1. DETAILED APPROACH

In the EPR design there are, in principle, two approaches to severe accident mitigation:

- identifying the most likely scenarios leading to core melt using probabilistic methods and postulating preventative or mitigating measures for these scenarios: these scenarios will represent a high percentage of conceivable scenarios.

- determining the containment response in representative core melt sequences and ensuring containment integrity by means of suitable design measures (the phenomenological or deterministic approach).

The EPR, as a priority, follows the deterministic approach, which aims at improving the design to virtually eliminate the possibility of a radioactivity release should such an event occur. This approach involves design measures to prevent early containment failure due to the transient event, and to maintain long-term containment integrity.

It is subsequently demonstrated that the probabilistic approach supports the mitigation measures developed using the deterministic approach (level 2 PSA, see Chapter R.3).

An early failure of the containment in a severe accident situation could have major consequences for the public in terms of radiological dose. The principal design objective of the EPR therefore is to eliminate, insofar as possible, any risk of early containment failure by implementing design measures against the following energetic phenomena:

- Hydrogen detonation:
  Implementation of rupture foils and dampers between the two zones of the containment (two-zone concept), and installation of recombiners, which in combination with the primary system discharge into the containment atmosphere, ensure good mixing in the containment volume, high steam concentrations, and a reduction in the amount of hydrogen.

- Direct containment heating (DCH):
  - measures to depressurise the primary system to avoid a high-pressure failure of the reactor vessel;
  - design of the reactor pit to avoid direct flowpaths for corium debris to reach containment structural concrete

- Ex-vessel steam explosion:
  Ex-vessel steam explosions are prevented by avoiding the presence of large quantities of water in the reactor pit at the time of pressure-vessel failure and during corium ejection from the RPV, and in the corium spreading compartment before corium spreading.

Measures taken to avoid containment bypass events that could cause significant radiological impact, rapid reactivity insertion accidents, and fuel damage in the spent fuel pool, are presented in Chapter S.2.4.

To meet the EPR radiological goals (see Chapter S.2.3 for a precise definition), the integrity of the containment must be ensured. Features are integrated in the safety concept of the EPR to cope with the following challenges:

- Ablation of the containment foundation raft:
  avoidance by spreading and stabilising the corium in a shallow core catcher, which has a surface area of 170 m^2. The decay heat is removed from the melt's upper surface by flooding and quenching the melt from the top and at the melt's underside and lateral boundaries by the cooling structures of the core catcher. The coolant water necessary for heat removal is drawn from the IRWST either in a passive way or by the EVU [CHRS] pumps..

- Containment over-pressurisation:
  avoidance by containment heat removal by the EVU [CHRS] system, which provides containment spray and cooling using water from the IRWST which is circulated outside the containment. The containment free volume in combination with the containment structural heat sinks allow a period of at least 12 hours before the EVU [CHRS] is required to be operated.

- Containment leakage:
  a leak rate of 0.3% of the containment volume per day is achieved (based on a conservative evaluation) at a pressure of 5.5 bar. Depressurisation of the containment using the EVU [CHRS] spray system (a reduction to 2 bar in 24 hr) further reduces the leak rate.

In the following sections, the above technical features are described in more detail, together with the calculations of accident progression. The engineering solutions for the most significant design features (strategy for controlling combustible gases, protection for the foundation raft, and containment heat removal), are described in detail in the sub-chapters relating to the containment (see Chapters F.2.4, F.2.6, and F.2.7, respectively).

Within the framework of the deterministic approach, scenarios (see Chapter S.2.2.1***)*** have been selected in order to analyse the following:

- hydrogen control

- depressurisation of the primary circuit

- protection of the containment foundation raft

- containment heat removal.

These scenarios are chosen to address all significant phenomena and are representative of the limiting cases for the specific problems to be considered.

For each of the scenarios, the accident progression has been calculated up to reactor vessel failure, using the MAAP integrated computer code to define the boundary conditions needed to calculate the containment response and to justify the appropriateness of the dedicated depressurisation. These calculations are described in the following sections (see Chapters S.2.2.1 and S.2.2.2).

The consequences for the containment design, with specific reference to the design of two-zone concept and the metal liner, are described in the following sections (H_2 control and P-T conditions).

1.1. CONCEPT OF HYDROGEN CONTROL

a) Hydrogen production (see Chapters S.2.2.1 and S.2.2.2).

In order to adequately design the method of hydrogen control, it is necessary to specify representative and bounding scenarios which define the rate and amount of hydrogen released. The large-break LOCA is not bounding for pressurisation of the containment due to hydrogen combustion, given the high steam concentrations in the containment in this scenario. Similar considerations apply to LOOP. It is the amall-break LOCA sequences that provide the most limiting scenarios.

Four representative scenarios, selected primarily for their occurrence frequency, are used to demonstrate the adequacy of the hydrogen control design:

- a 2-inch cold leg SB-LOCA with partial secondary cooldown
- a 2-inch cold leg SB-LOCA with fast secondary cooldown
- a 2-inch hot leg SB-LOCA with fast secondary cooldown
- a 3-inch SB-LOCA in the upper part of the pressuriser with fast secondary cooldown.

