for operation during a SA. The recombination capacity is such that:

- the system can recombine hydrogen faster than it is generated during the molten core concrete interaction phase of a SA;
- during the initial (in-vessel) phase of the accident, the hydrogen concentration in containment is limited to approximately 10 vol.% at any location.

The 'in-vessel' hydrogen production determines for the concentration of hydrogen in the containment because this hydrogen is produced on a short time scale at a rather high rate (order of magnitude 100 g/s). The 'ex-vessel' hydrogen production (mainly MCCI) occurs later during the accident and at a much slower rate (order of magnitude 1 g/s). Therefore, with the installation of the PARs (recombination rate in the order of 10 - 100 g/s), the "ex-vessel' hydrogen source can be effectively encountered, while the 'in-vessel' hydrogen concentration can be limited.

Hydrogen production caused by molten core-concrete interaction is a smaller problem for the Borssele PWR NPP than for other nuclear power plants because of the relatively high content of carbonates in the concrete compared to, for instance, most German PWRs. This leads to a relatively higher production of carbon dioxide, which has an inerting effect.

For the **Spanish** NPPs, there are several measures to deal with hydrogen risk within containment:

- For SA conditions, Trillo NPP (PWR Siemens-KWU) installed PAR from FRAMATOME ANP (now AREVA) in 2002.

- Implementation of PAR system has been required to the rest of Spanish NPPs after the "stress test" by the end of 2016.
- The primary containment of BWR-GE Mark I is inerted by nitrogen.
- The containment of BWR-GE Mark III is equipped by AC-power igniters. After "stress test" an alternative electric supply to the existing igniters was installed by the end of 2012.
- Thermal recombiners are installed in all Spanish NPPs containments apart from Mark I design. They consist of two redundant devices of 100% capacity for DBA. SAMG has recommended not using them in case of SA (H_2 dry concentration over 6 vol.%).

For the **Swedish** BWRs and PWRs, the hydrogen control is achieved through:

- All BWRs have nitrogen inerted containment during full power operation,
- In all PWRs containments Siemens/AREA PARs were installed in 2007. The number of PARs is between 20 and 25 depending on reactor and the nominal total capacity at 4 vol.% H_2 is 106 kg/h for Ringhals 2 and 126 kg/h for Ringhals 3 and 4.
- All BWRs and PWRs have thermal recombiners to mitigate hydrogen during DBA.

In **Switzerland,** there are 5 reactor units, of 4 different types. In the BWR Mark 1, the mitigation is based on drywell inerting and also on the use of thermal recombiner. In the BWR Mark III, are used glow-plug igniters. In the PWR Westinghouse type, are used PAR and in the PWR-Siemens-KWU type are used thermal recombiners, Moreover all the Swiss plants are equipped with FCVS. Moreover all the utilities are evaluating if additional mitigation measures are to be implemented.

In the **United States**, the hydrogen control is achieved through:

- BWR Mark I/IIs have nitrogen inerted conditions inside the containments;
- BWR Mark IIIs and PWRs with ice condenser containments have hydrogen igniter systems installed in the containments;
- BWRs and PWRs could also have thermal recombiners.

3.1.2 Outside the Containment and Other Places

3.1.2.1 National Requirements

During an accident, hydrogen generated from the reactor core may migrate from the containment to other places. Examples are the annular space between the reactor building and the containment in a PWR, the multiple rooms inside a BWR reactor building, or other non-accident containments for multi-unit stations. During a loss of coolant accident in a spent fuel pool, hydrogen may also be generated from Zr-steam reaction and released into the spent fuel pool building. Hydrogen mitigation outside the containment as well as in the spent fuel pool building needs to be considered.

Currently, no specific requirements are adopted for hydrogen mitigation measures outside the containment in most countries except France and Japan.

In **France**, the adapted requirements aim to avoid flammable atmosphere formation outside the reactor containment (secondary building, spent fuel pool building, etc.). It is required that the mean hydrogen concentration must stay below the lower flammability limit (4 vol.% in air).

In **Japan**, the national requirements applied for the secondary building are:

- Prepare equipment and procedures to prevent damage to the reactor building and containment vessel annulus due to accumulation and explosion of hydrogen in the event of severe core damage.

- To ensure the above requirement, the following measures shall be taken:
  a) Installation of hydrogen concentration control equipment (to ensure no threat of hydrogen explosion inside the reactor building) or hydrogen discharge equipment (i.e., filtered containment venting system).
  b) Installation of monitoring equipment that can measure hydrogen concentration during postulated accidents.
  c) This equipment shall be capable of connecting to alternative power sources (available AC or DC power).
  d) Other measures having the same or better effectiveness can also be employed.

And the national requirements for spent fuel pool building applied in **Japan** are:

- During the SA transient, degradation of fuel structures, unshielding or criticality shall be avoided in the spent fuel pool.
- To avoid degradation of fuel structures, unshielding or criticality, accident managements with alternative cooling measure shall be prepared.
- In order to confirm the above requirement, one has to assume the case of loss of water injection or heat removal or leak of water due to siphon off in the spent fuel pool.

3.1.2.2 Mitigation Measures and Expected Performance

The considerations for hydrogen mitigation outside the containment are described in the following.

In **Belgian** NPPs, the annular space between the inner containment and the reactor building is not equipped with hydrogen mitigation measures. The risk of hydrogen explosion in the annular space has been considered negligible as long as the containment is intact and the design leakage is not exceeded. In the building containing the spent fuel pool, no hydrogen mitigation measures have been implemented either. As results of the “stress-test”, the potential hydrogen risks will be examined.

For the **Canadian** multi-unit station, all the reactor vaults are connected together by the fuelling machine duct and pressure relief duct. This shared containment volume is isolated from the vacuum building by pressure relief valves. An accident, such as LOCA, in one of the reactor vaults causes the pressure in the shared containment volume to rise. The pressure difference between the reactor vault and the vacuum building automatically opens the relief valves, connecting the reactor vaults and ducts to the vacuum building. Steam, air and hydrogen gases released in the accident are swept into the vacuum building. No hydrogen mitigation measure has been implemented in the vacuum building due to the large containment volume. In addition, the operation of EFADS will maintain the vacuuming building at sub-atmosphere allowing hydrogen in the reactor building to vent to the vacuum building. However, EFADS are not designed to handle SAs. Following the Fukushima event, some multi-unit stations have been considering to install FCVS for SAs.

The SFPs of CANDU reactors are located outside of the reactor buildings in a confined building at ground level. The SFP is double-walled to provide double containment against leakage and to prevent loss of water in case of accident or failure. The SFP is lined with stainless steel. No hydrogen mitigation measures have been recommended for the SFP area under the current regulation. However, in response to the Fukushima accident, the CNSC convened a Task Force to evaluate the lessons learned and the operational, technical and regulatory implications for Canadian NPPs, and to develop a strategy for prioritization and implementation of corrective measures. Cooling of spent fuel storage in SA scenarios has been identified as one of areas to be addressed by the CNSC Task Force review. It has been required that industries should systematically verify the effectiveness of, and supplement where appropriate, the existing plant design capabilities in BDBA and SA conditions, including control capabilities for hydrogen and other combustible gases, including installation of the hydrogen management capability and sampling provisions for spent fuel bays and any other areas where hydrogen accumulation cannot be precluded.

Concerning possibility of hydrogen presents at other places than containment itself, the **Czech** NPPs have very different conditions. The principal additional source is the SFP, which is located inside of the containment at the Temelin NPP (VVER-1000/320 type of units), so the hydrogen removal system designed for the SA is capable to cover also sources of hydrogen in case of the accident initiated in the SFP. At the Dukovany (VVER-440/213 type of units), the SPF is located in the reactor hall, which is common for two units (multi-unit issue) and is not part of hermetic zone (containment). Recently study on hydrogen behaviour in the reactor hall is under preparation.

The **Finnish** NPPs do not have installed hydrogen mitigation devices outside the containment. However, for Olkiluoto 1 & 2 BWRs, the possibility to install latches, which could be opened during an accident for venting of hydrogen from the reactor building to the environment, is being considered.

Until now, secondary buildings in **French** NPPs are not equipped with mitigation measures. The risk of hydrogen explosion had been considered negligible due to the large volume of those buildings and slow and low hydrogen mass release rate.

Until now, in **German** NPPs – the annular space between the reactor building and the containment in a PWR and the multiple rooms inside a BWR reactor building - are not equipped with hydrogen mitigation measures. The risk of hydrogen explosion had been considered negligible, as long as the containment is intact and the design leakage is not exceeded or other significant hydrogen sources exist outside the containment. The German AM concept consists of measures to ensure the containment isolation in case of an accident. Containment bypass sequences typically show a relatively small contribution to the probability of case leading to core melt, as many actions are implemented in the operational procedures to prevent core melting. After the Fukushima accident the BWR-72 utilities decided to install PARs as well above the SFPs [3.20] in the reactor building to deal with enlarged containment leakages mainly through the containment head coverage. Details are not yet presented.

SFPs are located inside the containment for German PWRs while are located in the upper part of the reactor building for the BWRs.. After the Fukushima accident, additional mobile measures are under discussion to maintain the SFP cooling. For PWRs, the existing PAR concept will provide protection also in case of a SA in the SFP, although it has not been analysed in detail.

For both the PWRs and BWRs in **Japan**, as measures at discharge paths connected to the outside of the containment, PARs or other hydrogen discharge equipment can be installed. For PWRs, as measures inside the containment vessel annulus, PARs or other hydrogen discharge equipment can be installed. For BWRs, as measures inside the reactor building, PARs can be installed.

The standby gas treatment system (SGTS) in BWRs can be used to mitigate hydrogen in the reactor building under DBA conditions.

Under the new regulatory requirements, Japanese ABWRs adopted PARs in the reactor building. More than fifty PARs are used in each ABWR unit. Expected performance is 0.25 kg/h per unit at 100°C in atmospheric pressure and 4 vol.% H_2.

Moreover, alternative water injection measures to the spent fuel pool are prepared in Japanese NPPs.

In **the Netherlands**, the Borssele NPP consists of a single unit. The secondary buildings are not equipped with hydrogen mitigation measures. However, to cope with external hazards, important safety systems like emergency core cooling, spent fuel pool cooling, reactor protection system and the emergency control room are installed in “bunkered” buildings. These buildings are qualified to withstand earthquake, flooding, gas cloud explosions, aeroplane crash and severe weather conditions. The SFP is located inside the Borssele containment. There is no separate fuel storage facility outside the containment. There are two PARs installed above the SFP for control of hydrogen in case of hydrogen release from the SFP.

In **Spain**, no specific measures are installed outside the containment. As results of the “stress-test”, analysis of potential hazards in other buildings surrounding the containment has been required by the end of 2013. These studies will be undertaken considering that PAR systems have been installed and work properly in the containment building.

In **Sweden**, no specific equipment is installed in the reactor buildings or any other adjacent building for hydrogen mitigation measures in the **Swedish** NPPs. However, the possibility and consequences of accumulating hydrogen in the reactor building including necessary instrumentation and management is being investigated, as one of the requirements in the Swedish National Action Plan following EU Stress Tests.

In **Switzerland**, the possible benefit of installing hydrogen mitigation device, e.g. PAR also in secondary buildings, is under investigation by various utilities.

In the **United States**, no specific system is installed outside the containment for hydrogen control; however, existing ventilation systems can be used. Mitigation strategies will be pursued and assessed.

3.2 Considerations of Systems and Events on Hydrogen Behaviour

Several engineering systems (e.g., spray, local air cooler, venting systems, blow-out panels, etc.) have been installed in many NPPs to reduce containment pressure and temperature during an accident. However, operation of these systems can have an impact on hydrogen distribution and combustion if ignition occurs. They may reduce maximum hydrogen concentration due to enhanced mixing or increase in the total volume, but on the other hand, they may increase the hydrogen concentration due to steam removal. Various requirements and considerations have been defined by some countries in use of these systems in their SAMGs.

3.2.1 Spray System

Spray systems are installed in many NPPs. The requirements for operation of spray systems are shown in Table 3-3. Their application details are presented in the following.

In **Belgian** PWRs, the primary function of the spray system is to reduce pressure and temperature in the primary containment. Spraying is also used to reduce the concentrations of radioactive particles and iodine in the inner containment atmosphere. In the Doel units, containment pressure control is also achieved by means of fan coolers.

During a SA, the reduction of the steam concentration in the containment using the spray could lead to a containment atmosphere composition with a mean hydrogen concentration exceeding the hydrogen flammability, however, the PARs are designed to sufficiently reduce the mean hydrogen concentration and prevent large hydrogen combustions leading to containment failure. In the SAM procedures of the Doel plants, it is recommended to reduce the containment depressurisation rate in order to avoid severe hydrogen burn. As results of the “stress-test”, such strategies will also be added to the SAMG of the Tihange plants.

For CANDU multi-unit stations in **Canada**, the vacuum building dousing system does not play a significant role following a small break. Dousing spray will be initiated in the short-term for the larger breaks sizes, but the vacuum reserve in itself is sufficient to quickly return containment to a sub-atmospheric pressure. Dousing spray does not affect the duration of short-term containment overpressure, but will extend re-pressurization time by reducing the temperature and corresponding vapour pressure in the vacuum building. In the long term, EFADS operation is intended to maintain containment at a sub-atmospheric pressure during DBAs. The use of a filtered and monitored venting path eliminates the possibility of uncontrolled leakage from containment. During normal operation the dousing tank level is maintained at 7.4 m (nominal). The water in the tanks is heated during the winter to prevent freezing. Typically the water temperature ranges from 5°C to 12°C over the period of one year. The heat sink effect of the water is what prolongs the vacuum in the vacuum building following a LOCA. The initiation timing for the dousing depends on the accident scenarios. Dousing system is credited for short term hydrogen mitigation.

For CANDU 6 single unit stations, dousing is one of the design responses to suppress the pressure peak generated inside containment as the result of a break in any of the high energy pipes. The dousing tank is located in the dome of the reactor building. The design dousing flow rate of ~4000 kg/s is provided by any four of the six downcomers passing water together. Containment overpressure does not exceed 124 kPa (differential) after a LOCA with this flow. A total spray flow of 6804 kg/s can be provided with all valves open. At the dousing “on” setpoint of 14 kPa (g), the isolation valves open and a flow of water, from the dousing tank to the spray nozzles, will be established by gravity. When the pressure drops below the dousing “off” setpoint of 7 kPa (g), the isolation valves close and the dousing flow is stopped. Depending on the conditions of the break, the system may cycle on and off until the dousing water is exhausted (the water level decreases below 2.3 m), or it may only cycle once. According to the current SAMGs, one of the options for mitigation of hydrogen challenge is to pressurize containment with steam by turning off cooling to inert the atmosphere or dilute the hydrogen volumetric concentration; dousing can be terminated by manually closing the dousing spray valves. The major concern for this option is condensation of steam in the long term raising the hydrogen concentration to a level that may challenge the containment integrity.

In **French** PWRs, the spray primary function is to remove heat and condense steam in order to reduce pressure and temperature in the containment building. In case of hydrogen release inside the containment, sprays homogenize the hydrogen distribution and may lead to “de-inerting” of the mixture through the condensation of steam on water droplets.

In case of ignition, the water sprays can affect the flame propagation. Two antagonist effects could be expected:

- FA due the turbulence induced by spray actuation
- Flame quenching due to the cooling effect of water spray

Indeed, the role of water sprays on premixed flame propagation is complex and depends strongly on several parameters such as the liquid water fraction, droplets distribution, droplets size. R&D activities are still needed to identify conditions leading to FA or flame quenching when spray is actuated.

For these reasons and in order to keep the containment atmosphere inert during the in-vessel hydrogen production phase, the French SAMG recommend postponing the spray system activation at least 6 hours after the beginning of core degradation. During this time, hydrogen concentration would be reduced by recombination.

In **German** PWRs, no spray systems are installed. A full pressure containment design is used. The SAMG to be used in PWR are currently under development. Further measures to limit the containment pressure increase may be described there.

In German BWRs, no safety relevant spray systems are installed. The spray systems in the drywell and wetwell may be used to reduce the airborne aerosol concentration and to limit the containment pressure. The related measures are described in the emergency hand book. The SAMG to be used in BWR-72 are currently under development. Influences of the use of the spray system on the PAR performance may be discussed there.

In **Japan**, as mentioned in Section 3.2, activation of the containment spray is assumed in assessing effect of measures since it causes not only mixing but also condensation of the steam and enables conservative estimate of the hydrogen concentration. Criteria of the dry conditioned hydrogen concentrations as shown in Table 3-1 take this effect at most. Influences of the flame propagation are explicitly addressed in neither the regulatory requirements nor guidelines. However, it is requested to assess a degree of the flame propagation based on the conservative model and the partition wise hydrogen atmosphere conditions. Considering complexity of relevant phenomena, further discussions based on the up-to-date experimental and analytical knowledge base are necessary to establish deliberate operational procedures of the containment spray in SAMG. Postponing the spray activation, controlling spray flow, combining with natural circulation systems, etc. with supported by monitoring the containment atmosphere are recognized as important elements to be included in the procedures.

In **Korea**, the PWRs and PHWRs have the containment spray system in order to reduce pressure and temperature in the containment building. But actuation of the spray system may lead to an increase in the hydrogen concentration and threaten the integrity of the containment building due to hydrogen combustion for some cases. So, the SAMGs recommend controlling spray mass flow rates or isolating ignition sources which can make electrical sparks.

In **the Netherlands**, the containment spray can be used as an additional measure to reduce the containment pressure and temperature. The spray system is, however, primary designed to wash-out radioactive products. A possible disadvantage of reducing the containment pressure is that it might increase hydrogen concentration. To aid in the diagnosis of the SA conditions and the selection of appropriate strategies for implementation, graphical computational aids (CAs) have been developed for the Borssele NPP. One of the CAs is CA-6 (hydrogen flammability in the containment), which presents a containment depressurisation limit to avoid any possible hydrogen severe challenge or hydrogen burn when depressurising the containment. This limit is taken into account and respected.

In **Spain**, the Westinghouse-PWR and GE-BWR plants are equipped with a containment spray system, in order to reduce pressure and temperature within the containment building. The spray system improves mixing of the containment atmosphere and reduces containment pressure. However, the negative effect of spray system actuation is a reduction in steam concentration by condensation and increased atmosphere turbulence. In the Westinghouse SAMG when containment pressurization is recommended in order to prevent hydrogen combustion, the strategy is to switch off sprays and open PRZ and RPV PORVs. In the GE-BWR Mark III SAMG sprays are started, in case of AC igniters are working, when pressure and hydrogen concentration within containment reaches a limit curve. In case of AC igniters are not working and the hydrogen concentration is indefinite, hydrogen is controlled by switching off thermal recombiners, switching on sprays and venting the containment without limitations.

All **Swedish** NPPs (BWRs and PWRs) have containment spray systems. In addition, a back-fitted “independent” spray system has been installed in late 80’s at all NPPs as one of the measures to mitigate SAs. The question of when to use the spray from a hydrogen stand point is mainly an issue for PWRs and is a part of the SAMG. Since PARs were installed in PWRs, the SAMGs have been updated to accommodate new strategies for the use of both PARs and spray.

In **Switzerland**, the Beznau PWR Westinghouse reactor is equipped with containment spray system in order to reduce pressure and temperature inside containment. The system consists of 106 spray nozzles in the upper part of the containment and is designed to inject 110 kg/s borated water at 3.1 bar (g) containment pressure. During a postulated station black-out there is the possibility to inject water via accident management nozzles with mobile pumps from the fire brigade.

In the **United States**, most BWRs and PWRs utilize spray systems to reduce pressure and temperatures inside the containments.

3.2.2 Containment (Filtered) Venting

Containment venting is another system that has been considered as a key measure to avoid containment pressurization beyond the design pressure during an accident. It also allows retention of fission product by filtered venting. Its application details for different countries are addressed in [3.21]. The aim of this paragraph is to highlight the applied SAMG regarding the effect that the venting system can have on hydrogen combustion. The requirements for the use of FCV of the applied countries are summarized in Table 3-4.

The **Belgian** PWRs have a large dry containment without FCV. In the framework of the long-term operation of the Tihange 1 unit and since the “stress-test” after the Fukushima accident, a containment filtered venting system has been required in all Belgian PWRs. Installation and commissioning is planned in the period of 2015-2017 (except for Doel 1-2 which will be permanently shut down after 2015).

In **Canada**, all NPPs have the means to vent the reactor building (usually referred to as containment for CANDUs) to protect containment structural integrity. However, not all NPPs can filter the vented gases in SAs. Multi-unit CANDU reactors have a “negative pressure design” achieved by the provision of a large vacuum building supported in the long term by EFADS during DBAs. This type of design can be effective in limiting the release of radioactive material in DBAs as any leakage is into the containment. However, these systems are not designed to handle the large gas volumes that a SA may generate. Moreover, electrical power is required for controlled venting of the containment to maintain a negative pressure.

FCV is a complementary design feature intended to protect the containment envelope if the internal containment pressure approaches the containment strength limit and to remove radioactive materials from any gases vented from the containment in a SA. Such a system has been installed in a refurbished CANDU 6 unit at Point Lepreau; it is manually actuated, does not require an external source of power and is used to relieve containment pressure for the conditions that could be present in a SA. FCV uses a high-efficiency scrubber and filtration unit to filter out the vast majority of fission products so radiation exposure to the public would be limited to acceptable levels in the event of a release. The Westinghouse FCV dry filter method design has been proposed to be implemented in the four units at the Darlington multi-unit station by the end of 2015 and it is presently in its conceptual design stage. Assessment for other multi-units NPPs are ongoing and due by the end of 2014.

In **Finland**, Olkiluoto 1 & 2 BWRs and Olkiluoto 3 EPR have FCV systems but Loviisa 1 & 2 VVER-440 reactors do not. The venting line is inerted with nitrogen and sealed with a rupture membrane to prevent hydrogen burns inside the system.

For **French** PWRs, the opening of the containment heat removal system (CHRS) is mandatory when the pressure inside the containment reaches 5 bar. Before the opening, CHRS components are heated to avoid steam condensation and to keep gas inert inside the CHRS components. This conditioning is lost in the event of total loss of the electrical power supplies. Although measures are taken to limit the risk of hydrogen combustion in the venting line (pressure reduction upstream of the line limiting the risk of condensation), recombiners substantially limit the hydrogen concentration.

After the Fukushima, the French nuclear Safety Authority asked EDF to re-examine the possibility of hydrogen combustion and its possible impacts on the venting setup. ASN considers that this examination must focus in particular on the impact of the oxygen already present in the pipe and on the risk of hydrogen deflagration and its possible consequences at the venting line outlet.

In December 1986, the **German** RSK specified the requirements for design, operation and construction of a containment venting system for the German PWRs and in June 1987 for the German BWRs [3.8].

Within the framework of the safety reviews of the NPPs and the concomitant final report as of 23.11.1988, [3.6], the RSK recommended to give priority to the containments global integrity and to the prevention of high pressure containment failure of German NPPs. Thus the point of time for opening the corresponding relief valves of the venting system is determined by a critical pressure build-up in the containment. The FCV system was one of the systems which was recommended and installed very early [3.20].

In the emergency hand book, the measures are described to limit the maximum pressure in the containment by spraying into the wet well or drywell (BWR) or by FCV. The later one can be used in BWR-72 as well in case the heat removal from the wet well fails in an accident.

In **Japan**, BWRs adopted a hydrogen mitigation strategy that combines inerting with nitrogen and filtered venting. Furthermore, an additional filtered venting system is required for both PWRs and BWRs as a “specialized safety facility” to cope with extremely harsh external events such as the intentional aircraft crashes, etc. For a long term operation of the filtered venting systems, build-up of hydrogen is anticipated due to inflow from the containment and slow but steady generation by radiolysis inside a scrubbing pool. It is also required to prevent hydrogen explosion inside a filter and at discharge paths connected to the outside of the containment vessel.

The **Korean** regulations stay in line with the US regulatory on the containment venting issues. After the TMI accident, some Korean PWRs (e.g., PWR1000) provided a 3-foot diameter containment penetration according to the 10 CFR 50.34(f) [3.10]. After the Fukushima accident, the Korean PHWRs (CANDU-type) have planned to install containment filtered venting system.

The only FCV system installed in the one PHWR unit (PHWR679) is provided by AREVA. It has not been in operation due to the delay for the life extension of PHWR679. The remaining Korea PWRs and PHWRs plan to install FCVs by the end of 2015 for which an open tender is being prepared.

In **the Netherlands**, the Borssele NPP is equipped with a FCV line. Filtered venting is used as a last resort to control the containment pressure. The SAMGs envisage venting being carried out before a 3.6 bar over-pressure is reached. The venting system is kept inert with nitrogen and uses a wet scrubbing filter that is qualified for SAs. No electric supply is needed to operate the filtered venting system as the valves can be opened manually from the outside.

Similar to spray operation, a containment depressurisation limit is expected to avoid high hydrogen concentrations when depressurising the containment by venting.

In **Spain** there are no containment filtered venting systems installed in any NPPs. BWR-GE Mark III and Mark I plants are equipped with hardened venting. After the “stress-test”, the installation of containment filtered venting has been required in all Spanish NPPs.

All the **Swedish** NPPs have installed FCV. This installation was performed during late 1980’s as a result of a Government decision. The filter is called a Multi-Venturi Scrubbing System (MVSS) and utilises a wet scrubber. The filter is actuated automatically by a rupture disc or manually by operators. The overall requirement for the scrubber is that no acute lethal effects due to radiation is accepted and that the releases of Cs-134 and Cs-137 has to be limited to 0.1% of the core inventory of a 1800 MWt reactor. The release limit is about 200 TBq, using a typical core inventory from the time when the decision was issued. The reference to the 1800 MWt is based on Barsebäck 1 and 2 BWRs, which are now decommissioned, but which had the first Filtered Containment System installed in Sweden in 1985 (gravel-bed filter).

The MVSS is filled with nitrogen to provide an inert atmosphere and prevent conditions for hydrogen combustion. After a release through the scrubber nitrogen is refilled. The scrubber has a dedicated stack for releases.

In **Switzerland,** all the nuclear power plants are equipped with FCV system since the early 1990. The venting systems can be actuated manually or passively at specified pressure which is different for each plant.

In the **United States**, containment venting capabilities are limited. The USNRC have required that BWR Mark I/IIs install reliable hardened vents and are currently studying the need to include an in-line filter. Other containment types will be assessed in the future for the need of reliable vent capabilities along with the consideration to include filters.

3.2.3 Local Air Coolers/Mixing Fans

Local air coolers are used in many nuclear reactors for heat removal during normal operating conditions by air cooling, but they also provide heat removal following accidents by steam condensation. Questions have been raised concerning the use of air coolers during an accident because it may result in higher local hydrogen concentrations after steam condensation; however, it may also mitigate the risk of hydrogen clouds due to enhanced mixing. The details of each country that has local air coolers installed in their NPPs are discussed in the following. The requirements for the use of local air coolers in these countries are summarized in Table 3-5.

The **Belgian** PWRs at the Doel site are equipped with fan coolers, located close to the upper level of the cylindrical part of the containment (bottom level of the hemispherical dome), as an additional safety system used to achieve containment pressure control after an accident (DBA or SA) and for mixing the containment atmosphere to avoid hydrogen accumulation and stratification (SA).

In CANDU multi-unit stations of **Canada**, the reactor vault and fuelling duct local air coolers in each unit can remove ~2,000 kW of sensible heat under normal operating conditions and ~38 MW of sensible heat and latent heat under LOCA conditions. Eight axial flow fans, four operating and four on standby, each rated at 35.4 m^3/s and 16 air-cooling heat exchangers are located in each reactor vault. The cooling units are seismically qualified and are capable of operating under post LOCA conditions.

For CANDU 6 stations in Canada, local air coolers are provided at strategic locations inside the reactor building to keep the air temperature at acceptable levels under normal operating conditions. During post-accident conditions, the local air coolers provide a long-term heat removal capability and prevent a possible containment re-pressurization. In the long-term, after an accident, their operation will reduce containment pressure to about atmospheric pressure. Local air coolers in the fuelling machine vaults and boiler room are connected to Class III busses. The fan motors are rated to handle the steam/air mixture and the temperatures existing under accident conditions.

According to the current SA management guidelines, the preferred heat sink isolation option is to isolate the cooling water to the containment air coolers while allowing the fan motors to continue to run to assist with hydrogen dispersion and aerosol plate-out. The major concern for this option is condensation of steam in the long term raising the hydrogen concentration.

In the **German** PWRs, a special air mixing system is installed for DBAs to mix the atmosphere between the upper and lower containment part. Its use under SA conditions was not credited during the PAR concept setup. It may get recommended in the SAMGs being under final development to use it in addition to the PAR concept.

In German BWR-72 plants, no other specific strategies besides the PARs and the inerted wetwell exist. SAMGs are under final development.

In the dry-type PWR containment of **Japan**, the natural convection coolers are activated by supplying water from the component cooling water system (CCWS) when the containment spray fails and the containment pressure increases. Typically four cooler units are installed. In some older PWRs such as Mihama 1 and 2, the containment is cooled by the external spray system which is supplied water from the snow melting system. Both of these systems cool down the containment atmosphere by global and local condensation of steam, which raises a concern of global and local increase of the hydrogen concentration. A combination of instrumentations and detailed analyses are required to monitor the hydrogen concentration in the containment.

The ice-condenser type PWR containments such as Ohi-1 and 2 have relatively small volume and condensation is a main mechanism to decrease the containment pressure. To prevent the hydrogen explosion, electrical re-combiners and igniters have been already installed.

In BWRs, the excess steam in the drywell is transferred to the wet well and is condensed there. The containment is ordinary cooled by the containment spray driven by RHRS. When RHRS fails, the drywell cooler, the alternative heat removal by the reactor water clean-up system and the venting system is prepared as the accident managements. Because of inerting, there is not a significant hydrogen risk in the containment.

All the **Korean** PWRs and PHWRs are equipped the RCFC (Reactor Cooling Fan Cooler) for removing heat in the containment to maintain containment pressure and temperature below the limits. Because of operating fans, the steam may be condensed and hydrogen concentration can be higher than before. There is a likelihood of flammability in the RCFC duct. In case of using RCFC, flammable possibility should be checked by monitoring containment pressure and hydrogen concentration.

In **the Netherlands**, the Borssele NPP is equipped with air coolers. A containment depressurisation limit is respected to avoid high hydrogen concentrations when depressurising/cooling the containment.

The **Spanish** Westinghouse-PWR1000 plants are equipped by fan coolers, apart from Almaraz NPP. The rest of Spanish NPP (PWR-KWU and GE-BWR) are not equipped with fan coolers credited for DBAs.

All **Swedish** NPPs have fan coolers installed for normal operating conditions. They are not credited under accident conditions.

In the KKB PWR of **Switzerland**, 4 fan coolers (for each of the two blocks) are installed in the upper half of the single compartment containment. Two of them are on emergency power supply (diesel generator and hydro power station).

In the **United States**, fan coolers are used in BWRs and PWRs, which are credited in some pants during accident analyses. Moreover, for some containment types, e.g., PWRs with ice condenser containments and BWR Mark IIIs, there are dedicated mixing fans used during post-accident conditions to mix large containment compartments.

3.2.4 Suppression Pool

Suppression pool is a general design feature for BWR, VVER, and EPR. During SA, the steam released into the containment may condense on the water surface of the suppression pool leading to an increase in local hydrogen concentration. The considerations for the hydrogen risk in the suppression pool for the BWRs in Finland, Germany, Spain, Sweden and Switzerland are discussed in the following.

In **Finland**, the Olkiluoto 1 & 2 BWRs have suppression pools in the wet well. Because the containment is inerted with nitrogen, condensation in the pool does not cause a hydrogen risk.

In **Germany**, the water temperature of the suppression pool of BWR-72 is limited during normal operation and accidents to a maximum of about 60°C. This is guaranteed by operational and safety systems with heat removal. The pool surface area is ~450 m². The air ventilation system of the drywell and the wet well with coolers are not operated during accidents. In the emergency hand book (Notfallhandbuch) several measures are described to limit the maximum pressure in the containment by spraying into the wet well or drywell or by FCV. The later one can be used in SAs and as well in case the heat removal from the wet well fails in an accident.

The **Spanish** GE-BWR Mark I and Mark III are equipped by suppression pools. In the Mark I containment, it is located in a toroidal carbon steel vessel connected to the drywell by 8 circular vents and it is inerted with nitrogen. In the Mark III containment, it is located both in the drywell and the outer containment and three rows of horizontal circular vent communicate drywell with outer containment.

All **Swedish** BWR containments are based on the P/S design and thus they are equipped with a suppression pool in the wet well. The dry well and wet well compartments are connected with blowdown pipes, similar to the GE-BWR Mark II containments. Elevated concentrations of hydrogen are expected in the wet well during SAs, but since the containment is inerted by nitrogen any potential hydrogen combustion is prevented. The production of oxygen by radiolysis in the wet well has been addressed but analysis demonstrates that oxygen concentration will be below the limits for hydrogen combustion.

The **Swiss** BWR Mark I (KKM) and Mark III (KKL) are equipped by suppression pools. In the Mark I reactor, the wetwell (torus) has 6 vent pipes which link the drywell to the wetwell (together considered as primary containment). The primary containment is equipped with a nitrogen inerting system. In case of pressure build-up in the primary containment, the containment depressurization system will actuate and reduce the pressure. Moreover, the secondary containment includes a so called outer torus which has the function to condense steam which may be released in the secondary containment during a postulated accident. In the Mark III reactor, the suppression pool has a cylindrical shape. Part of the suppression pool water is inside the drywell, retained by the cylindrical concrete retaining wall (weir wall). The major part is outside the drywell between the outer drywell wall and the inner containment shell wall (steel). Water from the pools in the upper containment can be released into the suppression pool by opening valves under emergency conditions.

In the **United States**, all BWRs are equipped with suppression pools for pressure suppression. During postulated SAs, the hydrogen released into the suppression pool will exit from the pool surface in the wetwell region. If the hydrogen release rates are high enough, the igniters will initiate standing flames on the pool surface.

3.2.5 Latch System, Blow-out Panels and Doors

Venting is one of the common practices for pressure relief to protect the structure of a confined space. This method has been applied in some reactor designs to reduce the containment pressure during an accident, such as latch systems, blow-out panels or doors. As a result, hydrogen can also be transported to the adjacent spaces by venting. The application details are discussed in the following. Requirements of these countries in using the venting technique are defined in Table 3-6.

In the **Canadian** CANDU 6 units, there are blow-out panels to separate the accessible areas from the inaccessible areas. The panel strength is usually 6.9 kPa (differential).

In some of the Canadian CANDU multi-units, there are explosion vents for venting of the hot gases that may arise from the combustion of a fuel-air mixture within an enclosure. Explosion vents do not prevent the occurrence of an explosion, but are intended to limit the damage from the quasi-static overpressure generated by a fire or deflagration. Generally, vents cannot mitigate the consequences of a detonation, which proceeds too quickly for any venting to have effect. However, explosion vents may prevent a deflagration from transitioning to a detonation and for certain extreme geometries actually quench a detonation.

In Loviisa 1 & 2 VVER-440 of **Finland**, hydrogen mixing is ensured by forcing open the ice-condenser doors when entering into the severe accident phase. In Olkiluoto 3 EPR in Finland, hydrogen mixing is enhanced by rupture foils in the ceiling of the steam generator room and by hydrogen mixing dampers at a low elevation between the annular space and the in-containment refuelling water storage tank (IRWST).

In **France**, the latch system or blow-out panels are not applied in the current operating French NPPs. The EPR design includes a number of features that promote mixing. Heat and gas transfer to the containment heat sinks is promoted by the CONVECT system. The CONVECT system consists of rupture foils, convection foils, mixing dampers, and related instrumentation and control equipment. Many rupture foils and convection foils are placed in the ceiling of each steam generator compartment. More than half of the foils are convection foils. The dampers are located in the lower part of the containment in the IRWST wall above the water level. There are eight of these. Opening of the foils and dampers is designed to set up circulation patterns in both the equipment space and the service space. The rupture foils are passive components which will burst open if the pressure differential on the foils exceeds a predetermined value. The rupture foils burst in either direction. The convection foils are rupture foils placed in a frame. The frame is kept in the closed position by a fusible link. Should temperature rise to a set level, the link will melt with a short delay, and the frame will swing open by gravity. The result is a foil that will burst open on a pressure differential and will also open if the compartment temperature reaches a certain level.

The design of the **German** PWR containments comprises of the inner component compartments and the outer dome and peripheral area. Both are isolated from each other by air ventilation systems during normal operation. Access is only possible to the outer area during normal operation. There are typically a large number of burst membranes or burst flaps (in older NPPs) installed in both steam generator ceilings with a failure pressure of ~30 - 50 mbar, and in addition several others with larger failure pressure within openings of the missile protection cylinder, typically near or above doors. Those burst elements are designed to fail at lower pressure differences to prevent a door failure itself.

The missile protection cylinder is a heavy concrete structure separating the component rooms from the periphery of the containment.

Dependent on the accident scenario and the pressure spike in the beginning caused by a leak from the reactor circuit, the number of failed burst membranes may vary. This has a significant influence on the overall gas convection within the containment and the gas distribution and possible inerting by steam. Active measures to open flaps are known only for older NPPs being out of operation now.

In the **Japanese** ice condenser PWRs, the inlet door at the bottom is opened under accident condition where a significant amount of steam is released in the containment. The high temperature steam is cooled and condensed by a huge amount of ice. This operation will enhance mixing of hydrogen while condensation will increase the local hydrogen concentration. As is already mentioned, electrical re-combiners and igniters have been already installed.

In BWRs, a blow-out panel is attached on the reactor building wall. This panel is designed to fall off when a significant pressure acts on it under those situations such as tornados and static pressure caused by the steam line break.

In **the Netherlands**, in order to reduce the hydrogen risk in the Borssele NPP, the relief latches between the installation area and the operation area of the containment can be opened passively or manually. This enhances mixing and reduces the probability of high local hydrogen concentrations.

In **Spain**, the Trillo (PWR Siemens-KWU) NPP is equipped with blow-out panels in the steam generator rooms. These panels burst when the pressure difference between these rooms and the dome is higher than 0.5 bar. The BWR-GE Mark I and Mark III NPP are equipped by vacuum breakers to prevent excessive differential pressure between the drywell and wet well.

In the **Korean** PWR1400 (APR1400), the IRWST is used as a sink/source for feed/bleed operation during a DBA. Pressure-relief dampers (swing-panel type) are installed at the vent exits of the IRWST to remove a pressure difference between the IRWST and an annular compartment above the IRWST. Because of a possibility that a highly concentrated hydrogen/air mixture could be developed in the IRWST free volume, a 3-way valve is installed to change the flow path during a high-pressure accident such as a SBO. When a SAM for the plant is initiated, the flow path from a pressurizer to the IRWST is changed to a steam-generator compartment by turning the 3-way valve actively (pilot operated). By doing so, the hydrogen is not accumulated in the IRWST and released into the S/G compartment.

In the **United States**, the PWRs with ice condenser containments use doors connected to the ice chest. For most BWRs and PWRs, vacuum breakers are used to equalize pressure differences.

3.2.6 Containment Leakage

During an accident, potential hydrogen build-up in the reactor buildings or space adjacent to the containment due to containment leakage is also a concern for hydrogen risk management.

In the **Belgian** PWRs, the ventilation system of the annulus is equipped with a filtered air suction system which is activated during an accident leading to containment pressure increase and containment isolation. This filtered air suction system keeps a sub-atmospheric pressure in the annulus and filters the design leakages from the inner containment (using absolute filters and carbon filters) before release into the environment. Confirmation of the containment isolation is one of the immediate actions on entry into the SAMG. During a SA, the hydrogen concentration in the annulus should be low as long as no significant containment failure (enlarged leak or rupture) has to be assumed.

For the CANDU 6 reactors in **Canada**, to assist in leakage control and cleaning, the containment structure has an internal lining comprising a flexible coating applied to the inner surfaces of the pre-stressed concrete dome, floor and walls. The containment structure and all other parts of the containment boundary are pressure tested for leakage before first criticality and are periodically retested. Standby active components of the system such as isolating valves, dousing valves and their control logic are tested regularly to demonstrate their availability and leak tightness. All containment closures are given functional tests during operation. Functional operation of all airlock doors is checked on a regular basis. The leakage rates of airlocks and ventilation system isolation dampers is checked periodically by pressurizing the inter-space between doors or dampers and measuring the pressure rundown to confirm leakage limits are satisfied The operating limit on containment leakage is at 0.5% free volume per day at the design pressure of 124 kPa (d).

For the CANDU multi-units in **Canada**, the concrete containment envelope, including the reactor vaults, fuelling duct, fuelling machine head removal area, fuel handling and service areas, pressure relief duct, floor of the vacuum structure, bottom 6.5 m of the vacuum structure perimeter wall, are steel-lined to reduce leakage. The containment envelope leakage rate limit is 1% (operational target) of the total enclosed free volume per hour at an internal pressure equal to the containment positive internal design pressure. This is a reasonably achievable rate based on past plant experience and is well below the leakage rate required to meet the dose limit.

In the Loviisa 1 & 2 VVER-440 and Olkiluoto 3 EPR in **Finland**, containment leakages are collected in the reactor building, which acts as a secondary containment. From there the release goes to the stack through filters.

In **France**, for the reactors in service, confirmation of the isolation of the containment penetrations is required as part of the immediate actions on entry into a SA situation. The activity is monitored so that restoration measures can be implemented if necessary. The U2 operating procedure (continuous monitoring of containment integrity) which is part of incident/accident operating procedure is applicable in a SA situation. Its aim is to monitor the containment integrity under accident conditions and if necessary restore the reactor containment (by isolating the areas concerned, reinjection of highly radioactive effluents, etc.). On the EPR, the containment and the peripheral buildings are designed such that there is no direct leakage path from the reactor containment to the environment. The building ventilation systems are backed up by the main diesel generators and the ultimate backup diesel generator sets.

The French PWR1300 and PWR1450 containments have a double concrete wall without metallic liners. The annular space between the two concrete walls is kept below atmospheric pressure and filtered using a specific device.

During the ex-vessel phase, hydrogen and carbon monoxide may be transported through the cracks from the inner containment to the annular space and lead to the formation of flammable atmosphere. Until now, no hydrogen mitigation means was installed in the annular space.

On completion of the Complementary Safety Assessments (CSAs), EDF undertook to study the hydrogen risk in the other peripheral buildings of the reactor containment. The study of the hydrogen risk in the inter-containment space of the 1300 MWe reactors is in progress as part of the PSR associated with their third 10-year inspection.

The **German** PWRs have a large annulus between the containment and the reactor building where the ECCS components are located, e.g., a special filtered air suction system is installed, to get activated in an accident after the regular ventilation systems have been switched off. This keeps a sub-pressure in the annulus and allows a filtered release of aerosols released by the containment design leakage. The hydrogen concentration should stay low as long as no significantly enlarged containment leakage has to be assumed.

There is a big reactor building surrounding the German BWR type 72 NPPs. Two specific systems exist to keep a sub-pressure in the reactor building during accidents and to control the leakages of the containment.

In **Japan**, the hydrogen explosion in the Fukushima Dai-Ichi NPS Unit-4 showed importance of focusing on the expansion of hydrogen risk due to gas transportation among adjacent facilities. Venting operation procedures should include prevention of inverse flow via other parallel lines such as the SGTS in case of BWRs. Sharing a common stack should be avoided as far as possible. When it is shared, inverse flow should be avoided by adding no-return valves and duly closing them before venting. The same level of care should be taken for other discharge paths connected to the outside of the containment including the FCV system lines where a significant amount of hydrogen will be built up under long-term accident scenarios.

In **Spain**, containment leak rate is limited by Performance Technical Specifications (PTS). There is no specific requirement for leakages limits in SA. In the framework of the “stress-test”, analyses of potential hazards in other buildings surrounding the containment have been performed. These studies consider that PAR systems have been installed.

The Beznau PWR NPP in **Switzerland** has a double-containment. The large annulus between the primary steel containment and the outer, secondary steel lined concrete safety building is equipped with an emergency power supplied containment pump back system with 2x100% capacity, sufficient to pump back any leakage from the primary containment within the limits of the technical specifications. Other plants have different measures to maintain leakage within the limit of technical specification.

In the **United States**, periodic leak rate tests are performed to assure that the flow rates are below technical specification limits.

3.3 Hydrogen Measurement Strategies

The implementation of hydrogen measurement systems has been considered (or is being considered) as a part of hydrogen management strategies in many countries, particularly following the Fukushima accident. Hydrogen concentration measurements can be useful:

- For the administrative authorities to have knowledge about the accident progression,
- For the crisis teams to avoid an inadvertent CHRS spray actuation for example that could lead to a risk of loosening the containment tightness in case of hydrogen deflagration.

In some countries active measures are applied in the accident management concepts.

In the **Belgian** NPPs, hydrogen concentration measurement systems have been installed inside the containment for DBA. In addition, sampling for hydrogen concentration measurement during a SA, with a measurement range up to 20 vol.% of hydrogen, is installed and used in the SAM procedures of the Doel units.

For the **Canadian** NPPs, a SA sampling/monitoring system has been put in place in the refurbished Point Lepreau NPP (CANDU 6 station). Sample air from the reactor building can be collected in a sample line, brought into the sampling building via a penetration, sampled, analyzed, and finally exhausted back into the reactor building. Iodine and noble gases can be analyzed from the extracted sample and in-line monitoring of hydrogen is also performed. The on-line gross gamma monitoring system will be capable of measuring high range gamma gross dose rates present in the reactor building sample air during and after a SA. Significant thought was given to the installation location based on the fact that the components and sampling lines will be extremely radioactive following an accident. Easy access to the room is essential in order to transport the sample to a suitable location for analysis. The system is designed for BDBA and SA. During a SA, it is assumed that the containment atmosphere will be homogenous making sample extraction from one point acceptable. The gross gamma monitoring system and the hydrogen monitoring system can be in continuous operation for the 90 day mission time following a SA. Periodic gaseous and iodine samples will be extracted for spectrometric analysis at a remote location. Implementation of hydrogen sampling system in other Canadian NPPs is under consideration in response to the Fukushima accident**.**

For the Olkiluoto 3 EPR in **Finland**, hydrogen concentration in the containment can be measured by a sampling system. The measurement system consists of two redundant subsystems with four separate sampling locations each. The system is designed to withstand SA conditions.

For the Loviisa VVER-440 in Finland, there is a containment hydrogen concentration measurement system, which is not qualified for hydrogen burning conditions (high temperature), but additional protection of the measurement device against hydrogen burns has been introduced.

For the Olkiluoto 1 & 2 BWRs in Finland, there is a system for measuring hydrogen concentration in the containment, but the system cannot operate if the containment pressure exceeds 2 bar. In that case a sample can be taken from the drywell by means of a syringe.

In **France**, in order to ensure fission product confinement, there is a common agreement to avoid sampling for hydrogen concentration measurement purposes during SA. Consequently, no precise measurement of hydrogen concentration in the containment is available.

Nevertheless, it was decided to equip two PARs per containment with thermocouples. The increase of temperature signal delivered by those thermocouples will inform on a tendency of the hydrogen production and hydrogen concentrations inside the containment

In **German** NPPs, probe sampling systems and hydrogen and oxygen concentration measurement systems have been installed inside the containment. The systems are especially designed for SAs and have a measurement range of up to 10 vol.% of hydrogen. In German PWRs, about 10 different measurement locations exist inside the containment.

In **Japan** under the new regulatory requirements, it is required to monitor and record temperature, pressure, hydrogen concentration and dose rate under accident conditions in the containment vessel and reactor building (annulus of the containment vessel for PWRs). In PWRs, it is proposed to install a mobile hydrogen meter based on the electrical conduction method. This system is connected to the existent gas sampling lines. Other monitoring methods such as the gas chromatography connected to the sampling line can be applied. Solid electrolyte type hydrogen meters are also under development for PWRs and BWRs under SA conditions. In addition, a new concept of hydrogen monitor based on ultrasonic wave technology is being developed. A risk of fission product leakage and reliability of limited sampling locations are major concerns of these methods. Monitoring PAR temperature can be a supplemental method.

The **Korean** PWRs and PHWRs are equipped with PARs in the containments to reduce hydrogen concentration under SAs. During SAs, hydrogen concentration is monitored with other related information such as pressure and temperature of the atmosphere in the containment to understand the status of a plant. Though PARs are operating automatically with change of hydrogen concentration, the operations of other related systems are accompanied such as a spray system, a containment (filtered) vent and local air coolers depending on the status of a NPP.

In **the Netherlands**, the Borssele NPP has a containment hydrogen measurement system (TS090) with sample points in the operational area and in the installation area. The hydrogen measurement instrumentation is harsh environment qualified. Besides this, the use of hydrogen measurement instrumentation, backup systems such as TV090 hand sampling and TV061/062 systems is included in the SAMG strategies.

In **Spain**, the Trillo (PWR Siemens-KWU) NPP is equipped by 11 post-accident hydrogen detectors, located in the dome, loop rooms, pressurizer room and pressurizer valves room. They are based on catalytic reaction and are supported by the DC system. The BWR-GE Mark III NPP are equipped by two redundant sampling channels for hydrogen concentration measurement, one from the drywell and the other for the outer containment, they are supported by AC system. Westinghouse-PWR1000 plants are equipped with 8 detectors supported by AC power, and they are located in representative regions. Additionally this containment design, in general, is equipped with a hydrogen sampling system.

The **Swedish** PWRs have instrumentation for on-line hydrogen measurement. In BWRs there is instrumentation for on-line hydrogen measurement during normal operation, while during accidents a Post-Accident Sampling System would be used for determining gas composition of containment atmosphere.

In **Switzerland**, the KKL NPP (Mark-3 Containment) is equipped with redundant Containment Atmosphere Monitoring (CAM) systems that measure (by sampling) the hydrogen content in Drywell and Containment during accident conditions, in the range of 0-10 vol.%.

In the KKM (Mark I containment) plant, the hydrogen concentration is continuously measured in the containment (drywell and wetwell). Furthermore as direct consequence from the TMI accident a Post-Accident Sampling System has been installed. This system allows collection of samples of water (RPV or wetwell) and air (drywell, wetwell and reactor building/secondary containment) independently from sampling system for normal operation.

The Swiss PWR NPP Beznau is equipped with five hydrogen detectors inside containment. Two of these detectors are part of the accident instrumentation of the plant. An additional post accident sampling system allows taking samples manually at three different elevations (top, middle, and bottom) of the containment and is also available after SBO.

In the **United States**, all the NPPs utilize appropriate hydrogen measuring techniques for operator actions.

3.4 Lessons Learned from Fukushima for Hydrogen Management

Every country has taken (or is planning to take) some actions to address questions raised after the Fukushima accidents. The details are described in the following.

In **Belgium**, there are many lessons learned from the Fukushima accident and the subsequent “stress test”. Some lessons learned are related to hydrogen management during a SA. The existing PARs are considered sufficient for hydrogen management in the inner containment during SAs. Nevertheless, the measurement of hydrogen concentrations during a SA and the management of hydrogen flammability will be included in the SAMG of the units where this strategy was not implemented. As part of the national action plan of the “stress test”, hydrogen risks in the SFP area or due to hydrogen leaks from the inner containment to other peripheral buildings will also be investigated. Furthermore, installation of a FCV is now required and its design should prevent any additional hydrogen risk in the venting path.

In **Canada,** in response to the Fukushima accidents, the CNSC immediately performed inspections of all NPPs and other nuclear facilities in Canada to assess the readiness of mitigating systems, including seismic preparedness, firefighting capability, backup power sources, hydrogen mitigation and irradiated fuel bay cooling. It also established a Task Force to evaluate the operational, technical and regulatory implications of the accident and the adequacy of emergency preparedness for NPPs.