In addition, to demonstrate the robustness of the design, two further bounding scenarios are selected to allow for additional aggravating effects on the hydrogen risk:

- a 2-inch cold leg SB-LOCA with partial secondary cooldown and late depressurisation
- a 2-inch cold leg SB-LOCA with fast secondary cooldown and flooding of the reactor vessel.

The LOOP sequence with recovery of four ISMP [MHSI] is also studied to evaluate the thermal effect of recombination, because the amount of hydrogen produced is large.

The hydrogen is generated in three phases, mainly due to the nature of zirconium/steam reaction. The first phase, and the most significant one, is the degradation of the core in-vessel, involving oxidation of large surface areas of metallic Zr with a temperature excursion accelerated by the exothermic oxidation reaction. Hydrogen may also be generated during the in-vessel phase of corium relocation, due to interaction with the water in the bottom of the vessel. The production of hydrogen during this phase depends on the mode of degradation of the core and the internals of the reactor vessel (particularly the core support plate) and on the mass of water available at the bottom of the vessel. Finally, some hydrogen is likely to be produced ex-vessel, as the corium draining from the vessel interacts with the sacrificial material in the reactor pit and spreading area.

Oxidation of the total amount of zirconium present in the core would yield approximately 1600 kg of hydrogen, of which oxidation of the fuel cladding accounts for approximately 1400 kg. These values however are not reached during the in-vessel phase (see Chapter S.2.2.1).

According to the MAAP calculations for these representative scenarios, taking into account the large range of break sizes and different modes of release into the containment, a quantity of hydrogen of approximately 700 kg will be released during early core degradation over a very short period.

High release rates, up to several kg/s for a very short period of time, can result from scenarios with recovery of the safety injection system or with delayed RCP [RCS] depressurisation, causing late water injection from the accumulators, with up to 1000 kg of hydrogen being generated (corresponding to 60% Zr oxidation).

Ex-vessel hydrogen production, which continues for approximately an hour and is governed by the erosion of sacrificial concrete in the reactor pit, generates 500 to 800 kg of H_2. In the late phase, when the metallic corium comes into contact with the sacrificial material, the hydrogen release rate is relatively high for a short period of time, as in the initial phase of core degradation. Due to the high gas temperatures (exceeding 2000°C) which are well above the temperature of spontaneous combustion (500 to 600°C), this hydrogen burns immediately in the presence of oxygen, probably together with any hydrogen produced in-vessel which has not yet been recombined, independently of any mitigation means (see Chapter F.2.4.).

b) Means of hydrogen control (see Chapter S.2.2.3)

Considering the production of hydrogen as specified above and the locations of its release within the containment, the hydrogen control system must satisfy the following requirements:

- with regard to the local risk: it must safely prevent local hydrogen detonation or DDT (transition from deflagration to detonation): the development of hydrogen clouds at average concentration beyond a critical size (is determined by the 7λ criterion) must be avoided.

- with regard to the global risk: to practically eliminate the risk of global hydrogen detonation, the average volume concentration of hydrogen in the entire containment must be kept below 10%, in dry air conditions. Values up to an equivalent of 13% under dry air conditions may be allowed if there is partial steam inerting (which would be equivalent to 10%, accounting for the steam content). Achieving the local hydrogen concentration limit using the recombiners automatically ensures that the global concentration limit is met.

- Global deflagration of the hydrogen present in the containment at any given time should not lead to a pressure exceeding the containment verification pressure (6.5 bar - see Chapter F.2.1);

- the long-term hydrogen concentration must be maintained below the combustion limit of 4%

- recombination or possible combustion must not lead to unacceptable temperatures at the containment walls.

To achieve these goals, mitigation methods described below are employed in the EPR:

- **a sufficiently large containment volume (a free volume of approximately 80,000m^3)** to reach the global concentration target using the recombiners.

- **installation of rupture foils and dampers** designed to open sufficiently quickly in the course of an accident to change the geometric configuration from a two-zone to a one-zone containment, to promote natural convection and consequently to improve mixing.

- **installation of catalytic recombiners:** in order to reduce the amount of hydrogen in the containment atmosphere, recombiners are installed, mainly in and near the primary system compartments. The recombination rate depends on the hydrogen concentration, the pressure, and the size of the recombiner. Based on their size and the containment pressure, the that rate of recombination varies from about 1 to 8 kg/hr. The number of recombiners required is determined by the need to keep global hydrogen concentration below 10% at all times and to reduce the average hydrogen concentration to below the combustion limit in a dry atmosphere in a time of less than 12 hours based on the complete oxidation of the Zr.