The CNSC Task Force has a number of recommendations for strengthening each layer of defence built into the Canadian NPP design and licensing philosophy. In particular, the CNSC Task Force recommends that certain design enhancements for SA management – such as containment performance to prevent unfiltered releases of radioactive products, control capabilities for hydrogen and other combustible gases, and adequacy and survivability of equipment and instrumentation – should be evaluated and implemented wherever practicable.

For CANDU multi-unit NPPs, the previous analysis was based on a computer model that represents only a single unit. The effect of a SA simultaneously in four units is approximated by modelling the shared part of the containment as one quarter of its true size. This approach is accepted to give broadly representative results of the accident source term but would not be capable of calculating, for example, the effects of different times of core meltdown in the different units. In some effects, the current modelling may be conservative. For example, all four units would be very unlikely to experience identical failures, such as vessel failure, at the same moment, thus the corresponding containment pressure increases may be over-predicted. Other effects may not be captured, such as higher than average local hydrogen concentrations. The CNSC has required the licensees to develop the capability to model SAs in multi-unit NPPs.

The spent fuel pools at most NPPs were not designed to accommodate boiling in the pool; therefore a loss of cooling can be tolerated for about 16 hours before the structural design temperature is reached. Above this temperature there is an increasing risk of structural cracking which could lead to leakage from the bay. The CNSC Task Force recommends that licensees develop a strategy to mitigate these concerns by demonstrating that procedures and equipment are in place to provide make-up water that will compensate for possible leakage.

In **Finland**, installation of latches in the Olkiluoto 1 & 2 BWRs, which could be opened during a spent fuel pool accident for venting of steam out of the reactor building, is being considered. These latches could be used also to vent hydrogen out of the reactor building, if needed.

In **France**, following the Fukushima accident, a stress-test review was conducted. Based on the IRSN review and position endorsed by the French Safety Advisory Group, the French Safety Authority (ASN) established some requirements (see http://www.ensreg.eu/EU-Stress-Tests/Country-Specific-Reports/EU-Member-States/France for additional details). Some of them are given in the following.

The ASN has asked the EDF to examine the risks linked to the build-up of hydrogen in the buildings other than the containment, especially the fuel building. ASN in particular asked EDF to identify:

- The phenomena capable of generating hydrogen (radiolysis, zirconium/steam reactions);
- The possible build-up of hydrogen;
- The means implemented to prevent hydrogen explosion or detonation.

As part of the CSAs, EDF states that the presence of fuel assemblies in the spent fuel pool can lead to the production of hydrogen in normal operation by radiolysis of the water and that an additional analysis is being initiated to assess the possible risk in the absence of ventilation. EDF also states that oxidisation of the cladding by steam would lead to the production of hydrogen in sufficiently large quantities to exceed the flammability threshold, but that bearing in mind the means used to prevent uncovering of the fuel assemblies, the risk of hydrogen production by oxidisation of the zirconium cladding is ruled out. EDF therefore considered undertaking thermal-hydraulic studies on the fuel storage pools during the period after the Fukushima Dai-ichi accident, taking account of the different behaviour of the various areas of the spent fuel pool. In accordance with the hydrogen risk studies, particular steps may need to be taken depending on the result of these studies, such as the installation of PARs in the fuel building. These studies cover both the NPP fleet in service and the EPR. ASN considers these studies to be necessary in order to determine the material and organisational measures that could be taken on the NPPs in operation and on the EPR, such as the installation of PARs in the fuel building".

In **Germany**, there are many lessons learned from the Fukushima accident, some of which are related to hydrogen management [3.22]. The existing provisions (PARs, inerting) are sufficient for accidents within the containment. It was recommended to recheck the filtered venting systems, especially its off-gas systems related to the hydrogen combustion issue.

SA in spent fuel pools is a new area. In German PWRs the spent fuel pool is located inside the containment. Even so SAs within the spent fuel pool have not been considered for PAR concept development, the installed PARs will protect the containment as well in such cases.

In BWRs the spent fuel pool is located inside the reactor building above the containment. After the Fukushima accident the BWR type 72 utility decided to install PARs as well above the spent fuel pools [3.22] or [A.2]. Details are not yet presented.

In **Japan**, the new regulatory standards taking the Fukushima accident experience into the consideration had come into force on July 2013. Major causes of the Fukushima accident were a coincident with a station black out and loss of a heat sink due to tsunami attack after the earthquake.

Therefore, all NPPs to be considered to restart are required to prepare alternatives of a power supply and cooling measures. BWRs are required to prepare a FCV in order to suppress the containment pressure increase in early stage of the accident [3.23].

Further investigation of the Fukushima accidents suggested the following additional measures.

- Preventing a backward flow of hydrogen due to the PCV venting
  Unit-3 and Unit-4 share the same stack as neighbour plants. In the operator manual, there was no instruction of isolating the line connected to another unit when conducting the PCV venting. In addition, check valves preventing a backward flow were not implemented for Unit-4. It is therefore necessary to install equipment to prevent a backward flow to the building through the connecting pipe under the PCV venting.
- Ensuring independence of pipes belonging to PCV venting
  In order to prevent hydrogen inflow into R/B through other parallel lines or the vent line of different units, it is desirable to make vent lines independent from them. Independency of venting lines among different units is ensured by prohibiting sharing the same stack. Under accidental conditions, it is necessary to examine influences of sharing components among different units.
- Preventing hydrogen explosion in the reactor building
  In order to prevent hydrogen explosion in the reactor building, it is effective to add openings at the top and the ground floor of the building. However, opening of the blowout panel leads to direct release of radioactive materials to environment. As a consequence, a course of countermeasures needs to be studied with expecting the utilization of the hydrogen discharging facilities for decreasing total amount of radioactive release or the hydrogen re-combiner systems for decreasing concentration of hydrogen.

In **Korea**, following the Fukushima accident, safety inspection was carried out on all Korean NPPs by Korean government. As a result, 50 recommendations for safety improvement were suggested. Hydrogen management was included among the recommendations. According to the recommendations, PARs will be installed in all operating and under-construction plants by the end of 2013.

The **Netherlands** fully participated in the post-Fukushima Daiichi activities led by ENSREG and by the IAEA (under the CNS umbrella). The Licence Holder (EPZ), operating the Borssele NPP, also participated in two post-Fukushima WANO self-assessment exercises. The results of these assessments are included in the 'stress test' results. The lessons learned with respect to hydrogen management are the following:

- Re-evaluation and extension of hydrogen analysis of the containment (Evaluate H2-combustion vulnerability and H_2-management in several cases), including 3D CFD analyses.
- Examination of measures to remove hydrogen in other places/buildings than the reactor building
- Develop specific SAMG for spent fuel pool

In **Spain**, after the Fukushima accident, actions in the framework of so-called "Stress tests" were initiated [3.24]. The aim was to verify the safety measures at their plants. Related to hydrogen the following actions have been stated:

- Implementation of actions to cope with prolonged SBO, including new equipment (fixed or mobile) to make up water to RCS and spray system.
- Implementation of containment filtered system.
- Implementation of PARs.
- Analysis of hydrogen potential hazards in other buildings surrounding the Containment.
- Analysis of the impact of existing strategies for containment flooding over equipments (instrumentation necessary for Accident Management) placed inside it.
- Analysis of Level 2 PSA in "other modes of operation".
- Implementation of an alternative electric supply to the existing containment hydrogen igniters in BWR-Mark-III.

In the **Swedish** National Action Plan, which is a response to the ENSREG Action Plan within the Peer Review of the European Stress Tests, a number of issues have been identified which require investigations in order to determine and consider which measures are fit for purpose, how they shall be implemented as well as the point in time for this. It is expected that the majority of necessary technical and administrative measures will be implemented between 2015 and 2020. The following hydrogen related issues are listed in the Action Plan:

- The management of hydrogen in the containment and in the reactor building in a long-term.
- The possibility and consequences of accumulating hydrogen in the reactor building, including necessary instrumentation and management.
- SAM measures to cope with hydrogen production in the SFP.
- SAM measures for handling hydrogen risk in BWRs during outage with non-inerted containment. The risk contribution is assessed and quantified in the PSA Level 2 analysis for shutdown conditions.
- Re-evaluations of emergency procedures and implementation of possible updates regarding hydrogen venting using FCV system.
- Decision support for handling hydrogen in a lengthy accident sequence. Among the issues to be addressed are gas composition in different volumes in the containment, re-inerting of the containment after filtered venting and elimination of hydrogen gas by combustion.

In the aftermath of the Fukushima, the **Swiss** Federal Nuclear Safety Inspectorate (ENSI) required the Swiss NPPs:

- to reassess the hydrogen risk. These reassessments are ongoing, among others, specifically for analysing hydrogen measurements, reviewing FA and DDT, a more detailed analysis of the distribution of hydrogen in the containment, analysing migration of hydrogen out of the primary containment into adjacent buildings including scenarios like Bypass-LOCAs, loss of containment isolation and situations during shutdown of the plant, and analysis of hydrogen explosions in the FCV path.
- to develop a concept for the installation of PARs in the containment, in combination with igniters where applicable. NPPs already equipped with PARs have to review their concept given the current state of the art.
- NPPs with a primary containment which is nitrogen-inerted do not have such a requirement.

In the **United States**, current USNRC regulations focus on hydrogen mitigation within the primary containment only. However given the events at the Fukushima, the USNRC has embarked on lessons learned campaign:

- Assessment of hydrogen control measures and potential hydrogen ingress into adjacent buildings
- Evaluate the Fukushima Dai-ichi accident sequences with particular emphasis on hydrogen generation from all sources and timing.
- Review information that becomes available in the near term on potential containment release pathways for hydrogen ingress into the respective Fukushima damaged reactor buildings.

Assess the technical basis for NRC’s existing hydrogen generation and control requirements in [3.1] against the results of the above three bullets.

Following the Fukushima accidents, the European Council requested a comprehensive safety and risk assessment performed on all EU nuclear plants, including “stress tests” performed at national level complemented by a European peer review [3.25]. Fifteen EU countries with nuclear power plants as well as Switzerland and Ukraine performed the stress tests and were subjected to the peer review. The peer review concluded that all countries have taken significant steps to improve the safety of their plants, with varying degrees of practical implementation. In spite of differences in the national approaches and degree of implementation, the peer review showed an overall consistency across

Europe in the identification of strong features, limitations and possible ways to increase plant robustness in light of the preliminary lessons learned from the Fukushima disaster.

For hydrogen management, it is concluded that high priority must be given to installing means for hydrogen mitigation designed for SAs, in order to practically eliminate containment failure due to hydrogen combustion. Installation of PARs seems to be the preferred option for future upgrading.

The peer review also identified a number of possible measures to increase plant robustness, including two for hydrogen management [3.25]:

- Since hydrogen flammability depends on the composition of the containment atmosphere, which in turn depends on the operation of other systems such as the containment spray system, qualified monitoring of the hydrogen concentration must also be available to avoid such operation when concentrations that allow explosion exist.
- The potential for migration of hydrogen into spaces beyond where it is produced in the primary containment, as well as hydrogen production in SFPs, has to be carefully analysed and adequate countermeasures adopted if necessary.

3.5 Hydrogen Assessment Methodology and its Application to PSA

The methodology adopted by different countries to assess the hydrogen risk in the containment as well as in the reactor or auxiliary buildings has many similarities, but difference exists in the numerical tools and correlations used as well as in the accident scenarios considered, examples of these methodologies are the following references [3.26], [3.27] to [3.30]. In Reference [3.26], the authors proposed a methodology to evaluate the impact of PARs on the hydrogen risk in French 900 MWe reactor containment in case of a SA and its application to a Level 2 PSA. A slightly modified methodology is presented in the following, which is developed based on reference [3.26] and the methodologies implemented by the participants. The purpose is to provide an example for hydrogen assessment that can be considered by the member countries.

3.5.1 Hydrogen Assessment Methodology

This proposed methodology consists of the following seven steps:

Step 1: Plant design

The starting point of any analysis is the selection of the plant and geometrical modelling of the containment or the reactor and auxiliary building. This step is aimed to well describe the building shape and volume as well as the safety systems activated (e.g., PARs, spray, coolers, etc.).

Step 2: Selection of relevant scenarios

The representative or bounding SA scenarios for hydrogen assessment are generally identified from PSA or other analysis. The evaluation of the associated hydrogen production rates and release into the containment or reactor building is usually derived from parametric code calculations with best estimates for still uncertain hydrogen production processes.

Step 3: Evaluation of the containment atmosphere condition

During the accident transient, the temperature, pressure and gas composition in the different regions and volumes of the containment or reactor building are determined accounting for the presence of mitigation systems. For this purpose, both LP and 3D codes are often used to simulate the transient. Usually, 3D codes are used for deeper analysis in short-term temporal windows, e.g., hydrogen release phase, scenarios with local accumulation and stratification expected.

Step 4: Evaluation of the time evolution of flammable hydrogen-air-steam cloud

The flammability of the containment or reactor building gas mixture depends on its temperature, pressure and composition. However, in practice, the point representing the mixture's composition (hydrogen, air, and steam) on the Shapiro diagram (see Figure 1.2.3-2) is used to determine whether the mixture is flammable. In this diagram, the flammability and detonation zones are, respectively, delimited by the exterior and interior curves.

Step 5: Evaluation of the propensity of a premixed flame to propagate inside the building

Under the effect of hydrodynamic instabilities and turbulence (caused primarily by obstacles in the flame's path), an initially laminar deflagration (with a flame velocity around 1 m/s) may accelerate. Fast combustion regimes may also develop, involving rapid deflagration (a few hundred m/s), DDT and detonation (over 1000 m/s). These combustion regimes may generate high pressure loads which could endanger the containment integrity.

To define the transition from slow to fast combustion regime, two types of criteria are considered:

i. The "σ" criterion related to FA.
σ stands for the mixture's expansion factor, a ratio of unburnt verse burnt gas densities at constant pressure. It is an intrinsic property of the mixture. The critical value σ^*, beyond which FA is possible, depends on initial gas composition and temperature and flame stability (see Figure 1.2.3-1).

ii. The "λ" criterion related to DDT.
Similarly, prerequisite conditions have been defined for characterizing the transition between deflagration and detonation regimes. They are based on comparing a characteristic dimension of the geometry with detonation cell size λ (see Figure 3.5.1-2).

FA and DDT criteria are based on the results of numerous experiments at various scales and in various geometries [1.50] and are considered as prerequisite criteria, i.e., conditions required for the various combustion modes.

Step 6: Evaluation of pressure and thermal loads generated by combustion

Two configurations are distinguished:

i. If the FA criteria are not met, dynamic pressure loads are excluded and the pressure load is evaluated by considering the adiabatic isochoric complete combustion (AICC) process.

ii. If FA criteria are met, the induced combustion loads are evaluated using the most appropriate combustion models, mainly CFD codes.

Step 7: Evaluation of the structure response to pressure and thermal loads generated by combustion

Structural integrity of the containment or the reactor building can be evaluated using dedicated dynamic structural response codes.

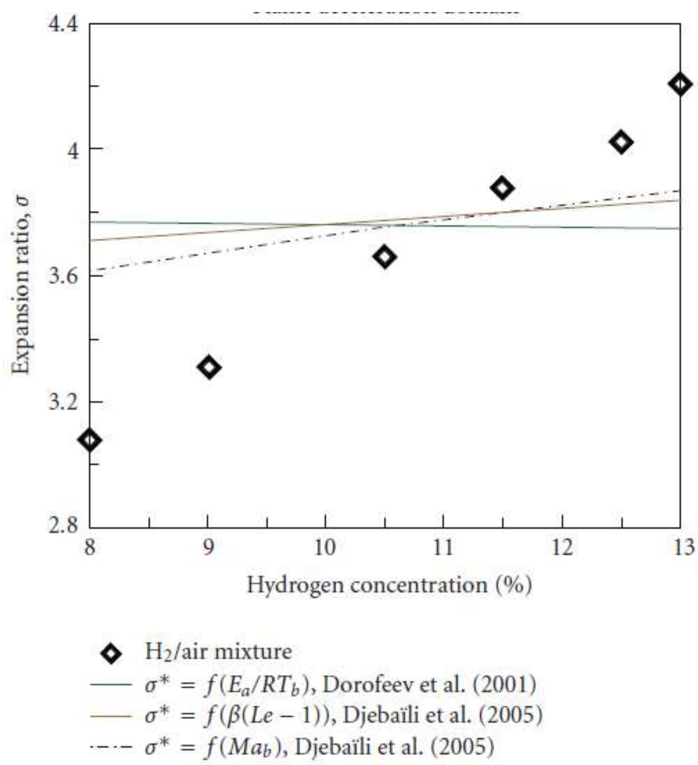

Figure 3.5.1-1 Critical value σ* as a function of hydrogen concentration (reproduced from [3.26]

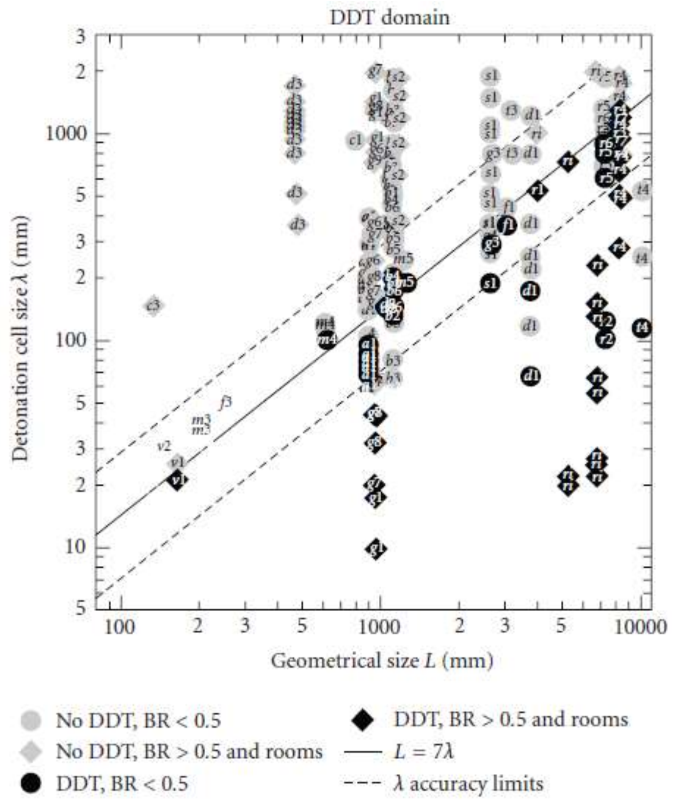

Figure 3.5.1-2 DDT criteria (reproduced from [3.26])

3.5.2 PSA Level 2 Assessment of Performance of Mitigation Measures

In France, the demonstration of the safety of NPPs is based firstly on a deterministic approach, by which the operator guarantees the resistance of the installation to reference accidents. This approach is supplemented by PSA based on a systematic examination of the accident scenarios to assess the probability of arriving at unacceptable consequences. They provide a global view of safety, integrating the resistance of the equipment and the behaviour of the operators. The PSAs help to determine whether the measures adopted by the licensee are satisfactory or not. They enable the safety problems relating to the design or operation of the reactors to be prioritized, and constitute a means of dialogue between the licensees and the administration. For the existing reactors, the PSAs are carried out and updated during the 10-year reviews. For this purpose and in framework of Level 2 PSA (PSA-2), several SA scenarios had been investigated with and without PARs in order to evaluate the impact of PARs on hydrogen risk. The analysis [3.31] shows that the use of PARs permits, in most cases, to avoid the formation of flammable atmosphere, see Figure 3.5.2-1 .

Despite the installation of PARs, the same analysis shows that it is difficult to prevent, at all times and locations, the formation of a combustible mixture potentially leading to local FA. Moreover, the ranking of the L2 PSA scenarios, see Table 3-7 for PWR 1300, using a basic risk metric: accident frequency × consequence amplitude, shows that the containment failure due to hydrogen combustion is still non-negligible even with the use of PARs.

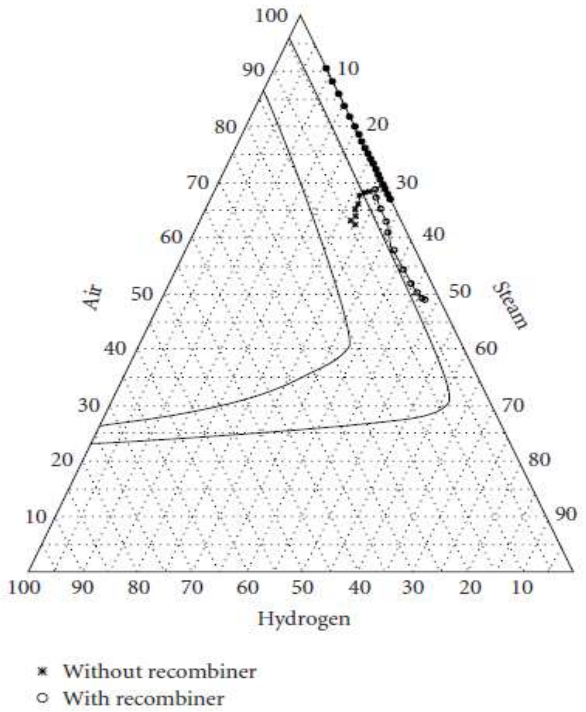

Figure 3.5.2-1 Effect of PARs on gas mixture flammability (stars: gas composition without PARs; circle: gas composition with PARs)

A similar approach is also adopted in **Finland** where the safety regulations require that “the level 2 PSA shall analyse amount and timing of the occurrence of hydrogen generated during various accident sequences, the spreading of hydrogen in the containment, and the likelihood and impact of hydrogen combustion and burning”.

A similar approach is also adopted in **the Netherlands**, where the effect of the PARs is taken into account in the PSA-2 assessments of the Borssele NPP

In **Japan**, feasibility of measures against hydrogen risk is proved based on the graded approach that is composed of the initial safety assessment and the periodical safety improvement assessment. In each assessment, probabilistic and deterministic approaches are combined. Possible containment failure modes are identified for each plant damage state based on the level 1 and level 2 PSAs. In extracting representative accident sequences for each containment failure mode, frequency and consequence are recognized as major indices. For the hydrogen explosion in PWRs, a combination of large break LOCA (failure of high and low pressure injection systems), early core degradation, and success of containment spray, is extracted. Activation of the containment spray is recognized as a conservative factor since it will cause condensation of steam and larger water inventory will result in larger amount of radiolysis. Effect of proposed measures such as PARs, igniters and N_2 inerting is evaluated based on the deterministic approach based on the above-mentioned scenarios. Additional sequences can be added unless the representative sequences will envelop other postulated sequences. The best-estimate computer models such as MAAP, MELCOR and GOTHIC can be applied in this evaluation. An adequate level of conservatism should be taken into account in input parameters and initial and boundary conditions to compensate for uncertainties. Before and after statuses of plant safety are compared by level 1 and level 2 PSAs. By conducting this PSA basis periodical assessment, the state-of-knowledge of the hydrogen risk management can be timely reflected in the plant safety improvement activities by utilities.

In **Spain**, the potential risk reduction in PWR-KWU-1000 by PAR implementation was quantified in the PSA Level 2 analyses. The overall risk (base on the so-called risk activity) decreased by almost a factor of 10.

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Table 3-1
National Requirements for Hydrogen Management inside the Containment

Country/NPPs	Requirements
Belgium/PWRs	Avoid combustions challenging the containment integrity Design criteria: mean H_2 < 4 vol.% for DBA, mean H_2 < 5 vol.% for SA, no criterion for local H_2 concentration.
Canada/CANDU 6	mean H_2 < 6 vol.% for DBA, 8 vol.% for BDBA/SA, P<3.35 bar (a) (failure of airlock seals); demonstrate containment function to be maintained.
Canada/Multi-unit	mean H_2 < 4 vol.% for DBA, 8 vol.% for BDBA; no hydrogen concern during short term hydrogen release to avoid global combustions challenging the containment integrity.
Czech Republic	Design of hydrogen removal system based on evolution of hydrogen concentrations, criteria for FA and DDT, and AICC pressure.
Finland	Gas burns that may jeopardise containment leak tightness shall be prevented.
France/PWR900	mean H_2 < 8 vol.%, local H_2 <10 vol.%, PAICC < 5 bar
France/PWR1300	mean H_2 < 8 vol.%, local H_2 <10 vol.%, PAICC < 4.8-5.2 bar
France/PWR1450	mean H_2 < 8 vol.%, local H_2 <10 vol.%, PAICC < 5.3 bar
Germany/PWR	Avoid global combustions challenging the containment integrity
Germany/BWR-69	Oxygen concentration below 4 vol.% in wetwell and drywell
Germany/BWR-72	Oxygen concentration below 4 vol.% in wetwell Avoid global combustions challenging the containment integrity by PARs
Japan/PWRs/BWRs	Criteria for preventing the destructive detonation are interpreted as maintaining the mean and local hydrogen concentration at 13 vol.% or less without steam condition or the mean and local oxygen concentration at 5 vol.% or less.
Korea/PWRs/PHWRs	Mean H_2 < 10 vol.%; local H_2 concentration should be low to avoid wide-scale FA or DDT. For static combustion load (AICC), KEPIC requirements for containment integrity (such as the Factored Load Category of ASME, Sec. III) should be satisfied
Spain	Eliminate the possibility of deflagration or detonations that threaten the containment integrity
The Netherlands/PWR 500	Avoid global combustions challenging the containment integrity
Sweden	The oxygen content has to be below 1-2 % in the BWR containments during normal operation. No specific requirement for hydrogen management. A general requirement regarding the robustness of the containment in accident situations, including SAs, implicitly addresses the hydrogen threat.
Swiss	Prevent hydrogen concentrations endangering the containment by an ignition
US/BWR Mark I/IIs	Conditions not to support combustion
US/BWR Mark IIIs/ PWR Ice Condensers	Install system to accommodate hydrogen equivalent to 75% MWR of the active cladding
US/PWR large dry containments	No system required
US/ALWR containments	Inert the containment atmosphere, or limit hydrogen concentration to less than 10 vol.% following an accident that releases an equivalent amount of hydrogen from a 100% fuel clad-coolant reaction, and maintain structural integrity.

Table 3-2
Summary of Mitigation Measures inside Containment

Country/NPPs	Hydrogen Mitigation Measures
Belgium/PWRs	AREVA PARs (~25 in the 440 MWe units ; ~40 in 1000 MWe units)
Canada/CANDU 6 (Point Lepreau)	19 AECL PAR; Local air cooler and dousing
Canada/Bruce (A &B)	64 glow plug with 22 AECL PAR each station; Local air cooler and dousing
Canada/Pickering A	24 glow plug with 20 AECL PAR; Local air cooler and dousing
Canada/Pickering B	64 glow plug with 120 AECL PAR; Local air cooler and dousing
Canada/Darlington	232 glow plug with 34 AECL PAR; Local air cooler and dousing
Finland/Loviisa 1&2 (VVER-440)	154 AECL PAR & glow plug igniters in the lower compartment
Finland/Olkiluoto 1&2	Containment inerted with nitrogen
Finland/Olkiluoto 3 (EPR)	50 AREVA PAR
France/PWR900	24 AREVA PARs (111.6 kg/h at 1.5 bar and 4 vol.% H_2)
France/PWR1300	116 (50 without chimney and 66 with chimney) AECL PARs (109 kg/h at 80°C and 4 vol.% H_2)
France/PWR1450	
German PWR KONVOI	65 AREVA PARs (~190 kg/h at 1.5 bar and 4 vol.% H_2)
German PWR Pre-KONVOI	90 NIS PARs (~190 kg/h at 1.5 bar and 4 vol.% H_2)
German BWR-72	78 NIS PAR in drywell and wetwell (~133 kg/h at 3 bar and 4 vol.% H_2) wetwell inerted by N_2
Japan/PWRs	5 AREVA PAR (1.2 kg/h per unit at 1.5 bar and 4 vol.% H_2) 11 Glow plug type igniters.
Japan/BWRs	BWR5 / ABWR: ~50 PARs in the reactor building (0.138kg/h per unit at 1.0 bar and 2 vol.% H_2)
Korea/PWR587	18 AECL PAR (72kg/h at 8vol.% H_2)
Korea/PWR650	24 Ceracomb (Korea) PARs (126k g/h at 8 vol.% H_2)
Korea/PWR950	23 Ceracomb (Korea) PARs (136.8 kg/h at 8 vol.% H_2)
Korea/PWR1000	24 Ceracomb (Korea) PARs (129.6 kg/h at 8 vol.% H_2)
Korea/PHWR700	31 Ceracomb (Korea) PARs (104.4 kg/h at 8 vol.% H_2)
Korea/PWR1000	21 KNT (Korea) PARs (93.6 kg/h at 8 vol.% H_2)
Korea/PWR1400	30 KNT (Korea) PARs (158.4 kg/h at 8 vol.% H_2) and 10 glow plugs
Korea/PHWR679	27 KNT (Korea) PARs (86.4 kg/h at 8 vol.% H_2)
The Netherlands PWR 500	22 Siemens/KWU PARs (119 kg/h at 1 bar)
Spain/PWR-KWU-1000	32 AREVA PAR (142.6 kg/h at 1.5 bar and 4 vol.% H_2)
Spain/BWR-GE-MarkI	Containment inerted with nitrogen
Spain/BWR-GE-MarkIII	36 igniters to deal with a metal-water reaction involving 75% of zirconium in the core
Sweden/ Asea-Atom BWRs	Nitrogen filled containment during full power operation
Sweden/PWRs	25 AREVA PARs (106 kg/h at 4 vol.% H_2) in Ringhals 2, 29 PARs (125.8 kg/h at 4 vol.% H_2) for Ringhals 3 and 4
Swiss/KKG (PWR)	Combination of dilution by passive mixing, 2 thermal recombiners used for ignition (4-6 vol.% H_2), venting and melt cooling by injection

Country/NPPs	**Hydrogen Mitigation Measures**
Swiss/KKL (BWR)	5o hydrogen ignition system (operation as long as oxygen is available)
Swiss/BWR-GE-MarkI	Containment inerted with nitrogen
US/BWR Mark I/IIs	Containment inerted with nitrogen
US/BWR Mark IIIs/ PWR Ice Condensers	~50 -100 igniters installed through-out containment
US/PWR large dry containments	Large and robust containment structure
US/ALWR (PWRs)	47 PARs throughout the containment for US-EPR designs 64 igniters throughout the containment and 2 PARs for US-AP1000 designs
US/ALWR (BWRs)	Containment inerted with nitrogen

Notes:

(1) In Canada, all AECL PARs are required to self-start at 2% H_2 and 100°C regardless of catalyst degradation level, self-stop at 0.5-0.6% H_2 with capacity of 0.8 kg/h/per PAR at 20°C and 4% H_2.

(2) The containment and reactor building are a single combined structure for CANDUs with a single unit, but for multi-unit CANDUs, the reactor building includes the reactor vault of a specific unit, the common vacuum building and the fuelling duct. The mitigation measures are installed inside the containment/reactor building for CANDU 6 and inside the reactor vault and fuelling machine duct for multi-units.

Table 3-3
Requirements for Operation of Spray System

Country/NPP	Nominal water spray rate (kg/s)	Criteria for spray actuation and termination	SAMG recommendation
Belgium/PWR1000	Varies among units, ~125-150 kg/s per train (3 trains)	Varies among units, between P > 2.1 bar(a) and P > 3.1 bar(a)	
Canada/CANDU 6	~6800 kg/s (maximum) with all six headers on; decrease with an decrease in header pressure	14 kPa (g) on; 7 kPa (g) off	
Canada/multi-units vacuum building	Varies among stations; as high as 5750 kg/s	Pressure increase in the vacuum building caused by opening of the pressure relief vault, ~2.5 kPa(d)	Licensees have criteria (entry conditions) for which such systems are used during a SA
Finland / Loviisa VVER-440	500 kg/s for internal spray; 35 kg/s for external spray	P > 1.17 bar(a) for internal spray; P > 1.7 bar (a) for external spray	
Finland / Olkiluoto 1 & 2 BWRs	~70 kg/s	Manually activated in accident situation	
Finland / Olkiluoto 3 EPR	180 kg/s (if both pumps are operating)	Manually activated in accident situation	
France/PWR900		P > 2.6 bar	6 h after core degradation start
France/PWR1300		P > 2.5 bar	
France/PWR1300		P > 2.5bar	
Germany/BWR-72	~110 kg/s for drywell spray at 0.9 bar; ~45 kg/s for wetwell spray at 1.5 bar	containment (drywell/wetwell) pressure	Existing emergency hand book procedures; SAMG under development
Korea/PWR587/ PWR650/PWR950	103 kg/s (PWR587) 140 kg/s (PWR650) 233 kg/s (PWR950)	P>0.28 bar (g)	Spray is preferred to lower containment pressure. Not used for hydrogen concentration control.
Korea/PWR1000/ PWR1400	264 kg/s (PWR1000) 338 kg/s (PWR1400)	P>2.4 bar(a) or manually activated in accident situation	Spray is preferred to lower containment pressure. Not used for hydrogen concentration control.
Korea /PHWR679/PHWR700	4535kg/s (PHWR679) 4535kg/s (PHWR700)	P>0.14 bar (g)	N/A
The Netherlands PWR 500	50 m^3/h	Manually activated in BDBA	Spray is preferred to lower containment pressure in case of aerosols in the atmosphere. Not used in case of high H_2 concentrations.

Sweden/BWR	75 – 100 kg/s per train depending on reactors		
Sweden/PWR	~200 kg/s per train		
Spain/PWR W-1000	~100 kg/s	P > 1.26 bar	Switch off sprays, when containment pressurization is recommended in SAMG in order to prevent H_2 ignition.
Spain/BWR GE-Mark I		Wetwell: P > 1.6 bar	Stop when venting criteria are fulfilled
	~300 kg/s	Drywell: P_{WW} with spray > 1.6 bar T_{DW} > 138 ºC P_{WW} > 1.6 bar	Stop when venting criteria are fulfilled
Spain/BWR GE-Mark III	~300 kg/s	P > 1.65 bar; P_{DW} > 1.12 bar	
Swiss/KKB (PWR)	110 kg/s borated water	p > 1.31 bar (g)	During a SBO, there is a possibility to inject water via accident management nozzles with mobile pump from fire brigade.
US/existing NPPs	Nearly all types of plants have sprays, there are some exceptions, flow rate varies	Variety of pressures setpoint and manual actuation	
US/New NPPs (EPR deign)	Single spray train	Manually actuated	Used for containment pressure control following a SA
US/New NPPs (AP-1000 design)	Single spray train		Could be used in SA, but not credited in accident analyses

Table 3-4
Requirements for Operation of Containment Venting System

Country/NPPs	Nominal mass flow rate (kg/s)	Criteria for CHRS opening	SAMG recommendation
Canada/CANDU 6 (FCV)		300 kPa(a) ON; 200 kPa(a) OFF	
Canada/multi-unit (EFADS)	~149 m^3/h (peak to 790 m^3/h depending on subatmospheric hold-up period)	100.325 kPa(a) ON	To consider FCV upgrades as in CANDU 6
Finland/Olkiluoto 1 & 2 BWRs	At 3.5 bar, corresponds to steam generated by decay heat power 24 h after scram.	Venting line opened passively by rupture disk at 5.5 bar. Can be opened earlier manually.	
Finland/Olkiluoto 3 EPR		Opened manually if necessary. Not needed if other SAM systems work properly.	
France/PWR900	3.5	P > 5 bar	At least 24 h after core degradation start and after heating CHRS component
Germany/PWR	~3-5 kg/s	P > 6.5 - 7 bar (a)	Procedure included in emergency hand book; to be used after 2 – 3 d after IE; pressure reduction to ½ of design pressure within 48 h according to RSK
Germany/BWR-72	~14 kg/s	P > 7 bar (a)	Procedure included in emergency hand book; to be used not before 4 h after IE; pressure reduction to ½ of design pressure according to RSK; duration in the order of 10 – 20 min
Korea/PHWR			Under development
The Netherlands PWR 500	4 kg/s steam at 4.8 bar (a) and 138°C.	P > 4.8 bar (a)	Prevention of containment failure
Spain/BWR-GE-MarkIII	15 – 20 kg/s	P > 1.93 bar	Keep containment pressure < 4.13 bar
		H_2 > 0.2 vol.%	Discharged activity(*) < 1.1E-9 Ci/cm^3
		H_2 > (5.2 vol.% - 12.5 vol.%)	No activity limits
Spain/BWR-GE-MarkI		P > 3.67 bar or H_2 > 6 vol.% and O_2 > 6 vol.%	Keep containment pressure < 4.86 bar
Sweden		Venting line opened passively by rupture disk at ~5.5 bar. Can be opened earlier	

		manually	
Swiss KKB (PWR)	5.2 kg/s	p>4.2 bar (g) by a rupture disk (passive path)	Venting can be initiated earlier or later over the active path of the FCVS on behalf of emergency staff and/or in accordance with the Swiss emergency organisations (ENSI, NAZ, etc.)
United States			BWR Mark I/IIs installing reliable hardened vents capable of functioning during a SA; need for filters currently under assessment. Vent capability for other designs to be assessed.

(*) Ci/cm^3 is the unit for specific activity that is measured with monitors in the hardened vent lines.

Table 3-5
Requirements for Operation of Local Air Coolers

Country/NPPs	Cooler location	Criteria for cooler actuation and termination	SAMG recommendation
Belgium/PWR1000 (Doel)	Primary containment, peripheral or upper zone	P>1.3 bar (a)	
Canada/multi-units	Reactor vault	~38 MW heat removal during a LOCA	
Canada/CANDU 6	Fuelling machine vaults, boiler room, dome and other accessible areas	To maintain temperature in the inaccessible areas between 41°C–55°C	
Korea/PWR587/950		P>0.34 bar (g)	Non safety grade
Korea/PWR650		P>0.27 bar (g)	Non safety grade
Korea/PWR1000/1400		P>0.13 bar (g)	Non safety grade
Korea/PHWR679/700		P>0.0345 bar (g)	Non safety grade
The Netherlands/PWR 500	Inside containment	Cooling of containment atmosphere during normal operation	Heat removal by air coolers is limited or terminated in presence of hydrogen
Spain/PWR-W-1000 (apart from Almaraz NPP)	Within containment, outside of the missile shield.	Safety Injection signal	
Sweden	Within containment, only during normal operation		
Swiss/KKB	during normal operation and accident conditions	P > 0.131 bar (g), safety injection signal	Use of mobile fire water pumps for water supply of fan coolers over Accident Management nozzles in penetration room of auxiliary building
United States	Plant specific if part of plant design		

Table 3-6
Requirements for Latch System, Blow-out Panels and Doors

Country/NPPs	Type of devices	Criteria	Passive/Active
Canada/CANDU 6	Blow-out panels between accessible and inaccessible area	ΔP>6.9 kPa	Passive
Canada/Multi-units	Explosion panels		Passive
Finland/Loviisa VVER-440	Forcing open the ice-condenser doors (lower-inlet, intermediate-deck and top-deck doors)	Core exit temperature > 450 °C	Manually opened with pressurized nitrogen cylinders (no electricity needed)
Finland/Olkiluoto 3 EPR	Rupture foils in the ceiling of steam generator room	ΔP > 50 mbar or T > 90 °C	Passive
Finland/Olkiluoto 3 EPR	Hydrogen mixing dampers between annular space and IRWST	Loss of AC power or ΔP > 35 mbar or P > 1.2 bar	Passive
Germany/PWR	Burst membranes between the component compartment and the dome and periphery of the containment	Differential pressure	Passive
Korea/PWR1400 (APR1400)	Swing panels at the vent exit of the IRWST to annular compartment	ΔP > 0.034 bar	Passive
Japan/PWRs	In the ice condenser PWRs, the inlet door at the bottom is opened under accident condition where a significant amount of steam is released in the containment.		
Japan/BWRs	In BWRs, a blow-out panel is attached on the reactor building wall.		
The Netherlands/PWR 500	Burst membranes between the primary component compartments and the dome	Differential pre. In case of H_2, 12 panels can be operated (opened) by hand	Passive Active
Spain/PWR-KWU-1000	Blow-out panels (steam generator rooms to dome)	ΔP > 0.5 bar	Passive
Spain/BWR-GE-MarkI	Vacuum breakers (wet well to dry well)	ΔP > 0.034 bar	Passive
Spain/BWR-GE-MarkIII	Vacuum breakers (containment to dry well)	ΔP > 0.013 bar	Passive
Swiss/KKB	Not applicable because of single compartment containment		
US/PWRs or BWRs	Vacuum breakers		
US/PWR Ice Condensers	Ice chest doors		

Table 3-7
Contribution to Global Risk of Containment Failure Modes (Simplified Ranking) for PWR 1300

Containment failure mode	Contribution to global risk (freq x cons)
I-SGTR (Induced steam generator tube rupture)	29.0 %
Reactor containment isolation failure	16.1 %
Reactor containment failure after hydrogen combustion during in-vessel phase	14.4 %
Reactor containment bypasses (heterogeneous dilutions, initial SGTR, Interfacing LOCA, bypass through equipment latch access)	10.0 %
Ex-vessel steam explosion	8.7 %
Spent fuel pool accident	6.5 %
Reactor containment failure after hydrogen combustion during ex-vessel phase	4.0 %
Reactor containment failure after direct containment heating	2.7 %
Secondary containment failure after hydrogen combustion in the annulus	2.0 %
Filtered containment venting	1.7 %
Basement penetration	1.3 %
Long term containment over-pressurization	1.0 %

4 CODES AND VALIDATION

This chapter provides an overview of various codes that are being used for SA analyses for various NPPs related to hydrogen issues by the member countries (or organizations), and the codes capabilities and validation status in terms of modelling hydrogen related phenomena (e.g., generation, distribution, combustion and mitigation). It may not cover all the codes being used by these countries and the information, therefore, may not be complete. The focus is to identify the gaps and future needs for model development and validations of most commonly used codes.

4.1 Codes Used for Hydrogen Analysis

There are many different codes that are being used for SA analyses for various NPPs related to different hydrogen issues by the participating countries. The application involves research and development, safety analysis related to the implementation of hydrogen mitigation measures, licensing, or periodic safety assessments. In some countries, the same sets of codes are used by their utilities, research institutions and/or regulatory bodies, but in other countries, different codes are used by different organizations.

The codes that are typically used for the hydrogen analysis by the participating countries are listed in Table 4-1. Depending on the code capabilities, some codes are used to simulate all the hydrogen aspects (i.e., generation, distribution, combustion and mitigation), but some are only used for hydrogen generation, which provide boundary conditions (e.g., hydrogen source term) for the codes that simulates hydrogen distribution, combustion and mitigation. Some codes are specially designed to simulate hydrogen combustion, particularly detonation, which requires input for hydrogen distribution calculated by other codes.

The participating countries provided descriptions for eleven codes that are being used by many organisations, such as ASTEC, MAAP/MAAP-CANDU, MELCOR, SPECTRA, COCOSYS, TONUS, GOTHIC, GASFLOW, ANSYS CFX, ANSYS FLUENT, and ANSYS AUTODYN[1]. Detailed descriptions of these codes are presented in Appendix B. The other codes mentioned in Table 4-1 are not further described in this report.

The focus of this chapter is to assess the code capabilities and their validation status in terms of hydrogen phenomena as well as their strengths, limitations, improvements and application methods. These codes are continuously under development and review and are continuously validated against new experimental data. A larger user community has been established, providing useful feedback for further model developments and improvements as well as for appropriate code application for hydrogen analysis.

The above mentioned codes can be broadly classified into two groups:

- Lumped parameter (LP) codes

  ASTEC, MAAP/MAAP-CANDU, MELCOR, SPECTRA, COCOSYS, and GOTHIC are the typical LP codes.

  ASTEC, MAAP/MAAP-CANDU, MELCOR, SPECTRA are often referred to as integral or system codes [1.32]. These codes cover all aspects of in-vessel and ex-vessel SA phenomena, such as thermo-hydraulic response in the reactor coolant system and containment, core heat up, core degradation and relocation, fission product release and transport, DCH, MCCI, etc. These system codes employ simpler physics, models and calculation methods (1D approach) which make them capable to simulate long time transients in acceptable computation times. Assumptions and model simplifications can lead to larger uncertainties.

[1] Hereafter referred to as CFX, FLUENT, and AUTODYN

COCOSYS and GOTHIC are mainly developed for containment analyses and are based on the fundamental assumption that spatial differences of thermo-hydraulic variables are neglected within a control volume. Energy and mass conservation equations are defined in each control volume. The control volumes are linked to each other via a transport of mass and energy described by flow paths or junctions, which is time-dependent. Most typical phenomena of SAs are also modelled in these codes. These LP codes are relatively fast-running and tend to be structured so that integrated models can be readily incorporated. However, they don't model turbulence and cannot capture local effects due to relative large size of the control volumes.

It should be noted that GOTHIC's control volumes can be modelled with a LP approach or subdivided into 1D, 2D or 3D grids. Many 3D features have been built into the recent version of GOTHIC. Its 3D capability has been widely used for hydrogen analysis, so it is hereafter referred to as a 3D code in this report.

- 3D codes

  TONUS, GOTHIC, GASFLOW, CFX, FLUENT, and AUTODYN are 3D codes. In these codes, fluid motion and heat transfer is computed in 3D high resolution over finite volumes/elements. 3D codes represent the state-of-the-art computational capabilities for analysing flow and heat transfer problems in complex geometries. The main strengths of 3D codes are the local resolution/detail and the capability to model turbulence. Main drawback of these codes is that they require a relatively large computational effort. It is, therefore, recommended to apply the 3D codes complementary to LP codes for hydrogen risk assessment ([1.44], [4.1]).

  TONUS, GOTHIC and GASFLOW are special purpose 3D codes designed for containment analyses. CFX and FLUENT are commercial multipurpose 3D (or often called Computational Fluid Dynamics - CFD) codes. AUTODYN is specially designed for simulations of detonations and the resulting interaction between pressure waves and the structural behaviour. For CFX and FLUENT, special models, dedicated to hydrogen mixing, mitigation and combustion phenomena are generally developed, implemented and validated by the users.

4.2 Code Capabilities Regarding Hydrogen Phenomena

4.2.1 Phenomena List

To assess the code capabilities in modelling hydrogen related phenomena for SAs, a phenomena list was developed based on References [1.31], [1.32] and [4.2]. These phenomena cover four aspects of hydrogen behaviour.

4.2.1.1 Hydrogen Generation

During the development of a SA, potential hydrogen sources can come from in-vessel and/or ex-vessel processes, such as:

- G1: Water radiolysis in the sump
- G2: Metal (Zr, steel) steam reaction (oxidation within the core)
- G3: Metal corrosion (zinc, steel, aluminium)
- G4: Molten corium concrete interaction (MCCI)
- G5: Oxidation in molten pools or debris beds within the core/RPV
- G6: Metal (Zr, steel, B4C) oxygen reaction within the core

The phenomena G2, G5 and G6 occur during in-vessel process and G1, G3 and G4 are typical ex-vessel sources.

4.2.1.2 Hydrogen Distribution

Hydrogen distribution can be significantly affected by the containment configuration, hydrogen release location and mass flow rate, and its mixing with other gases [1.32]. The typical containment thermal-hydraulics phenomena [4.2] that are related to hydrogen distribution are:

- D1: Stratification
- D2: Momentum induced mixing
- D3: Buoyancy induced mixing
- D4: Condensation on surfaces
- D5: Turbulent flow characterisation
- D6: Liquid/water in films and pools

Density variation due to temperature difference or mixture composition can lead to stratification into layers. Hydrogen stratification (D1) is a safety concern because pockets of high hydrogen concentration could lead to a deflagration or detonation, which might challenge the integrity of containment buildings. Modelling hydrogen stratification by lumped parameter codes typically requires a more detailed and specific nodalization of the containment and therefore larger user experience; it is not a code feature itself. Recommendations on how to prepare an appropriate nodalization can be derived from analyses of different experiments and benchmarks (ISP-47, HM-2 benchmark) performed at the THAI facility ([1.44], [4.1]). This phenomenon has strong link to the other five phenomena (D2 to D6). To capture the hydrogen behaviour consistently, a code must have the capability or consideration must be taken for the above six phenomena.

4.2.1.3 Hydrogen Combustion

There are several types of combustion modes that are likely to occur in a post-accident containment depending on various parameters (e.g., mixture composition, pressure, temperature, geometry of the confinement, turbulence level, etc.). The phenomena for hydrogen combustion are distinguished into the following 9 types [4.2]:

- C1: Deflagration
- C2: FA
- C3: DDT
- C4: Detonation
- C5: Quenching of detonations
- C6: Diffusion flame
- C7: Strong ignition/jet ignition
- C8: Combustion with water droplets
- C9: H_2-CO combustion

Deflagration deals with combustion with flame speeds on the order of several meters per second to several hundred meters per second. The burning rate can be affected by initial conditions (mixture compositions, pressure, and temperature), geometry of the confinement, location of ignition, and turbulence level. Nevertheless, the maximum deflagration pressure is bounded by the adiabatic isochoric complete combustion (AICC) pressure.

Depending on the mixture properties and boundary conditions, the interaction of a flame with turbulence in the unburned gas can lead to either weak FA within relatively slow, unstable, turbulent flame regimes, or to strong FA resulting in fast flames that propagate at supersonic speeds. The mixture expansion ratio is the key parameter that defines the border between weak and strong FA [1.50]. Within a sufficiently large round-up distance, supersonic combustion regimes can be developed.

It is vital that a code (or a set of codes) can predict the outcome of various hydrogen combustion modes with reasonable accuracy, thus the impact for the containment structure or equipment can be estimated.

4.2.1.4 Hydrogen Mitigation

For most operating NPPs (e.g., PWRs, VVERs and PHWRs) as well as next generation reactors (see Chapter 2), PARs and/or igniters are the major measures that have been (or are being) implemented as part of the plants accident management concepts to mitigate hydrogen risk, while the smaller containment of BWRs is typically inerted by N_2. Engineering safety systems, such as fan/local air coolers and spray systems are often included in the design in many NPPs and are also used in SAs for heat and/or fissional product removal. In addition, FCV systems are implemented as Accident Management provision in many NPPs to prevent long-term containment over-pressure failure. Activation of these systems would have an influence on the local flows and turbulence level and thus on the hydrogen distribution and/or possible combustions. The hydrogen mitigation related phenomena are listed as follows:

- M1: H_2 recombination by PAR
- M2: H_2 ignition by PAR
- M3: H_2 ignition by igniter
- M4: CO recombination by PAR
- M5: Filtered venting system
- M6: Fan/local air cooler
- M7: Spray system

4.2.2 Code Capabilities

The codes capabilities are evaluated in terms of whether the above phenomena are modelled and the assessment results are shown in Table 4-2. The observations are summarized as follows:

4.2.2.1 Hydrogen Generation

- Only the four integral codes (ASTEC, MAAP, MELCOR, and SPECTRA) out of the eleven codes described in the report are capable of calculating hydrogen generation within the reactor core during fuel cladding / steel oxidation, core degradation and molten pool formation. In addition, theses codes are capable of calculating other non-condensable gas generation due to MCCI and DCH after the RPV failure.
- Other well-known thermal-hydraulic LP codes like the German ATHLET-CD [4.3] or the US code RELAP/SCDAP [4.4], which are not described in this report, are able and specialised for the calculation of the reactor circuit behaviour under accidents and SAs and include models for the core degradation and hydrogen generating phenomena.
- All the 3D codes (TONUS, GOTHIC, GASFLOW, CFX and FLUENT, AUTODYN) have no capability to model hydrogen generation. The hydrogen source is generally calculated by other codes and specified as a source term. For the LP code COCOSYS, it is a containment code, so the hydrogen generation due to MCCI is included.
- None of the codes calculate hydrogen production from radiolysis within the containment sump water. Metal corrosion (i.e., low temperature Zr oxidation by water) can be modelled in MELCOR and SPECTRA based on a user input for the reaction coefficients, and it is under development in some other codes. This is not a significant concern because these contributions are relatively small when performing postulated SA analyses.

4.2.2.2 Hydrogen Distribution

- All the mentioned codes, except AUTODYN which is dedicated for detonation, are capable of modelling hydrogen distribution in the containment as it is a general transport process. The accuracy of the hydrogen distribution calculation depends to a large degree on the users choice of an appropriate nodalization scheme and the consideration of plant specific features (doors, burst membranes, spray systems, fan coolers, etc.).

  This is particularly true for LP codes and the user must pay attention on how the containment is nodalized. Some effects, such as stratification, momentum or buoyancy induced mixing originating from plumes of steam and non-condensable gases rising from a steam generator, can be quite satisfactory modelled even with LP codes when special care is taken. The user impact can affect the result in the same order as the general code validation [4.5], thus the users must follow carefully the available best user practices and results from international benchmark activities ([4.1], [1.44]) in order to obtain reliable results.
- In general, LP codes average the properties of a fluid within a given control volume, thus they tend to amplify mixing for coarse nodalization concepts. 3D codes have shown certain advantages as compared to some LP codes because they allow investigation of hydrogen mixing in complex geometric structures [1.37]. However, large discrepancies can exist due to user effects, particularly in using the commercial CFD codes (e.g., FLUENT, CFX), as not all physical phenomena, especially condensation processes, can be satisfactorily modelled by the CFD codes unless specific user functions are developed. It is highly recommended that a user closely follows the best practice guidelines for the use of CFD in nuclear reactor safety applications [4.6].
- Even though the available CFD codes (e.g., FLUENT, CFX) allow simulation of multiphase flows, their application to hydrogen mixing and mitigation in containment compartments is still limited to a single (gas) phase approach due to the computational effort. Consequently, heat and mass transfer with the liquid phase (i.e. fog, condensate on walls or structures as well as sumps and pools) is neglected; the condensate is simply removed by the code and it is not automatically reinserted back to the calculation (e.g., in case of a boiling sump). This is a concern for the late accident phase, where it can significantly affect the global water-steam balance. Furthermore, this can affect the general behaviour of safety systems which rely on the condensed water collected in the sump. Nevertheless, this is not a critical issue as long as CFD codes are generally used to perform short term calculations.
- Production, transport and dissipation of turbulence are not considered by the LP codes, but are taken into account by the 3D codes. The local turbulence level is an important factor for the prediction of 3D hydrogen mixing and a critical initial condition for flame propagation and acceleration.

4.2.2.3 Hydrogen Combustion

- None of the mentioned codes can simulate the entire range of combustion modes (deflagration, DDT, denotation, diffusion flame). There are many numerical combustion models, but the range of applicability generally is limited to a specific type of event/regime. For instance, TONUS is able to simulate slow and fast turbulent deflagration and detonation. AUTODYN is specially designed for simulation of detonation and the resulting interaction between pressure waves and the structural behaviour only. Most other codes are dedicated for slow deflagration only. GASFLOW and SPECTRA can predict whether a hydrogen-air-steam mixture is flammable or detonable, and if it has the potential risk of FA. However, they do not simulate FA, DDT, and detonation.
- All the LP codes mentioned here have a built-in simple hydrogen deflagration model. Turbulent combustion is not treated in a specific way by any of these codes. The accuracy of the combustion calculation depends to a large degree on the user's choice of an appropriate nodalization scheme. Some impressions can be gained from ISP-49 exercise [1.53].

- The flame speed strongly depends on the initial turbulence level and the turbulence generated during flame propagation. Turbulence is not computed by the mentioned LP codes and requires the application of 3D codes. It is common to apply a user defined burn enhancement factor to mimic the flame surface area increase due to turbulence effect.
- Other codes like COM3D [4.13] and DET3D [4.14], which are not described in this report, are developed and applied at KIT Germany for simulation of gaseous combustions in complex, 3D geometries (COM3D) and gaseous detonations in complex, 3D geometries (DET3D), with its main field of application being detonations involving hydrogen.
- Most of the mentioned codes employ a simple diffusion flame model that allows for the burning of hydrogen-rich mixtures upon entry into volumes containing oxygen. Note that the diffusion flame model can be used to model the burning during DCH while leaving the bulk burn parameters at their nominal values.
- None of the mentioned codes have been assessed for H_2 combustion during spray operation. Research programmes on this topic are ongoing (e.g., OECD-THAI2) in order to generate experimental data for developing and validating these specific combustion models.
- H_2-CO combustion is modelled by ASTEC, MAAP, MELCOR, COCOSYS and TONUS.

4.2.2.4 Hydrogen Mitigation

- The status of the codes related to modelling of hydrogen recombination by PARs is different. Often "black-box" models, namely, hydrogen recombination efficiency and/or recombination rate are calculated using manufacturer correlations by most codes. For the commercial codes FLUENT and CFX, some users have implemented their own PAR models [4.7]. Hydrogen and carbon monoxide recombination is treated by some codes (based on manufacturer correlations). Some codes (e.g., ASTEC, COCOSYS [4.8]) have implemented a one-dimensional detailed PAR model for box-type components (AREVA, AECL) based on a diffusion approach. The PAR is subdivided into three main segments along the height of the box: entry, catalytic plates and chimney, while the plate region itself is further subdivided. The thermal hydraulic conditions are calculated for each segment. The model considers the catalytic plates as well as the structure of the housing. The heat transfer between the surrounding gas and the structures is calculated by means of free and forced convection, condensation and radiation.
- A mechanistic PAR model (REKO-DIREKT [4.10]) with complete description of chemical reaction has been developed by JUELICH (Germany) [1.56], describing in detail the operational behaviour of a recombiner by employing a mass transfer approach for the reaction kinetics. The model has been implemented in the CFX code and preliminary validation has been performed with encouraging results [4.11]; a coupling to COCOSYS is currently being performed. This model is capable of handling H_2 and/or H_2-CO recombination.
- The SPARK code, developed by IRSN (not described in this report), considers a full description of the gas-phase and surface chemistry and is dedicated to detailed investigations on PAR performance [4.12]. Research is on-going to evaluate PARs efficiency and hydrogen ignition limit for any operating conditions.
- There is no model built into any of the described codes to predict the onset of PAR operation as well as hydrogen ignition by PARs. Most codes allow users to start or stop the hydrogen recombination at a defined hydrogen concentration. In order to consider ignition due to hot PARs, most of the mentioned codes allow the user to initiate a hydrogen combustion in the zone containing a PAR at a given hydrogen concentration. In ASTEC, there is a correlation implemented based on the SPARK code results. Ignition due to PAR operation is investigated in the frame of the OECD-THAI 1&2 project, but no mechanistic models for LP codes are developed yet.
- Most LP codes including GOTHIC have sub-models (or components) to simulate spray systems, air coolers and fan systems, and FCV systems.

4.3 Status of Codes Validations

In general, certain degrees of validations are performed by the code developer before the codes are released for use. Users often perform their own independent validation using their experimental data or data obtained through international collaboration projects. However, none of the codes hereby described is fully validated regarding the complete list of phenomena either due to a lack of experimental data or due to a limited application range. In addition, assessing the existing uncertainties is extremely difficult. Engineering judgement and a large degree of know-how on code application is therefore needed in order to obtain realistic results.

The codes validation status is assessed in terms of whether any validation has been performed for the hydrogen related phenomena either by the code developer, the user or both. The results are show in Table 4-2. The findings are summarized as follows:

- Validations for hydrogen generation from metal-steam and/or metal-oxygen reaction exist for the four codes (ASTEC, MAAP, MELCOR and SPECTRA) that have these capabilities. Validations do not exist for some of these codes for hydrogen generation from metal corrosion, oxidation in molten pools and MCCI.
- Hydrogen distribution in the containment is quite well validated by almost all the mentioned codes except AUTODYN (not dedicated for this application) and lots of experimental data are available [4.2].
- Hydrogen deflagration is the phenomena where many of experimental data exists and the codes are widely validated. Other phenomena such as fast deflagration and FA are only modelled by some codes, but validations are limited or large uncertainty exists. The other phenomena such as DDT, detonation and quenching of the detonation are not fully treated and no validations or very limited validations are performed.
- All the mentioned codes, except the commercial multi-purpose ANSYS codes (FLUENT, CFX and AUTODYN), have simple PAR models and validations have been performed using the well-known experiments performed at facilities like H2PAR, KALI, HDR, BMC, and THAI. The validations are reasonably good for normal conditions (sufficient air), but limited for challenging conditions (e.g. oxygen starvation).
- Validations for spray and local air cooler models exist for most codes, but they largely lacking for filtered venting system, CO recombination and hydrogen (and/or H_2-CO) ignition by PAR.

It should be noted that the recently published OECD/NEA report on containment validation matrix [4.2] has identified a large number of experiments applicable for validation of hydrogen distribution, combustion and PAR recombination, which can be considered for future code validation by the code developers and/or users.

4.4 Recent International Benchmarks on Hydrogen

In Chapter 1.3, the major international experimental and numerical activities on hydrogen related phenomena (i.e., ISP-42, ISP-47, ISP-49, etc.) were reviewed. This section summarizes some major findings of these benchmarks or analytical activities. The codes that are assessed in this chapter and used for the benchmark simulations are shown in Table 4-3, including the key phenomena that were modelled.

4.4.1 Hydrogen Generation

In 1993, the ISP-31 benchmark [4.15] was performed against the CORA-13 experiments performed at Kemforschungszentrum Karlsruhe in 1990. The major objectives of the experiment were to investigate the behaviour of PWR fuel elements during early core degradation and fast cool-down due to refill. Hydrogen generation was one of the measurements. Five codes were used in the benchmark, including SCDAP/RELAPS, FRAS-SRF, ICARE2, KESS-III and MELCOR. The hydrogen generation was over-predicted in the early transient, but under-predicted in the later pre-quench phase. None of the codes could predict the intensive hydrogen generation during refill.

4.4.2 Hydrogen Distribution

Since 1990, several ISP benchmarks have been performed against many thermal-hydraulics and hydrogen mixing experiments, including ISP-23, 29, 35, 39, 42, and 47.

In the ISP-42 ([1.39], [1.40]) benchmark using the gas mixing tests performed in the PANDA facility, SPECTRA, COCOSYS and GOTHIC were three of the six codes used. The best results were obtained by the LP code SPECTRA, but other codes also gave globally acceptable results. Condensation of steam in presence of non-condensable was predicted well by all the LP codes. The only 3D contribution was the GOTHIC simulation, which showed successful results for mixing and stratification, but due to its limitation on modelling specific equipment (e.g., passive containment cooling), its 3D capabilities were not well reflected as compared to the LP codes. Separate effect experiments for specific phenomena related to containment multi-compartment geometries would be more appropriate to assess the 3D models and their advanced features (e.g., turbulence).

In the ISP-47 benchmark using the gas mixing tests performed in the THAI, TOSQAN and MISTRA facilities [1.44], participants from various institutions used the ASTEC, MELCOR, COCOSYS, TONUS, GOTHIC, GASFLOW and CFX codes. The observations and recommendations from the ISP-47 activities are described as follows:

- Although LP codes have some inherent limitations due to simplification of physical processes, appropriate user modelling can often overcome these limitations. Thus, development of general guidelines (with specific aspect to nodalization) for LP including specific requirements for user manual is also highly recommended.
- Combined use of both LP codes and 3D/CFD is recommended. LP can be used as the basic tool for containment analyses, whereas 3D/CFD codes are used for local detailed analyses for selected accident scenarios.
- For the commercial CFD codes, further improvements regarding the modelling of condensation, turbulence, and wall treatment are needed.

In the frame of the OECD-THAI project [1.45], benchmarks were performed using one of the test series - helium/hydrogen material scaling (HM) tests. The codes used for this benchmark included ASTEC, MELCOR, COCOSYS, GOTHIC, GASFLOW and FLUENT codes. It has been concluded that erosion of hydrogen cloud is possible even if the density of the plume is higher than the density of the cloud; however, large uncertainties existed in the prediction of mixing time for all the codes during the blind calculation stages. The benchmark showed strong user influence for both the LP and 3D codes. It was recommended that sufficient nodalization in the vertical direction is needed to capture the hydrogen stratification for the LP codes. For some of the 3D codes, the results were sensitive to the choice of turbulence models, wall functions, bulking condensation and rainout models.

In the frame of the SETH-2 project [1.46], the experimental programme was accompanied by code simulations. The observations are:

- LP codes require dedicated mesh adjustments to simulate high-momentum jets, which may cause breaking up of atmosphere stratification. Such conditions are normally addressed by the use of 3D/CFD codes.
- Although 3D/CFD codes are based on first principles, calculations performed with them still carry uncertainties and also require users to have adequate knowledge of simulated phenomena. In addition, the use of 3D/CFD codes necessitates lengthy model developments and calculations, LP codes are still expected to be used for overall predictions of the behaviour of a non-homogeneous containment atmosphere for the time being.

The SARNET2 'Generic Containment' LP code-to-code comparison [4.5] investigated the user effect as well as impact of the fundamental differences in the underlying models on the thermal hydraulics and hydrogen distribution on the basis of a well-defined generic containment nodalization and accident scenario. Even though this problem was uniquely defined and the calculations were performed by skilled users, the uncertainty of calculated results due to different modelling approaches and users was much higher than expected. Furthermore, users of the same codes produced noticeably different results. This implies that user training and guidelines are in the same order of importance as code validation. Especially, the heat and mass transfer with dead-end compartments was predicted with a huge uncertainty among the codes. This resulted in differences of the local hydrogen concentration but also the distribution of the available oxygen which limits PAR operation.

In the ongoing ERCOSAM-SAMARA projects [1.47], extensive analytical activities are also performed using ASTEC, COCOSYS, GOTHIC, GASFLOW, CFX and FLUENT. Post-analysis and model evaluation is in progress.

4.4.3 Hydrogen Combustion

In the frame of SARNET project [1.41], benchmark for FA and DDT was performed against the experimental data generated in IRSN's ENACCEF facility. TONUS and FLUENT were two of the codes used. It was found that all codes were able to predict flame speed and pressure for a homogenous hydrogen-air mixture as well as for mixtures with a positive hydrogen concentration gradient, but the flame speed was generally over-predicted for the mixture with a negative hydrogen concentration gradient. Model improvement is needed for hydrogen combustion in mixtures with non-uniform hydrogen concentrations.

The ISP-49 benchmark [1.53] was performed with hydrogen combustion tests performed in the THAI and ENACCEF facilities covering slow deflagration and FA range. It was concluded that:

- The simulations with LP codes utilizing the *FRONT* combustion model demonstrated satisfactory prediction of the flame speed and can be considered as perspective basis for further model development,
- The approach realized that the burning velocity models based on the two-equation RANS turbulence models in the CFD codes can be considered as quite successful for the purposes of the containment combustion analysis, although the observed lack in a stability of the predictions has to be addressed by further improvement of the coupled combustion and turbulence models.
- Model development for prediction of FA in the geometrical configuration typical to the NPP environment is needed, e.g., in partially enclosed volumes with the irregular obstruction, in flat layer of hydrogen-air mixture, and in the large volumes having walking grids.

The experience acquired in the course of the ISP-49 revealed that the contemporary level of the numerical tools developed for the simulation of the combustion processes under SA conditions of a NPP requires further improvement to provide high quality blind predictions. The existing combustion models demonstrated that the quality of the prediction of FA reached moderate level of accuracy in the tube-like geometries with the regular obstruction, however, in other geometrical configurations (e.g., in partially enclosed volumes with the irregular obstruction, in flat layer of hydrogen-air mixture, in the large volumes having walking grids, etc.), there were no clear proof of their conformity to the numerical code validation requirements.

4.4.4 Hydrogen Mitigation

In the frame of the SARNET project, PAR interaction studies have been performed in order to generally investigate adopted models and approaches. Seven organizations were involved in the PARIS-1 benchmark using CFX, FLUENT, GASFLOW and TONUS, and four organizations were involved in the PARIS-2 benchmark using CFX and GASGLOW. The first numerical benchmark (PARIS1) investigated the behaviour of two PARs in opposite arrangement inside a simple 2-D geometry [4.16]. The global flow behaviour and hydrogen depletion rates were captured by the codes, but code-to-code comparison exhibited local differences, e.g., at the PAR outlet where jet or plume behaviour were both predicted. Moreover, the characteristic hydrogen layer below the PAR was more or less stable. The second step (PARIS-2) was devoted to PAR-mainstream (upward or downward) interaction and numerical mesh size effects [4.17]. The benchmark showed that the PAR efficiency could be influenced by local effects, such as atmosphere mixing and mobilisation, local hydrogen concentration and gradients, and local atmosphere temperature and composition. Only CFD approaches could give detailed fields and thus provide useful complementary information in addition to the LP approaches. Future R&D activities are needed for development of enhanced models and strategies to investigate local effects of PAR operation and hydrogen distribution. In addition, it should be identified if conditions exit where local PAR behaviour plays an important role in the course of an accident.

In the frame of the SARNET2 project [4.18], a benchmark exercise was organized by GRS to investigate different modelling of PAR devices. Two PAR experiments performed in the THAI facility under dry air conditions and different initial pressures were simulated using ASTEC, COCOSYS, GASFLOW, FLUENT. The purpose of the benchmark was to investigate how well different codes simulated pressure increase and decrease, thermal stratification and recombination rate. CFD codes appeared to be more adequate to model the thermal stratification than LP codes.

In the frame of the OECD/NEA THAI projects [1.45], many participants conducted simulations against the PAR tests using MELCOR, COCOSYS, GOTHIC, GASFLOW, and CFX. Most codes used the simple "black-box" PAR model except CFX users who employed the REKO-DIREKT PAR model developed by JUELICH, Germany or the AREVA PAR model implemented by GRS, Germany.

During the OECD/NEA THAI2 project, benchmark simulations were performed against a PAR test (HR-35) performed under extremely low oxygen concentration using AREVA's PAR unit [4.19] by participants using COCOSYS, GOTHIC, GASFLOW, and CFX. The main features of the simulated recombination and its effect on the surrounding atmosphere revealed significant differences among the codes and the models applied. After the onset of the catalytic reaction which was specified as experimental time as well, the simulations using the REKO-DIREKT approaches performed best in predicting the recombination rate. For models which rely on the AREVA correlation, there is a substantial need for improvement under low oxygen conditions.

4.5 References

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Table 4-1
List of Major Codes Used for Hydrogen Analysis by the Member Countries

Country	H_2 Generation	H_2 Distribution	H_2 Combustion	H_2 Mitigation
Belgium	MELCOR	MELCOR	MELCOR	MELCOR
Canada(1)	CHAN, CATHENA, TUF, MAAP-CANDU	GOTHIC,MAAP-CANDU	GOTHIC, MAAP-CANDU, DDTINDEX	GOTHIC, MAAP-CANDU
Czech-Rep.	MELCOR	MELCOR	MELCOR	MELCOR
Finland	MELCOR	COCOSYS APROS MELCORFLUENT	FLUENT	COCOSYS APROS MELCOR FLUENT
France(2)	ASTEC, MAAP	ASTEC, MAAP, TONUS, SATURNE, FLUENT	ASTEC, TONUS, FLUENT, EUROPLEXUS	ASTEC, TONUS, FLUENT, SPARK
Germany(3)	MELCOR, ATHLET-CD, ASTEC	COCOSYS, ASTEC, MELCOR, CFX, GASFLOW	COCOSYS, ASTEC, MELCOR, GASFLOW, COM3D, DET3D	COCOSYS, ASTEC, MELCOR, GASFLOW, CFX
Italy(4)	MELCOR	MELCOR	MELCOR, OPENFOAM	MELCOR
Japan	MELCOR	MELCOR, FLUENT	MELCOR, AUTODYN	MELCOR
Korea	MAAP4 MELCOR	GASFLOW	COM3D	MELCOR GASFLOW
The Netherlands	MAAP4, MELCOR, RELAP, TRAC	MAAP4, MELCOR, SPECTRA, WAVCO, FLUENT, CFX	SPECTRA, FLUENT	SPECTRA, FLUENT
Poland(5)	MELCOR, RELAP5/SCDAP	MELCOR, FLUENT, OpenFOAM	MELCOR, FLUENT, OpenFOAM	MELCOR, FLUENT, OpenFOAM
Sweden	MAAP4, MELCOR	MAAP4, MELCOR, GOTHIC	MAAP4, MELCOR, GOTHIC	MAAP4, MELCOR, GOTHIC
Switzerland	MELCOR, RELAP/SCDAP	MELCOR, GOTHIC, FLUENT	MELCOR, GOTHIC, FLUENT	MELCOR, GOTHIC, FLUENT
Spain(6)	MAAP, MELCOR	MAAP, MELCOR, GOTHIC, CFX	MAAP, MELCOR, CPPC	MAAP, MELCOR
USA	MELCOR, MAAP	MELCOR, MAAP, GOTHIC	MELCOR, MAAP, GOTHIC	MELCOR, MAAP, GOTHIC

Notes:

(1) CHAN, CATHENA, TUF are thermal-hydraulic codes used for CANDU reactors. DDTINDEX is an AECL developed code to determine the likelihood of FA and DDT based on hydrogen distribution.

(2) The SPARK code, developed by IRSN, considers a full description of the gas-phase and surface chemistry and is dedicated for detailed investigations on PAR performance [4.12].
(3) ATHLET-CD is a thermal-hydraulic code for RCS and core behaviour under accident and SA conditions developed at GRS, Germany [4.3]. COM3D and DET3D are 3D codes for hydrogen combustion and detonation analysis developed at KIT, Germany [4.13], [4.14]
(4) There is no reactor built in Italy yet and OPENFOAM is considered for future use.
(5) There is only one research reactor built in Poland and the listed codes are considered for future use only. RELAP/SCDAP [4.4], developed by US is specialised for calculation of reactor circuit behaviour under SAs and include models for core degradation and hydrogen generation.
(6) CPPC is a code developed by Polytechnic University of Madrid for FA&DDT criteria, AICC pressure and effective static pressure from expected dynamic pressure. MELCOR was modified with a detailed chemistry PAR model [3.29].

Table 4-2
Summary of Codes Capabilities and Codes Validation Status for Modelling Hydrogen Generation, Distribution, Combustion and Mitigation

Category	ID	Phenomena Description	LP codes					3D codes					
			ASTEC	MAAP/MAAP-CANDU	MELCOR	SPECTRA	COCOSYS	TONUS	GOTHIC	GASFLOW	CFX	FLUENT	AUTODYN
Generation	G1	Water radiolysis in sump											
	G2	Metal (Zr, steel) steam reaction (oxidation within the core)	♦	♦	♦	♦							
	G3	Metal corrosion (zinc, steel, aluminium) (1)			◊	◊							
	G4	Molten corium concrete interaction	♦	◊	♦	◊	♦						
	G5	Oxidation in molten pools or debris beds within the core	◊	◊	♦	♦							
	G6	Metal (Zr, steel, B_4C) oxygen reaction within the core	♦	♦	♦	♦							
Distribution	D1	Stratification (2)	♦	♦	♦	♦	♦	♦	♦	♦	♦	♦	
	D2	Momentum induced mixing (2)	♦	♦	♦	♦	♦	♦	♦	♦	♦	♦	
	D3	Buoyancy induced mixing (2)	♦	♦	♦	♦	♦	♦	♦	♦	♦	♦	
	D4	Condensation on surfaces (3)	♦	♦	♦	♦	♦	♦	♦	♦	♦	♦	
	D5	Turbulent flow characterisation(4)						♦	♦	♦	♦	♦	
	D6	Liquid/water in films and pools(5)	♦	♦	♦	♦	♦		♦	♦		◊	
Combustion	C1	Deflagration	♦	♦	♦	♦	♦	♦	♦	◊	♦	♦	
	C2	Flame acceleration(6)	◊					♦			♦	♦	
	C3	DDT(6)											
	C4	Detonation(6)						♦					♦
	C5	Quenching of detonations											◊
	C6	Diffusion flame	♦	◊	♦	◊	♦		◊		♦	♦	
	C7	Strong ignition/jet ignition									◊		
	C8	Combustion with droplets											
	C9	H_2-CO combustion	◊	◊	◊		◊	◊					
Mitigation	M1	H_2 recombination by PAR(7)	♦	◊	♦	♦	♦	♦	♦	♦	♦	♦	
	M2	H_2 ignition by PAR(8)	◊		◊		◊	◊					
	M3	H_2 ignition by igniter	◊	♦	◊	♦	◊		♦		◊	◊	
	M4	CO recombination by PAR(9)	◊			◊	◊	◊			♦		
	M5	Filtered venting system(10)	◊	◊	◊	◊	◊		◊				
	M6	Fan/local air cooler	♦	♦	♦	♦	♦		♦		♦	◊	
	M7	Spray system(11)	♦	◊	♦	♦	♦	♦	♦	♦	◊	♦	

◊ - code has this capability (but no validations exist); ♦- code has this capability and validations have been performed against relevant experiments.

Notes:

(1) Metal corrosion can be modelled in MELCOR and SPECTRA by defining a user input for the reaction coefficients.

(2) Modelling of D1, D2, and D3 by LP codes requires specific and detailed nodalization schemes.

(3) For CFX and FLUENT, the condensation models are mostly developed by users (i.e., France, Germany, the Netherlands, etc.). While CFX provides a basic model for condensation, there is condensation model available in the general FLUENT package provided by the code developer.
(4) The code developers performed fundamental qualification/verification of the turbulence models of their codes against small scale tests with turbulence characteristics measured; the users performed benchmarks against large scale integral tests where turbulence took a significant role in mixing, but the turbulence characteristics were not quantified.
(5) Even though CFX and FLUENT can simulate multiphase flows with heat and mass transfer, to limit the computational effort, containment simulations are performed in single phase. The water in films and pools are either neglected or modelled using a Lagrangian approach. User models for heat and mass transfer in the sump are under development for CFX in Germany. User models for heat and mass transfer with water films on the wall and in the sump are developed and being testing for FLUENT in the Netherlands.
(6) GASFLOW and SPECTRA can predict whether a hydrogen-air-steam mixture is flammable or detonable, and if it has the potential risk of FA. However, the code does not simulate the FA, DDT, and detonation.
(7) Detailed PAR models in CFX and FLUENT are developed by code users (i.e., Germany, the Netherlands) and are not available in general package obtained from the code developer. Simple models to employ manufacturer correlations can be easily implemented in both codes with general packages.
(8) ASTEC, COCOSYS and MELCOR allow the user to trigger the ignition in zones where PARs exist, but no specific model exits.
(9) All codes except AUTODYN can potentially use the similar approach for H_2 recombination to model H_2-CO recombination, but such application is only done for the labelled codes. Validation of PAR models for CO-recombination is only partly done by users as no enough data are publically available.
(10) Similar models for filtered venting system are built into these codes, which employ user defined parameters (i.e., removal efficiency, decontamination factor, etc.). For CFX and FLUENT, similar approach can be used, but no application has been performed.
(11) A general spray model is available in CFX, but it has not been applied for containment spray modelling yet. Some users (i.e., the Netherlands) have developed their own spray model in FLUENT.

Table 4-3
Summary of Codes Used for the Recent International Benchmarks

Benchmarks/Projects	Key Phenomena	LP codes					3D codes					
		ASTEC	MAAP/MAAP-CANDU	MELCOR	SPECTRA	COCOSYS	TONUS	GOTHIC	GASFLOW	CFX	FLUENT	AUTODYN
ISP-31 [4.15]	G2			x								
ISP-42 [1.39], [1.40]	D1, D3, D4			x	x	x		x				
ISP-47 [1.44]	D1, D3, D4	x		x		x	x	x	x	x		
THAI HM2 [4.1]	D1, D3, D4	x		x		x		x	x		x	
SETH-2 [1.46]	D1, D3, D4, M6, M7	x					x	x		x	x	
ERCOSAM-SAMARA [1.47]	D1, D3, D4, M6, M7	x				x		x	x	x	x	
SARNET [1.41]	C2, C3						x				x	
ISP-49 [1.53]	C1, C2	x		x			x			x	x	
SARNET PARIS-1 [4.16]	M1						x		x	x	x	
SARNET PARIS-2 [4.17]	M1								x	x		
SARNET2-PAR2 [4.18]	M1	x				x		x			x	
THAI HR tests [1.45]	M1			x		x		x	x	x		
THAI2 HR35 [4.19]	M1			x		x		x	x	x		

5 CONCLUSIONS AND RECOMMENDATIONS

This report is one of the post-Fukushima activities initiated by the OECD WGAMA group in December 2012. The purpose is to review the approaches for hydrogen management under SA conditions within the OECD member countries, including safety requirements, mitigation systems and their implementation status, analysis codes and their validation status, and SA management strategies. This report mostly covers the information provided by participants from Belgium, Canada, Czech Republic, Finland, France, Germany, Italy, Japan, Korea, the Netherlands, Poland, Sweden, Switzerland, Spain, and USA.

Chapter 1 of this report provides some general background on hydrogen behaviour and control in SAs, and includes a brief literature review on early US NRC sponsored research (1980-1998) and recent international OECD and EC sponsored research (1999-present) on hydrogen behaviour (generation, distribution, combustion and mitigation). The significant factors are:

- It has been recognized that (even though not further detailed in this report) the in-vessel hydrogen generation processes are the main factors influencing the hydrogen risk, which is generally well simulated when the core geometry is still intact, but some knowledge is needed for the late phase of the core degradation and reflooding, which is addressed by the former CORA experiments and the on-going QUENCH experiments at KIT. One international benchmark on hydrogen generation during core heat-up and quenching was organised (ISP-31) against the CORA-13 experiments.
- Hydrogen distribution under various thermal-hydraulic conditions has been widely examined in various scales of experimental facilities. Recent research is directed to provide data for the development and validation of 3D codes. For this purpose, the experimental facilities are well-instrumented using multiple measurement techniques (e.g., THAI, SETH/SETH2 projects). The research is also directed to examine the effect of engineering systems (i.e., spray, local air cooler, thermal effect of PAR) on hydrogen distribution (e.g., ERCOSAM and HYMERES projects).
- Many international benchmark simulations on hydrogen distribution (ISP-23, 29, 35, 39, 42 and 47) have been performed using various codes, including LP and 3D codes. User guidelines for LP and 3D codes exist. Advantage and drawbacks of LP and 3D codes for hydrogen distribution are well recognized.
- The early experimental research on hydrogen combustion sponsored by US NRC has provided a foundation for understanding of hydrogen combustion behaviour in various regimes. The recent experimental research on hydrogen combustion is directed to provide new 3D data for model development as well as to examine the effect of spray on combustion (i.e., THAI project).
- International benchmark simulations on combustion (i.e., ISP-49, SARNET) have been organized in recent years. Uncertainties were observed in predicting flame speed for mixtures with non-uniform hydrogen concentrations. Future model development and validation need to consider geometrical configurations typical to NPP environment (e.g., partially enclosed volumes with the irregular obstruction, flat layer, etc.).
- A large number of experimental data exist on PAR performance under a wide range of conditions relevant to SAs. The recent research is directed towards examining PAR performance under specific conditions (i.e., extremely low oxygen concentration, PAR induced combustion). Benchmark simulations against PAR experiments were performed in the frame of OECD-THAI project and within the OECD-THAI-2 project. For the PAR model, most codes employ simple engineering correlations to calculate the PAR recombination rate or efficiency. Progress has been made in developing more sophisticated PAR models in some countries (i.e., France, Germany).

Chapter 2 and Appendix A describes containment design features of the various reactors, including Western PWRs, BWRs, Eastern European VVERs, and PHWRs (CANDUs), to demonstrate their relation to hydrogen management measures. The observations are as follows:

- The choice of hydrogen mitigation strategy depends primarily on the design of the containments.
- For NPPs with large dry containment (e.g., PWR, PHWR, and VVER-1000), the strategy is mostly a combination of a large free containment volume with the use of many PARs, and/or glow plug igniters.
- For BWRs with Mark III containments or the US PWR and Finnish VVER-440 both with ice condenser containments, the deliberate ignition concept was explored (due to the specific conditions within the containment and the design parameters) leading to the installation of glow plug igniters (active systems).
- For NPPs with small containments (e.g., BWR type 69 in Germany or BWR Mark I), nitrogen inerting in the whole containment is typically applied. PARs are used in addition if the dry-well cannot be inerted as for example in the German BWR type 72.
- For VVER-440 different means exist depending on the containment design (bubble condenser tower, ice condensers).

Chapter 3 provides a detailed description of national requirements on hydrogen management and hydrogen mitigation measures inside the containment and other places (e.g., annulus space, secondary buildings, spent fuel pool, etc.). Hydrogen analysis approaches, consideration for application of safety systems (e.g., spray, containment ventilation, local air cooler, suppression pool, latch systems) in hydrogen measurement strategies as well as lessons learnt from the Fukushima accident are discussed. The findings are as follows:

- The national requirements of the member countries on hydrogen management for SAs in the various NPPs under operation vary in details. Many countries define maximum mean and local hydrogen concentrations (typically PWR) for the design of igniter or PAR concepts or a maximum oxygen concentration (typically BWR) for N_2 inerting-concepts.
- The optimal location for PARs installed and the number required in a NPP is plant specific. General recommendations and guidelines exist in some countries (i.e., PAR system implementation procedure in Germany for large dry PWR containments).
- In response to the Fukushima accidents, hydrogen mitigation systems, particularly PARs, are now required to be installed in most of the countries inside the containment if there was no mitigation concept required before.
- The existing hydrogen mitigation systems are being evaluated and considered to be enhanced under SA conditions in such countries where they are designed for DBAs only.
- The Fukushima accident showed that hydrogen transport from the inerted BWR containment to the surrounding reactor buildings could be another important subject. The hydrogen accumulation in the reactor building in Fukushima and the hydrogen combustion processes led to a strong destruction of the reactor building structure of 3 out of 4 units. Investigations on possible hydrogen transfer to buildings adjacent to the containment became the focus of additional activities after the Fukushima accidents. This can be caused by significant leakages from the containment through failures at penetrations, latches, and other larger containment openings and/or from a containment vent line in case that its integrity is breached.
- Most countries have not yet adopted specific national requirements for hydrogen mitigation measures outside the containment (e.g., annulus, reactor or secondary building, etc.) or the spent fuel pool areas. Due to the Fukushima accidents, many countries have started to study SA conditions within such areas and to consider hydrogen management outside of the primary containment (i.e., reactor building) and at the spent fuel pool area. Question remains open regarding the need of hydrogen management outside the containment and decision has to be made whether additional mitigation measures are required.
- The Fukushima accident provided some insights for hydrogen mitigation strategies outside of the containment. For example for the BWR Mark I reactor building, the hydrogen mitigation scheme could consist of many levels, such as (1) reliable containment venting to limit potential leakage into the reactor building, (2) installation of PARs in the reactor building

based on hydrogen source from leakage, SFP, (3) monitoring of hydrogen concentration in the reactor building, and (4) choice of actions to be integrated into the SAMGs.

- In most countries, the requirements for operation of the engineering systems (e.g., spray, containment ventilation, local air cooler, suppression pool, latch systems) are defined based on their primary purpose (e.g., heat removal or depressurization). It is expected even though not further substantiated in this report that these requirements may be (or have to be) defined/updated to take into account their effect on hydrogen behaviour.
- Hydrogen measurement systems (e.g., gas sampling/monitoring) have been installed in some NPPs. It is under consideration by many countries now, but the implementation details vary.

Chapter 4 and Appendix B assess various codes capabilities and validation status in terms of hydrogen related phenomena (e.g., generation, distribution, combustion and mitigation). Eleven codes are evaluated in this report, including ASTEC, MAAP/MAAP-CANDU, MELCOR, SPECTRA, COCOSYS, TONUS, GOTHIC, GASFLOW, CFX, FLUENT, and AUTODYN. The conclusions are:

- Most countries are using lumped parameter codes (e.g., integral or system codes with mechanistic models) for full plant long term SA analysis combined with 3D-like or 3D codes for detailed short-term and/or local hydrogen analysis (e.g., hydrogen distribution, combustion and mitigation).
- Amongst the 11 codes assessed, only the integral or system codes are capable of calculating hydrogen generation in the reactor core and/or from MCCI in the cavity. No codes evaluated in this report have models to calculate hydrogen sources from water radiolysis which is generally calculated by other codes and implemented to these codes as a source term. The application to spent fuel pool SAs has started after the Fukushima accident and this needs further attention.
- All the LP codes have capabilities to model hydrogen distribution, combustion (only deflagration and/or diffusion flames) and mitigation systems (i.e., PAR, N_2 inerting, igniter systems) and engineered safety features (containment spray, cooler, etc.), and they are mostly based on mechanistic models. Development of sophisticated models with CFD codes for hydrogen combustion (i.e., deflagration, FA or DDT) and recombination (e.g., detailed PAR modelling) are being performed by some countries.
- Even though the LP codes cover a broad range of phenomena and allow performing integral calculations, attention must be paid when assessing the hydrogen behaviour in the containment. There are still uncertainties regarding the hydrogen generation in-vessel and ex-vessel and consequently the hydrogen release into the containment. Therefore, code sensitivity studies are recommended and necessary to capture inherent uncertainties, including consideration of various potential RCS release paths into the containment.
- LP codes employ simpler physics, models and nodalization as compared to 3D codes. Therefore, some phenomena such as hydrogen stratification, mixing, FA and detonation are difficult to predict. Applying LP codes typically requires a more detailed and specific nodalization of the containment, thus user experience is important. It is highly recommended that appropriate nodalization be derived from analyses of different experiments and benchmarks. Nevertheless an experienced user can sometimes overcome remaining limitations imposing ad-hoc nodalization and criteria.
- 3D codes are generally more capable of modelling hydrogen distribution, combustion and mitigation in complex geometries than LP codes, but its relative high computational cost limits its application for real plant analysis. In addition, the standard commercial CFD codes (i.e., FLUENT, CFX) don’t take into account the specific phenomena relevant to hydrogen analysis during a SA. Hence, these codes need to be customized by applying user functions. Significant progress has been made in several countries (i.e., France, Germany, the Netherlands, etc.) in complementing FLUENT and CFX with adequate models for condensation, evaporation, sprays, PARs and combustion.

- Most codes discussed in this report have simple models for PAR to calculate hydrogen recombination rate and/or efficiency using manufacturer correlations. Progress has been made in some countries (i.e., France, Germany, etc.) in developing PAR models to capture detailed operational behaviour of a PAR and to predict PAR induced ignition limits.
- Validations performed by the code developers or users for hydrogen related phenomena (generation, distribution, combustion and mitigation) vary largely. Some countries have performed extensive validations using their own experimental data and/or international benchmarks, but some heavily rely on code developers.
- None of the codes hereby described is fully validated regarding the complete list of phenomena either due to a lack of experimental data or due to a limited application range. Engineering judgement and a large degree of know-how on code application is therefore needed in order to obtain realistic results.
- For hydrogen generation from in-vessel metal-steam and/or metal-oxygen reaction, validations have been performed for all the codes that have these capabilities. No validations (or limited validations) exist for some codes for hydrogen generation from containment metal corrosion, oxidation in molten pools and MCCI.
- A large number of validations exist for the codes that are dedicated for simulation of hydrogen distribution and deflagration. Validations are limited or large uncertainties exist for fast deflagration and FA. Validations for PAR modeled with simple approach have been performed for most codes using tests performed under normal conditions (i.e., sufficient air). No validations have been performed for CO recombination and hydrogen (and/or H_2-CO) ignition by PAR.
- The user impact can affect the result in the same order of importance as the general code validation, thus the users must be trained and follow carefully the available best user practices and results from international benchmark activities in order to obtain reliable results.

In conclusion, R&D efforts to date have already significantly enhanced the understanding of the phenomena governing the distribution of hydrogen gas mixtures and their potential combustion. Research will continue to better understand SA conditions as a result of interactions to equipment and components inside containment in an effort to reduce uncertainties and provide insights to refine the SAMGs. Regarding computational tools, although they have clearly reached a degree of maturity, their predictive capacity still needs to be reinforced.

It is noteworthy that prior to the Fukushima accidents, SA analyses of BWR Mark I containments also revealed that failure of the primary containment would cause sufficient amounts of combustible gas to migrate into the reactor building, and subsequently, the building would fail due to combustion. Similarly, a large amount of combustible gas entered into the Fukushima reactor buildings causing significant combustion events, e.g., a strong deflagration or DDT. Furthermore, overpressure due to even a weak deflagration would also likely fail the reactor building. A hydrogen combustion event did not occur in Fukushima Unit 2, even though the reactor core was significantly damaged. The blowout panel of the top floor of the reactor building in Unit 2 had been possibly opened by the Unit 1 hydrogen explosion, which might have prevented accumulation of hydrogen within the reactor building due to venting. However, it shall be noted that opening of the blowout panel may lead to subsequent direct release of radioactive materials to environment.

Extensive research activities on PAR have been done by the main PAR vendors in the late 80’s and 90’s. Additional research is being performed in national and international levels, e.g., PAR operation under various SA relevant conditions including experiments to determine PAR induced ignition limits as well as PAR operation under oxygen starvation or in the presence of H_2 and CO as it is typically for ex-vessel situations. Implementation of the knowledge gained through R&D into model development is in progress.

Efforts are still needed to close research gaps, enhance code capability, and reduce code uncertainty, such as:

- Research on the effect of engineering systems (e.g., containment spray, air cooler, etc.) on hydrogen distribution is on-going in some countries. It is not assessed in this report how the knowledge gained through the R&D has been implemented in the plant safety analysis and how they are considered in the current SAMG.
- None of the codes evaluated in this report have models to calculate hydrogen generation due to radiolysis. During the post-Fukushima accident analysis, it has been revealed that some uncertainties existed in predicting hydrogen generation due to radiolysis in the spent fuel pool. Further model development/validation may be needed. Nevertheless, this is not a significant concern because the dominating hydrogen source would be produced from clad oxidation in postulated loss of coolant SFP accidents.
- Studies performed in framework of PSA level 2 by some countries showed that FA cannot be ruled out even with the use of PARs. Therefore, the effect of pressure loads due to combustion of hydrogen and/or carbon monoxide mixture on containment and equipment (especially those qualified important for safety) needs to be assessed under in-vessel and ex-vessel conditions, but this may be plant specific.
- Concerns are raised on hydrogen measurement strategies. In most NPPs, it is a single point or a few (e.g., up to 10) points measurement. It is not analysed or discussed in the report how the measurement locations are determined and how the progress of the accident based on the limited number of hydrogen concentration measurements is determined, and whether the information is sufficient to provide guidance for SAM, e.g., decisions for activation of safety systems, such as spray.
- Uncertainties in modelling fast (or turbulent) combustion in mixture with non-uniform hydrogen concentrations remain large by both LP and 3D/CFD codes as revealed in the ISP-49 and SARNET benchmarks. The tests used for these benchmarks were all performed in tube-like geometries with regular obstructions. Geometric configurations typical to NPP environment (e.g., partially enclosed volumes with the irregular obstruction, flat layer, etc.) need to be considered in the future model development and validation.

In summary, addressing the above questions will allow the recommendations highlighted in the European stress tests report [3.25] published after the Fukushima accident to be answered.

The present report is an adequate basis for reviewing SAM strategies for hydrogen management. It is recommended that assessment of the SAM strategies or guidelines as well as advantages and drawbacks of the various hydrogen mitigation approaches be pursued as a follow-up activity.

A DESCRIPTION OF CONTAINMENT DESIGNS

A.1 PWRs

A.1.1 German PWR of Type KONVOI

There are 3 units of PWR type KONVOI **(**see Figure A.1-1) in operation in Germany and 4 units with a very similar design, the so called pre-KONVOI design. These are the latest developments made by the former Siemens-KWU. Earlier developments are described below in Section A.1.2 for the Spanish 3-loop PWR and in Section A.1.3 for the Netherlands 2-loop PWR. Those plants are still in operation while similar units in Germany have been shut off after the Fukushima accident due to a decision of the German government [A.1].

These German NPPs have been put into operation between June 1982 and April 1989. The power of the plants is between 1345 MWe and 1485 MWe. The differences are caused by different power uprate measures. All of the operating NPPs are single units. For three NPPs the older neighbouring unit have been shut off after the Fukushima accident in 2011.

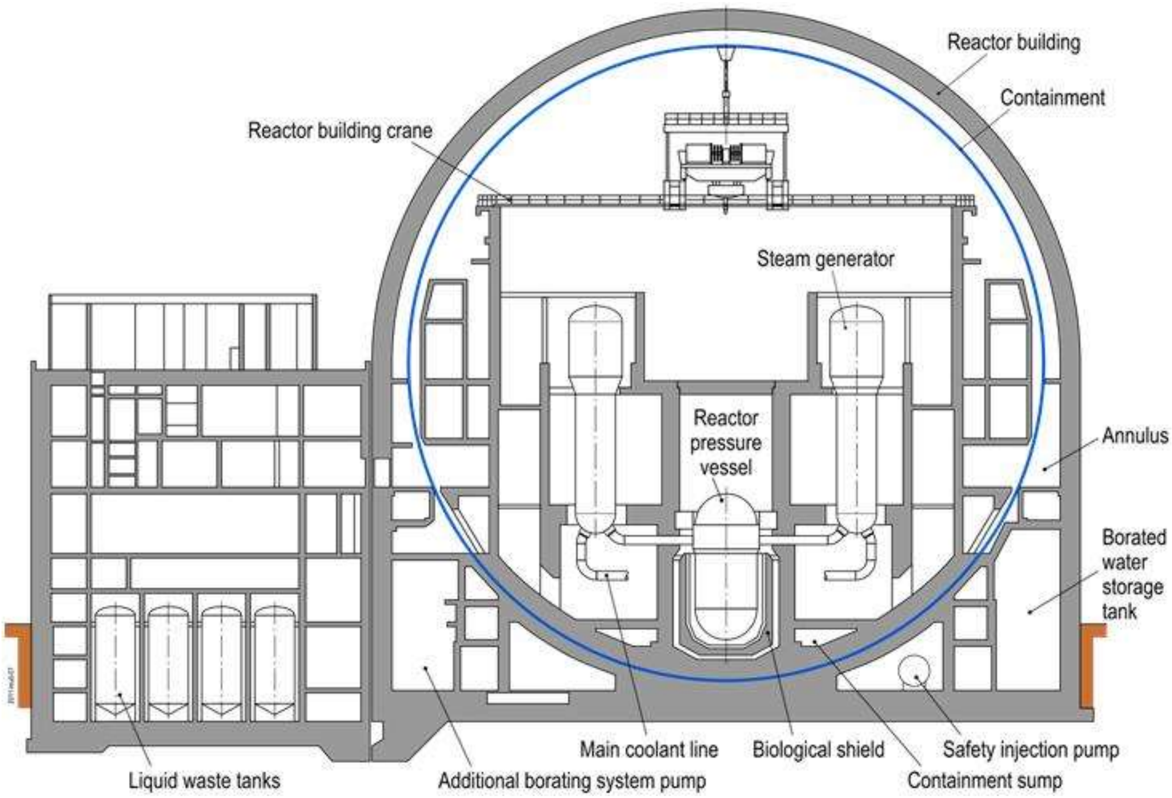


Figure A.1-1 German four loop PWR type Konvoi NPP [A.1]

The PWR KONVOI units consist of a steel containment and a surrounding reactor building made of concrete. The containment provides a barrier against the release of radioactive substances. It consists of a spherical steel vessel with a diameter of 56 m and a wall thickness of typically 38 mm and is designed against pressures and temperatures occurring during DBAs. The lower spherical part rests on a concrete foundation; apart from that, the containment is self-supported. The containment contains the entire reactor coolant system which is under operating pressure, the SFP and parts of the directly connecting safety systems and reactor auxiliary systems. During operation, the containment is continuously ventilated and the rooms not containing the main components are accessible so that inspections, preparatory work for inspections or fuel handling in the SFP located in the upper containment area may take place during plant operation.

The reactor systems are located in the containment, while the Emergency Core Cooling (ECC) systems are located in the reactor building annulus and in the reactor auxiliary building. The main reactor systems, particularly those important to safety, are volume control system, extra borating system, coolant treatment. The containment related systems are hydrogen mixing system with thermal recombiners, exhaust system, and nuclear ventilation system. There is no containment spray system neither installed nor needed. The containment design pressure is in the range of 6.3 bar (a) and its net volume is about 7000 m^3.

The reactor building, which consists of a hemispherical dome and a cylindrical base, surrounds the containment. The reactor building has a wall thickness of approximately 1.8 m and rests on a foundation. It protects the containment against external hazards, such as aircraft impact and explosion pressure waves. The area between the lower cylindrical part of the reactor building and the containment forms the annulus where parts of the safety systems are designed redundantly, and where parts of the reactor systems are located. In case of an accident with pressure or temperature increase in the containment, the containment isolation and the annulus isolation are triggered (delayed by some minutes). Air ventilation of the annulus is stopped and the emergency sub-atmospheric pressure system in the annulus is started. This system has the task to retain the sub-atmospheric pressure in the reactor building annulus and to filter potential leakages from the containment vessel before discharge into the environment. For external events, the time frame for which no operator action is required is at least 10 hours.

As part of the accident management concept, a large number of PARs have been installed in the containment to prevent global hydrogen combustions challenging the containment integrity. Typically up to 65 AREVA PARs of different size are installed. Selected SA scenarios with core degradation and MCCI in the dry cavity have been used for the analysis of the PAR system design. Accidents in the SFP have not been considered. No hydrogen mitigation measures are foreseen in the annulus. For long-term over-pressure protection of the containment a FCV system was installed as part of accident management concept. Dry and wet filtration is used. SAMG have been recommended after the Fukushima accident and will be implemented end of 2013 respectively early 2014 according to the national action plan of Germany [A.2].

A.1.2 Spanish PWR 1000 (Trillo)

Trillo NPP was designed by Siemens-KWU, Germany, and has been operational since 1988. An overview of the Trillo NPP is shown in Figure A.1-2. It is very similar in general to the German PWR of type KONVOI described in the chapter above.

Trillo NPP is a three loop PWR with a thermal output of 3010 MWt and 1066 MWe of electricity production. Trillo has a large dry containment which provides the ultimate barrier against radioactive products releases. It consists of a spherical steel vessel with an inner diameter of 53 m and a wall thickness between 31 to 50 mm. The containment free volume is 61100 m^3 and the containment design pressure is 6.38 bar (a) [A.3].

The reactor coolant system and the SFP are located within the containment building. The ECCSs are located in the annulus building and the auxiliary building. As shown in Figure A.1-2, the containment is surrounded by a concrete reactor building with annulus in between. It consists of a hemispherical dome and a cylindrical base which remains at sub-atmospheric pressure after a DBA due to a special air suction system, as described for the German KONVOI plant.

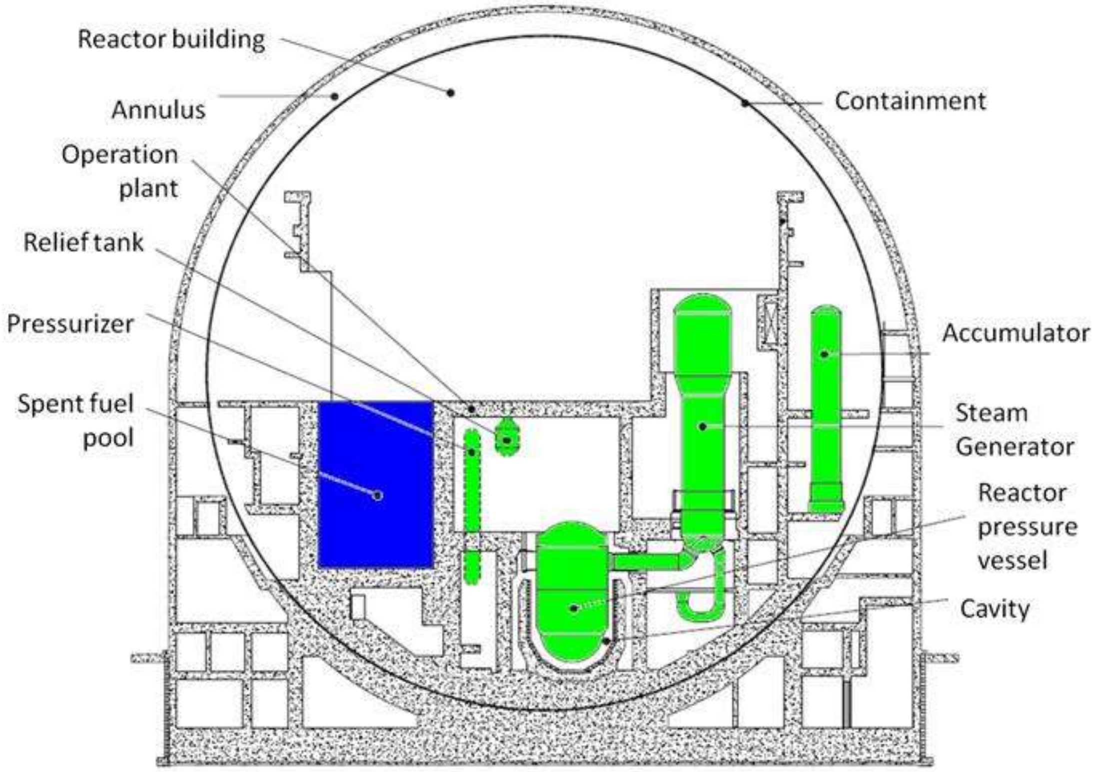


Figure A.1-2 Spanish three loop PWR with Siemens-KWU containment

The containment hydrogen control systems for DBA are: hydrogen detection system, gas mixing system with opening explosion latches at the steam generator rooms and hydrogen thermal recombiners. A PAR system was installed to manage the hydrogen to deal with the hydrogen risk beyond DBA. There is no containment spray system and no containment venting system. After the so-called "stress tests", the installation of a containment filtered venting has been required [A.4]. Currently, the plant has procedures for normal conditions, emergency conditions and beyond DBA. SA Management Guidelines (SAMGs) will be developed in 2016.

A.1.3 The Swiss - PWR 1000

The Swiss Gösgen (KKG) NPP is a PWR of Siemens-KWU design with a reactor unit which has been upgraded several times over the plant life and has today a gross electric power of 1035 MWe (Figure A.1-3). The nuclear section of the plant comprises the reactor building with the steel containment, the reactor auxiliary building and the external spent fuel storage building, completed in 2008, which together form a closed controlled area.

The reactor's spent fuel storage pool, together with the plant components containing radioactivity that are at reactor operating pressure, are enclosed by a spherical steel shell – the containment. The steel containment is spherical and is in an off-centre position inside the reactor building. The steel containment has an inside diameter of 52 m and has design pressure of 5.89 bar (a). The reactor building has an outside diameter of 63.6 m.

Fuel storage pools are located inside the containment in proximity of the reactor. A new storage building, serving as spent fuel building, was built outside the existing building, in the direct proximity of the reactor auxiliary building. This new building comprises a tract for all the control systems with a skywalk to the reactor auxiliary building and two dry cooling towers. The internal structures of the building are separated from the exterior walls, and the spent fuel storage pool is protected against tremors by springs and damping elements. The outer structures of the spent fuel storage building are at least 1.5 m thick. This ensures that the building is protected against exceptional events, such as earthquakes, flooding and an aircraft impact.

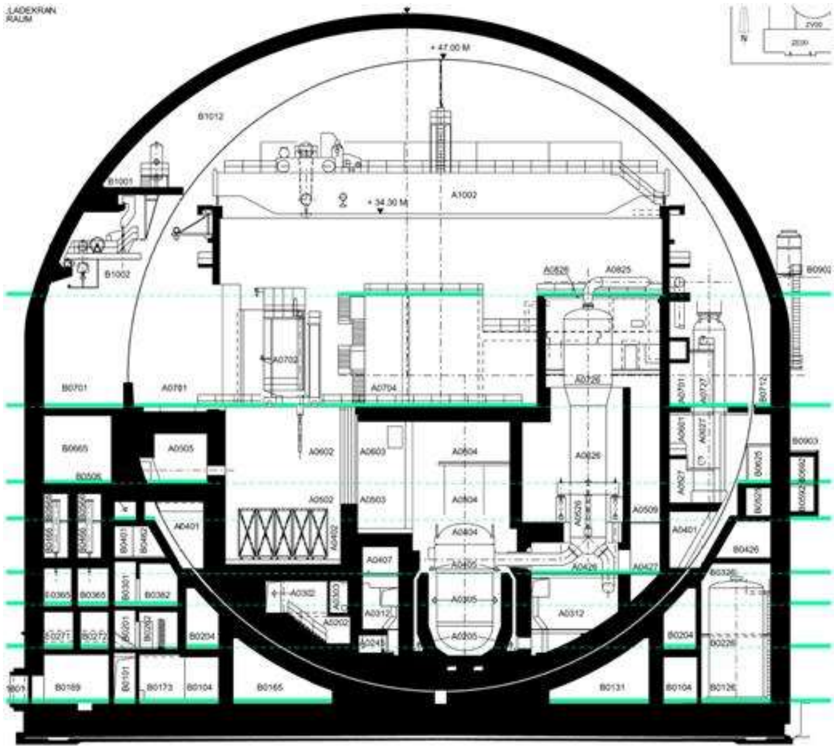


Figure A.1-3 Schematic of the KKG containment (Siemens-KWU design)

Hydrogen mitigation during a postulated SA is obtained mainly by the high containment structural capacity. Hydrogen mitigation is achieved with the use of two thermal recombiners (design equipment) used as igniters under SA conditions (recommended to be used in SAMG), self-actuating passive mixing system (burst membranes between main compartments, e.g., in ceilings and connecting doors opened by shear bolts at minor pressure differences) and containment inerting by steam that occurs as a result of SA progression. Global hydrogen concentration or hydrogen concentration in the large containment compartments for an isolated containment is not exceeding the lower boundary concentration limits usually assumed as the threshold for DDT processes. DDT is also prevented physically by the compartmentalisation of containment (free runaway distances are small for most compartments). Moreover a FCV system is available and can be opened within a pressure range of 1.1 to 10 bar (a) to reduce hydrogen concentration. The venting system is inerted by nitrogen. Venting is combined with water injection into the reactor or (indirectly to the molten core after vessel breach) producing steam to support interting. The passive venting path (rupture disk) opens at 7.2 bar (a). Active venting is performed without the need of any external or internal power supply. The utility is considering implementation of PARs as part of the long term operation investment programme.

A.1.4 The Netherlands - PWR 500

The Borssele NPP was designed and built by Siemens-KWU, Germany and has been operational since 1973. An overview of the Borssele plant is given in Figure A.1-4.

Borssele is the only nuclear power plant operational for electricity production in the Netherlands. It is a 1365 MWt pressurized-water reactor with two loops, each with one primary pump and one steam generator. Currently the gross capacity is 512 MWe and the net capacity is 485 MWe [A.5]. In 2012 the long term operation of the Borssele NPP was extended from 40 to 60 years, till the end of 2033.

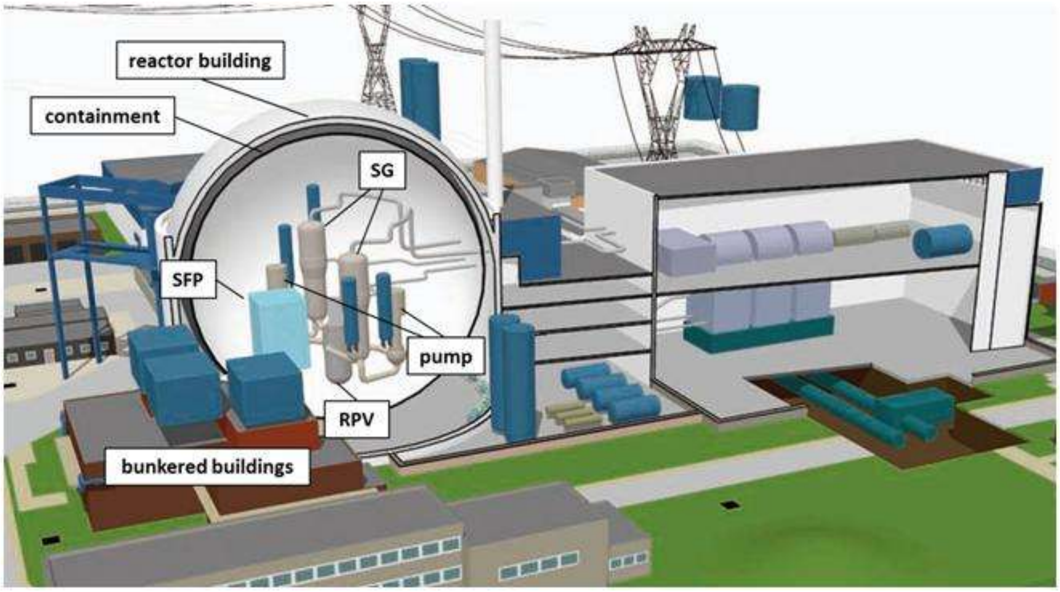


Figure A.1-4 The Borssele two loop PWR NPP in the Netherlands

The containment consists of a spherical steel shell (containment) with a diameter of 46 m and free volume of about 36000 m^3. The steel shell is encapsulated by the concrete reactor building. The containment is designed for 4.8 bar (a). In order to control the pressure, the containment is equipped with a spray system, air coolers, a filtered recirculation system and a filtered venting line (upgrade as part of accident management). PARs are installed in the containment to mitigate the risk of hydrogen during SAs. High concentrations of hydrogen inside the steam generator rooms can be reduced by opening explosion latches to other containment parts (dome, periphery).

In the Borssele NPP design the SFP is located inside the containment. Accidents in the SFP are currently under re-consideration in the light of the Fukushima accident [A.6]. The Borssele NPP design has no core catcher.

To cope with external hazards and beyond-design conditions, important safety systems, such as ECC, SFP cooling, reactor protection system and the emergency control room are installed in "bunkered" buildings. These buildings are qualified to withstand earthquake, flooding, gas cloud explosions, airplane impact and severe weather conditions. The reactor protection system is based on the principle that no operator action is required in the first 30 minutes after the start of the event, for design base accidents. For external events, the time frame for which no operator action is required is at least 10 hours.

Procedures for normal conditions, abnormal conditions and accident conditions are required by the regulatory body. At the Borssele NPP procedures are in use for all operational states (from shutdown to full power) to operate the plant in all possible plant (damage) states:

- Normal conditions (all operational states). Procedures for normal, undisturbed operation include plant and system operating procedures, checklists, surveillance requirement execution procedures, etc.;
- Abnormal conditions (all operational states). Procedures for likely deviations are the so-called S-instructions;
- Accident conditions (from hot-steaming to full power states). Emergency Operating Procedures are entered upon SCRAM and/or safety injection, and consist of:
  - Emergency Operating Procedures;
  - Function Restoration Procedures;
  - SAMGs.

SAMGs are entered upon criteria that identify imminent or occurring core melt (SA) conditions. The Borssele SAMGs are based on the generic WOG SAMGs. The SAMGs include guidance for using the pressure relief valves and various pressurizer spray options to control the RPV pressure.

A.1.5 French PWR

The nineteen nuclear power stations NPPs currently in operation in France are relatively similar. Each station includes from two to six PWRs, giving a total of fifty eight reactors. In addition to this, an EPR-type PWR (see Section A.1.5.2) is currently under construction at the Flamanville site, and an authorization application has been made for another EPR reactor at the Penly site.

French NPPs fleet comprises only PWRs and all have a common design. In addition to the reactor building, the nuclear island includes the main steam system (VVP) that removes the steam to the conventional island, and the building (BK) housing the fuel storage pit. Built adjacent to the reactor building, the BK building is used to store the fuel assemblies before and during the plant unit shutdowns and to cool the spent fuel.

A.1.5.1 French 900 MW to 1450 MW PWR

In spite the standardization of the French NPPs, a number of technological innovations have been introduced as the design and construction of nuclear reactors have progressed. Thus, six reactors groups called “series” can be distinguished:

- 900 MW series
  - the CP0 series with 6 reactors
  - the CPY series comprising the 28 remaining reactors (CP1 with 18 reactors and CP2 with 10 reactors)
- 1300 MW series
  - the P4 series with 8 reactors
  - the P’4 series with 12 reactors
- 1450 MW
  - the N4 series comprising 4 reactors.

The CP0 series consist of three reactor loops (see Figure A.1-5).

The CPY series has a different building design, an intermediate cooling system between the spray system in the event of an accident and that containing the water from the heat sink, and provides for greater management flexibility.

The plant series P4 and P’4 (Figure A.1-6) has significant changes with respect to the CPY series in the design of the circuits and systems protecting the core of the 1300 MWe reactors and the design of the buildings accommodating the installation. The increased power has resulted in a primary system with four steam generators (SG) offering a greater cooling capacity. The P’4 series reactors display a few differences with respect to the P4, notably the fuel building and the design of certain systems.

The N4 series with 1450 MWe reactors differ from the preceding reactors more particularly in the design of the SGs which are more compact, the design of the primary pumps, and the control room.

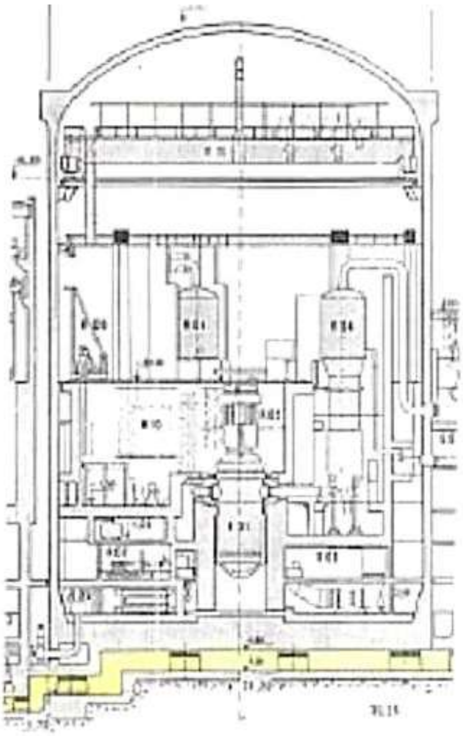


Figure A.1-5 The French three-loop PWR CP0 / CPY series

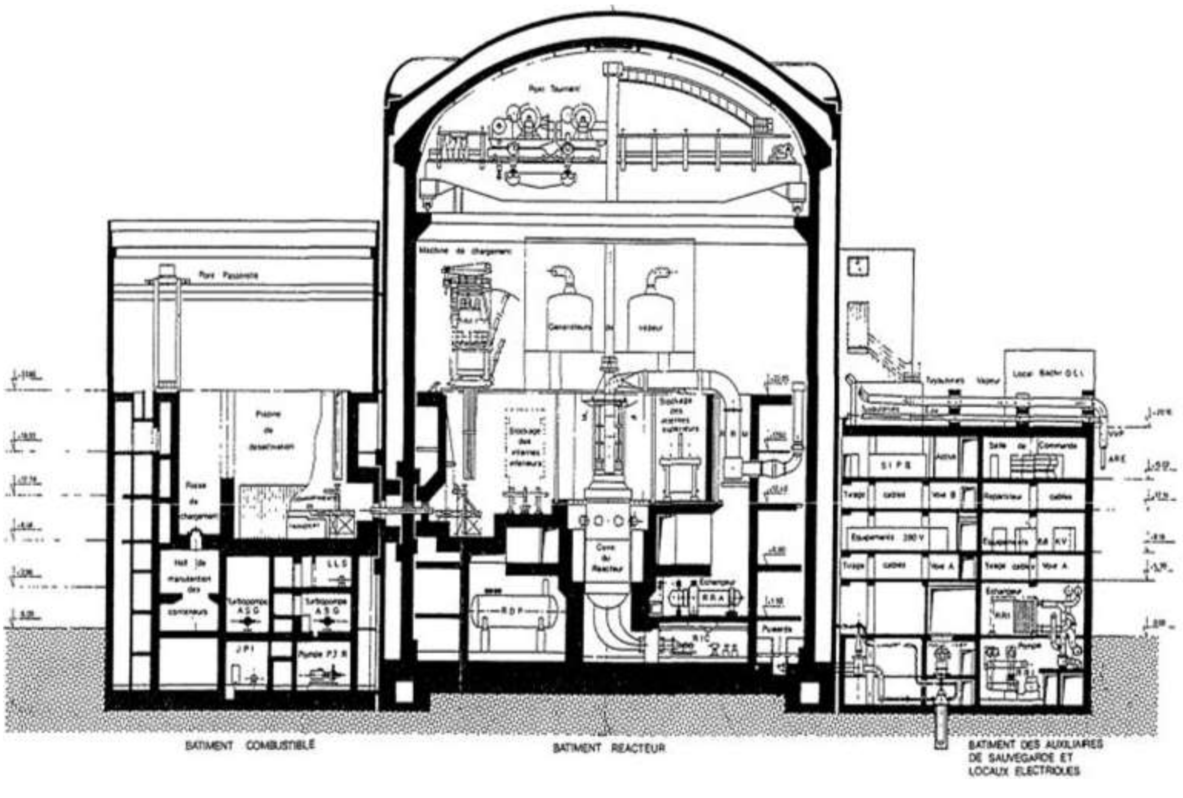


Figure A.1-6 The French four-loop PWR P4, P’4 and N4 series

The PWR reactor containments are designed to withstand the pressures and temperatures that could be reached in an accident situation, and offer sufficient leak tightness in such conditions. Two main types of the containments are then used:

- the 900 MWe reactor containments (see Figure A.1-5), consisting of a single wall of pre-stressed concrete (concrete containing steel cables tensioned to ensure compression of the structure). This wall provides mechanical resistance to the most severe design accident pressure and structural integrity against external hazards. Leak tightness is assured by a metal liner on the inside of the concrete wall. The containment failure pressure is 5 bar (a);
- the 1300 MWe and 1450 MWe reactor containments, comprising two walls, an inner wall made of pre-stressed concrete and an outer wall made of reinforced concrete (see Figure A.1-6). Leak tightness is provided by the inner wall and the ventilation system (EDE or AVS) which, in the annular space between the walls, channels any radioactive fluids and fission products that could come from inside the containment as a result of an accident. Resistance to external hazards is mainly provided by the outer wall. Furthermore, the reactor containment failure pressure is 4.8 bar (a) for the P4 series, 5.2 bar (a) for the P`4 series and 5.3 bar (a) for the N4 series.

All containments are equipped with a spray system, a filtered venting line (upgrade as part of accident management) and PARs.

A.1.5.2 French 1650 MW EPR

The EPR reactors under construction at Flamanville (Flamanville 3, BNI 167), and planned at Penly (Penly 3), are four-loop reactors with a unit electrical output of about 1650 MWe.

Compared with the existing power reactors operating in France, they are characterized by the fact that SA scenarios are integrated from the design stage. Based on the principle of a quadrupling (4 trains) of the safeguard systems (with a few exceptions) and, in addition to the presence of an aircraft impact-resistant shell (protecting the reactor building, the fuel building and two buildings housing two engineered safeguard trains) to counter external hazards, the EPR incorporates, for example, prevention measures:

- to prevent high-pressure core meltdown accidents ;
- to enhance the reliability of the on-site electric power supplies by adding two diversified diesel generator sets (ultimate backup);
- to protect the water supply of the safeguard systems cooling the reactor core and containment;
- by installing the water tank (IRWST tank) directly in the reactor building;
- by having an alternate heat sink based on the "reversed" use of the sea discharge channel, to take in water from the sea;
- by having mitigation measures such as a corium catcher, convect system which allows atmosphere homogenization in case of accident, or having a double-walled containment with a metallic internal sealing liner in the reactor building.

More details are provided in Section A.1.6 related to the EPR under construction in Finland.

A.1.6 EPR in France (Flamanville 3) and in Finland (Olkiluoto 3)

The plant under construction in Finland at Olkiluoto 3 has an electric power of 1600 MWe and thermal power 4300 MWth [A.7]. The plant will have four coolant loops with vertical steam generators.

Slightly different versions of the EPR are under construction in France and in China. In particular, the Flamanville EPR in France does not have a FCV system, but it has the possibility to operate the containment spray with mobile pumps.

The containment is shown in Figure A.1-7. The containment gas volume is about 80000 m^3. The containment design pressure is 5.3 bar (a), and leak tightness is maintained until about 9.6 bar (a). The containment walls are made of pre-stressed concrete with a steel liner. The reactor building (secondary containment) is made of reinforced concrete. The containment pressure is controlled by a spray system. If the containment spray is not available, the pressure can be controlled by a filtered venting system. The SFP is located in the fuel building, outside but adjacent to the containment.

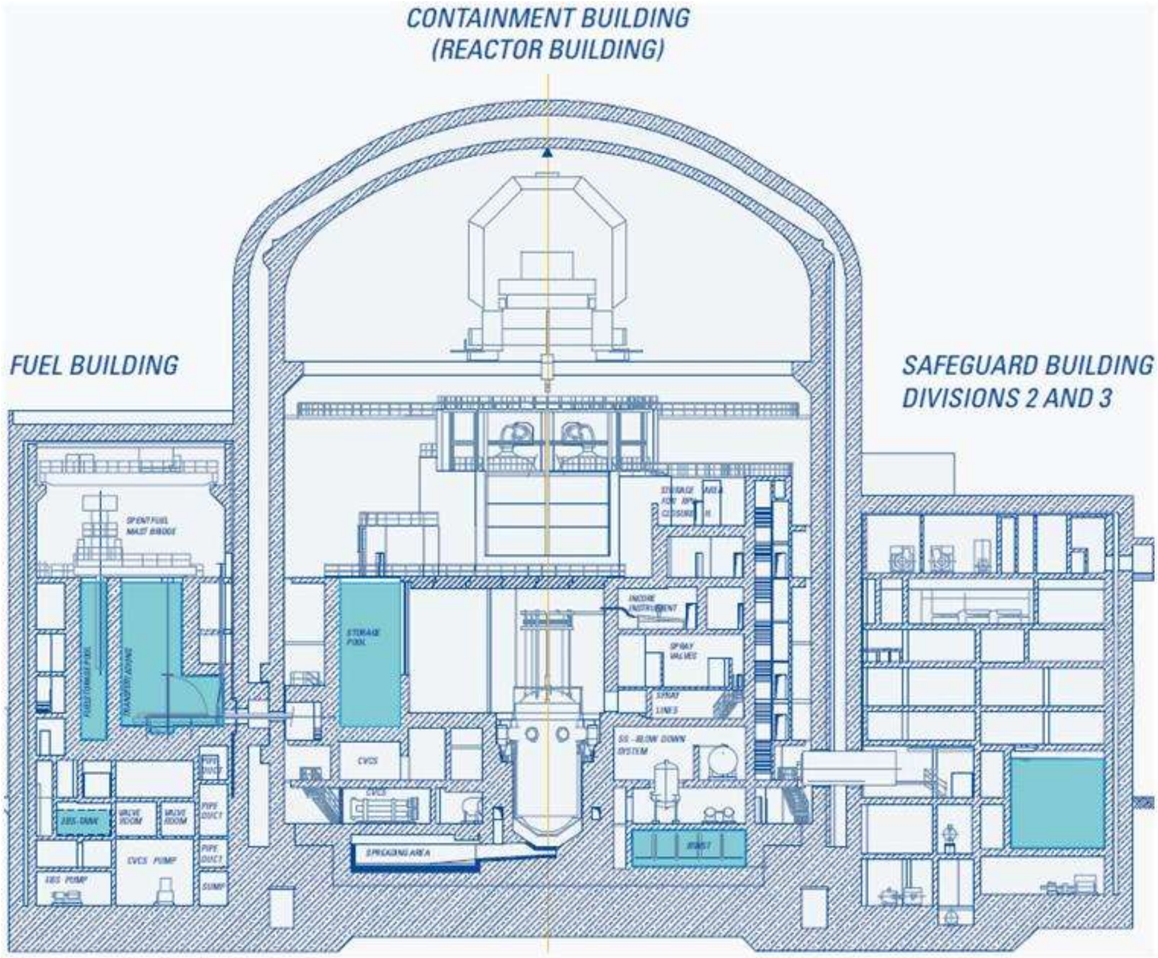


Figure A.1-7 EPR containment (Olkiluoto 3)

In the case of a SA, high pressure melt ejection would be prevented by reactor depressurization valves that are dedicated for SAs. After failure of the RPV, the corium would be temporarily retained in the dry reactor cavity, in order to ensure good conditions for melt spreading, and then released passively to the core catcher. Flooding of the core catcher from the IRWST would be triggered passively by spring-loaded valves.

The hydrogen management is based on efficient mixing and PARs. The hydrogen mixing is achieved by rupture foils in the ceiling of the steam generator room and by hydrogen mixing dampers at a low elevation between the annular space and the IRWST. The foils open by a pressure difference between the steam generator room and the containment dome, or by a high temperature. The hydrogen mixing dampers open passively if AC power is lost. A high pressure in the containment or a large pressure difference between the steam generator room and the rest of the containment also triggers opening of the dampers. Two exhaust pipes from the pressurizer relief tank to both sides of the steam generator room enhance hydrogen mixing if the operators depressurize the reactor. There are about 50 PAR units of AREVA design.

A.1.7 Westinghouse AP1000

The AP1000 [A.8] is a new reactor design from Westinghouse that is closely based on the AP600 and has been certified by the NRC. The NRC has issued four Combined Operating Licenses (COLs), and has several others under review. This design considers the inclusion of passive safety systems for core cooling and containment cooling that will protect the plant for the first 72 hours following an accident without operator action. Figure A.1-8 is a cut away of a generic AP1000 and Figure A.1-9 shows the containment layout.

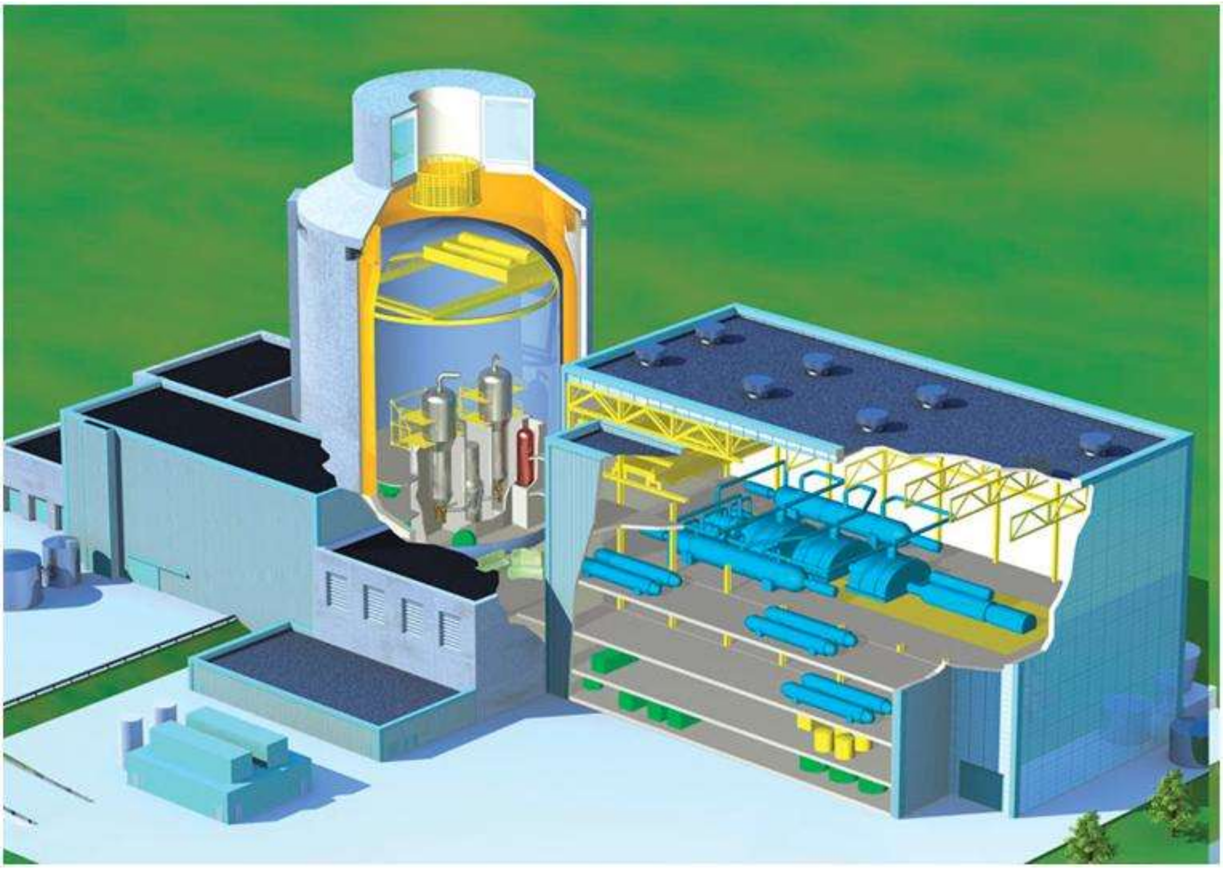


Figure A.1-8 Cut Away of AP1000 Reactor

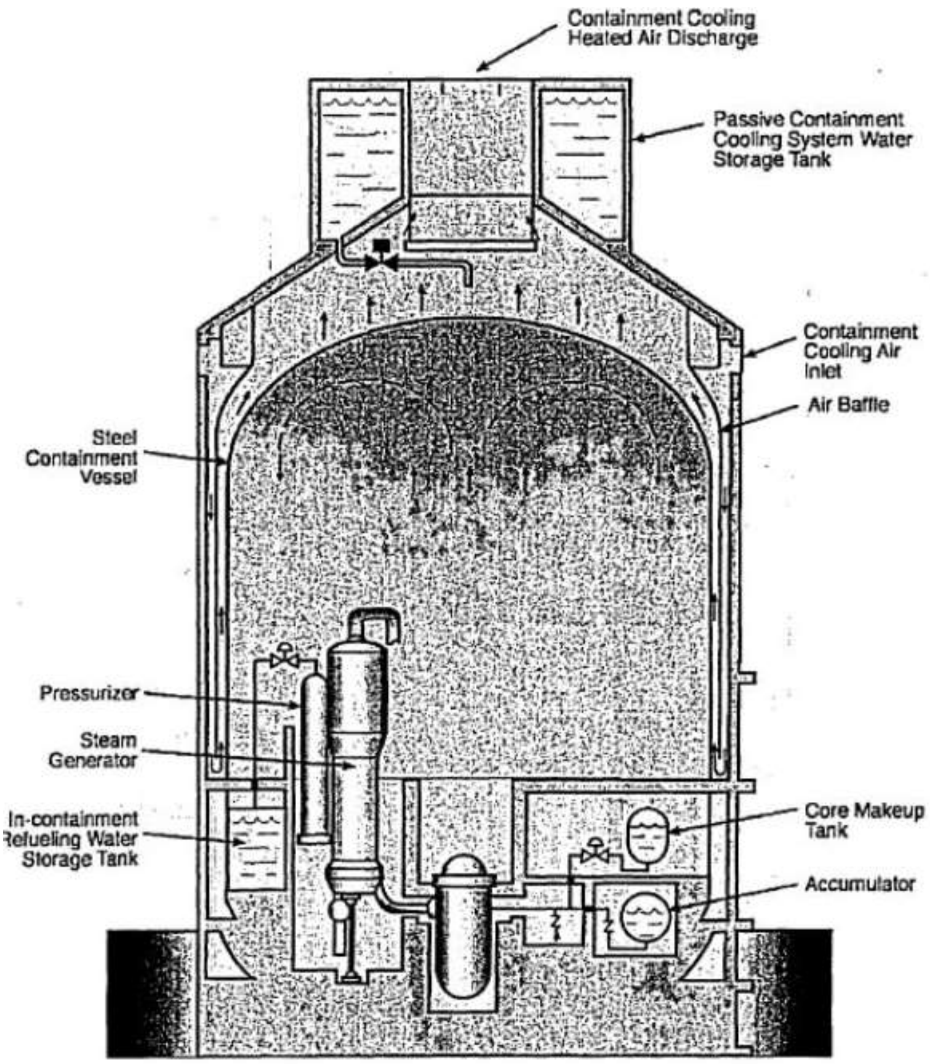


Figure A.1-9 AP1000 Containment Layout

The AP1000 reactor design is a Westinghouse 2-loop PWR with a rated core thermal power of 3400 MWt. The reactor coolant system (reactor vessel, 2 steam generators, reactor coolant pumps, etc.) and associated systems are housed inside the containment vessel. The passive core cooling systems (accumulator, makeup tanks, etc.) are housed inside the containment vessel. Note that the AP1000 does not have a safety-related containment spray system.

The containment vessel is a steel vessel which stands independently inside a shield building. It is a Seismic Category 1 vessel that is 39.62 m in diameter, has a top-to-bottom height of 65.63 m, thickness of ~4.5 cm, and with ellipsoidal heads. The containment bottom head is embedded in concrete. The purpose of the containment vessel is to contain radioactivity from a variety of postulated design basis accidents. Another feature of this vessel is that it is used as a heat sink to transfer energy to the surrounding environment.

The containment vessel is coated with safety-related materials both on the inside surface and outside surface. Inorganic zinc, with a top coat of epoxy, covers the inside surface of the containment vessel. The exterior of the containment vessel is also coated with inorganic zinc to aid with heat transfer. The containment has a net free volume of 58333 m^3, a design pressure of 508.12 kPa and a temperature of 149ºC. The design pressure is based on the containment peak pressure being below the design pressure for all break sizes up to and including the double-ended guillotine break of a reactor coolant pipe or a secondary pipe.

The shield building (reactor building) has multiple functions such as protecting the containment vessel from tornadoes and notably to support the primary cooling water storage tank. Its cylindrical portion has an outside diameter of 44.2 m and is 91.5 cm thick. This water tank is used to cool the outer surface of the containment from postulated high energy line breaks inside containment. There is a gap between the shield building and the containment vessel creating an annular air space, with a baffle, which allows for air flow. The shield building is comprised of both steel/concrete composite and reinforced concrete with the lower part of the shield building resting on a concrete foundation. The steel/concrete composite sections have steel plates on the inside and outside surfaces.

The SFP is located in the auxiliary building which is part of the nuclear island. The auxiliary building wraps half of the circumference of the shield building.

As part of the accident management feature, the AP1000 is designed with a containment hydrogen control system. This system has the functions of monitoring hydrogen concentrations and for hydrogen control during and following postulated severe core damage events. Hydrogen monitoring consists of three sensors that are sensitive to hydrogen concentrations between 0 - 20%, and are placed in the upper dome. These sensors are designed to serve as a post-accident monitoring function. This system is designed to limit hydrogen concentrations and promote hydrogen burning as soon as the lower flammability limit is reached. Burns are promoted primarily through the use of igniters.

The use of igniters is to burn hydrogen continuously at a low concentration and to control the rapid production of hydrogen. Rapid hydrogen generation is expected to occur during a degraded core or core melt SA event, where 100 % of the zirconium cladding (active fuel region) is assumed to have reacted with steam to produce hydrogen.

The AP1000 hydrogen control system includes 64 igniters that are distributed in the containment that include compartmented areas and the upper dome. Igniters are placed in the major areas of the containment where hydrogen maybe released, through which it may flow, and where it may accumulate. Enclosed regions have at least two igniters installed. Igniters are also fitted with a spray shield to protect the igniter from condensed water droplets.

The igniters are divided into two powered groups. Each group being powered by offsite power, and during events where offsite power is not available, will be powered by a non-essential diesel and finally for 4hrs by a battery.

To further defence-in-depth protection, there are two PARs installed inside containment and are located above the operating deck. These PARs are not credited in the safety analysis.

With regard to containment mixing, the AP1000 does not include safety-related containment sprays and does not inert the containment atmosphere. Containment mixing is performed via the promotion of natural circulation. However there is a non-safety related spray function in the form of a fire protection system that could be used for accident management but is not credited in the accident analyses.

A.1.8 Westinghouse PWR 365 (Switzerland)

The Swiss Beznau nuclear power plant (KKB), see Figure A.1-10, consists of two PWR units (Westinghouse) of gross electric power of 365 MWe each unit.

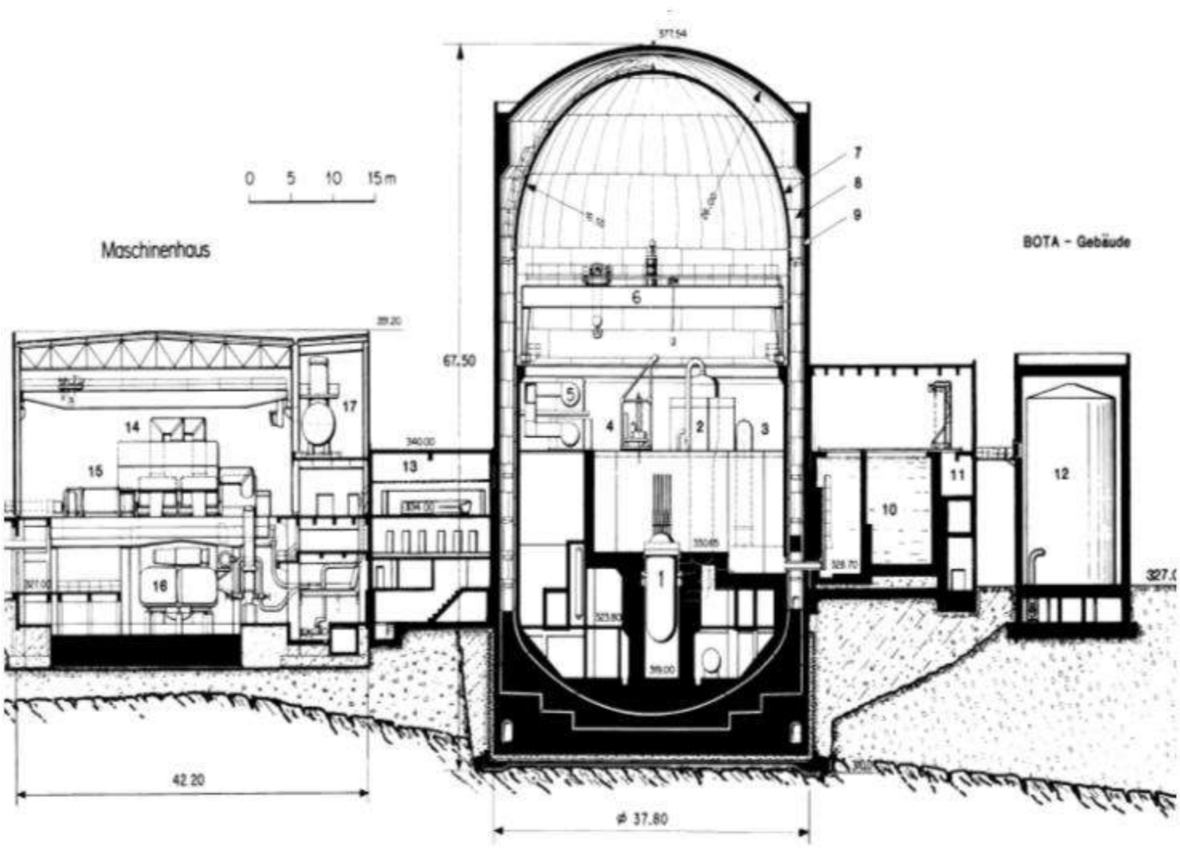


Figure A.1-10 Schematic of the KKB with Westinghouse containment

The containment structure consists of two layers: a large reinforced concrete reactor building, which has an external diameter of 37.8 m and a wall thickness of 0.9 m and an inner steel containment. The reinforced concrete reactor building is made of a cylindrical part and a dome. The inner wall of the concrete reactor building is sealed with a steel liner which has a thickness of 6 mm. It is therefore seen as a secondary containment, too. The containment inside the reactor building is made of steel, has a diameter of 33 m. The volume of the steel containment is 36380 m^3. Up to elevation 349 m (above sea level) the containment is equipped on the inner wall with a reinforced concrete missile shield of 0.6 m thickness. The containment is designed and qualified for a pressure of 4.1 bar (a).

Containment heat removal is at least comprised of spray system and fan-coolers (residual heat removal system and recirculation system etc. remove heat as well from the containment). The plant is equipped with containment venting filter which has the goal to protect the containment against loss of integrity and mitigate the (un-)controlled release of radioactivity in case of some postulated SA. The pressure between containment and annulus is below atmospheric pressure. Any leakage from containment to annulus is pumped back with the containment-pump-back-system into containment.

On the intermediate-term, the spent fuel is stored in a building located outside the containment building. After some years, the spent fuel is taken out of the pool, put into a transport and storage casks (like CASTOR) and is taken over to the plant internal intermediate storage (ZWIBEZ), before it will finally be taken into geological deep stock.

The KKB containment can handle some hydrogen deflagration due to its design. The amount of hydrogen and other flammable gases, which is generated due to MCCI is handled with PARs. The third option to protect containment against loss of integrity (hydrogen concentration close to detonation, high containment pressure) is the controlled venting over the wet scrubber. The containment-venting system is available and can be actuated manually from the emergency control room or locally in the filter-building when the containment pressure is above 4.2 bar (a) or passively through the opening of the rupture disk when the containment pressure reaches 5.2 bar (a).

A.1.9 Westinghouse PWR 1000 (Spain)

Presently, there are 5 reactors in operation in Spain designed by Westinghouse. Their operational dates range from 1981 through 1987. An overview of a typical Westinghouse PWR containment is shown in Figure A.1-11.

All Spanish Westinghouse NPPs are three-loop PWR with an original thermal output around 3000 MWt and 1000 MWe of electricity production, although a number of upgrades has increased their power. All NPPs are equipped with a concrete large dry containment, which provides the ultimate barriers against radioactive products releases. It consists of a concrete vertical cylinder and dome. The inner diameter of the cylinder ranges from 38.41 m to 40 m, the height of the cylinder ranges from 43.4 m to 59.06 m and the free volume ranges from 56900 m^3 to 62100 m^3. The inner face of the containment is covered by a steel liner of about 10 mm thickness. The containment design pressure ranges from 3.78 to 4.7 bar (a) [A.3].

The containment contains the reactor coolant system and the connection with the ECCS, which are located in the auxiliary building. The SFP is located outside the containment building.

The containment hydrogen control system for DBA is comprised of the hydrogen detection system and hydrogen thermal recombiners. Containment heat removal is comprised of containment spray and some plants are also equipped with fan coolers. Currently, there is no PAR system and no containment venting system installed in these plants to deal with the hydrogen risk beyond DBA, although they have been required after the “stress test” [A.3]. The plants have procedures for normal conditions, emergency conditions and SAs (SAMGs).

In SAMG for W-PWR it is defined: if no hydrogen mitigation systems have been installed, two strategies are recommended:

- Ignition by trying to initiate the ignition by switching on/off different operational systems.
- Containment pressurization by switching off sprays/fan-coolers and opening PRZ and RPV PORVs.
- Reduce hydrogen concentration by switching on thermal recombiners.

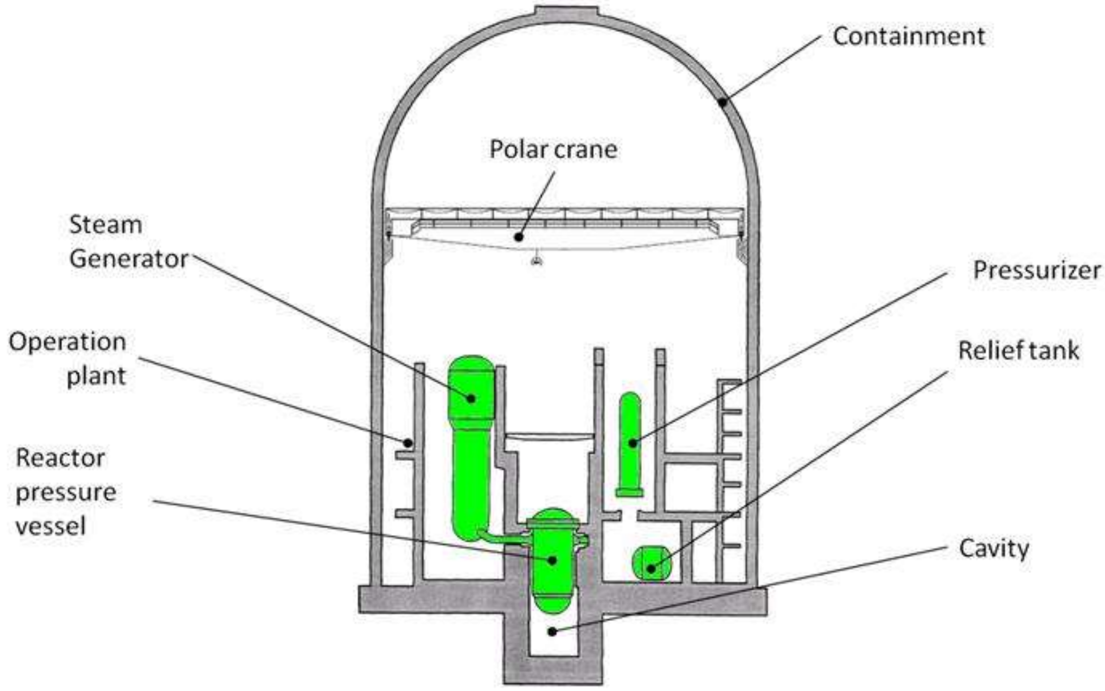


Figure A.1-11 Spanish three-loop PWR Westinghouse typical containment

A.1.10 PWR of Westinghouse or Framatome design (Belgium)

There are 7 units of PWR type in operation in Belgium, four at the Doel site located on the Scheldt River and three at the Tihange site located on the Meuse River.

Doel 1 and 2 are twin units of Westinghouse design and in operation since 1975 (2 loops per unit, 440 MWe per unit since SG replacement and power uprate).

All other NPPs are single units. Tihange 1 is a 3-loop unit of Framatome design and in operation since 1975 (960 MWe since SG replacement and power uprate). Doel 3 and Tihange 2 are 3-loop units of Framatome design and in operation since 1982-1983 (~1000 MWe since SG replacement and power uprate). Doel 4 and Tihange 3 are 3-loop units of Westinghouse design and in operation since 1985 (~1040 MWe, with SG replacement but no power uprate), see Figure A.1-12.

All reactor buildings are equipped with a double containment structure.

At Doel 1-2, the primary (inner) containment consists of a spherical steel containment. The secondary (outer) containment or reactor building consists of a reinforced concrete cylinder on which a reinforced concrete hemispherical dome is placed. The secondary containment encapsulates the primary containment and protects it against external accidents.

At the other Doel units and all Tihange units, the primary (inner) containment consists of a cylinder on which a hemispherical dome is placed. Both structures are made of pre-stressed concrete. This containment is internally covered with a steel liner. The secondary containment or reactor building consists of a reinforced concrete structure enclosing the primary containment and providing protection against external accidents.

In order to control the containment pressure after an accident leading to pressure increase, the inner containment is equipped with a spray system. Spraying is also used to capture radioactive particles and iodine in the inner containment. In the inner containment of the Doel units, containment pressure control is also achieved by means of fan coolers.

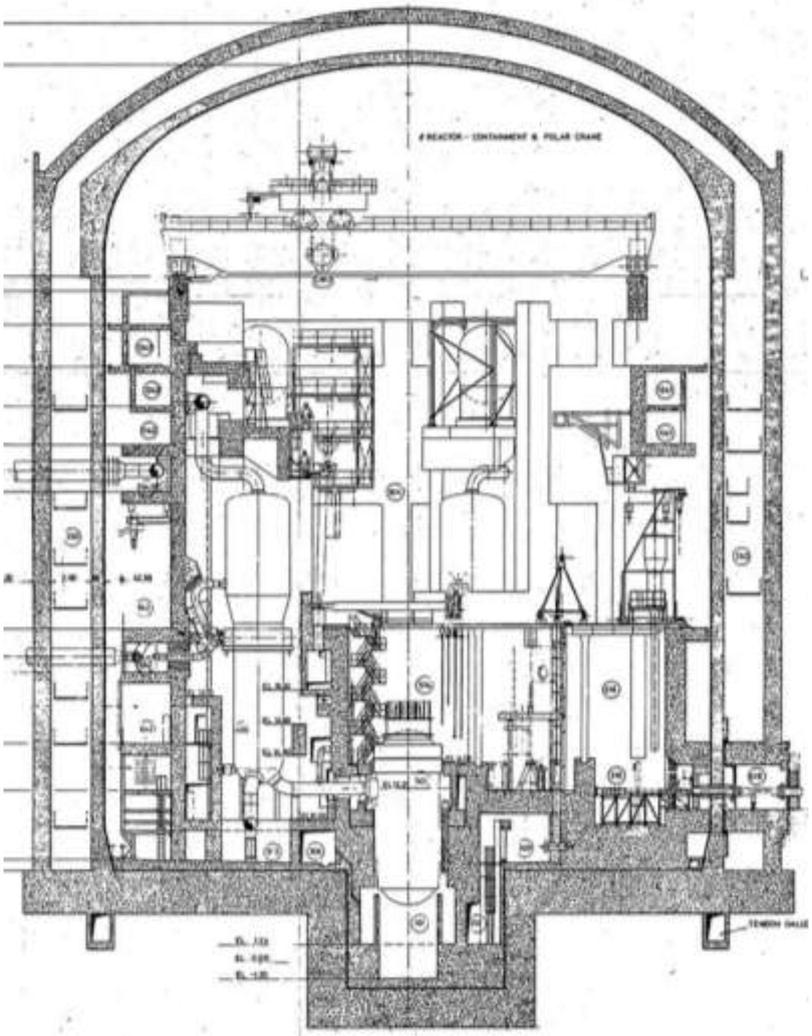


Figure A.1-12 Belgian 3-loop 1000 MWe PWR containment

In case of an accident leading to containment pressure increase, the containment isolation and annulus isolation is triggered and the ventilation of the annulus is automatically aligned towards the stack in order to maintain a sub-atmospheric pressure in the annulus and to filter potential leakages from the inner containment (using absolute filters and carbon filters) before release into the environment.

There is currently no FCV system for SA conditions, but such a system will be installed at each unit in the period 2015-2017 (except at Doel 1-2 which will be permanently shut down after 2015).

Hydrogen mitigation is achieved with the use of a hydrogen detection system and a thermal recombiner for DBA conditions, whereas AREVA PARs installed inside the containment are used for SA conditions. No hydrogen mitigation measures are foreseen in the annulus.

The spent nuclear fuel of Doel 1-2 is stored in shared pools housed in the nuclear auxiliaries building. The spent nuclear fuel of Doel 3 and Doel 4 is stored in pools on each unit. These pools, as well as the cooling systems, are housed in the bunkered nuclear fuel building.

After a sufficient cooling period in the pools, the spent fuel from Doel 1-2, Doel 3 and Doel 4 is transferred to the spent fuel container building (SCG). This reinforced concrete building functions as a dry storage where the spent fuel is placed in special shielded containers and under inert gas. The residual heat is evacuated via natural convection. The containers are designed to resist external accidents (aircraft crash, fire and earthquake).

Each of the three reactors at the Tihange site has a cooling pool designed for the temporary storage of spent nuclear fuel assemblies. This pool is located in the nuclear auxiliaries building of Tihange 1 and in the bunkered nuclear fuel building of the other units. After at least two years in the cooling pools of the units, the fuel assemblies are transferred to the DE building which houses 8 storage pools, each with a capacity of 465 spent assemblies. The power of this building is supplied by the Tihange 3 unit.

A.1.11 Westinghouse 3-loop PWR in Sweden

There are 3 Westinghouse 3-loop reactors operating in Sweden, see Figure A.1-13. These reactors are located at the west coast at the Värö peninsula and on the site there is also one ABB type BWR reactor. The reactors were taken into commercial operation 1975 (R2), 1981 (R3) and 1983 (R4). Ringhals 3 and 4 is of the same design. The initial power output was 870 MWe for R2 and 915 MWe for R3/R4. Ringhals 3 and 4 have since then undertaken power uprates and the present power is 1050 MWe for Ringhals 3 and 935 MWe for Ringhals 4. Ringhals 4 has applied for power upgrade license during January 2014 which will be slightly higher than for R3. All units have changed steam generators: Ringhals 2 during 1980’s, Ringhals 3 during 1990’s and finally Ringhals 4 during 2010.

The Ringhals PWR containments are all of a large dry type with a volume of approximately 52000 m^3 for R2 and 59000 m^3 for R3/R4. The containments are made by the pre-stressed concrete with steel liner embedded into the concrete. The containment design pressure is 5.14 bar (a)

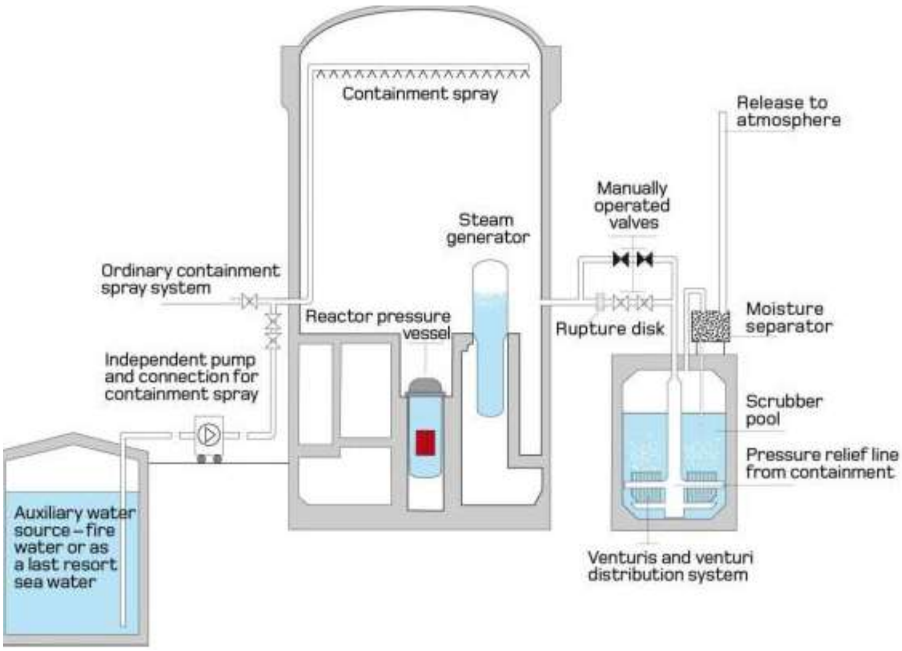


Figure A.1-13 Schematic of the Swedish three-loop Westinghouse PWR large dry containment and SA mitigation measures

During 1980’s and following the TMI accident, mitigation systems were installed. These systems consist of a filtered containment venting and an additional containment spray system. Early 2003 all PWR units implemented the Westinghouse SAMG package and some years later all units also installed AREVA type PAR’s for hydrogen management. Much of the analyses work for the installation of PAR’s was taken from the same work performed by EDF at their CPY/900 MWe reactors. The containment design and volume was considered to be so similar that the EDF analyse was valid also for the Ringhals containments. In total 25 PAR's has been installed in R2 and 29 units have been installed in R3/R4.

The spent fuel is stored outside the containment in the fuel building, and the fuel is transported to an interim storage after sufficient cooling at the site.

A.1.12 APR1400 in Korea

The APR1400 (Advanced Power Reactor 1400) [A.9] with 1400 MWe power has two primary coolant loops and each loop has one steam generator and two reactor coolant pumps in one hot leg and two cold legs arrangement. Two units are under construction currently in Korea (South).

The APR1400 containment is basically a cylindrical shape and has a free volume of about 90000 m^3, a height from the reactor cavity floor to the apex of the containment dome of 79.4 m, and an inner radius of the containment outer wall of 22.86 m (75 ft). The containment is made by the pre-stressed concrete with an inner steel liner to prevent a leakage of a radioactive material toward the environment. The containment design pressure is 5.1 bar (a) (60 psig). Although the primary shield wall and the refuelling pool look like a cube, the secondary shield wall is also cylindrical (Figure A.1-14).

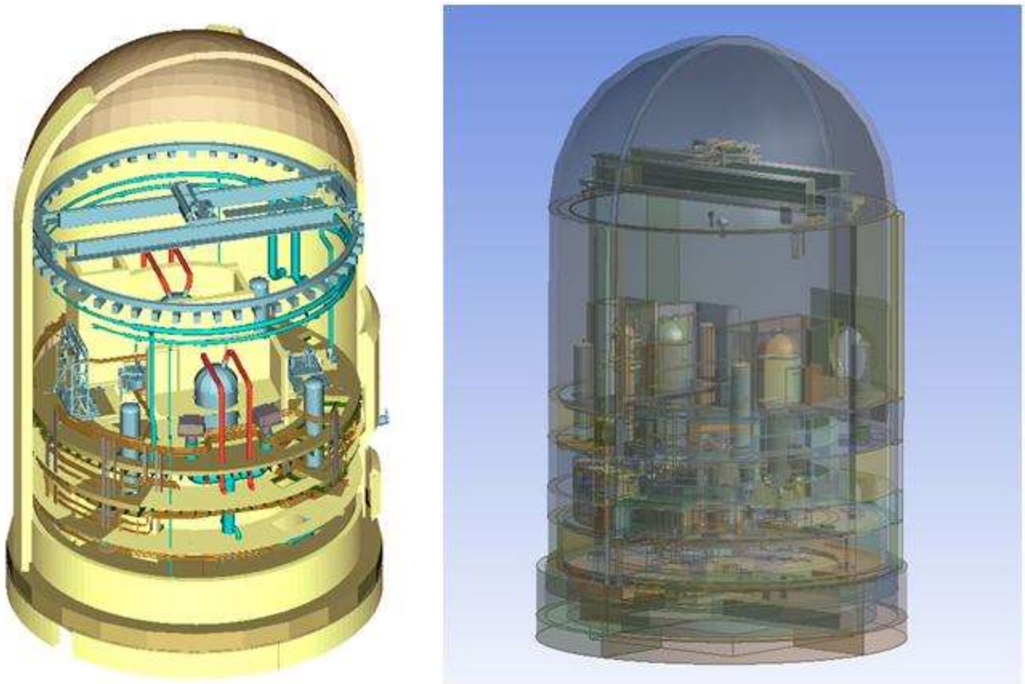


Figure A.1-14 Cross-section view of the APR1400 containment

For a loss of coolant accident (LOCA) of APR1400, the hydrogen generated in the reactor pressure vessel is released through a cold-leg or hot-leg break same as for other PWRs. But in the case of a high pressure accident such as a station black-out (SBO) in APR1400, its RCS is depressurized through an in-containment refuelling water storage tank (IRWST). The IRWST, which is an annular tank to store the refuelling water, located in the lower part of the containment is designed to be used as a water source for a cavity flooding and the containment spray system and also as a discharge location for the primary system's safety bleed operation. In order to prevent hydrogen from being accumulated in the IRWST, the release location is changed from the IRWST to steam generator compartment by a three-way valve when the core exit temperature exceeds 650^{o}C (1200 F).

The containment spray (CS) pumps of APR1400 are activated by a containment spray actuation signal (CSAS). The CS pumps take a suction of water from the IRWST and they discharge the water through the spray nozzles after reduction of the temperature at heat exchangers. The CSAS can be generated automatically by a pressure signal in the containment or manually by an operator in the control room. The main spray nozzles are attached to four spray rings on the dome wall. And the auxiliary spray nozzles are attached to the 2 spray rings around the annular compartment below the operating deck.

A Hydrogen Mitigation System (HMS) in APR1400 is designed to accommodate the hydrogen production from 100 % fuel clad metal-water reaction and limit the average hydrogen concentration in containment to 10 vol.% for degraded core accidents. The HMS consists of a system of PARs complemented by glow plug igniters installed within the containment. The current APR1400 has 30 PARs and 10 glow plug type igniters installed within the containment for the mitigation of the hazards of hydrogen generated and potentially released into the containment during a SA.

A.1.13 OPR1000 in Korea

The OPR1000 (Optimized Power Reactor 1000) [A.10] with 1000 MWe power has two primary coolant loops and each loop has one steam generator and two reactor coolant pumps in one hot leg and two cold legs arrangement. There are 12 units of OPR1000 currently operated in Korea (South). The geometrical configuration of the containment is a cylindrical shape with an upper hemispherical dome (see Figure A.1-15). The spent fuel pool is located in the fuel building besides the reactor containment building. The containment is made by the pre-stressed concrete with an inner steel liner to prevent a leakage of a radioactive material toward the environment. The containment has a free volume of 77220 m^3, a height from the reactor cavity floor to the apex of the dome of 75.28 m, and an inner radius of 21.94 m with a wall thickness of 1.06 m. The containment design pressure is in the range of 4.8 bar (a).

As a heat removal system of the containment, a spray and a fan cooler system are installed to maintain the pressure and temperature of atmosphere in the containment below the design limit for design basis accidents and SAs. The containment spray (CS) pump is operated automatically by a high pressure signal of the containment or a low pressure signal of the pressurizer. For a recirculation mode of the CS, the CS pump takes a suction of the water from the containment sump and transfers the water to the spray nozzles after reducing the temperature by the heat exchanger. The main spray nozzles are attached to four spray rings on the dome wall. The fan cooler system is installed at a middle elevation along the annulus wall in the containment. An atmosphere temperature of 48.9^{o}C is maintained by the fan cooler during the normal operation. As a Hydrogen Mitigation System (HMS), 24 PARs and 20 glow plug type igniters are installed to remove the hydrogen and to prevent the possibility of hydrogen combustion for SAs. The PARs installation was back-fitted to the OPR1000 after the Fukushima accidents in Japan. To prevent a leakage of fission products through the containment during SAs, an isolation system is installed and can be activated by closing valves located around the containment penetration parts.

As for an accident scenario related to the HMS, 100 % fuel clad metal-water reaction is assumed for SAs with a degraded reactor core. The hydrogen generated in the reactor pressure vessel is released into a steam generator compartment of the containment through a cold-leg or hot-leg break for a loss of coolant accident (LOCA). In the high pressure accident such as a station black-out (SBO) and a total loss of feed water (TLOFW), the hydrogen is released through the rupture disk of the reactor drain tank (RDT) installed in the annular compartment below the operating deck by an event of RCS depressurization. Finally, the HMS in the OPR1000 is designed to limit the average hydrogen concentration in containment to 10 vol.% for SAs.

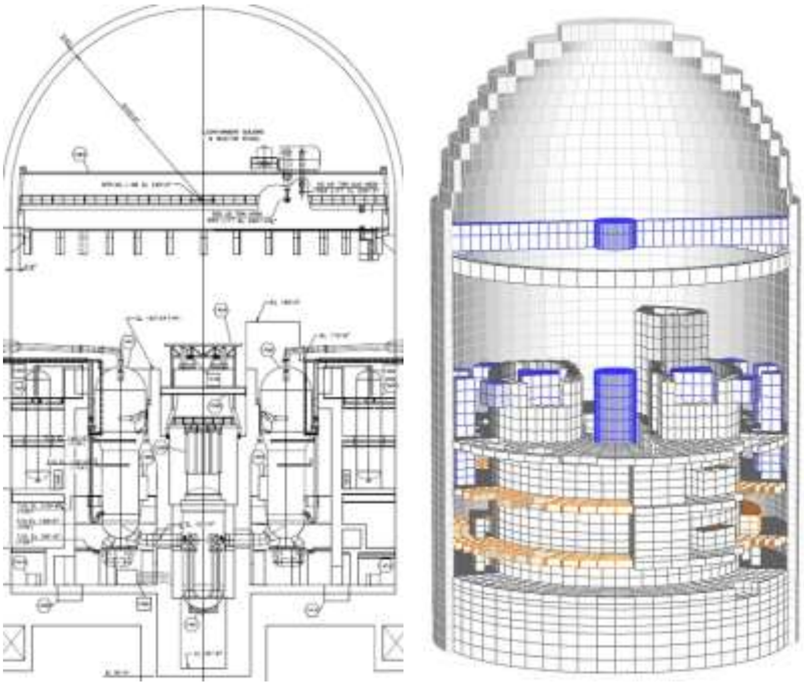


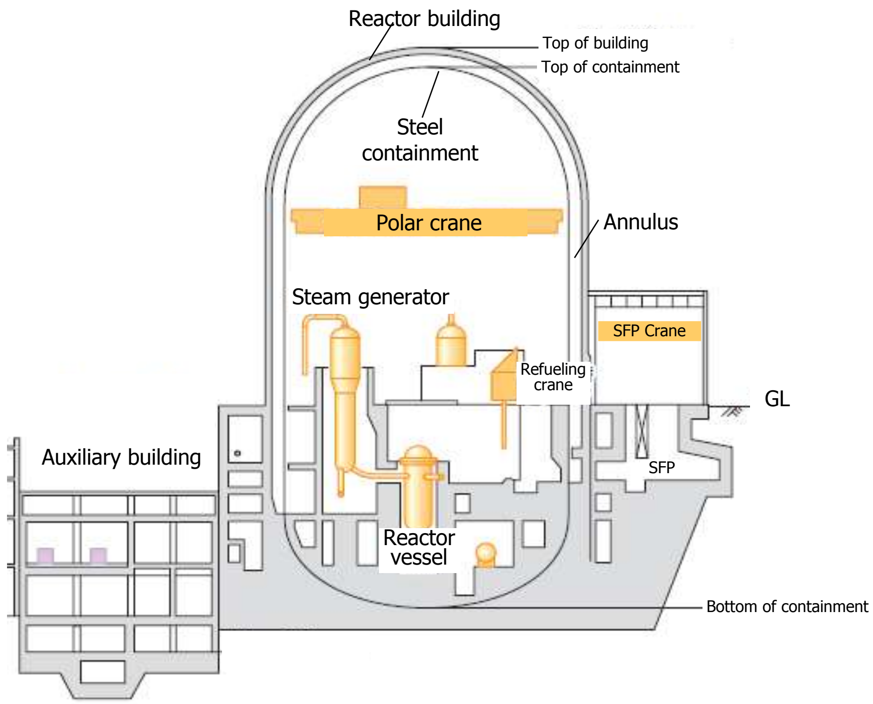
Figure A.1-15 Cross-section view of the OPR1000 containment (left) and its 3D model used for hydrogen analysis (right)

A.1.14 Mitsubishi PWR of Japan

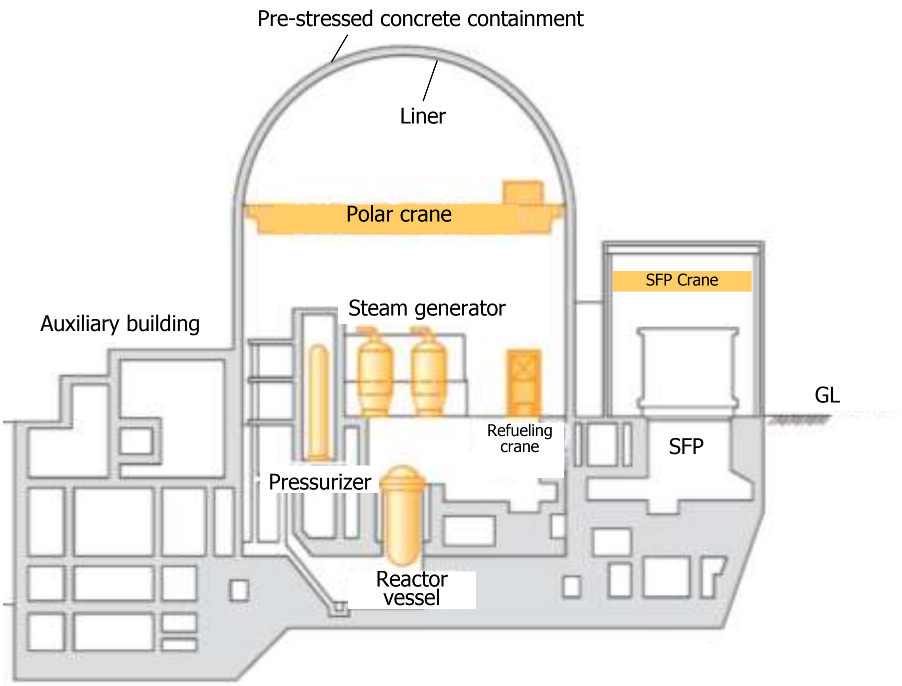
Mitsubishi PWR units range from two loop 1650 MWt units to four loop 3410 MWt units [A.11]. There are two types of the containment system in Mitsubishi PWR units in which one is the steel containment vessel (SCV) within a concrete reactor building and another is the pre-stressed concrete containment vessel (PCCV) with inner steel liner as shown in Figure A.1-16.

Three different NPP generations exist:

- Thermal power and electric power of the standard two loop units are 1650 MWt and 570 MWe. Two-loop units are applied the SCV containment with an inner steel vessel diameter of 35 m and a height of 65 m. The design pressure is about 3.6 bar (a). The containment net volume is about 42400 m³ surrounding with the annulus volume of about 3600 m³.
- Thermal power and electric power of the standard three loop units are 2650 MWt and 900 MWe. Three-loop units are also applied the SCV containment with an inner steel vessel diameter of 40 m and a height of 77 m. The design pressure is about 3.6 bar (a). The net volume is about 67900 m³ surrounding with the annulus volume of about 10000 m³.
- Thermal power and electric power of the standard four loop units are 3410 MWt and 1100 MWe. Four-loop units are applied the PCCV containment with the inner steel liner diameter of 43 m and a height of 65 m. The containment design pressure is about 5.0 bar (a). The net volume is about 73700 m³ surrounding with the annulus volume of about 15000 m³.



a) Steel Containment Vessel



b) Pre-stressed Concrete Containment Vessel

Figure A.1-16 Containment type of Mitsubishi PWR NPP

Before the Fukushima Dai-ichi accidents, there were no national requirements on installation of equipment and procedures to deal with SAs. Under the enforcement of a new regulatory guide, mitigation measures against hydrogen explosion inside the containment vessel are prescribed as the basic requirement, including installation of hydrogen concentration control equipment (i.e., PARs and/or igniters) and monitoring equipment that measures fluctuation of hydrogen concentration. It is also required to prevent hydrogen explosion at the discharge paths connected to the outside of the containment vessel as well as inside the containment vessel annulus, where installation of hydrogen control equipment such as PARs or hydrogen discharge equipment is required.

Both type of the containment system contains the ECCS system, the containment spray system and the annulus venting system. The ECCS system consists of the accumulators and the high pressure and low pressure injection systems which are motor driven powered by the emergency diesel generators in case of a station black-out. Water source of the high and low pressure injection systems and the containment spray system are the re-fuelling water tank. When a water level of the re-fuelling water tank became low it is switched to the containment sump pit operated with a re-circulation mode. The spent fuel pool is located outside of the containment below the ground level.

When a DBA such as the LOCA occurred, the ECCS injection starts and the containment spray starts to suppress the increase of the containment pressure due to discharged steam and the containment isolation and the annulus ventilation system is triggered to prevent to release radioactive materials to the environment.

A hydrogen mitigation measure was not installed in SCV nor PCCV type PWR units due to the enough containment volume for DBAs. But, after the Fukushima accident, PAR has been introduced to some PWR units as a hydrogen mitigation measure in SAs.

A.2 VVERs

A.2.1 VVER-440/W-213 – standard design

Reactors of the VVER-440 type are Russian design. Two main design types were constructed – older VVER-440/230 and newer VVER-440/213. The principal difference is in the level of safety assumptions – older type was practically without important emergency systems, and without the accident localisation system. This type of reactor is today only in operation in Russia (4 units) and in Armenia (1 unit).

The model V-213 has both the standard ECC system and additional accident localization features. NPPs of newer VVER-440/W-213 design are operated today in Russia, Slovakia, Czech Rep., Ukraine and Hungary.

All VVER-440 plants have a six loop reactor circuit, isolation valves on each loop and horizontal steam generators. Typically two 220 MWe steam turbines are used. Some of the NPP operators, like Czech Rep. Dukovany NPP, up-rated their units electrical power to 500 MWe.

The VVER-440/213 NPP has some specific features that positively affect the overall characteristics of the plant. It has relatively low operating parameters and medium reactor output, a primary coolant system layout with a large number of cooling loops, and the use of horizontal SGs and of two medium sized turbines instead of one, which have a positive impact on the safety of the plant. All plants are designed as twin units.

The VVER-440/V-213 NPP is equipped with a so-called bubble condenser containment (accident localization system with hermetic rooms, bubble condenser tower and air traps) that is intended to prevent the release of steam and fission products, and to facilitate the steam condensation, thereby reducing the pressure in case of a break of any single pipe of the primary circuit, including a double-ended break of the main circulation line with DN500 diameter. The containment system is composed of:

- Reinforced concrete accident localization structure (hermetic rooms), providing the confinement function of the system (SGs, inlet and outlet piping, pumps, isolation valves and the major portion of the reactor vessel);

- Pressure suppression system (bubble condenser tower, BC and air traps), providing the passive pressure suppression function;
- Water droplet spray system, providing the active pressure suppression function and the radioactivity removal function (called in this report the reactor building spray system).

The BC containment of a VVER-440/V-213 NPP consists of more than 40 hermetic compartments and rooms including the bubble condenser (hermetic rooms are highlighted in yellow colour in Figure A.2-1).

The containment free volume is about 55000 m^3. Its structures are composed of heavy reinforced concrete that serves as biological shield. A hermetic steel layer covers outer peripheral structures to reduce potential leakage. The thickness of the outer walls is 1.5 m. The thickness of the basement, under the reactor pressure vessel, is about 3.4 m. Thickness of the basement concrete in other parts of the containment is about 2.7 m. The hermetic carbon steel layer is 6 mm thick.

The pressure suppression system (bubble condenser) comprises gap-cap systems placed on 12 levels (floors). Each level contains water trays. These trays are filled with borated water at a concentration of 12 g H_3BO_3/kg H_2O. The designed water inventory inside the bubble condenser is about 1300 m^3. For each unit, the set of pressure suppression trays is located inside a separate building (the bubble condenser tower) constructed adjacent to the reactor building. The tower is connected to the steam generator compartment by a rectangular tunnel. The condenser tower also houses four large receiver volumes serving as air traps.

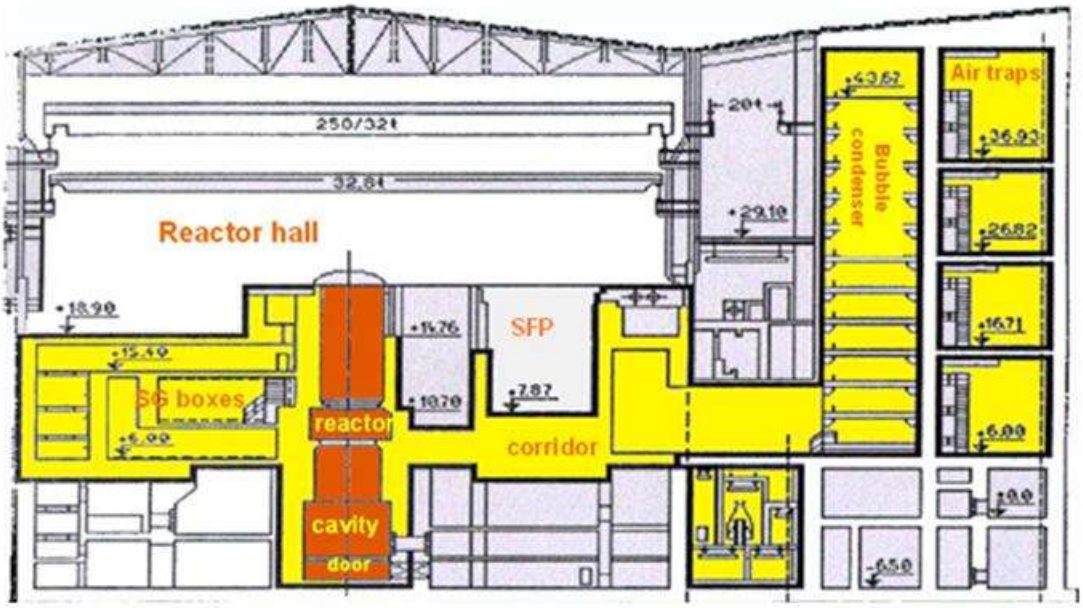


Figure A.2-1 VVER-440/V-213 standard design

A spray system is provided in the steam generator and pump compartments to condense steam during emergency conditions. The design pressure of the accident localization system is about 2.5 bar (a). Displacement of air from the localization compartments, followed by condensation of steam due to passive heat removal to cold structures and also due to operation of the spray system, permits the pressure to be reduced to sub-atmospheric values after 10 - 15 min (valid for a DN500 pipe rupture).

The reactor building (containment) spray system (RBSS) provides spray water to the reactor compartment following a LOCA or steam line break, to limit containment pressure and to minimize the release of radioactive iodine and particulates to the environment. The RBSS is composed of three identical and completely independent trains, each of them with a capacity of ~600 m^3/h.

In all countries operating VVER-440/213, PARs are already implemented or the implementation is foreseen. Another modernization measure related to SA mitigation is the flooding of the cavity to cool the RPV from outside. This measure was implemented first in Finland (see next chapter).

A.2.2 VVER-440 with ice-condenser containment (Loviisa 1 & 2)

Loviisa 1 & 2 are VVER-440 reactors of Soviet design [A.7]. They differ significantly from other VVER-440 standard units because they are equipped with US ice-condenser containments designed by Westinghouse. They are operated by Fortum in Finland. They started commercial operation in 1977 and 1981. They have electric power of 496 MW and thermal power of 1500 MW. They have six coolant loops with horizontal steam generators.

The containment is shown in Figure A.2-2. The containment gas volume is 58000 m^3. The containment design pressure is low, 1.7 bar (a), and the ultimate pressure capacity is above 3 bar (a). The containment walls are made of steel. Therefore heat can be removed from the containment with an external spray system. There is a spray system inside the containment, too. There is a SFP inside the containment. The reactor building, with walls made of concrete, acts as a secondary containment.

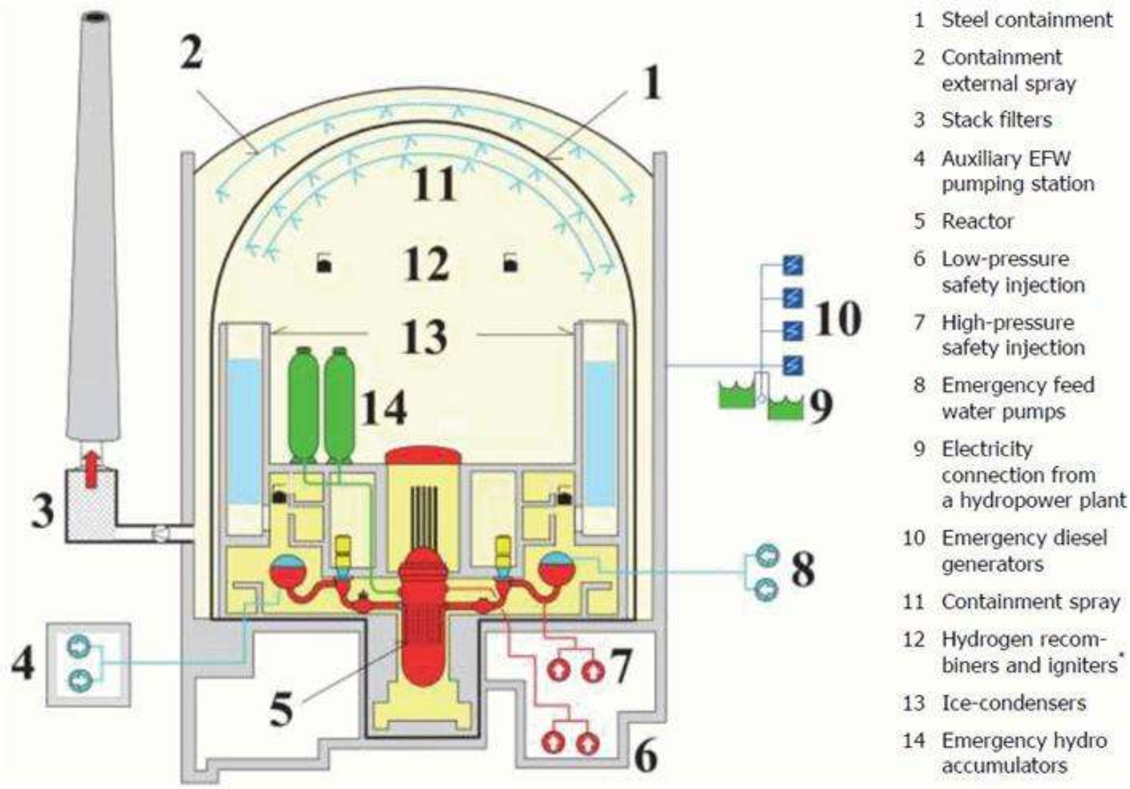


Figure A.2-2 VVER-440 with ice-condenser containment (Loviisa 1 & 2)

There are two ice-condensers in the containment. If the core exit temperature exceeds 450°C in an accident, the operators would open dedicated reactor depressurization valves and force open the ice-condenser doors. The ice would melt by the steam that is released from a LOCA or from the reactor depressurization. Condensation in the ice-condensers would limit the pressure increase in the containment. The water from the melting ice would fill the reactor cavity and submerge the RPV lower head.

The SA management relies on in-vessel melt retention by external cooling of the lower head. This is possible because: 1) melting of the ice-condensers generates sufficient water to submerge the lower head; 2) the flow paths around the RPV are sufficiently large; 3) the reactor power is so small that the heat flux through the lower head does not exceed the critical heat flux; and 4) there are no penetrations in the RPV lower head.

The hydrogen management is based on efficient mixing by forced opening of the ice-condenser doors by pressurized nitrogen. There are 154 PARs (84 in the steam generator room, 66 in the dome, and 4 in some dead-end spaces) made by AECL. There are also glow plug igniters installed.

A.2.3 VVER-1000/V-320 standard

There are more than 20 NPPs with VVER-100/V-320 reactor under operation. Most plants are operated in Ukraine and Russia. Two NPPs each are located in Czech Republic and Bulgaria.

During the design phases of the Russian VVER-1000 plants, great scientific efforts were made to meet the optimal core configuration based on the small series of V-187, V-302 and V-338 models, and large series of V-320 reactors [A.12]. These reactor series show design differences in the number of the control and protection system cluster rods (drives and bundles). The containment design was similar (Figure A.2-3).

After building the small series (V-187, V-302, and V-338) mainly the type V-320 was built for commercial operation. The reactor type VVER-1000/V-320 is a light water moderated and light water-cooled PWR with an electrical capacity of 1000 MW or a thermal power of 3000 MWt. The primary system consists of four main coolant loops with one horizontal steam generator and one main coolant pump each.

The containment encloses a hermetic system of 63 compartments in which the main components of the primary and secondary systems, equipment and pipelines are located. It is designed in a cylindrical shape with an internal diameter of 45 m, covered with a hemispheric dome, with a total free volume of ~61000 m^3. The lower elevation of the containment is +13.2 m and the upper elevation is +65.35 m (the highest point of the containment dome). The SFP is located within the containment as well. The emergency boron solution tank is located under the containment foundation slab at elevation +6.60 m and is also part of the containment hermetic boundaries. The containment has a pre-stressed reinforced concrete construction with an 8 mm internal steel liner. The containment design pressure is 5.1 bar (a), covering the pressure peak after a double-ended rupture of the main circulation pipeline with DN850 diameter. A negative pressure of a maximum of 200 Pa is maintained in the containment during normal operation.

The typical feature of the containment is that the containment shell is a part of reactor building with square ground plan and side length of 66 m. It is separated from the non-leak-tight lower parts of the building by a 3 m reinforced concrete plate, but connected to one rectangular leak-tight room in the lower part of the building housing the main ECC recirculation sump.

The ECC pumps and different supporting equipment are located in the lower part of the reactor building. The square building extends also above the containment shell base plate, up to about +40 m, protecting large part of the containment shell with the reactor against external impact and improving the primary system shielding. There is a narrow gap between the cylindrical shell and the cylindrical inner shaft in the square building.

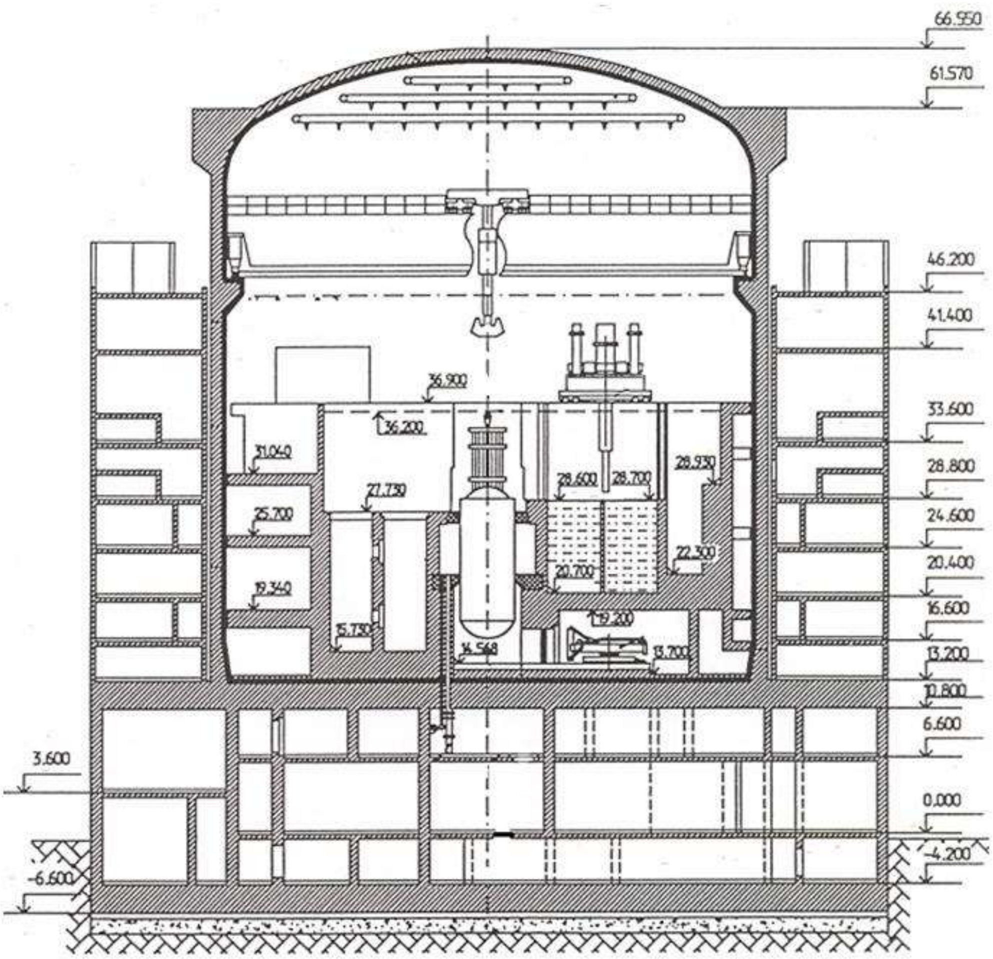


Figure A.2-3 Typical VVER1000/V320 NPP [2.3]

The containment spray system (CSS) is intended to control the containment pressure in case of a break of primary circuit pipelines or secondary circuit steam lines. The spray system consists of three independent trains, which are designed to withstand external impacts. The three trains are physically separated and emergency power supplied. They use the common boron solution tank (sump) of the emergency cooling system as a water source. The tank is connected to the three trains of the spray system by three pipes (DN600). During the accident it functions to:

- Depressurize containment
- Wash out air-borne fission products
- Discharge a part of the residual heat
- Ensure emergency filling of the spent-fuel pool via a connecting line to the pool cooling system.

As part of the accident management concepts different measures have been implemented in the countries using VVER-1000 reactors. An important engineering solution is the plugging of all instrumentation channels for the ionization chamber, located circumferentially inside the reactor cavity walls with removable steel cases to prevent early containment melt through. Installation of PARs is under discussion as well. In some NPPs with VVER-1000/V-320 reactor types a hydrogen reduction system with PARs was already installed for design basis events. For example, a hydrogen reduction system consisting of 8 PARs (AREVA type) was installed in the containments of units 5 and 6 of Kozloduy NPP (Bulgaria) to cope with the hydrogen generated during design basis accidents.

Another engineering solution is implementation of a FCV system to avoid containment overpressure and to reduce significantly the releases of radioactivity to the environment in case of a

SA. Containment filtered venting system was already installed in Kozloduy NPP (Bulgaria), units 5&6 and is presently under discussion in other countries, too.

As part of the accident management concept at the Temelin NPP (Czech Republic), a large number of PARs is expected to be installed in the containment to prevent global hydrogen combustions challenging the containment integrity. Project for installation of hydrogen removal system based on PARs was already initiated for the Temelin NPP and installation has to be finished before end of 2015 at both units. Selected SA scenarios including core degradation and MCCI in the dry cavity are used for the analysis of the PAR system design. Selection of scenarios took into account several factors – contribution of scenario to CDF in PSA-1, type of initiating event, location of possible hydrogen sources into the containment, operation of spray system. As an alternative of scenario development, melt spreading (after door melt-through) into neighbouring room to cavity was assumed as the room is connected with the cavity with a corridor closed by doors during normal operation. Accidents in the SFP are not yet considered, but the analyses of selected scenarios initiated in SFP are on-going and will be finished at the end of 2013 for Temelin NPP. For long-term over-pressure protection a FCV system is under consideration.

A.2.4 Next generation VVER-1000 and VVER-1200

Based on NPPs with VVER-100/V-320 design two new series have been developed in the late 1990s [A.13], [A.14], [A.15], [A.16]:

- AES-91 (VVER-1000/V-413) by Atom Energo Project (AEP) St. Petersburg with support by TVO and Fortum, Finland,
- AES-92 (VVER-1000/V-392) by AEP Moskau together with OKB Gidropress and Kurchatov-Institut Moscow.

The concept AES-91 is realised as VVER-1000/V-428 at NPP Tianwan, PR China (Figure A.2-4). The concept AES-92 is realised as VVER-1000/V-412 at NPP Kudankulam, India (Figure A.2-5).

Since 2005 AEP Moscow and AEP St. Petersburg continued and extended the design of VVER-1000 to VVER-1200 projects named:

- AES-2006 8VVER-1200/V-392M) – AEP Moscow
- AES-2006 (VVER-1200/V-491) – AEP St. Petersburg

These are different projects build at different locations. The AES-2006 is a VVER type reactor plant of 1200 MW power. Its design is based on experience feedback from several thousand reactor/years of VVER NPP operation; it incorporates the most recent VVER-1000 technologies, applied to Unit 3 of Kalinin NPP, Unit 2 of Balakovo NPP, Unit 1 of Volgodonsk NPP in Russia, “Tianwan” NPP in China, “Kudankulam” NPP in India [A.16]. Common is the use of passive safety features for core cooling and heat removal from the containment, core catcher concepts and a concrete containment with inner steel liner surrounded by a concrete reactor building providing protection against external events. Containment spray systems exist as in VVER-1000/V-320 units and a PAR system will be installed to prevent hydrogen combustion in SAs. More details are provided in [A.14], [A.15], [A.16].

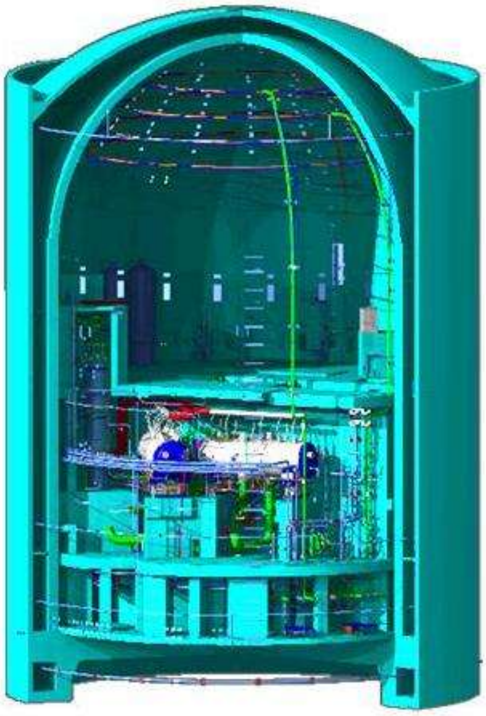


Figure A.2-4 VVER-1000/W-428 (AES-91), Tianwan, China [A.13]

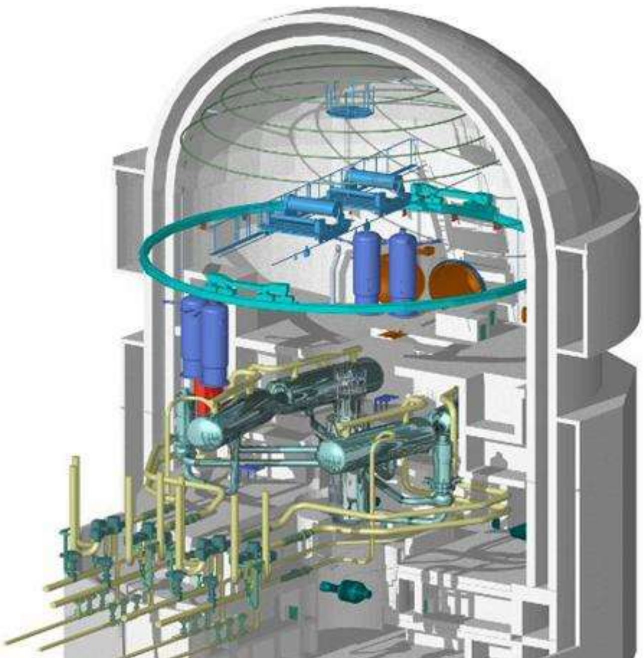


Figure A.2-5 VVER-1000/W-466 (AES-92), Kundankulam, India [A.14]

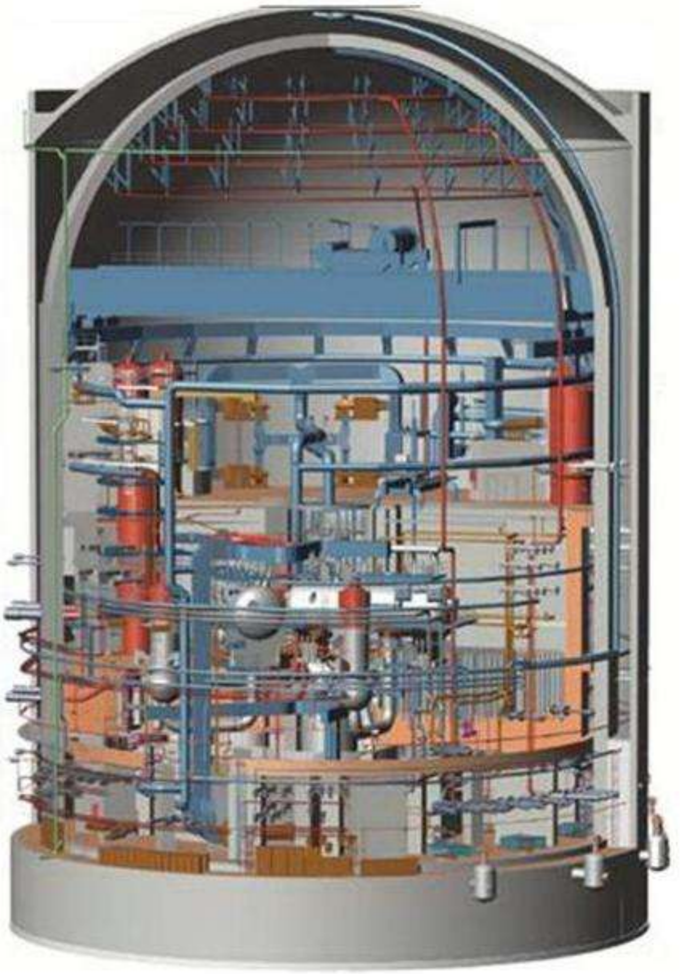


Figure A.2-6 VVER-1200 (AES-2006) [A.15]

A.3 BWR

A.3.1 German BWR of Type 72

There are two units of BWR type 72 in operation in Germany since July 1984 and January 1985. Their operation is scheduled until end of 2017 and end of 2021 respectively [A.12]. The power of each unit is 1344 MWe.

The containment concept of BWR type 72 consists (Figure A.3-1) of the internally located separate containment vessel (containment made of pre-stressed concrete with inner steel liner) and the surrounding reactor building. Both buildings are based on a common foundation plate with a diameter of 52 m and thickness of 3 m.

The containment vessel consists of pre-stressed concrete cylinder with an outer diameter of 29 m and a height of about 40 m. The inner surface is covered with a gas proof steel shell of approximately 8 mm thickness. The containment design pressure is about 4.3 bar (a). Inside the containment, there are the reactor pressure vessel and the pressure suppression system, which consists of the drywell and wetwell (suppression pool). The total volume is about 17800 m^3 while the wetwell has a water pool with approximate 3000 m^3 deionised water, to condense the escaping steam during the loss-of-coolant accident considered in the design (double-ended rupture of the main coolant line, the so called 2A break), thus limiting the pressure within the containment and the load to the building. During events which lead to increased activity release into the containment, a direct sealing is ensured because pipes penetrating the containment are equipped at least with two isolation valves, where one of these is arranged inside and the other outside the containment, unless it is not conflicting with safety related reasons (e.g., reactor scram). Thus the containment serves as an activity barrier for safe enclosure of radioactive material, which is also efficient during events with leakages from the reactor coolant pressure boundary.

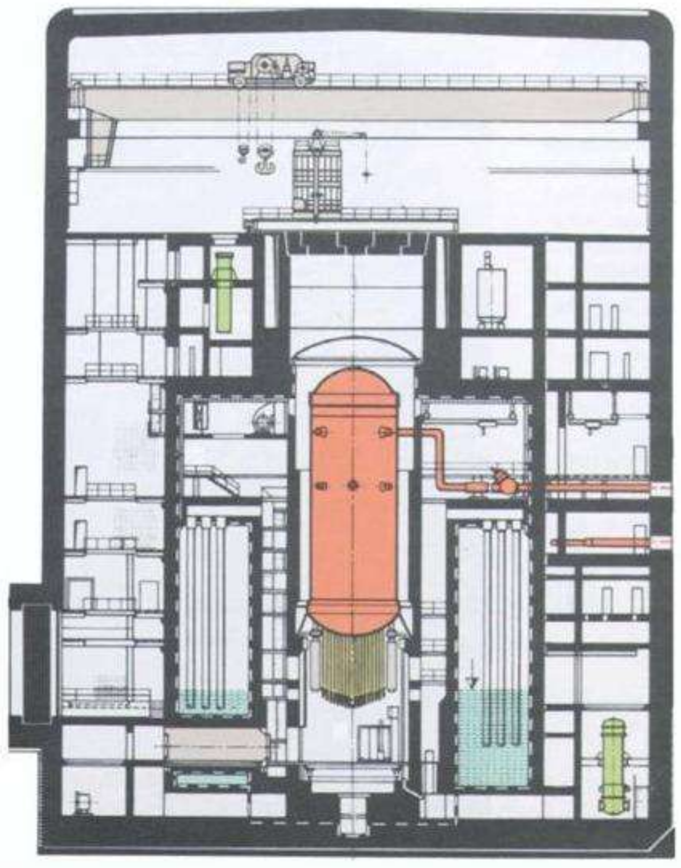


Figure A.3-1 German BWR type 72 NPP

The pressure suppression system has the task to condense the escaping steam in case of loss-of-coolant accidents, thus suppressing the pressure. Furthermore, it is considered as a passive part of the emergency cooling. The pressure suppression system consists of the wetwell, the condensate pipes from the drywell into the wetwell, and the check valves between the wetwell and the drywell. The water pool in the wetwell serves as the water supply for feeding the reactor pressure vessel for the emergency cooling and residual heat removal systems and as a substitute heat sink in case of any loss-of-coolant accidents where the main heat sink is not available.

The reactor building surrounding the containment consists of ferroconcrete with an outside diameter of 50 m and a thickness of 1.8 m and encloses the containment. It serves first of all as an additional shielding of the surrounding area against ionising radiation, furthermore it protects against external events caused by natural events, e.g., earthquakes and flood, as well as aircraft impact, fire, explosion blast wave and acts of sabotage. Additionally, the reactor building serves for retention of potential leakages from the containment so that these are controlled via the sub-atmospheric pressure holding system and released through suspended solids filter and activated carbon filter to the vent stack. In case of an accident with pressure or temperature increase in the containment, the containment isolation is triggered and the emergency sub-atmospheric pressure system is started. This system has the task to retain the sub-atmospheric pressure in the reactor building and to filter potential leaking from the containment vessel before discharge.

The SFP is located in the upper part of the reactor building above the containment. The containment head has to be removed for fuel loading.

The reactor protection system is based on the principle that no operator action is required in the first 30 min after design basis accidents. For external events, the time frame for which no operator action is required is at least 10 hours.

As part of the accident management the wetwell was inerted by nitrogen during normal plant operation. In addition a large number of PARs have been installed in the containment – both the drywell and the wetwell - to prevent global hydrogen combustions challenging the containment integrity. About 90 NIS PARs have been put into the containment. Selected SA scenarios including core degradation and MCCI in the cavity (control rod driving room) have been used for the analysis of the PAR system design. For long-term over-pressure protection a FCV system was installed connected to the wetwell gas part.

Accidents in the SFP have not been considered so far, but are under discussion after the Fukushima accidents. Primarily the installation of additional PARs in the SFP area is discussed (see German national action plan of 2012 [A.2]). As well SAMG are to be implemented after the Fukushima.

A.3.2 BWR Mark I Containment (Spain)

A General Electric BWR-3 Mark I (Santa María de Garoña) has been operating in Spain since March 1971 with at thermal output of 1.381 MWt. Figure A.3-2 in the next Section shows a diagram of the containment and the reactor building of the Japanese Mark I plants. This picture is a generic representation of a Mark I containment which is acceptable for the Spanish one.

The containment in the Santa María de Garoña NPP is a Mark I type, equipped with a double containment pressure suppression capacity. It consists of a drywell and a wetwell, both inerted with nitrogen. The drywell component is a steel “light-bulb shaped” vessel with a spherical lower portion and an upper cylindrical portion. A bolted head closes the top of the cylinder. Reinforced concrete walls enclose this vessel, providing additional shielding and resistance. The pressure suppression chamber, or the wetwell, is a toroidal carbon steel vessel that surrounds the lower portion of the drywell. Eight circular vent pipes interconnect the wetwell and the drywell [A.14].

The total containment volume is around 7056 m^3, evenly distributed between drywell (46.7 %) and wetwell (53.3 %). The wetwell contains an aqueous pond (suppression pool), which volume is 1856 m^3. This results into a containment free volume to power ratio of 4.5 m^3/MWt. The design pressure and temperature of the containment are 5.3 bar (a) and 138ºC.

The SFP is located outside the containment building within the reactor building on the top of the containment. It has a specific cooling and cleaning system.

The hydrogen mitigation measures are the inerting with nitrogen of the drywell and wetwell and a dedicated venting. Additionally, the containment is equipped by the Containment Spray System (CSS), the Suppression Pool Cooling System (SPCS) and the Containment Isolation System (CIS). All of them have two redundant trains and are seismic category I. The purpose of CSS and the SPCS is to reduce the pressure and temperature in the containment at DBA. The CIS prevents any fluid transported through by-pass lines from leaking outside the containment. As a result of the “stress tests”, a filtered containment system and a PAR system in the reactor building have been required to be installed in the plant [A.15]. The plants have procedures for normal conditions, emergency conditions and SAs (SAMGs).

A.3.3 BWR Mark I Containment (GE design), Japan

A General Electric Mark I containment is adapted to Japanese BWR NPPs ranging a thermal power from 1.070 MWt to 2.380 MWt [A.17]. Figure A.3-2 shows a diagram of the containment and the reactor building.

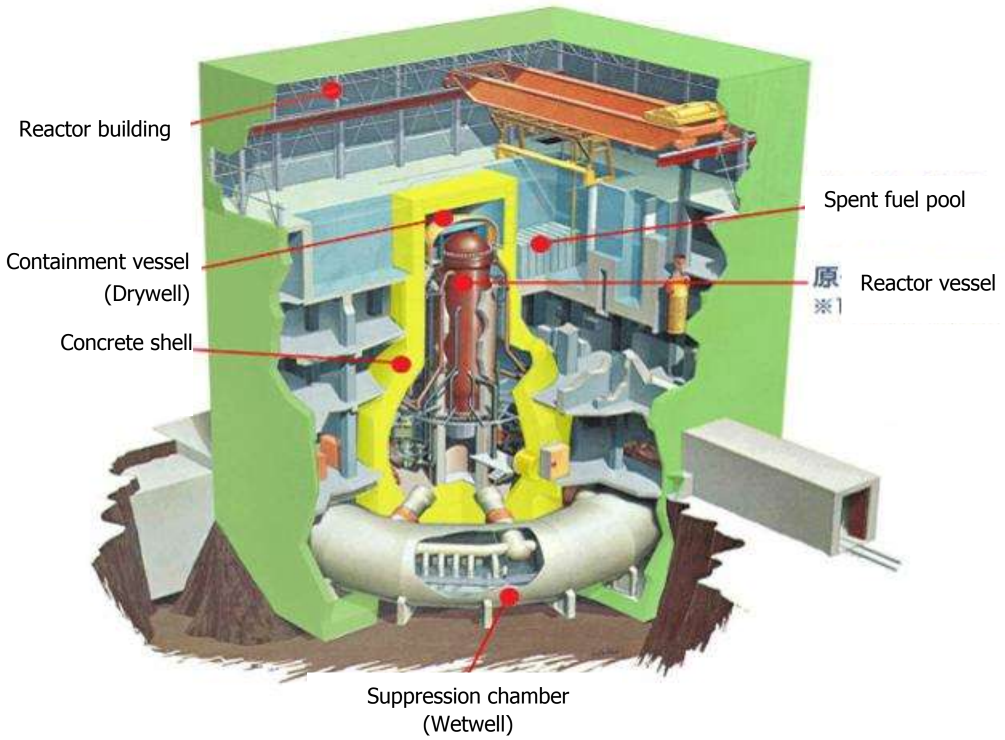


Figure A.3-2 BWR Mark I containment (Japan)

The thermal power and electric power of a General Electric BWR-3 Mark I type NPPs are 1380 MWt and 460 MWe respectively. The drywell inner diameter is about 10 m at cylindrical part, 18 m at spherical part and a height about 32 m. The wetwell is a toroidal component with a cross section of about 8 m and a diameter of the torus of about 30 m. The containment free volume is about 3.410 m^3 in drywell, 2620 m^3 in wetwell and 1750 m^3 in suppression pool. The design pressure of the containment is 5.4 bar (a).

The thermal power and electric power of a General Electric BWR-4 Mark I type NPPs are 2380 MWt and 784 MWe respectively. The drywell inner diameter is about 11 m at cylindrical part 20 m at spherical part and a height about 34 m. The wetwell is a toroidal component with a cross section of about 9 m and a diameter of the torus of about 34 m. The containment free volume is 4240 m^3 in drywell, 3160 m^3 in wetwell and 2980 m^3 in suppression pool. The design pressure of the containment is 4.9 bar (a).

The containment system contains the ECCS system, the containment spray cooling system, the flammable gas control system (FCS) and the standby gas treatment system (SGTS). The containment drywell and wetwell are inerted by nitrogen during normal operation.

The low pressure core injection and the containment spray cooling are one of an operation mode of the residual heat removal (RHR) system. After the core has been recovered, the LPCI mode is switched to the containment spray cooling mode which sprays the water from the suppression pool to the drywell and the wetwell. When a DBA such as the LOCA occurred, the ECCS injection starts. The containment is isolated. The ADS works to depressurize a core pressure with opening safety relief valves. Hydrogen and oxygen concentration in the containment is controlled below the combustible level by recombining them with the FCS system during DBAs. The SGTS system is triggered to prevent to release radioactive materials to the environment.

To prevent a long term containment over-pressure failure in case of SAs, the FCV system has been equipped as an accident management concept. The system is connected to the gas phase of the wetwell and the drywell of the containment and has a separate off-gas pipe towards the stack or the environment.

Before the Fukushima Dai-ichi accidents, there was no national requirement of preparing equipment and procedures to deal with the SA. However, as mentioned above, the containment is inerted to cope with DBA conditions. Under the enforced the new regulatory guide, measures against hydrogen explosion inside the containment vessel are prescribed as the basic requirement, such as inerting of the containment atmosphere, which is already installed, and monitoring equipment that will measure fluctuation of hydrogen concentration. It is also required to prevent hydrogen explosion at the discharge paths connected to the outside of the containment vessel. Measures against hydrogen explosion inside the reactor building are also prescribed as the basic requirement, where installation of hydrogen control equipment such as PARs or hydrogen discharge equipment is required. This guide applies to all Japanese BWRs (Section A.3.5 and A.3.9).

A.3.4 BWR Mark I Containment (Switzerland)

The Swiss Mühleberg nuclear power plant (KKM), see Figure A.3-3, is a BWR unit (General Electric) with gross electric power of 373 MWe. KKM has a double-containment and is a Mark I containment type. Other safety relevant systems are the isolation system, the vacuum breaker system, the hydrogen control system and the emergency system.

The containment is made of steel and consists of drywell, the wetwell (torus) and 6 vent pipes which link the drywell to the wetwell. The drywell volume is 3100 m³, the Wetwell volume is 4000 m³ (2100 m³ water volume and 1900 m³ free volume). The main goal of the containment is to mitigate pressure build-up, in case of a LOCA. The drywell and wetwell of KKM are inerted with nitrogen which has the main function to avoid hydrogen explosion during a postulated SA.

The cylindrical reactor building consists of thick reinforced concrete walls and has a volume of 40.000 m³. The pressure specification of the reactor building is about 340 mbar over pressure and 60 mbar under pressure. The reactor building includes as so called outer torus which has the function to condense steam which may be released in the reactor building during a postulated accident. The outer torus water volume is 1000 m³. The condensation to the external torus takes place passively when the pressure in the reactor building reaches about 60 mbar. The safety function of the reactor building is to protect the reactor from external events and also to confine (before filtering) any possible leak from the containment.

To protect the containment from failure it has various vacuum breakers (4 from the torus to the drywell; 12 from the drywell to the 6 vent lines) with different functions. Containment Depressurization System (CDS) consists of pipes from both drywell and wetwell to the water in the outer torus. In case of rising pressure in the containment, because of uncontrolled steam-production, the system will be manually actuated or automatically or passively at 7 bar through a rupture disk in order to reduce the pressure and prevent the failure of the containment. The exhaust is made of Venturi pipes to maximize the potential of the water to filter aerosols that could be present in the steam.

KKM is also is also equipped with a DSFS (Drywell Spray Flood System) for potential post core damage conditions. It aimed to flood the low part of the drywell in order to prevent potential MCCI after RPV lower head failure.

The spent fuel is stored in a pool outside the containment but inside the reactor building (secondary containment). The pool has been recently equipped with an emergency refilling system. The system doesn't require any action within the reactor building. The mitigation of hydrogen concentration build-up and formation of explosive mixture is achieved using thermal recombination system, inerting of the containment and various ventilation and depressurization systems.

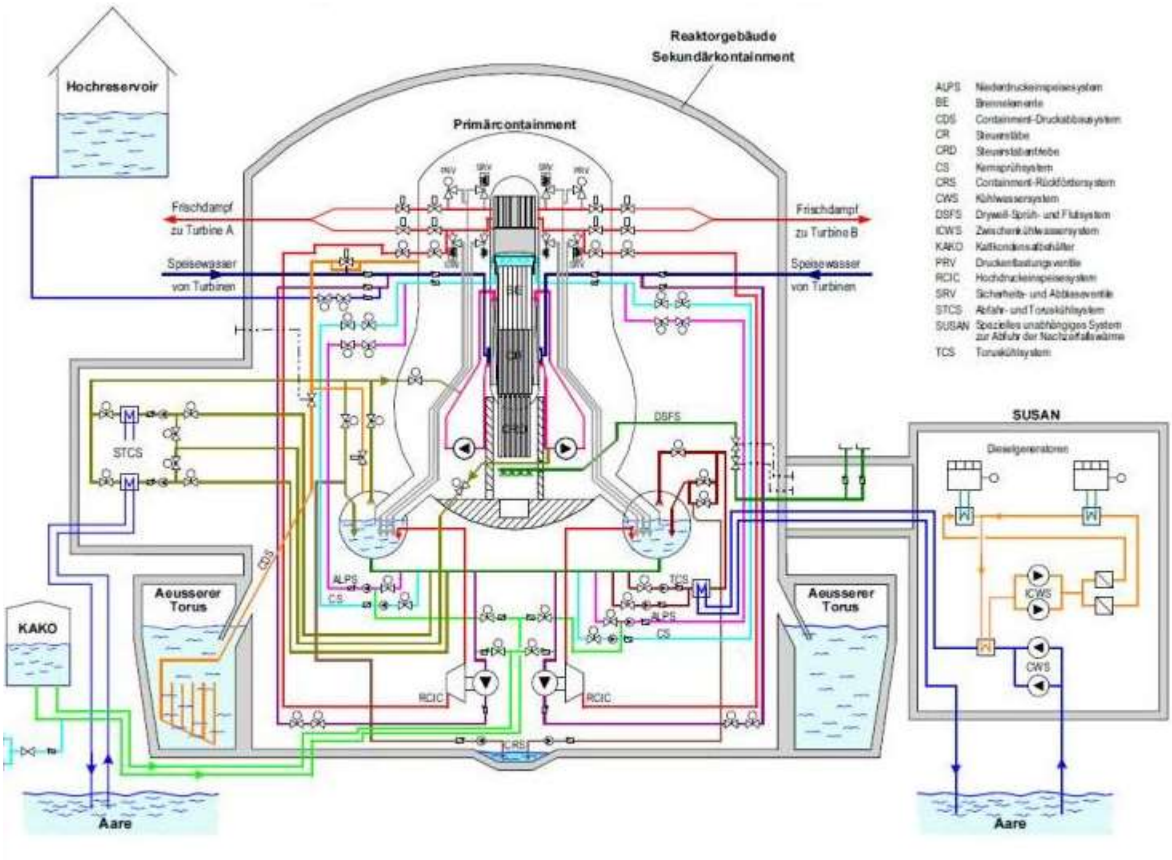


Figure A.3-3 Schematic of the KKM Mark I containment

A.3.5 BWR Mark II Containment (GE design), Japan

A General Electric Mark II containment is adapted to Japanese BWR NPPs. Figure A.3-4 shows a diagram of the containment and the reactor building.

The thermal power and electric power of a General Electric BWR-5 Mark II type NPPs are 3293 MWt and 1100 MWe respectively. The containment is made of a steel vessel which separated to a drywell and a wetwell by a diaphragm floor. Reactor building encloses this vessel, providing additional shielding and resistance. The wetwell contains a suppression pool filled with water. Vent pipes interconnect the drywell and the suppression pool. The containment inner diameter of cylindrical part is about 26 m and a height 48 m. The free volume is about 5700 m^3 in drywell, 4100 m^3 in wetwell and 3400 m^3 in suppression pool. The design pressure of the containment is 4.2 bar (a).

The containment system contains the ECCS system, the containment spray cooling system, the flammable gas control system (FCS) and the standby gas treatment system (SGTS). The containment drywell and wetwell are inerted by nitrogen during normal operation. The SFP is located outside the containment wall within the reactor building on the top of the containment.

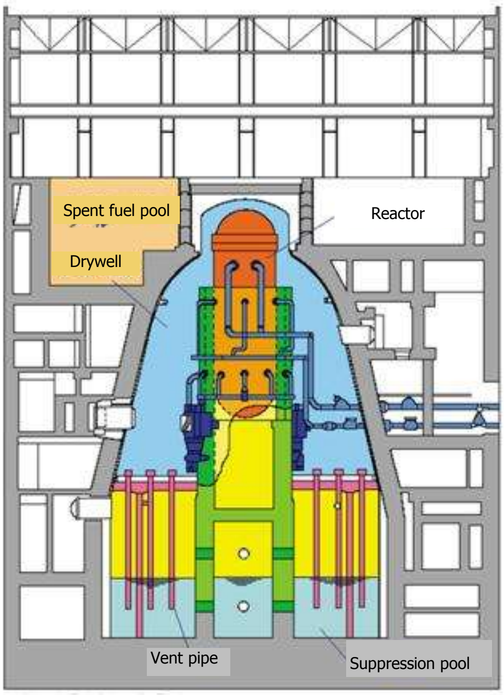


Figure A.3-4 BWR Mark II containment

The low pressure core injection LPCI and the containment spray cooling are one of an operation mode of the residual heat removal (RHR) system. After the core has been recovered, the LPCI mode is switched to the containment spray cooling mode which sprays the water from the suppression pool to the drywell and the wetwell.

When a DBA such as the LOCA occurred, the ECCS injection starts. The containment is isolated. The ADS works to depressurize a core pressure with opening safety relief valves. Hydrogen and oxygen concentration in the containment is controlled below the combustible level by recombining them with the FCS system during DBAs. The SGTS system is triggered to prevent to release radioactive materials to the environment.

To prevent a long term containment over-pressure failure in case of SAs, the unfiltered containment venting system has been equipped as an accident management concept. The system is connected to the gas phase of the drywell and the wetwell of the containment and has a separate off-gas pipe towards the stack or the environment.

A.3.6 BWR Mark III Containment (US)

BWRs with Mark III containment (Figure A.3-5) typically comprises of a concrete reactor building with a containment that completely surrounds the drywell. At the bottom of the containment, an annular suppression pool is located between the containment wall and drywell wall. Below the pool surface, horizontal vents are constructed in the drywell wall. A key variation of the Mark III containment design pertains to the make-up of the containment vessel; the design could be a free-standing steel vessel consisting of a vertical cylinder and a torus-spherical dome surrounded by a concrete reactor building (shield building), or it could be a steel-lined, reinforced concrete structure consisting of a vertical cylinder and a hemispherical dome. The diagram below provides an overview of the key features of the Mark III containment design representing a free standing steel containment vessel within the concrete reactor building (shield building).

Figure A.3-5 BWR Mark III containment

The total net free volume of the containment is approximately 35000 m³ and the drywell free volume is approximately 20 % of the total volume. The containment structure encloses the reactor system and is the final barrier against the release of radioactive fission products in the event of a breach of either the primary or secondary coolant system. Evaluations entail a variety of postulated design basis and beyond design basis (including core melt) events, involving accident progression and radiological source term calculations. The General Electric BWR/6 reactor coolant system is located inside the drywell, whereby air-steam mixtures from postulated LOCAs are channelled into the suppression pool. The drywell structure also supports the upper containment pool and provides shielding to reduce radiation levels in the containment to allow access during power operation. The containment design pressure is 2.0 bar (a), however the ultimate containment pressure capacity could be as high as 5.0 bar (a). An important attribute of the Mark III design is construction of the containment shell surrounding the drywell, effectively providing a double layer of protection. If containment failure were to occur, in many cases the containment would fail first, leaving the drywell and suppression pool intact. Any subsequent fission product releases would still be scrubbed as they passed through the suppression pool, greatly reducing the source term. Thus, the only accidents (other than bypass sequences) likely to produce large source terms must involve failure of the containment plus either loss of the suppression pool or failure of the drywell. Furthermore, the containment sprays/fan coolers can be used to remove fission products and cool the containment. Note that the SFP is situated in a separate building as compared to the earlier BWR containment designs which is located above the drywell structure.

For postulated SAs where copious amounts of hydrogen are produced, the Mark III design includes a hydrogen ignition system consisting of numerous glow plug igniters. The system design is similar to what is installed in other containment types.

A.3.7 BWR Mark III (KKL, Switzerland)

The Swiss Leibstadt Nuclear Power Plant (KKL) is a BWR Mark III unit (General Electric design) with gross electric power of 1275 MWe (Figure A.3-6).

KKL has a containment vessel completely surrounding a drywell and a pressure suppression pool. The containment vessel is a cylindrical steel structure with a spherical dome and flat bottom supported by a reinforced concrete foundation mat. The containment system is designated as Mark III Containment. The free air volume of the containment (including the suppression pool) is about 36000 m^3. The drywell is made by concrete and has a cylindrical shape. The suppression pool has an annular shape. Part of the suppression pool water lies inside the drywell, retained by the cylindrical concrete retaining wall (weir wall). The largest part lies outside the drywell in the containment between the outer drywell wall and the containment shell wall. Water from the upper pool is released into the suppression pool under emergency conditions.

Storage space is provided for temporary storage of spent fuel elements in a separate extension of the equipment storage pool in the reactor building (called containment pool) during the shutdown. Spent fuel shall not be stored in these racks during operation. The storage of spent fuel and channels at the site is in the pools located at the fuel handling building.

50 igniters are distributed symmetrically in the containment regions. A FCV system is available at KKL and can be actuated manually when the containment pressure is above 2.55 bar (a) or passively through the opening of rupture disk when the containment pressure reaches 3.1 bar (a).

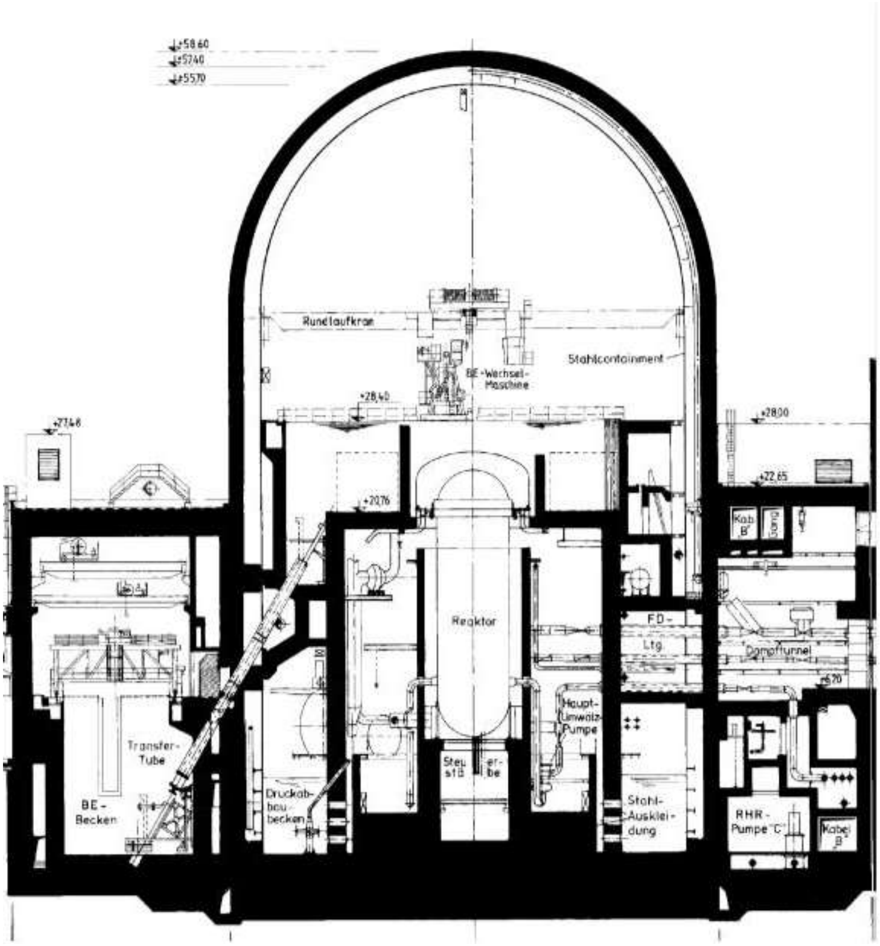


Figure A.3-6 BWR Mark III containment (KKL)

A.3.8 BWR Mark III (Cofrentes, Spain)

Spain has one General Electric BWR equipped with Mark III containment (Cofrentes) in operational since. An overview of a typical containment is shown in Figure A.3-7.

Presently, the thermal output is 3237 MWt, corresponding to 1092 MWe of electricity production. The containment prevents significant release of radioactive products and consists of a steel vertical cylinder and dome. The inner diameter is 34.8 m and the height is 55.5 m. It is divided into 2 main regions: the drywell and the surrounding containment rooms. The drywell contains the reactor pressure vessel and its free volume is 6851 m^3. The containment free volume is 32211 m^3. The suppression pool contains 3650 m^3 of water and is located both in the drywell and the containment. Three rows of horizontal circular vent pipes communicate between the drywell and the wetwell of the containment. The design pressure of the drywell and the containment is 3.1 bar (a) and 2.05 bar (a) respectively [A.3].

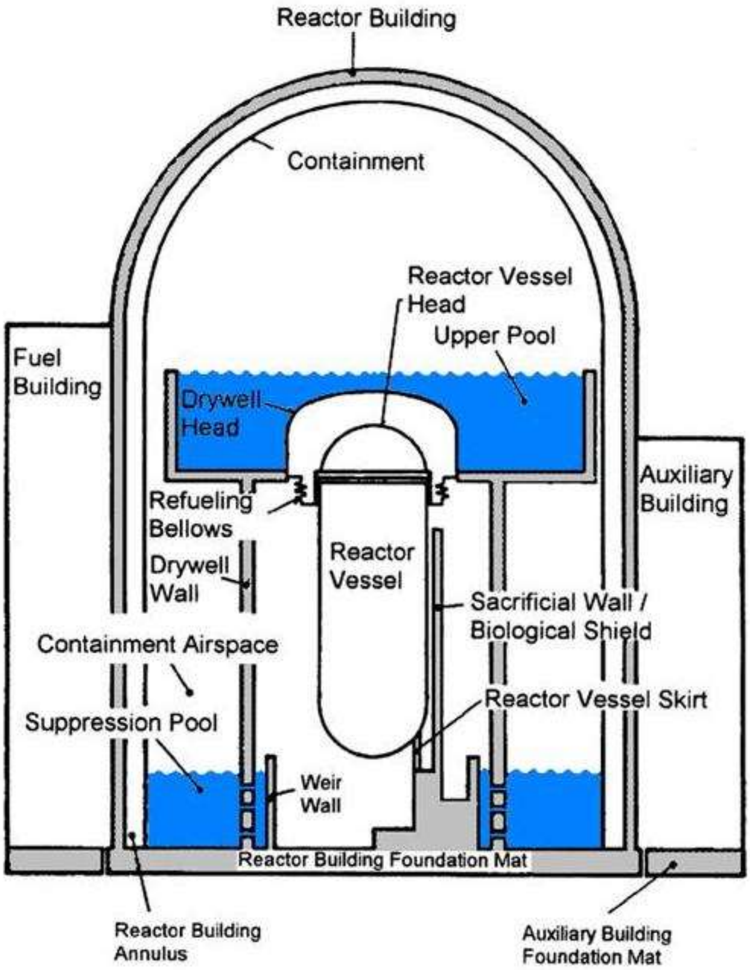


Figure A.3-7 Schematic of the Spanish BWR Mark III containment

There is an upper pool within the containment crossing the top of the drywell area. It provides water to the suppression pool in order to maintain the suppression pool water level above the first row of containment vents after DBA. The containment is surrounded by an additional concrete structure, the reactor building, whose free volume is 10760 m^3. It provides additional fission products retention by remaining at sub-atmospheric pressure after DBA. The SFP is located outside the containment and reactor building in a separate building adjacent to the containment.

The containment hydrogen control system for DBA is comprised of monitoring, mixing and thermal hydrogen recombiners. Additionally, the plant is equipped with hydrogen AC-power igniters that provide controlled burning of the hydrogen generated during beyond DBA. The containment can be cooled by a spray system with injection nozzles located in the upper dome. A hardened containment vent provides containment pressure relief for beyond DBA. As a result of the “stress tests”, a filtered containment system, a PAR system and an alternative electric supply to the existing containment hydrogen igniters system have been required to be installed in the plant [A.4]. The plants have procedures for normal conditions, emergency conditions and SAs (SAMGs). In SAMG for GE-BWR Mark III it is recommended: if AC igniters are not working, hydrogen is controlled by switching off thermal recombiners, switching on sprays and venting the containment without limitations.

A.3.9 ABWR Containment, Japan

ABWR NPPs exist in Japan where the thermal power and electric power are 3926 MWt and 1356 MWe respectively. Figure A.3-8 shows a diagram of the ABWR containment.

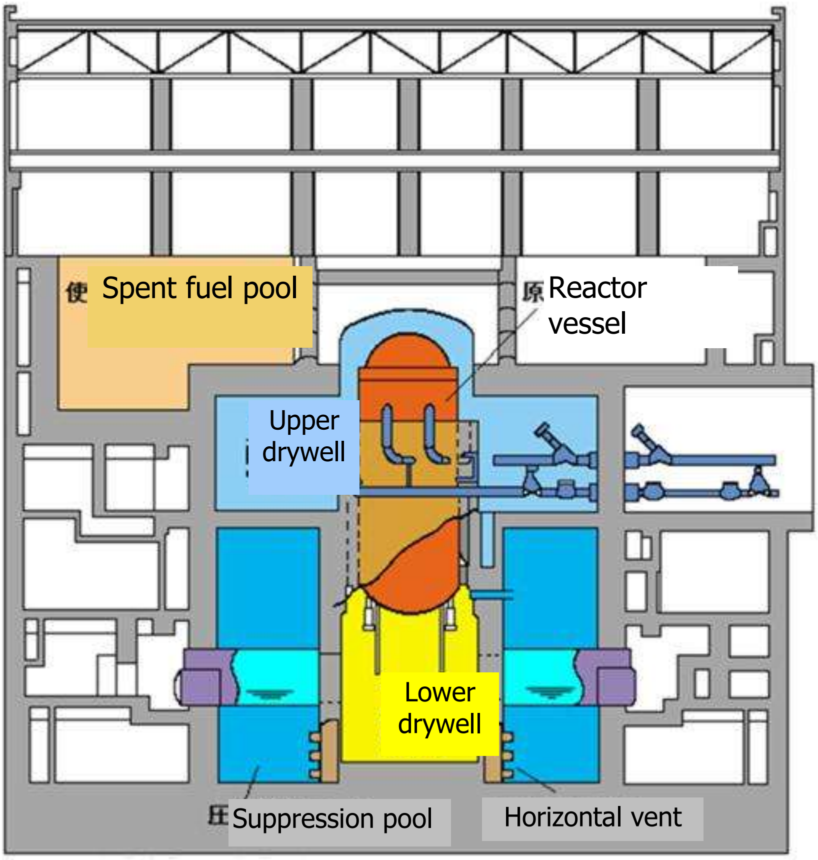


Figure A.3-8 ABWR containment

The containment is made of a reinforced concrete containment vessel (RCCV) with inner steel liner which is separated to a drywell and a wetwell by a diaphragm floor and a reactor pedestal wall. Reactor building encloses RCCV, providing additional shielding and resistance. The wetwell contains a suppression pool filled with water. Horizontal vents interconnect the lower drywell and the suppression pool. The containment inner diameter is about 29 m and a height 36 m. The free volume is about 7400 m^3 in drywell, 6000 m^3 in wetwell and 3600 m^3 in suppression pool. The design pressure of the containment is 4.2 bar (a).

The containment system contains the ECCS system, the containment spray cooling system, the flammable gas control system (FCS) and the standby gas treatment system (SGTS). The containment drywell and wetwell are inerted by nitrogen during normal operation. The SFP is located outside the containment wall within the reactor building on the top of the containment.

The low pressure core injection LPCI and the containment spray cooling are one of an operation mode of the residual heat removal (RHR) system. After the core has been recovered, the LPCI mode is switched to the containment spray cooling mode which sprays the water from the suppression pool to the drywell and the wetwell.

When a DBA such as the LOCA occurred, the ECCS injection starts. The containment is isolated. The ADS works to depressurize a core pressure with opening safety relief valves. Hydrogen and oxygen concentration in the containment is controlled below the combustible level by recombining them with the FCS system during DBAs. The SGTS system is triggered to prevent to release radioactive materials to the environment.

To prevent a long term containment over-pressure failure in case of SAs, the unfiltered containment venting system has been equipped as an accident management concept. The system is connected to the gas phase of the drywell and the wetwell of the containment and has a separate off-gas pipe towards the stack or the environment.

A.3.10 Asea Atom BWR (Olkiluoto 1 & 2)

The Finnish Olkiluoto 1 & 2 are BWRs designed by Asea Atom and operated by TVO [A.7]. They started commercial operation in 1979 and 1982. They have electric power of 880 MW and thermal power of 2500 MW. There are similar reactors in Sweden.

The containment is shown in Figure A.3-9. Gas volume of the containment is about 7300 m^3, and the condensation pool volume is about 2700 m^3. The containment design pressure is 4.7 bar (a), and the ultimate pressure capacity is about 10.1 bar (a). The containment walls are 1.1 m thick pre-stressed concrete with a steel liner embedded inside the concrete. The drywell head is made of steel. There is a containment spray system both in the drywell and in the wetwell. The containment is surrounded by the reactor building, which acts as a secondary containment. The SFP is located in the reactor building, above the containment.

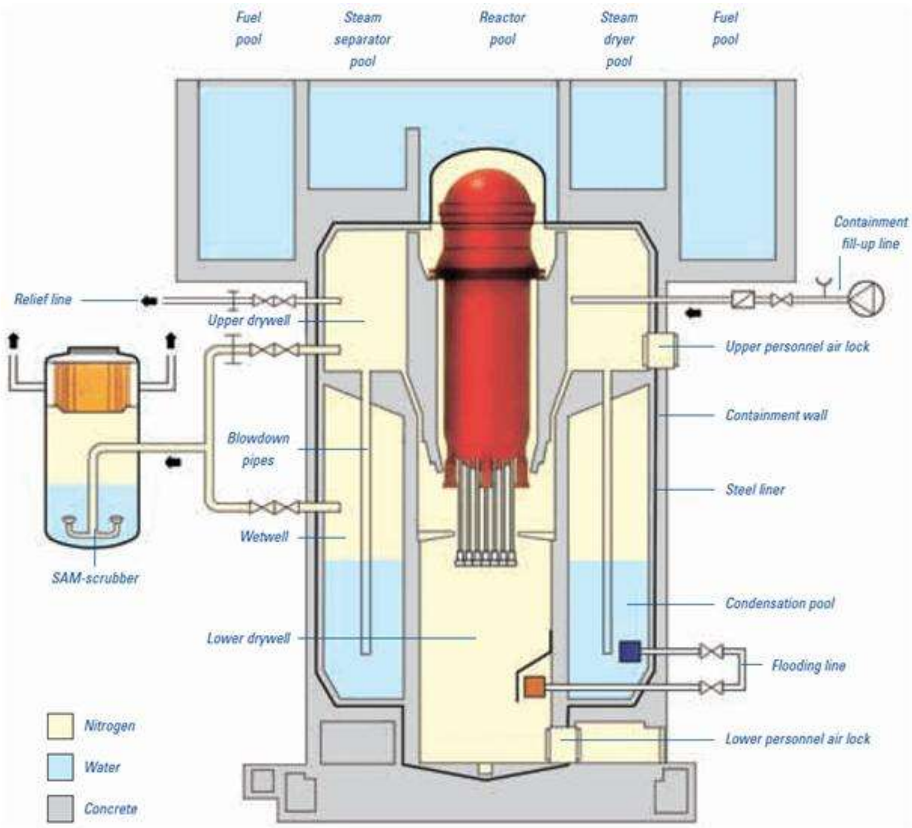


Figure A.3-9 Asea Atom BWR (Olkiluoto 1 & 2)

In the case of a SA, the lower drywell is flooded from the condensation pool by opening valves in the flooding lines. Upon failure of the RPV, the melt drops into a deep water pool. High pressure melt ejection is prevented by dedicated reactor depressurization valves.

Mitigation of SAs was not included in the original design basis. After the Chernobyl accident several SA mitigation systems were installed as backfitting. A containment filtered venting system has been installed. The filtered venting would normally be performed from the wetwell. The containment can be filled with water with firefighting pumps through the fill-up line. In this case, if the wetwell venting line becomes submerged, filtered venting can be performed from the top of the drywell. If the operators are not able to open the valves of the filtered venting line, the drywell filtered venting line automatically opens by bursting of a rupture disk at 5.5 bar. The filter unit is of venturi scrubber type. The decontamination factor is 1000 for particles larger than 0.3 µm and 100 for molecular iodine. The cleaned gas is released through a separate pipeline that has been installed inside the stack. The pipeline is normally inerted with nitrogen and sealed with a rupture membrane to prevent hydrogen burns inside the system.

To prevent hydrogen burns, the containment is inerted with nitrogen during power operation. There is currently no provision for dealing with hydrogen in the reactor building. However, the possibility to install latches, which could be opened during an accident for venting of hydrogen from the reactor building to the environment, is being considered.

A.3.11 Asea Atom BWR 75 (Forsmark 3 and Oskarshamn 3 Sweden)

The Swedish reactors Forsmark 3 and Oskarshamn 3 BWRs are designed by Asea Atom and operated by Forsmark Kraftgrupp AB (FKA) and Oskarshamns Kraftgrupp AB (OKG) respectively. These reactors are the latest constructed ABB reactors and started commercial operation in 1985. The initial electric power was 1050 MW with an thermal power output of 3000MW, and after several steps of power uprates the current power output is 1170 MW electric (3300 MWt) for Forsmark 3 and 1450 MW (3900 MWt) for Oskarshamn 3. There are five other BWR units in Sweden: Forsmark 1 and 2, Oskarshamn 1 and 2 and Ringhals 1.Forsmark 1 and 2 are of similar design as the Finnish Olkiluoto 1 and 2 mentioned above. Oskarshamn 1 and 2 and Ringhals 1 belong to the first generation of Swedish BWRs which have external recirculation pumps. The other BWRs have internal recirculation pumps. Hydrogen mitigation measures are similar at all BWRs – in particular all BWR containments are nitrogen inerted during normal operation.

The primary system in Forsmark 3 and Oskarshamn 3 is made up by 8 internal main circulation pumps and consists of 16 valves in total for depressurization with 4 valves classified for two phase flow. The blowdown function is supported by 24 pipes. The core consists of 700 fuel bundles and of 169 control blades.

The total gas volume of the containment is about 8300 m^3 distributed as 5600 m^3 in the upper drywell and 2700 m^3 in the lower drywell. The condensation pool volume is about 3200 m^3. The containment design pressure is 6.0 bar (a) and the ultimate pressure capacity is about 11bar (a). The containment walls are 1.3 m thick pre-stressed concrete with a steel liner embedded inside the concrete and the drywell head is made of steel. A containment spray system is installed both in the drywell and the wetwell. The containment is surrounded by the reactor building, which acts as a secondary containment. The SFP is located in the reactor building, above the containment.

As a SA management measure and in the event of a SA the released core debris from a reactor vessel rupture falls in a deep water pool. The water pool is established in the lower drywell by flooding water from the condensation pool. This is done automatically 30 minutes after containment isolation is activated.

Mitigation of SAs was not included in the original design basis, and during the 1980's a handful SA mitigation systems were installed as backfitting. This included a FCV, an additional containment spray, instrumentation qualified for SAs and shielding of some cable penetrations in the lower drywell.

The filtered containment venting consists of a multi-Venturi scrubber and the venting is performed from the upper drywell. The venting can be performed either automatically by a rupture disc released at 5.7 bar (a), or manually. The decision to vent is done by plant manager and the decision does not need to be confirmed by any government official or similar. The design decontamination factor is 100 for particles but has during experiments proved to be higher. The multi-Venturi scrubber system is housed in a separate building equipped with a dedicated stack with a height of 40 meters.

To prevent hydrogen burns, the containment is inerted with nitrogen during power operation. Nitrogen is also provided in the venture scrubber and in the connecting pipes. There is currently no provision for dealing with hydrogen in the reactor building. However, the possibility to install latches, which could be opened during an accident for venting of hydrogen from the reactor building to the environment, is being considered.

A schematic figure of the Asea-Atom BWR 75 containment with SA mitigation systems is presented in Figure A.3-10.

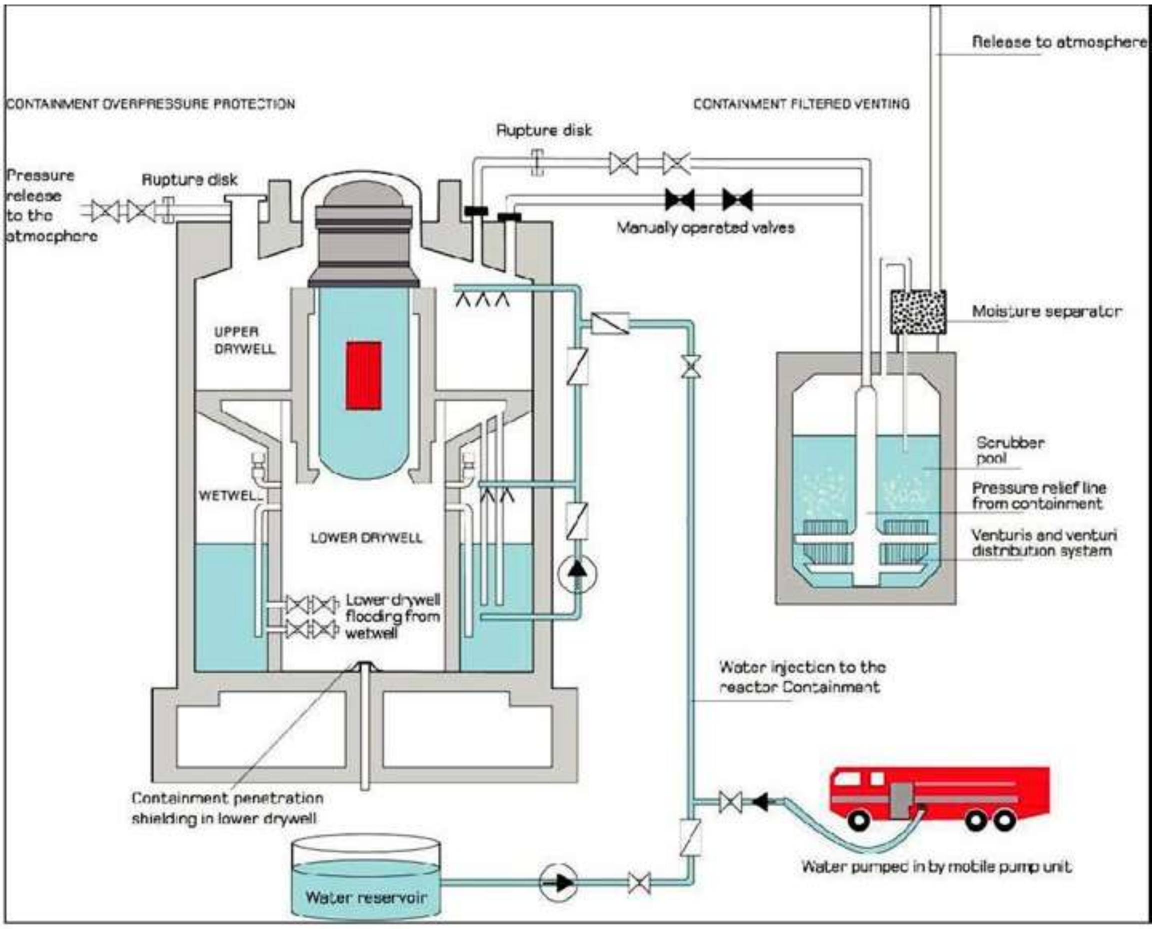


Figure A.3-10 Schematic figure of the Asea-Atom BWR 75 containment and severe accident mitigation systems

A.4 PHWRs - CANDUs

All CANDU reactors follow the same basic design (i.e., natural uranium with heavy water as coolant), although variations can be found in most units. The CANDU pressure tubes, holding fuel bundles immersed in high pressure (~10 MPa) and high temperature (~300°C) heavy water coolant, act like the reactor pressure vessel for LWR. They are located inside a calandria vessel (~220 m^3) filled with low pressure (near atmospheric) and low temperature (~70ºC) heavy water moderator. The calandria vessel is located inside a reactor vault (~540 m^3) built of concrete and filled with light water at ~38ºC, which functions as a biological shield under normal operating conditions and a passive heat sink under certain SA scenarios. Ex-vessel cooling, considered to be an important accident management programme in PWRs, is inherently in the CANDU design. A knowledge repository on CANDU technology is available in [A.18].

The NPPs with the CANDU design are operated today in Canada, South Korea, China, India, Argentina, Romania and Pakistan. Differences exist among these CANDU reactors. The following descriptions are more based on the operating CANDUs in Canada.

There are basically two types of installations of CANDU reactors; single or multi-unit stations. The multi-unit stations share a single containment system consisting typically of four units at each station with power output ranging from 540 to 940 MWe per unit, while the single units have a stand-alone containment with a power output in the 700 MWe range. All the single units, with the exception of the early units built in India and Pakistan, are of the CANDU 6 (C6) design. All CANDU units consist of two loops of heat transport system except one multi-unit station with only one loop for each unit.

A.4.1 CANDU Multi-Units

There are three multi-unit stations in Canada (Darlington, Pickering and Bruce). The containments (shape and dimension) vary from each other, but the general configurations are similar. The reinforced reactor building has a cylindrical shape in Pickering (Figure A.4-1), but rectangular shape in Darlington and Bruce. It serves as a support and an enclosure for the reactor and some of its associated equipment. The following parameters refer to the Darlington containment. The portion of the reactor building, forming part of the containment envelope, is called the reactor vault (a free volume of ~13000 m^3). The reactor vaults of all the units are connected with a common vacuum building (~80000 m^3) through a fuelling machine duct (~35000 m^3) and pressure relief duct. The containment positive internal design pressure is ~96.5 kPa (g) and the negative design pressure for the vacuum structure is zero kPa (absolute) and -53 kPa (g) for the rest of containments.

During normal operation of a multi-unit station, the reactor vault and the ducts are maintained under a slightly sub-atmosphere pressure to minimize any possible fission product release to the environment, and the vacuum building is kept at a very low absolute pressure. If an accident (i.e., LOCA) in one of reactor vaults causes the pressure in the shared containment volume to rise, the pressure difference between the reactor vault and the vacuum building can automatically (or manually) open the pressure relief valves, connecting the accident reactor vault to the vacuum building or other non-accident units depending on the scenario. In the vacuum building, the rise in pressure releases a large spray of cold water from a dousing tank in the upper dome of the vacuum building. The cold water condenses the steam and absorbs the energy released by the accident. The vacuum building is equipped with an emergency filtered atmospheric discharge system (EFADS) to vent the containment atmosphere following a DBA (maintaining the vacuum building sub-atmospheric). Air cooling units are installed in the reactor vault and they can be used for heat removal during an accident. Post-accident deliberate hydrogen ignition systems are installed in the reactor buildings of all multi-unit stations. PARs have been (or are being) installed for additional hydrogen mitigation.

A.4.2 CANDU 6 Single Unit

The reactor building of a C6 unit (see Figure A.4-2) has a free volume of ~50000 m^3. It is a pre-stressed and post-tensioned concrete structure consisting of three structural components: a base slab (~1.7 m thick), a cylindrical wall (41.5 m internal diameter with a minimum wall thickness of 1 m) and a spherical segmental dome (with thickness at the crown of 0.6 m). To assist in leakage control and cleaning, the containment structure has an internal lining comprising a flexible coating applied to the inner surfaces of the dome, floor and walls. The design pressure of a C6 unit is ~124 kPa (g) that can sustain a large steam line break with total dousing failure. The actual containment failure pressure, i.e., failure of airlock seals, is ~235 kPa (g) according to a station test.

For a C6 unit, the atmosphere energy removal consists of an automatic initiated dousing system, building air coolers, access airlocks and an automatically initiated containment isolation system. Containment filtered venting system has been installed in one of the Canadian C6 stations that completed refurbishment in 2012. This feature has been added as a new design for SA management. The dousing tank (~2000 m^3, including ~500 m^3 reserved for medium power ECC) is located in the dome of the reactor building and holds water for dousing spray and ECC injection. Dousing valves control the flow to six independent dousing headers located radially below the dousing tank. The spray system is not passive (although the flow itself is passive, the system has emergency power system backup and requires instrument air and electric power to operate the valves) and has no recirculation. Igniters are not equipped in the C6 units in Canada, but they are installed in the C6 units in Cernavada (Romanian), Qinshan (China) and Wolsong (Korea). In Canada PARs have been retro-fitted into the current stations and recommended for future designs.

The irradiated (or spent) fuel removed from CANDU reactors is stored under water in irradiated fuel bays (IFBs), equivalent to SFPs in LWRs. The IFBs are the size of ~25 m wide, ~50 m long and ~10 m deep. The walls and floors are constructed of carbon-steel-reinforced concrete approximately 2 m thick. Inner walls and floors are lined with a watertight liner consisting of stainless steel, a fibreglass-reinforced epoxy compound, or a combination of the two. The IFB structure is seismically qualified to maintain its safety function during and following a DBA. The IFBs are located outside of the reactor buildings in a confined building at ground level to prevent additional, cascading failures from events or accidents should they occur in the containment. No hydrogen mitigation system has been recommended for the IFB area.

A.4.3 Enhanced CANDU 6 (EC6)

The EC6 reactor is a Generation III CANDU design (see Figure A.4-3, [A.19]). While retaining the basic features of the CANDU 6 design, the EC6 reactor incorporates innovative features and state-of-the-art technologies that enhance safety, operation and performance. It has a target gross electrical output of between 730 and 745 MWe with two loops. The EC6 reactor building is a pre-stressed, seismically qualified concrete building and has been strengthened compared to previous CANDU 6 designs. It has a design pressure of 400 kPa (a) and the containment volume is nearly the same as CANDU 6. Pre-stressed concrete is reinforced with cables that are tightened to keep the structure under compression even when the forces it is designed to withstand would normally result in tension. The concrete containment structure has an inner steel liner that significantly reduces leakage rates in the event of an accident. The entire structure, including concrete internal structures, is supported by a reinforced concrete base slab.

The containment boundary consists of the reactor building structure, access air locks and a containment isolation system. Local air coolers remove heat from the containment atmosphere and are located to best maintain operating containment pressure and temperature. A hydrogen control system is designed to prevent the build-up and uncontrolled burning of hydrogen. It consists of PARs, hydrogen igniters, and a hydrogen monitoring system. In addition, the containment internal structures are arranged to promote natural air mixing inside containment. The provision of a spray system connected to the elevated reserve water tank will reduce reactor building pressures, if required, in the event of SAs. In the event of an accident, automatic containment isolation will occur to ensure that any subsequent release to the environment does not occur.

If severe core damage occurs, the mitigating features are provided to ensure it will not lead to the failure of the calandria vessel and the containment. This is achieved by:

- A large reactor vault water inventory surrounding the calandria vessel
- Using the shield cooling system to remove heat from the outside surface of the submerged calandria vessel to the ultimate heat sink via the shield cooling system heat exchangers or SA recovery and heat removal system (SARHRS).

This SARHRS system is designed and constructed to deliver cooling water to the reserve water tank, calandria vessel, calandria vault and the containment low-flow spray system following a BDBA. This system includes gravity-driven, passive water supply lines and a pump-driven recovery circuit.

The reserve water tank provides gravity-driven water in the short term while the SARHRS fresh water source is used in the interim prior to recovery and re-circulation from the reactor building sump.

The spent fuel bay is housed in the service building, which is a multi-level, reinforced concrete structure that is seismically qualified, Tornado missile and aircraft impact protected.

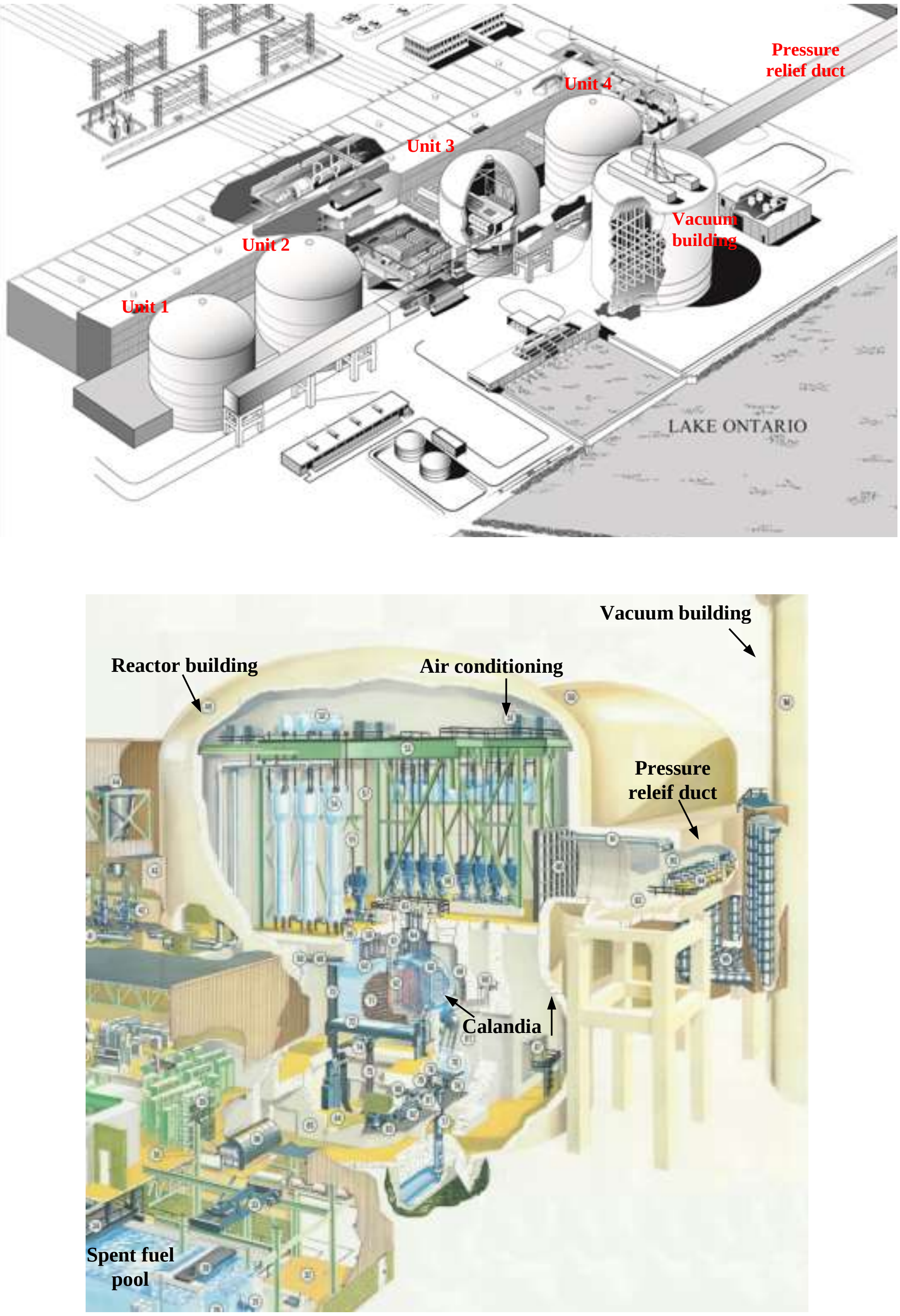


Figure A.4-1 Schematic diagram for Pickering multi-unit station.

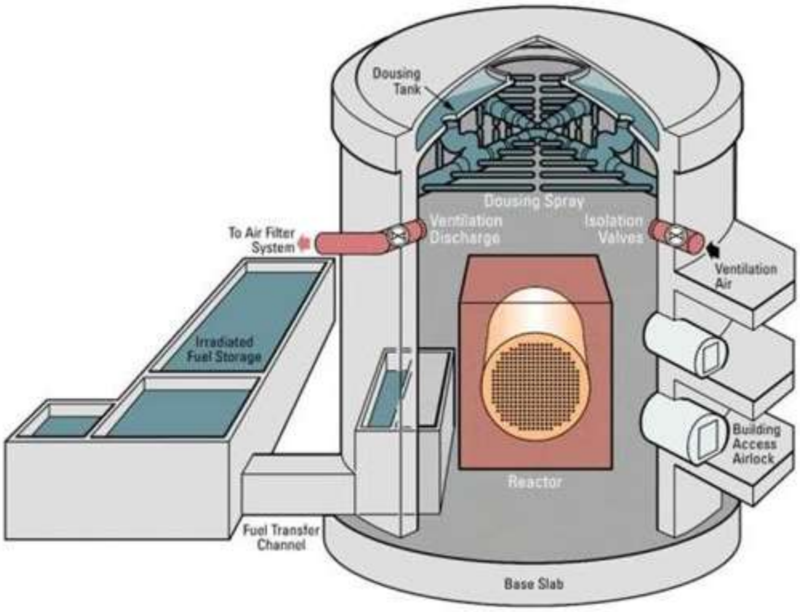
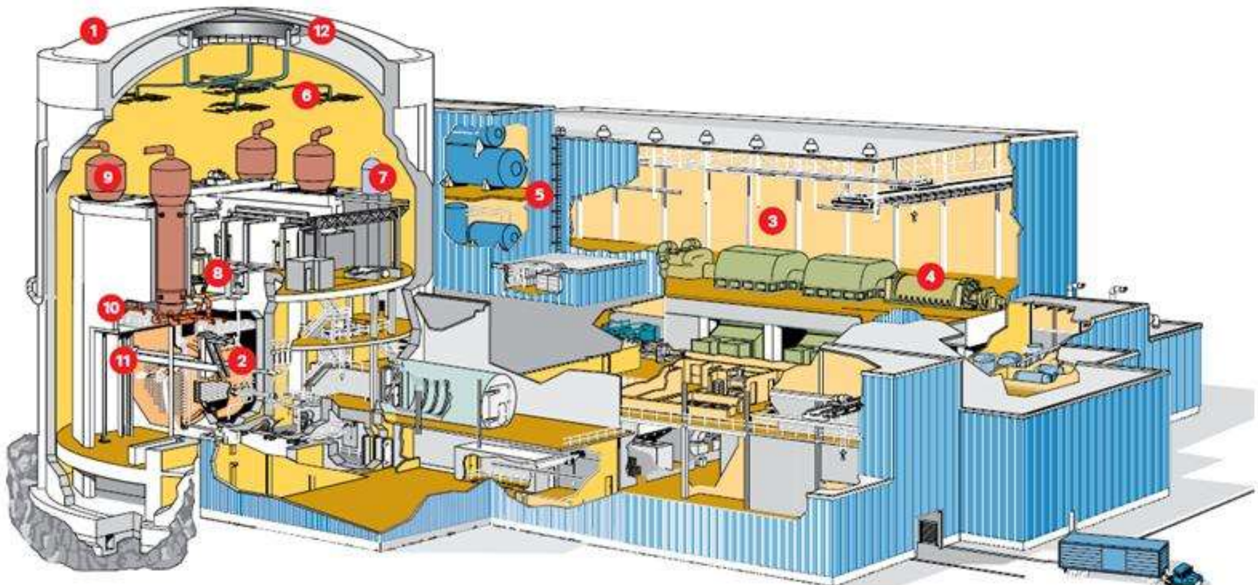


Figure A.4-2 Diagram of CANDU 6 single unit containment.



1-Reactor Building, 2-Calandria, 3-Turbine Building, 4-Turbine generator, 5-Service Building, 6-Spray system, 7-Pressurizer, 8-Heat transport pumps, 9-Steam generators, 10-Heat transport system, 11-Fuelling machine, 12-Reserve water tank

Figure A.4-3 Plant diagram of enhanced CANDU 6 single unit.

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B DESCRIPTION OF CODE CAPABILITIES AND VALIDATION

B.1 ASTEC[2]

Code developers	IRSN (France) and GRS (Germany)
Users	Finland, France, Germany, Spain, etc.
General capabilities	Simulate all relevant phenomena that occur during a SA in a water-cooled nuclear reactor (preferably PWR, VVER) and CANDU, from the initiating event at full power to the possible release of radioactive products (source term) outside the containment. BWR application is under development.
Applications	Safety analyses for nuclear reactors (e.g., EPR), PSA level 2 (French PWR, German PWR) source term evaluations (e.g., re-evaluation of the S3 source term for the French PWR), and development of SA management guidelines.
Reference	http://bit.ly/16uyP3Wm http://www.grs.de/en/astec

B.1.1 Code Capabilities

ASTEC has become the European reference software for SAs calculations thanks to the EC SARNET network of excellence. It is also used by organisations in Canada, Russia, India, Korea and China. ASTEC is widely used in IRSN level 2 PSA for PWR 900 and 1300 and EPR. It is also used in the preparation and interpretation of experimental programmes, in particular the Phébus FP integral test programme and in the tests carried out as part of the International Source Term Programme (ISTP).

ASTEC covers the entire phenomenology of SAs except steam explosion and the mechanical integrity of the containment. Its modular structure (Figure B.1.1-1) simplifies qualification of individual models by comparing the simulated results with those obtained experimentally.

B.1.1.1 Hydrogen Generation

The responsibility for model development and validation is split between GRS and IRSN. GRS is mainly responsible for the ASTEC CPA (containment) part while IRSN is responsible for the other modules. The code capabilities related to hydrogen generation are summarized as follows:

- Zr-Oxidation reaction with steam in the reactor core is fully treated while Zr oxidation reaction with oxygen in the core can only be accounted for in a simplified way in ASTEC V2.0.
- The lack of an adequate model to deal with Zr oxidation in air will be solved in the future ASTEC versions, by transferring the existing ICARE/CATHARE model into ASTEC.
- Fe and B_4C Oxidation reaction with steam only: No model to deal with Fe or B_4C oxidation by oxygen (air ingress conditions)
- Other chemical interaction: No direct impact on H_2 production, but creation of mixtures which could subsequently influence the course of oxidation processes.

[2]The description for ASTEC is provided by GRS and IRSN (http://www.sar-net.eu/node/42)

- Gas releases from MCCI and masses of elements (Al, C, Ca, Cr, Fe, H, Ni, O, Si, U, Zr, etc.) in different melt layers (oxide, metal) configuring the corium pool in the cavity are tracked. Species composition are calculated by the Material Data Base (MDB) package for each layer during the iteration for layer temperature, assuming stoichiometric equilibrium and a hierarchy of oxidation reactions uncertainty in evolution of pool configuration (e.g. mixed or stratified) impacts on hydrogen release, since the gas flow enters (and interacts with) the melt layer adjacent to the eroded concrete

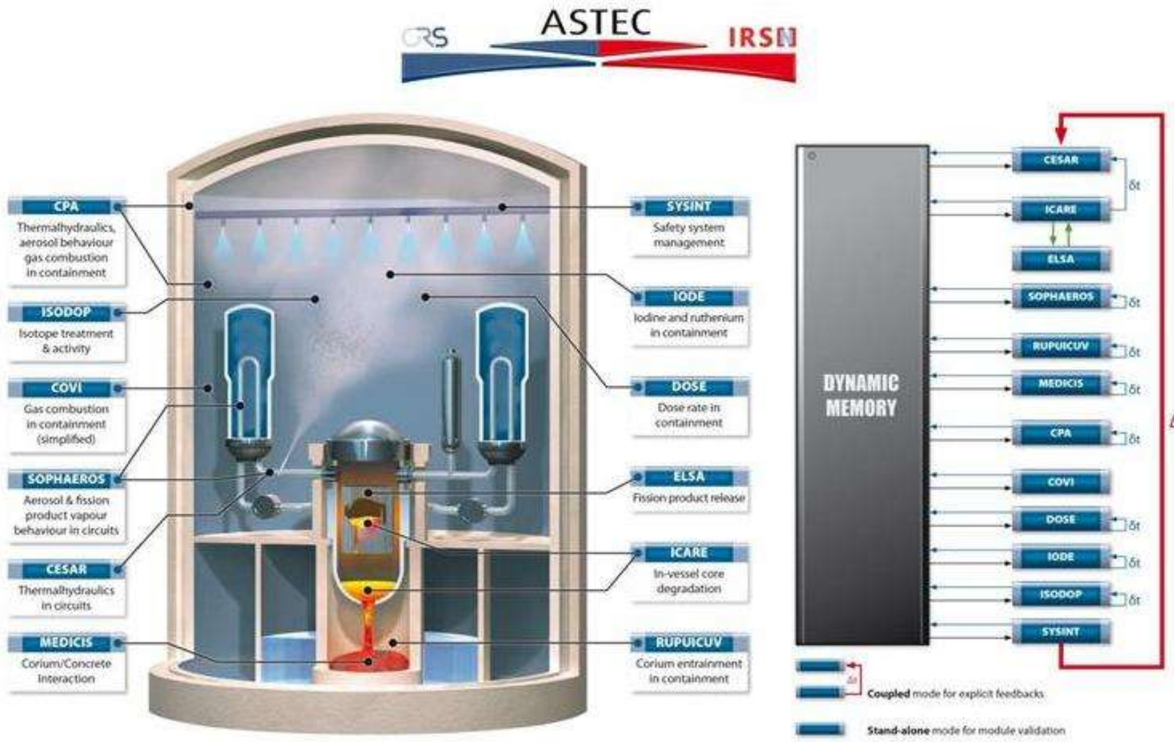

Figure B.1.1-1 Structure of ASTEC

B.1.1.2 Hydrogen Distribution

The typical non-condensable gases (H_2, He, N_2, O_2, CO, CO_2) are modelled in ASTEC. Different modules are responsible for the hydrogen transport in different parts of the reactor. Hydrogen is generated in the core by Zr oxidation, which is modelled by ICARE, or in the cavity, which is modelled by MEDICIS. The transport through the reactor cooling system is calculated by CESAR. The hydrogen is than released to the containment, which is simulated by containment part of ASTEC (CPA).

B.1.1.3 Hydrogen Combustion

Also within CPA, hydrogen deflagration is modelled with simplified parametric method. The combustion model COMB in the containment module CPA has been extended by FRONT model for the flame front propagation.

The burning model in ASTEC is applicable and validated for H_2-combustion. CO combustion is calculated but not validated due to lack of experiments.

Another model computes in a very simple way the maximal value of built-up pressure in adiabatic conditions of hydrogen combustion (AICC).

B.1.1.4 Hydrogen Mitigation Systems

A detailed PAR model simulated within a 1D atmospheric junction is implemented for all AREVA type PARs. Also a fast running parametric PAR correlation model based on the 1D detailed model for AREVA PARs is available. This simplified correlation does not give any information on PAR temperature or outlet concentration, on velocity, or mass flow rate. Concerning AECL PARs a detailed model is available only for the device used in OECD THAI 1 project. No detailed modelling and validation for AECL PARs with real size geometry (missing geometrical data) is available. A model for the simulation of NIS type PARs is under development.

Different containment spray models exist.

A model developed for VVER-440 bubble condenser, applicable for BWR and RBMK condensation pool exists.

A model using user-defined decontamination factors is implemented to model filters in containment venting systems. A detailed filter model is currently under development that accounts as well for hydrogen hazards in the filter lines.

B.1.2 Code Validations

The ASTEC package has been validated by over 160 tests, including:

- Analytical tests, either separate-effect or coupled-effect tests. For example, the VERCORS tests (CEA-IRSN-EDF) relating to fission-product release and transport, BETA (KIT –former FZK- Germany) concerning the molten corium-concrete interaction, and the ACE-RTF tests (Canada) on iodine behaviour in the containment;
- Integral tests, notably including the Phébus FP in-pile tests (IRSN) simulating an entire accident with real materials up to the source term in the containment, the out-of-pile CORA and QUENCH tests (KIT) representing a bundle of electrically heated fuel rods in the core, using simulant materials and the LOFT-LP-FP2 (INEL) in-pile test representing a 1/50 scale down model of a commercial 4-loop PWR.

Among these 160 tests, the OECD/NEA ISP exercises have been selected as often as possible as these constitute an international reference by virtue of the high quality of the measurements and their use as benchmarks when comparing software packages (PACTEL, VANAM, BETHSY, LOFT tests, etc.). The test matrix is continually being expanded by the results of international programmes, including: CCI-OECD (Argonne National Laboratories, USA), ISTP (EPICUR and CHIP at the IRSN), OECD THAI (Becker Technologies, Germany), and ARTIST (PSI, Switzerland).

The software has been tested against the TMI reactor accident in the USA (TMI-2) in 1979 with the aim of consolidating the results prior to application to actual reactor configurations while ASTEC applications to the Fukushima-Daiichi accidents are also carried out at IRSN in the frame notably of the on-going OECD-BSAF project. As the ASTEC CPA module and the GRS code COCOSYS are very similar, the model development and validation done by GRS was often done based on COCOSYS models. More details are presented in Section B.5.2.

B.1.3 Strengths, Limitations and Improvements

In common with the majority of software available worldwide, the ASTEC current model relating to the thermal-hydraulic coupling between CESAR and ICARE and the reflooding of degraded cores is still considered to be inadequate; R&D work on this model in order to improve it will continue within the CESAM project as a follow-up of the SARNET network. Such a process is of very high importance with respect to hydrogen risk issues since a late in-vessel reflooding could result in a violent oxidation of in-core remaining metallic structures (e.g. rod remnants or refrozen relocated corium) and could thus lead to get a large hydrogen peak. Another topic is the modelling improvement for BWR core degradation.

Regarding the main perspectives for the development of ASTEC, efforts are currently paid at IRSN to the development of a dedicated model to deal with the reflooding of degraded cores, taking full benefit of the on-going PRELUDE and PEARL IRSN experimental programmes. Other ASTEC current modelling efforts (with respect to hydrogen risk issues) are notably addressing corium concrete interaction with an emphasis on corium coolability aspects (both top cooling and bottom cooling are investigated) and DCH issues.

With view to code improvement, main focus at GRS will be on more realistic models for filtered venting systems in the CPA module, in particular Venturi scrubbers and aerosol or iodine filters. Besides, the modelling of aerosol and FP retention through pool scrubbing will be re-assessed and modelling improvements based on new experiments is expected. Finally, as concerns DCH phenomena, a more mechanistic model is currently under development in particular with view to the behaviour and chemical interaction of corium droplets entrained into the cavity and the adjacent containment zones.

B.1.4 Application Method

ASTEC can be used as an integral code with a complete simulation of the different parts of the power plant. This has the advantage that all phenomena are calculated in the core, the vessel, the reactor cooling system and the containment considering side effects from neighbour systems automatically. In addition, the transport of fission products is simulated starting with the detachment from the fuel up to the release into the environment with a single calculation. But due to the modular approach it is also possible to use single modules only for the simulation of single effect tests that simulate only containment or core degradation phenomena. This is also useful, if a focus is on a more detailed investigation of a single phenomenon during an accident sequence without any need to simulate the whole plant. ASTEC gives the user the freedom to choose the detail level that is needed for the investigation performed.

The ASTEC code has been applied for PSA level 2 studies for different NPPs. Plant specific input decks were developed and applied for different SA sequences either for individual accident phases or for integral calculation from initiating event until the releases of radionuclides into the environment is completed.

Uncertainty and sensitivity studies are also possible with ASTEC.

As well participation in EU sponsored projects is done, like EVITA, SARNET and others. In 2013 the new Euratom FP7 CESAM project (Code for European Severe Accident Management) started which concentrates on further development of the ASTEC code and on development of generic input decks for main types of European NPPs.

B.2 MAAP/MAAP-CANDU[3]

Code developer	Fauske and Associates Inc. (FAI), USA
Users	Canada, EDF, EPRI, Korea, Spain, Sweden, etc.
General capabilities	MAAP quantitatively predicts the evolution of a severe core damage accident starting from full power conditions given a set of system faults and initiating events through events such as core melt, primary heat transport system failure, vessel failure, shield tank/reactor vault failure, and containment failure
Applications	MAAP is an integral system analysis code for assessing SAs including small break LOCA, large break LOCA, LOCA/LOECI, and loss of heat sink in LWR and CANDU nuclear power plants. It is also capable of modelling steam generator tube ruptures, main steam line breaks and anticipated transient without scram initiating events. Accidents can be initiated when the reactor is at full power or when it is in a shutdown state.
Reference	http://www.fauske.com/nuclear/maap-modular-accident-analysis-program

MAAP is developed by FAI and owned by EPRI. In addition to the standard PWR/BWR versions of MAAP, a special version, called MAAP4-CANDU, has been developed for CANDU NPPs [B.1].

The MAAP-CANDU code can be broadly divided into three parts:

- Part I: the generic MAAP part, which models the PWR system with several generalized models;
- Part II: the CANDU-specific models, such as containment, ECC, are modelled using the capabilities of the generalized models of MAAP4;
- Part III: the CHANNEL model, which models the CANDU core and was developed by Ontario Hydro and is owned by the CANDU Owners Group (COG).

The following descriptions are based on MAAP-CANDU, but most of the features also apply to general MAAP.

B.2.1 Code Capabilities

MAAP-CANDU quantitatively predicts the evolution of a severe core damage accident starting from full power conditions given a set of system faults and initiating events through events such as core melt, primary heat transport system failure, calandria tank failure, shield tank/reactor vault failure, and containment failure. Models are included to represent the actions that could stop the accident by cooling the debris in the calandria tank or containment. It has a flexible input system allowing changes to many features of the reactor and associated systems, and allowing modifications for different CANDU reactor designs, but it is not very flexible with regards to the nodalization of the reactor coolant system or the types of safety systems that can be modelled as most of these features are hard coded.

B.2.1.1 Hydrogen Generation

The hydrogen generation from the following sources are modelled in MAAP-CANDU:

[3]The description for MAAP/MAAP4-CANDU is provided by AECL, Canada.

- Zr-steam reaction

  It includes hydrogen generation from inside the calandria tube (fuel sheath, pressure tube, inside of calandria tube), outside surface of the calandria tube, falling molten Zr in the calandria vessel, suspended debris and shield tank/reactor vault (during debris jet fragmentation). For Zr-steam oxidation in the suspended debris, MAAP-CANDU models the flow of steam through the suspended debris and assumes all the steam is consumed to oxidize the remaining Zr and produce hydrogen, given that the steam flows are likely to be low, the temperatures high, and the exposed Zr surface area large.

- Steel oxidation

  Hydrogen generation from the calandria vessel shell (inside/outside) and tube sheets (inside)

- Ex-vessel hydrogen formation (MCCI)

B.2.1.2 Hydrogen Distribution

MAAP can calculate the distribution of steam, main non condensable gases (H_2, N_2, O_2, CO, CO_2) and water. The hydrogen distribution related phenomena are modelled in MAAP as follows:

- Pressure-driven flows through openings
- Natural circulation flow inside the containment and buoyancy driven exchange gas flow through a single junction through flows between containment nodes
- To calculate stratification MAAP uses nodes that contain gradients or sub-nodes physics to simulate certain processes instead of well-mixed control volumes, but the sub-nodal physics model is rarely invoked by the code users, thus it has not been well tested.
- Phase change due to flashing and rainout, boiling of water pool due to decay heat of fission product or debris, steam/water absorption by suspended hygroscopic aerosols, spray droplet evaporation or steam condensation on the droplet surfaces, evaporation/condensation between vapour and uncovered heat sink surfaces (e.g., steam condensation on vertical or downward facing containment walls or on metal equipment is not modelled).

B.2.1.3 Hydrogen Combustion

In MAAP, detonation is not considered and only deflagration is modelled. Hydrogen and carbon monoxide combustion may occur according to different criteria:

- when the gas temperature exceeds a user-defined auto-ignition temperature regardless of the flammability of the mixture;
- when the mixture exceeds the flammability limit of H_2-CO-O_2-H_2O-CO_2-N_2 mixtures at the given gas temperature (less than the auto-ignition temperature) and pressure, and an igniter is active
- when a user-defined ignition criterion is met (the mixture exceeds the flammability limit with a user-defined shift) and no igniter is active

  Three types of burns are modelled phenomenally in MAAP:

- Global (or complete) burn: involve all the flammable gases in the containment due to auto ignition or the mixture exceeds downward flammability; The AICC temperature and pressure will be calculated when an complete burn is to be assumed to occur at any given time in the accident sequence for any given node;

- Local burn: initiated by deliberate ignition systems (e.g., igniters) and involves only part of the compartment gas volume;
- Continuous burn: when high temperature combustible gas enters a region with oxygen.

B.2.1.4 Hydrogen Mitigation Systems

MAAP has models for all the common containment engineered safety systems that are related to hydrogen mitigation. Their modelling approach is described as follows.

- PARs: Built-in correlation for hydrogen recombination rate as a function of hydrogen concentration, pressure and temperature for AECL and NIS type PARs
- Igniters: Glow plug type igniters to initiate combustion
- Filtered venting system (i.e., EFADS): Fan junction and standby gas treatment system (SGTS) features of the generalized containment model
- Containment fan or fan coolers: Heat removal and steam condensation in containment fan coolers
- Dousing spray: Activated whenever the pressure exceeds a user specified differential pressure threshold and water level in the storage tank must be above a user-input suction risers height
- Negative Pressure Containment System: Pressure relief valves connecting the containment to the vacuum building (using a vacuum breaker junction of generalized containment model)

B.2.2 Code Validations

B.2.2.1 Hydrogen Generation

Validations for hydrogen production from Zr-steam reaction were performed by the code developer using the CORA (severe core damage - clad oxidation) and TMI-2 accident (core degradation) tests. The MAAP model underestimated hydrogen production and consequent temperature rise because of premature liquefaction and relocation of fuel clad prior exposure to steam flow for the CORA tests. It was generally in good agreement with the overall plant response for the TMI-2 accident.

AECL performed validations of the channel model for MAAP-CANDU using analytical solution for CANDU fuel and fuel channel thermal response, and CHAN tests to demonstrate adequacy of MAAP-CANDU to model metal-water reaction and hydrogen production in fuel channel. The accumulated mass was in good agreement with the measurement.

B.2.2.2 Hydrogen Distribution

The code developer performed validations against the containment natural circulation tests (e.g. HDR, E11.2, CSTF, and NUPEC tests). The MAAP model provided an effective representation of the containment response under both natural circulation and forced flow conditions. In general, the predicted trend was similar to the measurement.

AECL performed code-to-code simulations CANDU 6 containment response to a station blackout event sequence using MAAP-CANDU against GOTHIC. The agreement was fairly good in terms of containment response.

B.2.2.3 Hydrogen Combustion

The code developer performed validations against the AECL, EPRI, VGES and Nevada tests. The combustion completeness model was extensively benchmarked against experimental data. A flame flux multiplier was recommended (2 for quiescent atmosphere and 10 in presence of turbulence).

B.2.2.4 Hydrogen Mitigation Systems

For the fan cooler model, the code developer performed validation against the TMI fan cooler tests. The model calculated heat removal and steam condensation rates for the PWR containment fan coolers (finned tube, cross-flow heat exchanger).

For the spray model, the code developer performed validation against steam dousing tests to examine the spray heat transfer. The general behaviour was consistent with the tests of water sprayed into a steam-air mixture in a large vessel, but it underestimated energy transfer for higher temperature spray water tests. It should be noted that MAAP only uses a single droplet diameter and it assumes actuation of spray immediately causes the spray to be available to the entire volume.

B.2.3 Strengths, Limitations and Improvements

The assumptions and limitations of the MAAP-CANDU are discussed as follows:

- The MAAP calculations are intended for SA calculations. There are inherently high uncertainties associated with SAs predictions because of the complexities (multiple system interactions) involved and the limited experimental data availability. Since the simulation time scale is typically in terms of days, the accuracy of the predictions of key events timings are expected to be in the orders of hours.
- It models and simplifies the physical system as lumped, nonlinear, first order, coupled differential equation in time. Most of the differential equations express mass and energy conservations. Momentum balances are considered in containment models only and assumed to be quasi-steady. Therefore, the conservation of momentum is not expressed as a differential equation. It models the primary heat transport system, moderator and shield tank/calandria vault thermal-hydraulics simply as containers where the water inventory, energy and phase changes are tracked. Detailed flows are not modelled.
- Only light water properties are used in MAAP-CANDU, but a CANDU reactor uses heavy water in the primary coolant and moderator.
- The fuel channels are usually distributed into 18 channel groups per PHTS loop. It is intended to group them according to channel power and elevation. The fuel elements are represented by multimode fuel rings. Detailed fuel and channel conditions are calculated only when the group of fuel channels is effectively dry. No temperature excursions will occur until the channels are deemed to be dry. Thus, computation time will be saved by deferring channel calculations. The initial fuel temperature and Zr oxide thickness are input.
- Predictions of fuel bundle slumping and core disassembly are determined by exceeding internally-calculated and user-input temperatures. The user input values are based on external calculations and/or engineering judgement. The timing of the overall core collapse (mass failure of channels) is based on the mass of suspended debris in the calandria vessel.
- The melt relocation in bundles and suspended debris beds are very complex phenomena, and are modelled in a simple way as there are virtually no experimental data available. Therefore, the best engineering judgment shall be used to estimate the process.

The assumptions in the MAAP-CANDU code simplify the solution and ensure reasonable simulation run times while maintaining a level of detail necessary to model an integrated SA. Assumptions and model simplifications in MAAP-CANDU can lead to uncertainties in the timing of major SA events, which will in turn affect:

- The production rate of hydrogen from steam-Zr reactions (e.g., models of steam flow and uncertainties in the core collapse/disassembly time)
- The combustion/recombination of hydrogen due to modelling simplifications of the complex flow patterns, especially within the containment volumes. Likewise, simplifications in the PAR or combustion models may affect results.
- The flow patterns and releases of hydrogen from the vessel into the containment volumes (e.g., the timing of the rupture discs bursting, simplified containment models that cannot track complex flow patterns and the timing and amounts of hydrogen production).

B.2.4 Application Method

The MAAP-CANDU code is intended to be used for Level 1 and Level 2 PSA and SA management evaluations for current and advanced CANDUs. It can be used for Level 1 analyses to determine whether a given specification of initiating events and recovery times leads to core damage or recovery or to both. It can be used in Level 2 analyses to determine containment response and fission-product release histories to the environment. MAAP-CANDU has been extensively used in Level 2 PSA and SA simulation of CANDU stations and is being applied to SAM activities.

One important application of MAAP-CANDU is to support consequence and risk assessment analyses in determining the timing and duration of certain stages of the accident progression. The timing and duration of SA progression are generally measured in terms of hours and days, which is relatively long compared to DBA analysis.

The postulated accidents involve the loss of heat removal following an initiating event (e.g. following a SBO). These accidents involve extreme fuel temperature excursions, damage to the core structure, and releases of fission products. The analysis of this accident category falls outside the design and licensing process for the plants.

MAAP-CANDU is unique among the CANDU industry computer codes because of its integral system and nuclear safety multi-disciplinary nature. For these reasons, MAAP-CANDU can be applied to applications other than SA, such as training, accident management, FLEX (diverse and flexible coping capability)/EME (emergency management equipment) activities support and SAMG validation.

B.3 MELCOR[4]

Code developers	Sandia National Laboratories for U.S. NRC
Users	Belgium, Finland, Germany, Japan, the Netherlands, Switzerland, Spain, USA, etc.
General capabilities	Simulate all the phenomena that occur during a SA in water-cooled nuclear reactors, from the initiating event to the possible release of radioactive products (the source term) outside the containment.

[4]The description for MELCOR is provided by NRC, USA.

Applications	Safety analyses for nuclear reactors, source term evaluations and the development of SAM guidelines.
Reference	http://melcor.sandia.gov/ http://www.nrc.gov/about-nrc/regulatory/research/comp-codes.html

B.3.1 Code Capabilities

The code structure of MELCOR is demonstrated in Figure B.3.1-1. Many MELCOR models are mechanistic; however, some are parametric, particularly those related to phenomena with large uncertainties where consensus is lacking concerning an acceptable mechanistic approach. Current use of MELCOR for deterministic safety analysis is often supplemented by uncertainty analyses and sensitivity studies. To facilitate this, many of the mechanistic models have been coded with optional adjustable parameters. These parameters can be varied one at a time as well as multivariate effects can be examined in a systematic manner. This does not affect the mechanistic nature of the modelling, but it does allow the analyst to easily address questions of how particular modelling parameters affect the course of a calculated transient.

B.3.1.1 Hydrogen Generation

MELCOR uses the classical parabolic kinetic model for oxidation, which applies quite well in most situations provided the geometry is intact. Once degradation occurs both the configuration and properties change. MELCOR uses a modified version of the intact geometry model to calculate oxidation of refrozen metallic that had melted and “candled” down. It really only considers the change on surface area, not the effect of material interaction. While not fully satisfactory, it is not significantly worse that the other simulations codes.

MELCOR is capable to predict the hydrogen generation during a SA, including

- Hydrogen generation due to Zr-steam reaction, Fe oxidation, and B_4C oxidation for the “ROD-LIKE” geometry.
- Hydrogen generation due to Zr-steam reaction, Fe oxidation, and B4C oxidation in the “LATE PHASE” configuration.
- Ex-vessel generation due to MCCI and DCH

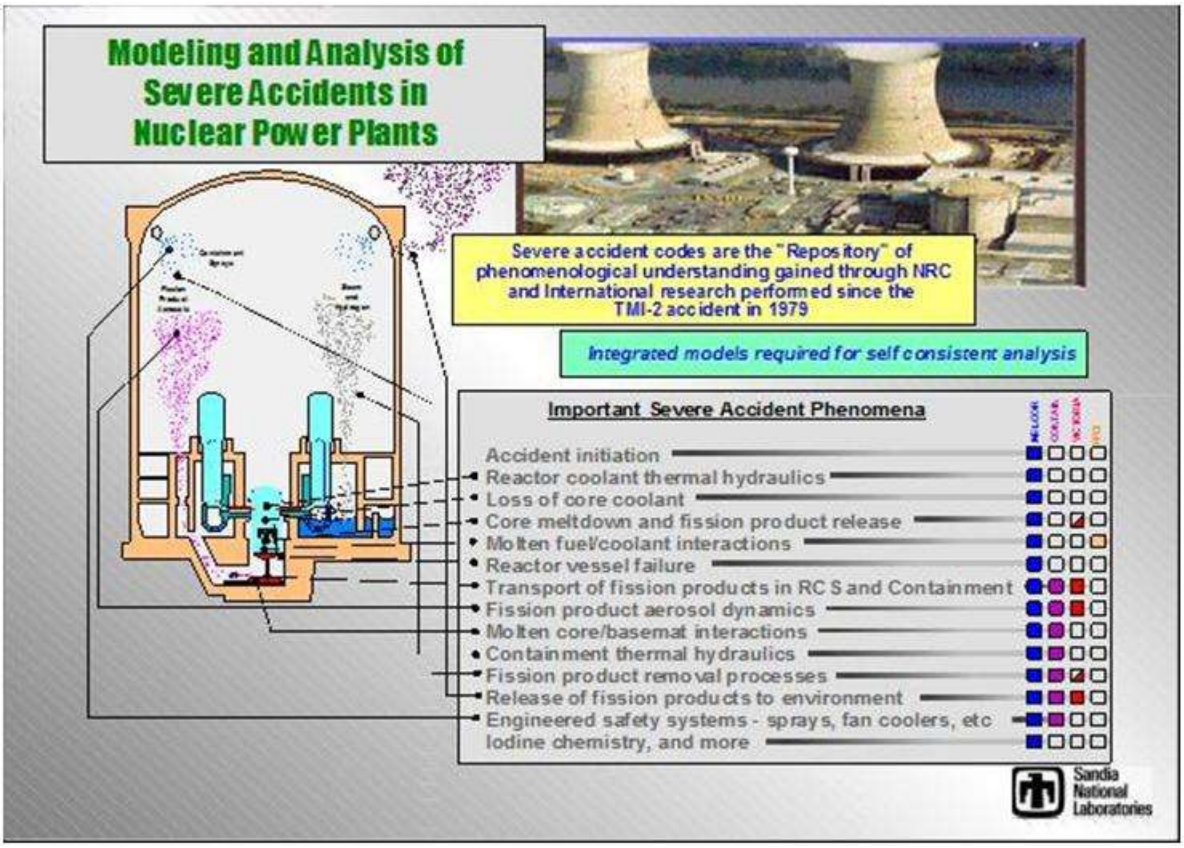

Figure B.3.1-1 Structure of MELCOR

B.3.1.2 Hydrogen Distribution

MELCOR uses a one dimensional model for transport process, including major typical non-condensable gases (H_2, N_2, O_2, CO, CO_2) and water. The hydrogen distribution related phenomena are modelled in MELCOR, such as:

- Pressure driven flows due to high energy break flows, activation of sprays or fans coolers
- Natural circulation flow inside the containment or buoyancy driven exchange gas flow through a single junction through flows between containment nodes
- Use of control volume nodes to simulate global/regional concentration gradients (stratification)
- Phase change due to flashing, rainout, boiling of water pool (e.g. decay heat of fission product or debris), steam/water absorption by suspended hygroscopic aerosols, spray droplet evaporation or steam condensation on the droplet surfaces, evaporation/condensation between vapour and uncovered heat sink surfaces (e.g., steam condensation on vertical or downward facing containment walls or on metal equipment).

B.3.1.3 Hydrogen Combustion

MELCOR models the combustion of gases in control volumes. The bulk burn models (deflagrations) consider the effects of burning premixed gases without modelling the actual reaction kinetics or tracking the actual flame front propagation. A simple diffusion flame model allows for the burning of hydrogen-rich mixtures upon entry into volumes containing oxygen. Note that the diffusion flame model can be used to model the burning during DCH while leaving the bulk burn parameters at their nominal values.

Deflagrations are ignited if the mole fraction composition in a control volume satisfies a form of LeChatelier's formula. Tests for sufficient H_2 and O_2 are performed, as well as an inerting test for the presence of excessive diluents (H_2O and CO_2). Deflagrations are propagated into adjoining control volumes if additional tests for the H_2 and CO mole fractions in those volumes are satisfied and if the flow path is open to gas flow.

The combustion rate is determined by the flame speed, the volume characteristic dimension, and the combustion completeness. The flame speed and combustion completeness can each be input as constant values, or they may be calculated from user-defined control functions or the default empirical correlations.

For user convenience, print messages are used to warn the user when the detonability criteria are satisfied in a control volume. However, only deflagrations are modelled; detonations are merely flagged.

B.3.1.4 Hydrogen Mitigation Systems

MELCOR has models for most of the containment engineered safety systems that are related to hydrogen mitigation. Their modelling approach is described as follows.

- Recombiners: Employ empirical correlation for hydrogen recombination rate
- Ignition from PARs: Use vendor data to specify ignition threshold
- Igniters: Glow plug type igniters to initiate combustion at lower flammability gas mixtures
- Filtered Venting: Flow pathway and aerosol filter model
- Containment fan or fan coolers: Heat removal and steam condensation in containment fan coolers
- Dousing spray: Sub-cooled spray droplets falls into control volume

B.3.2 Code Validations

Regarding generation of hydrogen, the MELCOR code development team has performed numerous validation assessments using results from integral tests such as Phebus FPT-1 & FPT-4, Phebus-B9, CORA-13, Quench-6, LOFT LP-FP-2, and PBF-SFD 1-4. Overall, MELCOR 1.8.6 & 2.1 provide a reasonable representation of hydrogen generation for these tests. However, it has been noted that MELCOR 1.8.6 underestimates the hydrogen generation burst during reflood that was observed in CORA-13 though recent improvement to modelling of oxidation of submerged surfaces in MELCOR 2.1 has demonstrated significant improvement in this regard. Furthermore, MELCOR 1.8.6 has been validated against the TMI-2 accident where the core is severely damaged and hydrogen generation is complicated by the process of fuel degradation and blockage of steam flow. MELCOR 1.8.6 underestimates the total hydrogen generation, though again, it is believed that this may be due to the hydrogen generation burst from submerged surfaces during the restart of the RCP 2B pump. Assessment of TMI-2 with MELCOR 2.1 is on-going.

Regarding hydrogen distribution, the code developer performed validations against the containment natural circulation tests (e.g., selected tests from HDR and NUPEC experimental programmes). The MELCOR model provides a reasonable representation of the containment response under both natural circulation and forced flow conditions.

Regarding hydrogen combustion, the code developer performed validations against the Nevada Test Site experimental data and the TMI hydrogen burn event. The empirical combustion models were derived from small scale experimental data, e.g., FITS and VGES.

Regarding Hydrogen Mitigation Systems the following have been performed by the code developers:

- Fan cooler: validation against vendor provided fan cooler data. The model calculated heat removal and steam condensation rates for the PWR containment fan coolers (finned tube, cross-flow heat exchanger).
- Spray: the code developer performed validation against steam dousing tests to examine the spray heat transfer. Spray separate effects validation was performed using the JAERI spray tests and large scale integral validation was performed using the CVTR tests.

B.3.3 Strengths, Limitations and Improvements

MELCOR is considered a state-of-the-art code for SA modelling and analysis, and it has reached a reasonably high level of maturity over the years as evidenced from its wide acceptability and its broad range of applications in regulatory decision support. Nevertheless, it is important to recognize the phenomenological uncertainties in MELCOR and their significance to MELCOR results. Moreover, it is important to understand the compounding effect of various uncertainties on the ultimate parameter of interest (e.g. source term) for all practical purposes. Some of the more important uncertainties are briefly discussed in this section.

The in-vessel melt progression modelling in MELCOR starting with the loss of intact core geometry to clad oxidation, in-vessel hydrogen generation, molten core relocation to lower plenum, and subsequent lower head failure are based on experiments which were conducted with the primary objective of gaining an understanding of these phenomena in relation to the observation and experience from plant accidents such as TMI-2. There are uncertainties associated with these phenomena. For example, the clad oxidation model in MELCOR is predicated on certain minimum thickness of pre-oxidized clad layer and certain minimum clad temperature. Any change in the values of these parameters may have an impact on the quantity of in-vessel hydrogen generation; melt temperature, and lower head failure timing.

MELCOR lacks a mechanistic model for evaluating fuel mechanical response to the effects of clad oxidation, material interactions (e.g. eutectic formation), Zircaloy melting, fuel swelling and other processes that occur at very high temperatures. The code uses a simple temperature-based criterion to define the threshold (or alternatively, a time at temperature failure model) beyond which normal ("intact") fuel rod geometry can no longer be maintained, and the core materials at a particular location collapse into particulate debris. The temperature-based criterion attempts to bound uncertainties in phenomenological processes that affect fuel rod integrity.

The rate of movement of radial molten and solid debris to the centre of the core and the time it takes the debris to move to the lower plenum are controlled by the relocation time constant parameter in MELCOR. This parameter is used as a surrogate for the broad uncertainty of debris relocation rate into water in the lower head. This, in turn, affects the potential for debris coolability in the lower head (faster relocation rates decrease coolability; slower rates improve coolability). Debris relocation in MELCOR occurs when the lower core plate in a ring yields. Molten material and particulate debris from the ring immediately moves towards the centre of the core and fall into the lower head.

As in the case of in-vessel melt progression, the ex-vessel phenomenological modelling is based on experiments which were conducted to gain an understanding of melt spreading on the floor, debris quenching in presence of water, and molten core-concrete interaction, among others. These uncertainties are currently captured in MELCOR in a parametric manner.

Model development and code assessments will continue to be performed to improve MELCOR to analyse current reactor designs, and also support the application of the code for new reactor designs. Moreover, in response to the accident at the Fukushima Daiichi nuclear power station in Japan, an accident reconstruction study is being performed as a means of assessing the SA modelling capability of the MELCOR code.

B.3.4 Application Method

MELCOR modelling is general and flexible, making use of a “control volume” approach in describing the transient response of the plant. No specific nodalization is provided, which allows a choice of the degree of detail appropriate to the task at hand. Reactor-specific geometry is imposed only in modelling the reactor core. The code has been modernized (source code upgrade to Fortran95) to provide an efficient code structure for ease of maintenance, resulting in the release of MELCOR version 2.1.

In order to predict the accident progression from thermal-hydraulic conditions inside the reactor coolant system to fission product release and transport to the environs, representative accident sequences are selected. Sources of phenomenological uncertainties should be identified and addressed in code sensitivity analysis. Thus the accuracy of the MELCOR predictions is commensurate with the uncertainties associated with the analysis.

In MELCOR the only specific step of the analyses related to H2 is that whenever H2 deflagration is predicted to occur or it is close to occurring, a sensitivity study should be carried out to assess the results sensitivity to nodalization.

B.4 SPECTRA[5]

Code developers	NRG
Users	NRG, South Africa, China, etc.
General capabilities	System code for thermal hydraulic analyses of accidents.
Applications	thermal-hydraulic safety analyses for LWR, HTR, LMFR, GCFR,
Reference	[B.2]

B.4.1 Code Capabilities

The code structure of SPECTRA is demonstrated in Figure B.4.1-1. The SPECTRA code is an accident analysis code developed at NRG since 1996 with original purpose to analyse reactors with passive safety cooling systems. SPECTRA (Sophisticated Plant Evaluation Code for Thermal-hydraulic Response Assessment) is a computer program designed for thermal-hydraulic analyses of nuclear or conventional power plants. The modelling approach is based on the Control Volume (CV) concept. A model of a certain physical system consists of Control Volumes connected by junctions (JN). Within a Control Volume, four fluid components may exist: two continuous components (atmosphere and pool), and two dispersed

[5]The description for SPECTRA is provided by NRG, the Netherlands.

components (droplets and bubbles). The gas components (atmosphere and bubbles) are assumed to be mixtures of non-condensable gases and steam. Non-equilibrium is assumed between each of the four fluid components, with the inter-component heat and mass transfer modelled in a mechanistic way.

B.4.1.1 Hydrogen Generation

SPECTRA is capable of predicting the hydrogen generation due to oxidation reactions. Reactions of Zircaloy (Zr) and steel (Fe) by steam are built-in the code. Other reactions are easily defined through a user-defined model.

B.4.1.2 Hydrogen Distribution

SPECTRA uses a one-dimensional model for fluid flows, including steam and non-condensable gases (built-in gases: H_2, He, H_2O, N_2, O_2, CO_2) transport. The general modelling scheme allows computing gas transport in a similar way as the other system codes. Special modelling features are available to account for stratification, tested in analyses of hydrogen distribution in HDR E11.2 (ISP-29) and NUPEC M-7-1 (ISP-35) experiments.

B.4.1.3 Hydrogen Combustion

The burn model distinguishes three different modes:

- slow deflagration
- fast turbulent deflagration
- detonation

Generally the hydrogen burn model, when activated, calculates the flammability of the gas mixture in a Control Volume, checks ignition criteria and, upon ignition, calculates H_2 and O_2 consumption, as well as generation of steam and heat of reaction.

The flammability limits are functions of gas composition as well as gas temperature. A method of constructing flammability limits proposed by Plys, Astleford and Epstein. The fast turbulent deflagration limits are constructed in a similar way as the flammability limits. Alternatively the "σ criterion" is available, which must be larger than a certain, temperature dependent critical value. The detonability limits are constructed in a similar way as the flammability limits. The values are based on data from Ciccarelli and Stamps. Alternatively the "λ criterion" is available, which must be larger than a certain value.

Ignition limits are calculated as a function of temperature, pressure, gas composition, and the characteristic dimension. When the ignition criteria are satisfied, then hydrogen begins to burn. The reaction rate depends on the flame speed; the model includes laminar and turbulent deflagrations. For laminar flow the correlation of Liu and McFarlane [B.3] is used. For turbulent deflagrations Klimov correlation [B.4] is used. In case of fast turbulent deflagration, SPECTRA uses a correlation developed based on data from [B.5]. In case of detonations, the code uses a correlation developed based on data from [B.6].

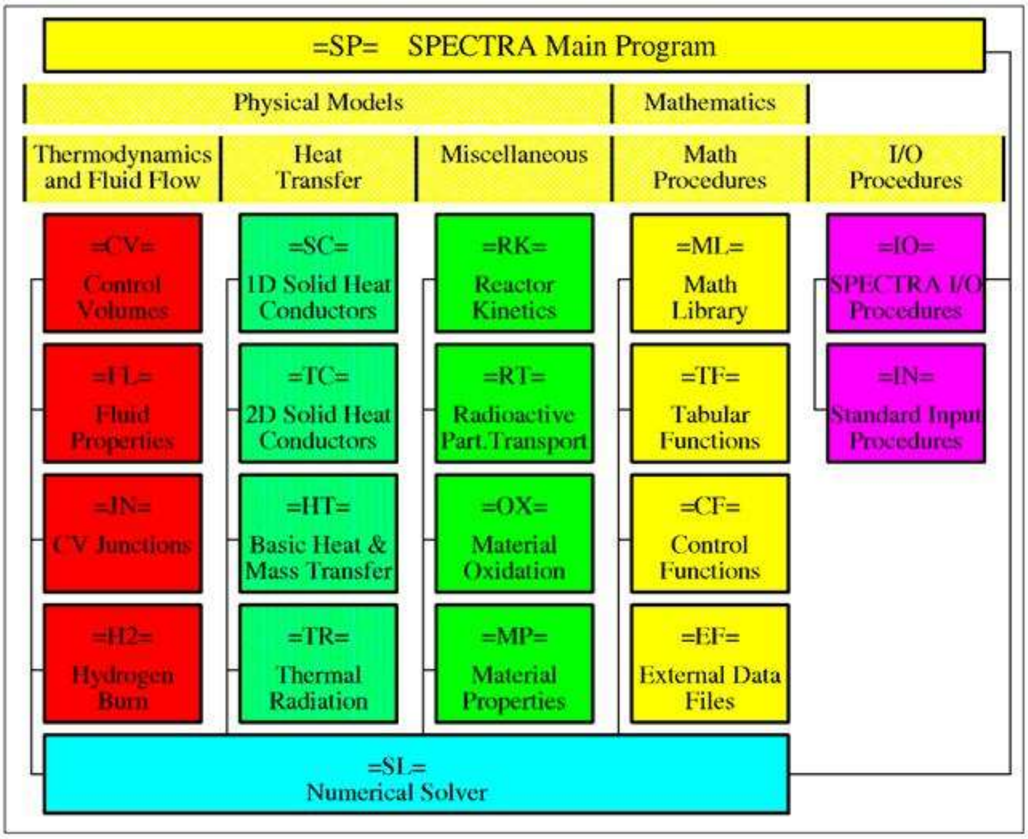

Figure B.4.1-1 Structure of SPECTRA

B.4.1.4 Hydrogen Mitigation Systems

General models of spray systems (non-equilibrium model, subcooled droplets), fan coolers, aerosol filters and fission product vapor filters, pool scrubbing. Hydrogen recombiners can be modelled with reaction kinetics computed from a correlation and defined via Control/Tabular Functions. This approach to model recombiners was tested by code-to-code comparisons within SARNET, Generic Containment project, where NRG participated with SPECTRA and MELCOR codes [4.5]. The same concept for recombiner modelling was used in both codes and very similar results were obtained. In the near future a mechanistic model of recombiners will be developed and tested.

B.4.2 Code Validations

Regarding hydrogen generation, verification and validation calculations were performed based on oxidation experiments of Zr oxidation in steam (Urbanic Heidrich), Zr oxidation in air (Natesan, Soppet and Benjamin, McCloskey, Powers, Dupree), steel oxidation in steam (White), steel oxidation in air, (Nanni, Buscaglia, Battilana), graphite oxidation in air (Roes), graphite oxidation in steam (K.-J. Loenißen).

Regarding hydrogen distribution, validation calculations were performed using the hydrogen distribution tests, HDR E11.4, E11.2 (ISP-29) and NUPEC M-4-3, M-7-1 (ISP-35), code-to-code comparisons within SARNET Generic Containment.

Regarding hydrogen combustion, verification and validation calculations were performed using experimental data and code-to-code comparisons. The experiments include on THAI experiments and hydrogen deflagration tests AECL's Combustion Test Facility and Large Scale Vented Test Facility.

Regarding hydrogen mitigation systems, verification and validation calculations were performed using experimental data and code-to-code comparisons. For hydrogen recombiners, spray systems, fan coolers code-to-code using MELCOR. For spray, experiments done within SARNET.

B.4.3 Strengths, Limitations and Improvements

SPECTRA is a fully integrated code, with very good input checking and diagnostics, which allows to eliminate most input errors at the early stage of model preparation. Currently there is no pre-processor; the input decks are prepared as text files. A very flexible and user friendly post-processor, Visor, is used.

The code is designed for thermal-hydraulic analyses and not SA analyses, so it is not possible to analyse core degradation, vessel failure and core concrete interactions.

B.4.4 Application Method

SPECTRA's modelling is general and flexible, making use of basic blocks, such as Control Volumes, Junctions, Solid Heat Conductors. No specific nodalization is provided, which allows a choice of the degree of detail appropriate to the task at hand. The models are general and allow applying the code to a variety of different designs, nuclear as well as non-nuclear. In the past the code has been used for:

- Light Water Reactors (thermal-hydraulic safety analyses)
- HTR/PBMR (thermal-hydraulic safety analyses, dust and fission product transport analyses)
- GCFR (thermal-hydraulic safety analyses - optimization for passive safety)
- Liquid metal reactors (ESFR, LEADER) - design transient analyses
- Chemical reactors (design analyses of reactor cooling systems)

B.5 COCOSYS[6]

Code developers	GRS (Germany)
Users	Germany, Finland, Russia, Bulgaria, Lithuania, China, etc.
General capabilities	COCOSYS provides a LP code system on the basis of mechanistic models for the comprehensive simulation of all relevant processes and plant states during design basis and SAs in the containments of LWRs.
Applications	Identification of possible deficits in plant safety; quantification of the safety reserves of the entire system; assessment of mitigation measures of SAM concepts; safety evaluation of new plant concepts.
Reference	http://www.grs.de/en/content/cocosys

[6]The description for COCOSYS is provided by GRS, Germany.

B.5.1 Code Capabilities

The code structure of COCOSYS is demonstrated in Figure B.5.1-1. The inner part of the code system consists of three main modules:

- Thermal-hydraulic main module THY for the simulation of the thermodynamic behaviour and transient processes like hydrogen deflagration, simulation of safety systems
- Aerosol fission product main module AFP for the simulation of the fission product behaviour, decay heat release and chemical reactions
- Core concrete interaction main module MEDICIS for the simulation of the core melt behaviour, concrete erosion, releases from core melt and chemical processes inside the core melt.

Furthermore it is possible to connect detailed models (e.g., LAVA for melt relocation) to the COCOSYS system. Using this concept it is possible to calculate local effects or specific processes (like melt relocation).

To simulate the integral behaviour of an NPP (e.g., the reactor and its cooling circuits together with the enclosing containment), the COCOSYS code can be coupled with the GRS program ATHLET-CD. Alternatively source data are provided by tables.

B.5.1.1 Hydrogen Generation

Hydrogen generation is not relevant for COCOSYS except for the non-condensable sources from the ex-vessel phase (MCCI). Results of in-vessel phase are provided by tables or due to coupling with ATHLET- CD [4.3].

Masses of some elements (Al, C, Ca, Cr, Fe, H, Ni, O, Si, U, Zr, etc.) in different melt layers (oxide, metal) configuring the corium pool in the cavity are tracked. Species composition is calculated for each layer during the iteration for layer temperature, assuming stoichiometric equilibrium and a hierarchy of oxidation reactions. Uncertainty exists in evolution of pool configuration (e.g. mixed or stratified) impacts on hydrogen release, since the gas flow enters (and interacts with) the melt layer adjacent to the eroded concrete.

B.5.1.2 Hydrogen Distribution

The thermal-hydraulic main module THY is designed for the simulation of the thermodynamic behaviour and transient processes like hydrogen deflagration, simulation of safety systems and other within the containment or other buildings of an NPP. The model for gas flow is one-dimensional, major typical non-condensable gases (H_2, He, N_2, O_2, CO, CO_2) are considered. There is a separate treatment of condensate/water flow.

B.5.1.3 Hydrogen Combustion

In COCOSYS combustion may occur according to two different criteria: user-input or crossover of flammability limits in the Shapiro diagram. For the crossover of flammability limits, four different flammability limits are defined on the ternary Shapiro diagram hydrogen-air–steam at atmospheric pressure and room temperature.

Deflagration of H_2 or H_2-CO combustible mixtures is modelled in COCOSYS by the FRONT model (similar model as in ASTEC CPA), but it is a simple model and can be used for first estimations. The H_2 combustion has been validated, but the CO-combustion is not validated due to lack of experiments.

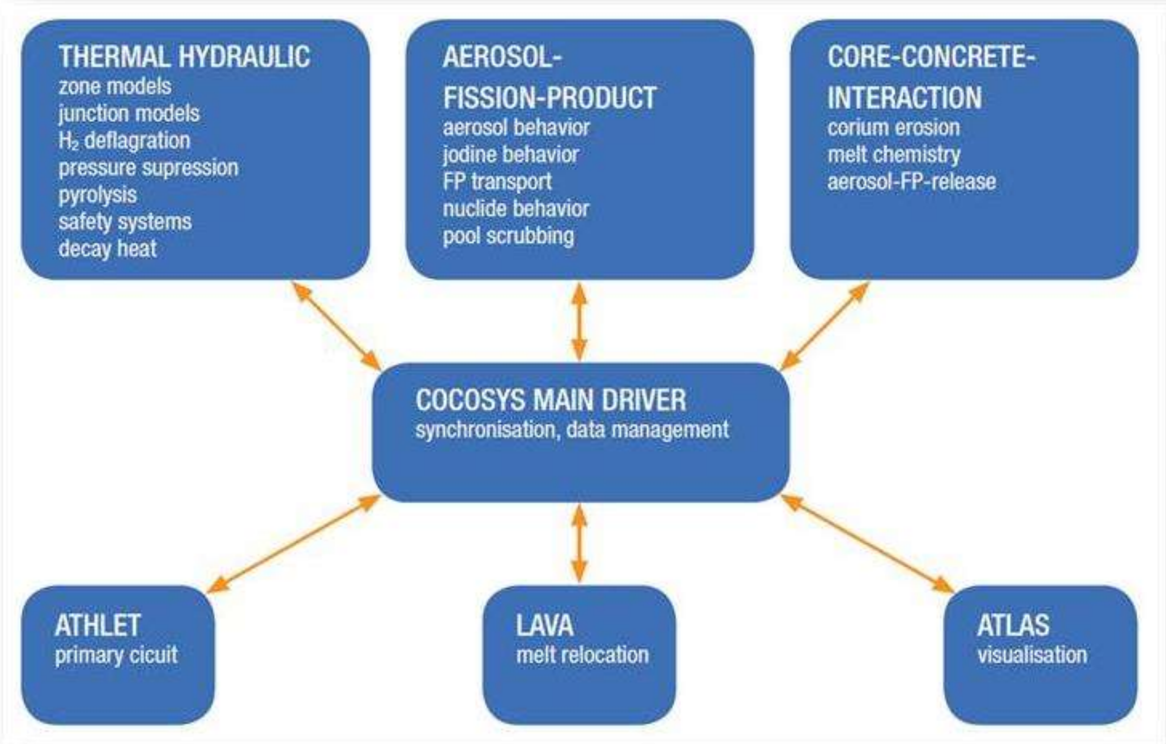

Figure B.5.1-1 Structure of COCOSYS

B.5.1.4 Hydrogen Mitigation Systems

COCOYSYS has models for the following containment engineering systems that are related to hydrogen mitigation:

- Spray model: use mono-disperse droplet size
- PAR 1D model:

  There is a detailed PAR model for all AREVA type PARs and one for AECL PARs that was used in OECD THAI project. There is no detailed modelling and validation for AECL PARs (missing geometrical data). A model for NIS PARs is under development.
- PAR parametric model:

  There is a fast running PAR model based on 1D model for AREVA PARs, but the model does not give any information on PAR temperature or outlet concentration, on velocity, or mass flow rate.
- Pressure suppression systems (DRASYS model)

  A model developed for VVER-440 bubble condenser, applicable for BWR and RBMK condensation pool.
- Filtered venting:

  A model using user-defined decontamination factors is implemented for filtered venting; detailed filter models are currently under development.

B.5.2 Code Validations

Regarding hydrogen combustion, the COCOSYS FRONT model validation and the work was performed together with ASTEC CPA FRONT model validation.

Regarding PAR modelling, the code was validated in late 90s to model AREVA PARs based on BMC experiments; recently the PAR model was renewed, re-evaluated against some of the PAR tests from the OECD THAI 1 project and extended for AECL PARs. Only one single experiment was used to test the AECL PAR model. More effort is ongoing.

Regarding gas distribution and stratification, many experiments were used to validate the THY model in COCOSYS and to gain experience i.e. in modelling gas stratifications and possible means for dissolution [4.1], [1.44], including:

- Gas and temperature distribution and stratification: **BMC** (Rx4); **HDR** (E11.2, E11.4), **THAI** (TH1, TH2, TH7, TH10, TH13, HM-2), **TOSQAN (**ISP47), **MISTRA** (ISP47, MICOCO), **PANDA-SETH (**T9, T9bis, T17)
- Heat transfer: **BMC -** F2, **VANAM-M3**, **THAI** (TH9)
- Short term effects: **BMC** (C13, D3, D6, D7), **HDR** (T31.3, T31.5)
- Plume/Jet: **BMC** (Jx2), **THAI** (TH7, TH10, TH13, HM-2)
- Recombination: **BMC**, **OECD THAI**

Regarding molten corium concrete interaction (MCCI), the work was performed together with ASTEC to validate the MCCI model MEDICIS taking into account concrete erosion (BETA, ANL-ACE, ANL-MACE, OECD-MCCI, COMETA tests), melt relocation (KATS, KATS-ECOKATS, COMAS, VULCANO tests) (core catcher) and DCH (DISCO: FH01, FH02, KH01, KH02 tests).

B.5.3 Strengths, Limitations and Improvements

At GRS COCOSYS is under development since 1997. The codes being the basis of COCOSYS are the former GRS codes RALOC and FIPLOC used before separately.

The development of the code is sponsored by the German ministry BMWi. The code system is based on mechanistic models for all relevant processes, phenomena and plant states during SAs in the containment of LWRs. It is used as know-how transfer & conservation of knowledge. COCOSYS is applied in 13 countries by about 40 national and international organizations (end of 2011). The interest in COCOSYS is still growing,

In the past the weakness of COCOSYS was mainly related to hydrogen combustion processes. The old combustion model DÉCOR was very seldom used and not ready for plant applications. This has changed recently with the new FRONT model being similar to the one developed for ASTEC CPA. Still this model needs large user know-how and is applicable only for combustions at lower hydrogen concentrations. Phenomena like FA, DDT or detonations are not treated.

Significant improvement has been made along the R&D progress on relevant SA phenomena. Especially the model of AREVA type of PAR was re-evaluated in the last years using the OECD THAI 1 results. As well the model MEDICIS used for MCCI calculation as in ASTEC CPA has made significant progress.

Further improvements are needed and are partially under way i.e. for passive features of Gen. III reactors (external cooling of RPV) or for other types of PARs (AECL, NIS) or for the modelling of different filter types in venting systems. As well model development / validation is needed for the current OECD THAI 2 experiments on combustion with spray operation.

B.5.4 Application Method

GRS's main strategic objectives are related to the code development and validation of the mechanistic codes COCOSYS (and ATHLET/ATHLET-CD) and the integral code ASTEC (together with IRSN) on one side and its application (e.g., for SAM concept development). Objectives in the past have been:

- Application of COCOSYS for detailed containment analysis of hydrogen distribution and recombination in SAs for PAR concept development of PWR and BWR-72 (about 1997 – 2001).
- Application of COCOSYS for detailed hydrogen recombination and combustion (e.g. initiated by PARs) analyses within PWR KONVOI containment – re-analysis of the earlier PAR concept work (2009 – 2001).
- Application of COCOSYS coupled to ATHLET/ATHLET-CD for detailed NPP safety analysis for the Fukushima accident (since 2012).

This work is mainly done through sponsored projects by German ministries BMU (PAR concept, Fukushima assessment) and BMWi (Fukushima OECD benchmark, code development and validation).

Besides this, GRS is working in many areas of nuclear reactor physics, thermal-hydraulics of nuclear power plants (reactor circuit and containment) under DBA and SA conditions. GRS also supports actively the experimental OECD projects like THAI, BIP, STEM.

Previously developed input decks for the German NPPs are always rechecked for a new application. While updating the input deck, the whole files where checked for coherence. A repetition of old analysis with a new code version was done for the PAR concept for German PWR. The older analysis results have been approved by the newer results and additional new insights have been provided.

Special quality assurance procedures exist for code development and validation.

B.6 TONUS[7]

Code developers	IRSN (France) and CEA (France)
Users	France
General capabilities	To model gas distribution and mixing processes, recombiners and combustion in a designed CFD tool recombiners
Applications	Best-estimate tool for predicting transport, mixing, and combustion of hydrogen and other gases in nuclear reactor containments and other facility buildings
Reference	[B.8]

[7]The description for TONUS is provided by IRSN, France.

B.6.1 Code Capabilities

TONUS is a containment thermal-hydraulics computer program for multi-component multi-phase flow. It solves the conservation equations for mass, momentum, and energy for multi-component, multiphase flow. The region of interest is modelled using a CFD mesh. Heat transfer between the fluid domain and one-or two-dimensional structures by convection/ condensation is calculated using a number of available semi-empirical modelling. Turbulence is available through a simplified mixing length model. Spray systems and recombiners are also modelled. For the combustion in TONUS-CFD, different types of models have been developed to cover slow deflagration regimes, accelerated flames and detonation and implemented in a fully compressible flow solver able to simulate shock wave propagation.

Since the objective of the TONUS code is to be applied to real plant applications such as EPR studies, numerical considerations such as mesh size and CPU run-time has constrained the choice of physical models and numerical algorithms.

B.6.1.1 Hydrogen Generation

Hydrogen generation is not modelled in TONUS. Hydrogen generation during the in-vessel and ex-vessel phase are provided by tables.

B.6.1.2 Hydrogen Distribution

Hydrogen distribution is very well addressed by TONUS being a CFD code and developed for it.

- 3D model for gas flow
- Mixing length model for the turbulence, a k-□ model available in a beta version
- major typical non-condensable gases: H_2, He, N_2, O_2, CO, CO_2
- Separate treatment for condensation on walls

B.6.1.3 Hydrogen Combustion

For the combustion in TONUS, different types of models have been developed to cover slow deflagration regimes, accelerated flames and detonation and implemented in a fully compressible flow solver able to simulate shock wave propagation.

The combustion model provides the reaction rates for the premixed gas chemical reaction. The rates appear in the source terms of the species transport.

B.6.1.4 Hydrogen Mitigation Systems

Even though the spray systems are not a mitigation system for hydrogen flammability, their operation will affect the H_2 distribution and this effect is taken into account in TONUS, using an Eulerian-Eulerian modelling.

For the recombiner we use a lumped-parameter approach, while the containment fluid volume as well as the inlet and the outlet of the recombiner are meshed. As a consequence, the recombiner model has to be supplied with space-averaged.

B.6.2 Code Validations

The strategy for the validation of the code followed a progressive approach in terms of physical phenomena, from separate effect tests to coupled effect tests. Much of the data used for the validation of the distribution models was produced in the TOSQAN and MISTRA containment facilities, which were specially developed to produce high quality detailed data suitable for CFD code validation with well-controlled boundary conditions and a dense mapping of sensors in the volumes. Test data from the ISP-47 is an example of such data. Besides TOSQAN and MISTRA data, tests from AECL's large-scale gas mixing facility, PHEBUS, THAI and PANDA facilities were used for validation purposes. Validation of mitigation models (spray, PAR) has also been performed. For the combustion part of the code, use was made of large-scale hydrogen combustion tests which are representative of containment geometries such as Battelle, HDR and RUT tests which cover all the combustion regimes from slow deflagrations to detonations. More detailed data produced in the IRSN-sponsored ENACCEF facility is also being used to investigate accelerated flame regimes.

B.6.3 Strengths, Limitations and Improvements

The strength of the code is the larger time step that can be used for distribution studies.

The main weakness is that the code is limited in the mesh size, and no parallel version is available. On the modelling aspects, the turbulence models are limited. The spray is modelled as a single injection and the use of an Eulerian model induces to enhance the number of solved equations for each droplet size considered.

B.6.4 Application Method

IRSN has used the TONUS code to support the hydrogen risk assessment for the EPR plants and to investigate the impact of the two-room concept on hydrogen distribution in the EPR containment. The application examples can be found in References [B.9] to [B.11].

B.7 GOTHIC[8]

Code developers	Numerical Applications Inc., USA
Users	Canada, Spain, Switzerland, etc.
General capabilities	Versatile software package for transient thermal hydraulic analysis of multiphase systems in complex geometries.
Applications	To solve a wide variety of fluid flow problems that may include single or multiple phases, fluid to solid heat transfer, hydrogen combustion, and numerous engineered components. A typical use is to model nuclear reactor containment buildings.
Reference	http://www.numerical.com

[8]The description for GOTHIC is provided by AECL, Canada.

B.7.1 Code Capabilities

GOTHIC is a general-purpose thermal-hydraulics computer program for multi-component multi-phase flow [B.12]. It solves the conservation equations for mass, momentum, and energy for multi-component, multiphase flow. The region of interest can be modelled with a combination of lumped and multidimensional volumes. Heat transfer between the fluid domain and one-or two-dimensional structures by convection/ condensation/evaporation is calculated using a number of available empirical correlations. Molecular and turbulent diffusion models are available for multidimensional analysis. Many specialized features are included to model and control equipment typically found in ventilation and hydraulic systems (pumps, fans, valves, heat exchangers, etc.).

B.7.1.1 Hydrogen Generation

GOTHIC has no model for hydrogen generation. The hydrogen source term calculated by other codes can applied as boundary conditions in GOTHIC for hydrogen analysis.

B.7.1.2 Hydrogen Distribution

Hydrogen distribution is well addressed by GOTHIC being a hybrid code and developed for it.

- It is capable to model the heat transfer, gas mixing and other thermal hydraulic behaviour of multi-component and multiphase flow; vapour (steam and a number of non-condensable gases), drops and continuous liquid (film, pool, etc.), and three secondary phases (mist, ice, and solid particles or another liquid or gas in the liquid phase) with simplifying assumptions.
- Heat transfer between a surface and the fluid: convection to the vapour/liquid phase, condensation on the dry portion of the surface and boiling on the wet portion of the surface with heat and mass transfer at the interface between the liquid film and the vapour
- Convection flow driven by buoyancy force (density or pressure difference) or active mixing measures (e.g. fans, air coolers).

B.7.1.3 Hydrogen Combustion

In GOTHIC, there is no model for detonation and only hydrogen deflagration is modelled. Deflagration can be initiated under the following conditions:

- when the gas temperature exceeds a user-defined auto-ignition temperature regardless of the gas mixture compositions;
- when the mixture exceeds user defined flammability limits (minimum H_2 and O_2 concentrations and maximum H_2O concentration) and an igniter component is active;

Burning will stop when the effective gas temperature of a computation cell below a user defined minimum gas temperature or the gas compositions below user defined limits.

Three types of burn modelled are available in GOTHIC:

- Discrete burn model (applicable to LP model only)

  To burn a specified fraction of existing H_2 within a volume when the mixture is within prescribed the limits and an igniter is active.
- Continuous burn model (applicable to LP model only)

To continuously burn hydrogen flowing into a volume from a junction when the mixture is within the prescribed limits.

- Mechanistic burn model (applicable to subdivided volumes only)

  Combustion of hydrogen is highly dependent on local H_2 and O_2 concentrations and turbulence levels. The reaction rate is determined from maximum of the laminar and turbulent combustion rates. The laminar flame speed is independent on steam concentration and gas temperature. A user defined multiplier is needed to account for turbulence effect.

B.7.1.4 Hydrogen Mitigation Systems

The following containment engineering safety systems that are related to hydrogen mitigation are modelled in GOTHIC:

- Hydrogen recombiners to model both forced and natural convection type of recombiners

  To be located on a flow path and convert a user defined fraction of hydrogen flowing through the flow path into steam.
- Igniter (glow plug type) to initiate a deflagration by providing a local high effective temperature
- Fan coolers

  The secondary (cold) side is assumed to be water and the primary (hot) side is determined by the vapour conditions at the upstream end of flow path. It can't be configured in a double heat exchange arrangement.
- Coolers (vapour or liquid phase) to act as a condenser if a volumetric flow is specified.
- Heat exchangers for common configurations of water to water heat exchangers. It allows users to model both single and double heat exchangers.
- Spray nozzles to be located on any flow path to convert some or all of the continuous liquid into drops. The drop diameter is specified by the user.

B.7.2 Code Validations

B.7.2.1 Hydrogen Distribution

For the phenomena related to hydrogen distribution, the code developer performed validations using the containment thermal-hydraulic and hydrogen behaviour experiments conducted between 1970s and 1980s, such as NUPEC, CSTF, Battelle Frankfurt, and HDR tests [4.2]. These tests examined gas entrainment caused by a high velocity jet as well as natural/forced convection mixing.

In Canada, a large number of validations have been performed against tests conducted in recent 10 years at AECL's two large scale gas mixing facilities as well as international facilities (e.g., ISP-47, OECD-THAI HM2, SETH-2). In general, the GOTHIC 3D model can accurately simulate buoyancy induced flow behaviour and properly capture stratified helium layer, particularly, helium mixing impeded by pre-stratified steam layer, enhanced by co-injection of helium and steam, and significantly affected by vapour condensation heat transfer. Larger uncertainties were observed when the helium injected at high elevations and GOTHIC tended to under-predict the stratification and over-predicted the mixing.

In Switzerland, an extensive validation activity has been performed by PSI based on PANDA tests within the OECD SETH and SETH-2 projects ([B.13] through [B.20]. The validation addressed stratification build-up and stratification erosion/break-up by means of various mass and heat sinks/sources.

Very good prediction of stratification build-up under various conditions was obtained, including vertical and horizontal jets and free and near-wall plumes with or without condensation. The only difficulty identified was the ability to predict steam distribution for cases where the steam concentration was dominated by re-evaporation of liquid films. Stratification break-up controlled by a spray and a heat source is well predicted. Large uncertainty existed in predicting gas distribution evolution controlled by the operation of a cooler. For the prediction of stratification erosion produced by coherent flow structures (jets and narrow plumes) very fine mesh are required, which cannot be performed with GOTHIC.

B.7.2.2 Hydrogen Combustion

For the phenomena related to hydrogen combustion, the code developer performed validations against the EPRI and Sandia FLAME tests. GOTHIC showed reasonable good agreement in predicting the effect of hydrogen concentration and transverse venting on flame speed, but showed large discrepancy in predicting the FA when obstacles were included.

In Canada, several validations were performed against the hydrogen deflagration tests performed at AECL's Combustion Test Facility and Large Scale Vented Test Facility (LSVCTF) as well as the THAI facility. The GOTHIC model quantitatively predicted the combustion overpressure well, but a user defined burn enhancement factor must be applied to account for the effects of turbulence on the burning rate. Larger uncertainties were observed for stratified mixtures and tests with higher steam concentrations as well as tests with obstacles.

B.7.2.3 Hydrogen Mitigation Systems

The validations performed for the mitigation models in GOTHIC are described as follows:

- PAR model (mechanistic approach)

  The GOTHIC code developer performed qualification of the PAR model using tests performed at AECL's LSVCTF. A set of key parameters were developed to represent an AECL PAR unit. AECL performed benchmark simulations against the OECD THAI HR tests using the recommended parameters. The prediction for PAR recombination rate, flow velocity through the PAR and hydrogen concentration agreed well with the measurement.

- Air cooler

  The code developer performed validation against the AAF TMI fan cooler tests. The GOTHIC model was run to a steady state and the differences between the measured and GOTHIC calculations were within measurement uncertainty. AECL performed validation against CANDU air cooler heat transfer tests. The heat removal capacity and outlet water temperature agreed well with the experimental measurements for the operating conditions with high steam concentrations, but large discrepancy for the low steam tests.

- Spray

  AECL performed validation against the CANDU dousing spray tests. Results of both the LP and 3D models showed good agreement with experimentally derived average temperatures and calculated thermal utilization

B.7.3 Strengths, Limitations and Improvements

GOTHIC can be used for design, safety and licensing, and operating analysis of nuclear containments and other confinements. It can be used to predict the thermal-hydraulic response of a full-scale containment building modelled as a collection of inter-connected volumes. Since GOTHIC 6.1version, there have been a number of major enhancements made to the GOTHIC code. These include improved gas mixing behaviour, new models such as near wall droplet behaviour, drop entrainment, and enhancements of component models, such as local air coolers and PARs.

AECL has performed extensive validation covering containment phenomena for containment thermal hydraulic and hydrogen mixing behaviour. Phenomena of liquid re-entrainment, standing flame and deflagration-to-detonation transition remain outside GOTHIC-IST modelling capabilities. GOTHIC is not recommended for fast deflagrations with obstacle-induced FA or hydrogen burn simulations where accurate flame speed results are required.

No major limitations are identified for using GOTHIC in CANDU containment analysis to address safety concerns and licensing issues. Specific limitations regarding code errors, applicability range, sensitivities, user options and/or uncertainties are discussed in the validation manual [B.21] for each individual phenomenon, and user guidelines are given where appropriate.

B.7.4 Application Method

GOTHIC is developed and maintained under NAI's QA programme conforming to 10CFR50, Appendix B requirements. The Canadian nuclear industry has adopted GOTHIC as the Industry Standard Toolset (IST) for containment thermal-hydraulics analysis. The primary objectives of these analyses are to estimate:

- Global pressure history to evaluate containment integrity, and estimate timing and initiation of pressure-dependent signals;
- Pressure differentials between adjacent compartments to assess internal structural integrity;
- Release path flow/energy rates from containment to the external atmosphere via various release paths for the purpose of calculating radionuclide releases from containment (subsequently used for dose analysis);
- Local temperature and humidity conditions for equipment qualification purposes;
- Steam/non-condensable gas distribution inside containment (especially hydrogen) to evaluate the potential for a hydrogen burn.

Objectives 1-3 can be obtained with sufficient accuracy by a lumped parameter code if a hydrogen burn can be precluded, since these codes can estimate pressures, temperatures and pressure differentials resulting from the blow-down reasonably well in the short term. In the long term, pressure differences within containment are small, and hence the release rates are usually estimated accurately by using a single uniform pressure. If a hydrogen burn is possible, objectives 1, 2 and 5 require distributed parameter capability. Objective 4 will also require distributed parameter capability, since temperature and relative humidity do not homogenize as quickly as pressure. In some applications, a hybrid model, consisting of both lumped and distributed parameter nodes, is desirable, for example, when assessing detailed conditions in a sensitive room that is connected to a homogenous compartment. To model hydrogen behaviour in more detail (i.e., local effect), additional models are also required (e.g., hydrogen release from a break, gas mixing and transport, hydrogen burning, hydrogen recombination by PARs etc.).

In support CANDU reactor safety analysis and licensing submissions, extensive validation exercise has been performed for GOTHIC by AECL and the CANDU Owner’s Group (COG) following the CSA N286.7-99 standards. The objective is to qualify the code for calculating containment thermal-hydraulic and hydrogen behaviour following a range of accidents.

B.8 GASFLOW[9]

Code developers	KIT, Germany
Users	Germany, China, Korea, Hungary, etc.
General capabilities	The GASFLOW computational fluid dynamics code can model the gas distribution and mixing processes in geometrically complex facilities with multiple compartments and internal structures in a multi-block computational domain connected by one-dimensional flow paths. It can simulate the effects of two-phase dynamics with the homogeneous equilibrium model (HEM), two- phase heat transfer to walls and internal structures, chemical kinetics, catalytic recombiners, and fluid turbulence.
Applications	Best-estimate tool for predicting transport, mixing, and combustion of hydrogen and other gases in nuclear reactor containments and other facility buildings
Reference	http://hycodes.net/downloads/gasflow http://hycodes.net/gasflow/documentation

B.8.1 Code Capabilities

GASFLOW is a finite-volume code based on proven computational fluid dynamics methodology that solves the compressible Navier-Stokes equations for three- dimensional volumes in Cartesian or cylindrical coordinates. Wall shear stress models are provided for bulk laminar and turbulent flow. GASFLOW has transport equations for multiple gas species and one for internal energy. Terms for turbulent diffusion of different species are included in the mass and internal energy equations.

GASFLOW can model geometrically complex containment systems with multiple compartments and internal structures within independent multiblock or multidomained computational regions. These independent regions may be connected on external boundaries by one-dimensional ductwork or piping systems. GASFLOW can calculate gas dynamic behaviour in low-speed, buoyancy-driven flows, as well as sonic flows or diffusion-dominated flows, and during chemically reacting flows. The code can model two-phase effects of condensation and/or vaporization in the fluid mixture regions within the assumptions of the homogeneous equilibrium model; two-phase heat transfer to and from walls and internal structures by convection and mass diffusion; chemical kinetics of hydrogen combustion with a generalized igniter model; effects of catalytic recombination; fluid turbulence; and transport, deposition, and entrainment of discrete particles. Heat conduction within walls and structures is one dimensional.

B.8.1.1 Hydrogen Generation

Hydrogen generation is not modelled in GASFLOW. Hydrogen generation during the in-vessel and ex-vessel phase are provided by tables.

B.8.1.2 Hydrogen Distribution

Hydrogen distribution is very well addressed by GASFLOW being a CFD code and developed for it.

[9]The description for GASFLOW is provided by KIT, Germany.

- 3D model for gas flow
- major typical non-condensable gases: H_2, He, N_2, O_2, CO, CO_2
- Separate treatment of condensate/water flow

B.8.1.3 Hydrogen Combustion

In GASFLOW combustion may occur according to two different criteria: user-input or crossover of flammability limits in the Shapiro diagram. For the crossover of flammability limits, four different flammability limits are defined on the ternary Shapiro diagram hydrogen-air–steam at atmospheric pressure and room temperature.

Chemical energy of combustion involving hydrogen provides a source of energy within the gaseous regions. A one-step global chemical kinetics model based on a modified Arrhenius law accounts for local hydrogen and oxygen concentrations.

Hydrogen is ignited using a generalized igniter model that represents both spark- and glow-plug- type designs. A catalytic hydrogen combination with oxygen is modelled using data from both the NIS and Siemens recombiner box designs.

B.8.1.4 Hydrogen Mitigation Systems

Even though the spray systems are not a mitigation system for hydrogen flammability, their operation will affect the H_2 distribution and this effect is taken into account by GASFLOW.

B.8.2 Code Validations

Each version of GASFLOW is tested with a Standard Test Matrix of more than 120 problems in four categories:

(1) feature tests for the computer science aspects of the code;

(2) functional tests for code algorithms, equations, logic paths, and decision points;

(3) comparisons with analytical solutions; and

(4) comparisons with data.

During the development of GASFLOW, many experiments were modelled and analysed. All 19 analytical solutions and 6 experiments are used to validate the code:

(1) the Bureau of Mines Spherical Test Chamber,

(2) the Sandia FLAME Experiment,

(3) Battelle Model Containment (BMC) Test GX6,

(4) Battelle Model Containment Test HYJET JX7,

(5) Heißdampfreaktor (HDR) Test T31.5, and

(6) Phebus Tests 4A, 4B, 6A, and 10A.

All of the problems in the Standard Test Matrix and in the initial set of assessments were executed successfully by GASFLOW without modification, and the results are in agreement with the analytical solution or the test data. New solutions from thermodynamic benchmarks have also been successfully evaluated with GASFLOW, e.g., SARNET and SARNET-II and so forth.

B.8.3 Strengths, Limitations and Improvements

The strongest point of GASFLOW code is that the models were fully verified and validated against theoretical solutions or experimental data, especially those nuclear containment models. The implicit algorithm in the code makes the solver able to advance in a relatively large time step, thus increase the computational efficiency especially in cases of nuclear containment simulations in a large scale with a significantly long term.

The weakest point of the code is that it is not parallelized yet. When a refined mesh is adapted to a large complex geometrical volume, the serial calculation can take significantly long time.

There are several aspects in the code which could be improved:

(1) parallelizing the code;

(2) the spray model: there is no drift velocity between the liquid droplets and gas phase;

(3) aerosol model: it is only one-way coupling between particle dynamics and gas dynamics. Namely, only gas motion can mobilize the particle, but not vice versa.

B.8.4 Application Method

GASFLOW has been used to calculate the distribution and control of hydrogen and noxious gases in complex nuclear containment and confinement buildings and in non-nuclear facilities. It has been applied to situations involving transport and distribution of flammable gas mixtures. It has been used to study gas behaviour in complex containment systems with low-speed buoyancy-driven flows, diffusion-dominated flows, and during deflagrations. The effects of controlling such mixtures by safety systems can be analysed.

B.9 CFX[10]

Code developers	ANSYS (US / Germany)
Users	Germany, the Netherlands, Spain[11], etc.
General capabilities	Comprehensive multi-purpose CFD toolbox
Applications	A broad variety of phenomena
Reference	http://www.ansys.com

B.9.1 Code Capabilities

CFX [B.22] is a commercial multi-purpose CFD package, based on the finite volume method, and is used in many industrial areas. CFX uses body fitted coordinates and can employ structured as well as unstructured numerical grids.

[10]The description for CFX is provided by JUELICH and GRS, Germany.

[11] CFX-4 was used for hydrogen distribution in the Polytechnical University of Madrid. CFX-4 was a multiblock structured CFD code, which was no longer under development or supported. It was a different code of the current CFX.

In the context of hydrogen analysis, CFX is not a stand-alone tool. Moreover, the overall aim of the CFD model development and application for containment analysis is to complement the LP code (e.g. COCOSYS) analysis by a detailed 3D simulation for dedicated / selected scenarios. Furthermore it allows investigating certain aspects in detail, such as,

- PAR positioning and operation inside single compartments
- Local flow and interaction with passive safety systems
- Buoyancy driven heat & mass transfer in complex 3D geometries
- Determination of flow resistances of complex 3D internals

and contributes to a detailed understanding and further development of LP models or inputs.

B.9.1.1 Hydrogen Generation

Hydrogen generation is not addressed by CFX and it is out of its purpose. Results of the in-vessel and ex-vessel phase hydrogen generation are provided by tables.

B.9.1.2 Hydrogen Distribution

Hydrogen distribution is very well addressed by CFX being a CFD code and developed for it. CFX allows considering the 3D transport and mixing of steam and non-condensable gases as well as the interaction with certain technical systems (ventilation, burst discs, doors, PAR) within the containment or other buildings of an NPP.

- 3D model for gas flow
- major typical non-condensable gases: H_2, He, N2, O_2, CO, CO_2
- Separate treatment of condensate/water flow

Only single (gas) phase simulation is addressed, specifically:

- no heat and mass transfer (HMT) with sumps
- thermal equilibrium assumption for condensation
- only steam (single phase) release from leak
- no interaction of flow with aerosols and FP (local heat sources)

B.9.1.3 Hydrogen Combustion

In CFX, combustion is addressed as follow:

- Burning velocity model: applicable and validated for H2-combustion
- Eddy Dissipation model: it considers only influence of turbulence (not the laminar burning velocity), model coefficients are case sensitive
- Extended Coherent Flame Model (ECFM): numerical instabilities (CFX13) may complicate application
- Arrhenius reactions: not applicable for large scale flame propagation

- CO-combustion (variable H_2/CO ratio) is not implemented and there is a lack of validating experiments.
- There are just valid experiments for turbulent accelerated flames, but not for simulation of DDT or detonation.
- Modelling of small obstacles needs to be clarified for combustion simulation in containment.
- Phenomena like flame instabilities (e.g., thermo-diffusive, acoustic) are not taken into account (ISP-49, THAI).
- No reliable quenching model (ISP-49 ENACCEF).

B.9.1.4 Hydrogen Mitigation Systems

CFX is capable to handle the following mitigation systems:

- PAR (correlation model): Detailed PAR model for all AREVA type PARs
- PAR (REKO-DIREKT model): detailed mechanistic model with detailed output (e.g., local reaction rates, catalyst temperatures) currently validated for AREVA PARs (extension to AECL and NIS type planned)
- Ignition due to PAR: Implementation of a criterion based on gas temperature planned for REKO-DIREKT
- Parallel CO recombination: Validation of REKO-DIREKT in stand-alone model
- Vents or rupture disks: Modelling by means of pressure boundary conditions and/or conditional interfaces
- Heat exchangers: Modelling by means of homogenized porous media approach
- Fans: Modelling by means of volumetric momentum sources

B.9.2 User-defined Code Capabilities

In Germany, a common R&D activity, coordinated by GRS, is performed to enhance the CFX capabilities to simulate hydrogen mixing, mitigation and combustion (among other NRS issues). In the field of containment flows, the developing partners consists of ANSYS (maintainer) and different partners from universities (e.g. University of Stuttgart, IKE, RWTH Aachen University), research institutions (e.g., FZJ, KIT) and industry (e.g., AREVA). This R&D activity has extended CFX in many areas relevant to containment analysis, including physical models (e.g., wall-condensation, turbulence models for buoyancy driven flows, combustion with multiple inert species, etc.) and technical components (e.g., doors, pressure relief flaps, burst discs, PARs, etc.).

In Spain, Polytechnical University of Madrid applied extensively the CFX-4 during the period 1998-2004, for hydrogen containment analysis. The following model development was done:

- Hydrogen distribution: Implementation of steam film condensation onto walls in presence of non-condensable gases (correlation and mechanistic models) and 1D heat conduction in walls.
- Hydrogen combustion: Standalone or CFD-coupled module for incorporating the FA/DDT criteria (SOAR de H_2) and pressure load (AICC, dynamic) within hydrogen distribution calculations.
- Hydrogen mitigation: MELCOR-coupled or CFD-coupled module for incorporating detailed mechanistic model for H_2/CO recombination in parallel-plate PARs.

B.9.3 Code Validations

In general, continuous validation against German THAI CFD-Validation (CV) tests, OECD/NEA THAI hydrogen mixing (HM), hydrogen deflagration (HD) and hydrogen recombiner (HR) tests is in progress.

In particular, regarding hydrogen combustion the code has been validated with a reasonable amount of experiments, taking into account slow combustion of a H2/steam/air mixture, fast turbulent combustion and turbulent combustion.

Regarding the PAR modelling, the code has been validated for:

- AREVA (Siemens) PAR operation und O_2 excess and dry atmosphere
- AREVA (Siemens) PAR operation under low O_2, humid atmosphere
- Catalyst ignition behaviour under well-defined steady state boundary conditions
- Database of catalyst performance tests under steady-state, well defined conditions (dry atmosphere, excess oxygen, steam atmosphere, low oxygen, CO interaction)
- Influence of chimney design on PAR efficiency

Regarding gas distribution, the code has been validated for:

- Gas stratification
- Momentum Induced Mixing in Gases
- Buoyancy Induced Mixing in Gases
- Condensation on surfaces

In Spain, the CFX-4 code had been extensively applied in the past by the Polytechnical University of Madrid during the period 1998-2004, including the validation against hydrogen distribution and mitigation experiments:

- Battelle BMC-Zx experiments within the participation in the HYMI/VOASM projects of the IV-EU-FWP
- MISTRA MICOCO test
- SBEP-01 (VI-FWP HySafe Network)
- Film condensation models (Dehbi and Anderson experiments)
- Contribution to the PARIS-1 benchmark with a PAR mechanistic model (coupled to CFX-4)

B.9.4 Strengths, Limitations and Improvements

CFX is a commercial code which is widely used in all industrial areas. This allows taking benefits from a state of the art solver with continuous enhancement of numerical schemes, models and other capabilities such as e.g. post-processor and scripting language. Furthermore the experience and improvements from other application field can be transferred applied to deal with NRS issues.

Using a multi-purpose code always comes along with the need to implement dedicated models for characteristic NRS or SA phenomena. This is successfully performed in the frame of a coordinated activity within the German CFD alliance and funded by the German Ministry for Economics and Technology (BMWi).

Recent achievements are e.g. the implementation of:

- an effective (single-phase) wall condensation model, based on a diffusion layer approach and a bulk condensation model based on a homogeneous equilibrium approach (ANSYS, GRS, RWTH Aachen, JÜLICH)
- a detailed two-phase bulk condensation and droplet transport model (IKE, University of Stuttgart)
- Implementation of the detailed mechanistic PAR model REKO-DIREKT and simplified models based on the manufacturers empirical correlations (GRS, JÜLICH)
- Enhancement of the burning velocity model by an inert gas model to consider combustion with variable mixtures of inert species (ANSYS, GRS)

Main drawback of CFD is the huge computational costs which still need to be reduced by effective subgrid scale models or by code coupling.

Since the OECD/NEA ISP-47, a significant improvement of the predictive capabilities has been achieved, but further extension of the model basis and validation is needed.

B.9.5 Application Method

The fundamental code validation is performed by the developer ANSYS. Due to this and the high number of users from different application areas serious programming or model errors can be practically excluded. The comprehensive documentation and training materials and a good usability of the software (GUI, consistency checks, etc.) leads to a reduced user error.

General and dedicated best practise guidelines, such as References [4.6], [B.23], [B.24], are available in order to reduce numerical errors:

In Spain, the methodology developed by CTN/UPM consisted of analysing a wide range of sequences and scenarios with a MELCOR detailed containment model. Afterwards, these results were used to perform CFX-4 analysis in short-term temporal windows. The following exercises were performed:

- Analysis of hydrogen distribution and FA/DDT criteria in Westinghouse PWR containment.
- Analysis of hydrogen distribution and FA/DDT criteria in KWU PWR containment.

B.10 FLUENT[12]

Code developers	ANSYS (USA)
Users	Finland, France, the Netherlands, Switzerland, Japan, etc.
General capabilities	Comprehensive multi-purpose CFD software
Applications	A broad variety of phenomena, including H2 mixing, combustion and mitigation.

[12]The description for FLUENT is provided by NRG, the Netherland.

Reference http://www.ansys.com

B.10.1 Code Capabilities

FLUENT [B.25] is a commercial general-purpose computational fluid dynamics (CFD) software package supplied by ANSYS, Inc. It is a state-of-the-art computer program for analysing steady-state and transient flow and heat transfer problems in complex geometries. It is based on the finite volume method. A wide range of physical models is available in FLUENT for modelling turbulence, multiphase flows and reacting flows.

FLUENT provides mesh flexibility and allows accurate representation of the geometry. FLUENT's parallel solver makes it possible to perform calculations on multiple processors, either on the same computer or on different computers in a network.

B.10.1.1 Hydrogen Generation

Hydrogen generation is not addressed by FLUENT and it is out of its purpose. Results of the in-vessel and ex-vessel phase hydrogen generation are provided as input data.

B.10.1.2 Hydrogen Distribution

Hydrogen distribution is very well addressed by FLUENT being a 3D CFD code and developed for it. FLUENT allows considering the 3D transport and mixing of steam and non-condensable gases as well as the interaction with certain technical systems (ventilation, burst discs, doors, PAR) within the containment or other buildings of an NPP.

- 3D model for gas flow
- major typical non-condensable gases: H_2, He, N_2, O_2, CO, CO_2
- Steam condensation and evaporation are not treated

The capabilities of the standard FLUENT code with respect to distribution and combustion of hydrogen are listed below:

- High, 3-dimensional resolution: local gas composition determines the burning mode of hydrogen and the operation of the PARs.
- Conservation of mass, momentum, energy and species: Pressure waves and oscillations resulting from combustion can be captured.
- Turbulence modelling: Turbulence affects the formation and mixing of stratified gas layers. Also the flame speed and combustion mode depends primarily on turbulence generation during flame propagation.
- Lagrangian trajectory calculation for dispersed phases: Spray modelling can be customized by user coding.

B.10.1.3 Hydrogen Combustion

The FLUENT solver has the capability to model chemical species mixing and reaction, including homogeneous and heterogeneous combustion models and surface deposition/reaction models. FLUENT can calculate the flame speed and combustion mode which depends primarily on turbulence generation during flame propagation; high resolution near flame front with feasible computing times is also possible.

B.10.1.4 Hydrogen Mitigation Systems

Mitigation systems are not explicitly addressed by FLUENT, but FLUENT can address the effect of fans, local air coolers and spray induced mixing in hydrogen distribution.

Mitigation systems can be modelled by built-in features, by user-coding or a combination of both. Below an overview of the mitigation systems that can be included by built-in features:

- Sprays can be modelled with the built-in Lagrangian approach for sprays.
- Vents can be modelled by defining an outlet or venting line.
- Fans can be modelled with the built-in lumped parameter approach for fans.
- Coolers can be modelled with the built-in lumped parameter approach for heat exchangers.

B.10.2 User-defined Code Capabilities

A customized version of FLUENT for containment applications is developed by NRG and used as best-estimate code for predicting the ex-vessel transport, mixing and combustion of hydrogen in the reactor containment during SAs. In order to capture the specific phenomena and processes that play a role during SAs, the general purpose CFD package is complemented with 'user-defined' sub-models. Specific sub-models for steam condensation/ evaporation on walls and in the atmosphere, multicomponent diffusion, conjugate heat transfer and thermal inertia of walls, hydrogen recombiners, sprays, and hydrogen combustion have been developed, implemented and validated by NRG. These sub-models have been employed successfully in several international benchmark exercises already ([B.26] through [B.30]). For a detailed description of the combustion sub-model check the references ([B.31] through [B.34]).

B.10.2.1 Hydrogen Generation

The FLUENT CFD code is customized by NRG for modelling the ex-vessel behaviour of hydrogen in the reactor containment. Hydrogen generation is, therefore, not calculated inside FLUENT. The hydrogen source term calculated with system codes is used instead as boundary condition for hydrogen distribution, mitigation and combustion analyses.

B.10.2.2 Hydrogen Distribution

The FLUENT code is customized by NRG to improve the modelling of hydrogen distribution in the reactor containment during SAs. The features defined by NRG are listed below:

- An appropriate model for multi-component diffusion is not available and is implemented by user-coding.
- An appropriate model for steam condensation and evaporation is not available and is implemented by user-coding. The behaviour of condensate deposited on walls is not treated yet (e.g. by means of film model). However, the amount of water on the walls is tracked and, depending on the near-wall conditions, it can re-evaporate again.
- The production (or dissipation) of turbulence by buoyancy is included in the k-epsilon turbulence models, but is implemented by user-coding in the k-omega turbulence models.

B.10.2.3 Hydrogen Combustion

Advanced turbulent flame speed closure (TFC) models for hydrogen combustion are not available in FLUENT and must be implemented by user-coding. The hydrogen combustion model implemented in

FLUENT by NRG covers the deflagration regime ([B.31], [B.34]) and is based on the combustion model of Zimont [B.35]. Models for detonation or DDT are not available yet. In addition, ignition models must be included by user-coding.

B.10.2.4 Hydrogen Mitigation Systems

The models for mitigation systems are mostly a combination of built in capabilities of the standard FLUENT code and user-coding (or user-specific choices). The code features used and defined by NRG for modelling the hydrogen mitigation systems are presented below.

- NRG's PAR model mainly uses build in FLUENT features. The PAR housing is modelled explicitly, while the catalytic plates are modelled as porous zone. The hydrogen recombination rate depends on pressure and local gas composition, and is implemented by means of a correlation provided by the PAR vendor. Onset and auto-ignition of PARs is user-defined.
- Igniters are modelled as multiple ignition sources. The flammability of the local gas mixture determines the subsequent flame propagation.
- NRG's spray model is based on FLUENT's built-in Lagrangian approach for sprays, which takes into account the drag between fluid and droplets (2 way coupling). Heat and mass transfer between fluid and droplets is modelled by a Nu-Re-Pr correlation and a Sh-Re-Sc correlations, respectively. The behaviour of spray droplets deposited on walls is not treated yet (e.g., by means of film model). However, the amount of water on the walls is tracked and, depending on the near-wall conditions, it can re-evaporate again

B.10.3 Code Validations

FLUENT is a commercial software widely used and it is validated by its users community on a daily basis; nevertheless it has been validated also in assessing the risk of hydrogen in the containment of a NPP.

- Distribution: validations have been performed for all distribution phenomena, including ISP-47, THAI, OECD/SETH and OECD/SETH2, TOSQAN and ERCOSAM;
- Combustion: validations have been performed for deflagration, FA and flame diffusion with the ENACCEF and THAI experiments;
- Mitigation: validations have been performed for the spray system, PAR, and fan-coolers.

B.10.4 Strengths, Limitations and Improvements

FLUENT is a commercial 3D CFD code which is widely used in all kinds of industrial areas. This allows taking benefits from a state-of-the-art solver with continuous enhancement of numerical schemes, models and other capabilities such as post-processor and scripting language. Furthermore the experience and improvements from other application fields can be readily applied to NRS issues.

The main drawbacks of the 3D FLUENT code is that it requires relatively long computation times and that it needs experienced users to customize the code for hydrogen analyses.

Its strengths are summarized as follows:

- Detailed thermal-hydraulic analyses with fine mesh in three-dimensional coordinates.
- Wide range of physical models for multi-phase flows, turbulent flows and so on.
- Customization capability via user-coding

- Flexible meshing
- Efficient parallel performance
- The standard k-ε turbulence model with buoyancy options provides quite good results; buoyancy option is a standard feature in this turbulence model (THAI HM-2)
- When obstacles in an experimental apparatus enhance turbulence and thus accelerate flame propagation, the standard eddy dissipation model can be used with quite good results (ENACCEF)

Its limitations are summarized as follows:

- Relative high computational effort
- Specific models for nuclear power plant components are not included
- Specific physical models for hydrogen analyses are inadequate or not included, e.g. models for steam condensation/evaporation, multi-component diffusion and combustion

B.10.5 Application Method

The fundamental code validation is performed by the developer ANSYS. Due to this and the high number of users from different application areas serious programming or model errors can be practically excluded. The comprehensive documentation and training materials and a good usability of the software leads to a reduced user error.

B.11 AUTODYN[13]

Code developers	ANSYS (USA)
Users	Japan, etc.
General capabilities	AUTODYN is a comprehensive multi-purpose FE/CFD code and it can model a broad variety of phenomena. In the context of the hydrogen risk management, AUTODYN can be categorized as the code intended for analysing explosion and resultant structural responses.
Applications	Mainly used for assessing loads on the containment due to hydrogen explosions and steam explosion. The code is also used for assessing dynamic responses of the containment, the reactor vessel and other important systems inside the containment
Reference	http://www.ansys.com

B.11.1 Code Capabilities

The AUTODYN program is based upon a coupled multi-solver approach: Euler, Lagrange, ALE, Shell, Beam and SPH are implemented. In JNES, the Lagrange/Shell solver is mainly applied in licensing and periodic safety assessment for evaluating dynamic responses of structures against the fast pressure loads caused by hydrogen explosion and steam explosion. In investigating causality of hydrogen explosions occurred in the Fukushima Dai-Ichi NPS, comprehensive plant level fluid-structure interactions

[13]The description for AUTODYN is provided by JNES, Japan.

were analysed based on the Euler/Lagrange coupling scheme, where a fluid part was modelled by the Euler solver while a structural part was modelled by the Lagrange/Shell solver.

B.11.1.1 Hydrogen Generation

Hydrogen generation is not addressed by AUTODYN and it is out of its purpose. Results of the in-vessel and ex-vessel phase hydrogen generation are provided by tables.

B.11.1.2 Hydrogen Distribution

Hydrogen generation is not addressed by AUTODYN and it is out of its purpose. In-vessel and ex-vessel phase hydrogen generations are evaluated by system level SA codes such as MELCOR and reflected in distribution analyses by CFD codes.

B.11.1.3 Hydrogen Combustion

Detonation is simulated by AUTODYN based on the Chapman-Jouguet (C-J) hypothesis. Propagation and reflection of detonation waves are calculated with or without considering ignition and growth. State variables at the C-J point (pressure, temperature, density, internal energy, velocity, detonation velocity, ratio of heat capacity, etc.) are evaluated by the detonation properties computer code AISTJAN based on the theory of thermo-chemical equilibrium of the pre-mixed hydrogen/air mixture. AISTJAN was developed by the National Institute of Advanced Industrial Science and Technology (AIST), Japan. Instead of applying the molecular statistical mechanism, thermodynamic properties are evaluated based on polynomials that are similar to those implemented in the NASA code, JANAF.

B.11.1.4 Hydrogen Mitigation Systems

Mitigation systems are not addressed by AUTODYN. However, the code is intended to assess dynamic structural responses including the fracture behaviours. For example, the discrete reinforcement model was implemented to facilitate to embed beam parts in Lagrange shell parts. The breakable bonded connection was implemented to facilitate to disconnect arbitrary connections according to a certain stress criterion. It is feasible to demonstrate efficacy of mechanical mitigation systems like the latch system.

B.11.2 Code Validations

The solution techniques used in AUTODYN software have been repeatedly validated by comparing the results generated with experimental data. Numerous comparisons by customers of experimental and simulated results are published annually in technical journals and presented at technical conferences.

High pressure injection line break accident due to detonation in the pipe with hydrogen and oxygen generated by water radiolysis at Hamaoka NPP unit 1 in Japan occurred on Nov. 2001. Detonation wave propagation in a high-pressure gas mixture (7 MPa) and the steel pipe fracture in a three-dimensional piping system caused by the detonation and pressure waves were checked with AUTODYN [B.35].

A series of explosion tests of hydrogen gas were conducted, and characteristics of hydrogen gas explosion and nonlinear response of reinforced concrete walls subjected to the explosive loads were studied. Simulated overpressures of blast waves obtained by AUTODYN shows good agreements with the test results. The simulations were performed in two dimensions [B.36].

B.11.3 Strengths, Limitations and Improvements

AUTODYN is a commercial code which is widely used in all industrial areas. This allows taking benefits from a state of the art solver with continuous enhancement of numerical schemes, models and other capabilities such as post-processor and scripting language. Furthermore the experience and improvements from other application field can be transferred applied to deal with NRS issues.

Main drawback of AUTODYN is the huge computational costs which still need to be reduced by effective subgrid scale models or by code coupling.

The strength of the code is:

- AUTODYN has been used over 25 years for nonlinear dynamic analyses.
- AUTODYN is a highly interactive computer program. The processes of setting up of problems, running the problems and viewing of outputs are seamlessly achieved by the interactive graphical user interface. Current results may be always displayed on a computer monitor.
- User-defined material models and boundary conditions can be implemented through FORTRAN based subroutines.
- Material data library is available. More than 250 materials are included.

The weakness of this code is that adequate experience and knowledge about the fluid/structure scheme of AUTODYN is required to obtain physically correct results when we model an architecturally complicated structure as a nuclear power plant which consists of many walls, floors and internal structures.

B.11.4 Application Method

The fundamental code validation is performed by the developer ANSYS. Due to this and a large number of users from different application areas serious programming or model errors can be practically excluded. The comprehensive documentation and training materials and a good usability of the software (GUI, consistency checks...) leads to a reduced user error.

In order to evaluate issues that are related to hydrogen and to the containment, JNES uses two methods: one is MELCOR for hydrogen generation, distribution and combustion; another is the combination of MELCOR for generation, FLUENT for distribution and AUTODYN for detonation.

JNES is preparing standard input decks for each representative NPP types to be run with MELCOR, FLUENT and AUTODYN respectively. Basis of input data of the standard input decks are documented and checked.

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