  Calculations carried out for typical installation of recombiners show that approximately 47 recombiners, 41 of them large and 6 small, (see Chapter F.2.4) are sufficient to limit the global concentration of hydrogen for all relevant accident situations, although they have little effect on local concentration peaks in scenarios involving hydrogen release rates which are high compared with the capacity of the recombiners. Recombiners installations which are sufficient for the early phase of a severe accident, are also sufficient for limiting the effects of hydrogen production during the long-term phase (as well as that in design basis accidents in which hydrogen is generated mainly by radiolysis, at a very low rate).

Note that the primary system discharge into the containment atmosphere results in an increase in the containment steam concentration and better mixing of gases in the containment.

The concept of mitigating the hydrogen risk is described in Chapter F.2.4, and the MAAP calculations for in-vessel hydrogen production are presented in Chapter S.2.2.1. Chapter S.2.2.3 presents calculations of gas distribution and thermal loads using the CFD GASFLOW code and dynamic load calculations using the COM3D code. Ex-vessel hydrogen production is evaluated in Chapter S.2.2.4.

1.2. CORIUM COOLANT INTERACTION

1.2.1. Introduction

Corium coolant interaction is a process in which the molten fuel transfers its thermal energy to the surrounding coolant, causing fragmentation of the corium, resulting either in the formation of a layer of coolable debris or in a corium coolant interaction which might cause an energetic steam explosion. The corium coolant interaction or steam explosion involves four sequential phases: pre-mixing, triggering, propagation, and expansion into the surrounding environment.

This process can occur by two modes:

a) a contact mode by means of flow, in which the corium pours into a volume of water: this mode can be realised in-vessel when the corium is relocated to the bottom of a vessel filled with water. In the event of corium flooding ex-vessel, design measures (see below and Chapter F.2.7) ensure that no water is present prior to ex-vessel corium relocation, preventing any possibility of interaction.

b) A water-injection or stratified-contact mode, when a mass of corium is flooded by water: this mode can occur in the EPR both in- and ex-vessel: in-vessel following flooding, when water is injected above the molten-core pool or later during the formation of the pool at the bottom of the reactor vessel, and ex-vessel due to passive flooding of the corium in the spreading area.

1.2.2. In-vessel phenomena

It is generally considered that the "flow" mode can potentially lead to the greatest loads and that flooding a mass of corium at an adequate water flow-rate does not lead to a strong corium coolant interaction. In fact, experiments indicate that injecting a low-density liquid into a much higher density liquid causes less energetic interactions than in the opposite case of corium draining into the water.

Evaluation by the NRC Steam Explosion Review Groups (SERG1 and SERG2) has concluded that the conditional probability of a steam explosion for a core damage scenario leading to containment failure, called mode α, is of the order of 10^{-3} to 10^{-5}. This corresponds to the formation in the pressure vessel of a slug of corium debris projected toward the pressure-vessel head, which, in the event of the head rupture, can form a projectile capable of damaging the containment [**Basu (1996)**]. This conclusion confirms that this phenomenon makes a negligible contribution the overall risk.

Since the above evaluation, there has been additional R&D which allows the key parameters to be quantified on a deterministic basis and a preliminary conclusion to be drawn on the risk to the reactor:

- experiments on the thermal-hydraulic behaviour of the corium in typical materials:
    - the mixing process has been studied, via film-boiling tests on spheres interacting with water (the QUEOS and BILLEAU tests) and tests on the relocation of corium in a typical material into saturated and subcooled water pools, with different geometries (PREMIX, [**Struve (1999)**], FARO, and FARO/FAT tests [**Mag (2001)**]);
    - explosion limits have been studied (in the KROTOS experiments, for example). To date, realistic mixtures of corium (UO_2, ZrO_2) have only displayed a low degree of explosivity compared with the tests with simulated materials on which the previous evaluations for steam explosion were based [**Piluso (2005)**]. No energetic steam explosion was produced in KROTOS [**Huh (1999)**] with corium involving typical materials, even under conditions of high water subcooling, superheating of the corium, or with a strong external trigger. Likewise, no steam explosion was produced with the same oxide mixtures during flooding tests of the corium in FARO. The rate reached for the conversion of thermal energy into mechanical energy was, at most, a tenth of that obtained with simulants. Accurate measurements of the transformation of thermal energy into mechanical are currently in progress at FZK (ECO) with thermite, and they indicate a yield lower than 1% [**Albrecht (2001)**, **Cher (2001)**, **Cher (2002)**, and **Cher (2003)**].
- experiments on the strength of the pressure-vessel head:

  - o the acceptable mechanical energy relative to the survival of the reactor-vessel head has been studied in the BERDA tests [**Krieg (2000)**], carried out in order to reproduce the phenomenon of a corium-slug projectile. These tests conclude that the full size reactor-vessel head is capable of resisting a liquid slug propelled with a force of at least 0.8 GJ without significant deformation of the head and that the acceptable energies would be much higher if significant deformation of the head were tolerated and if the various energy dissipation processes were taken into account. Likewise, the slug assumption seems conservative, given that it is highly unlikely for a compact slug would be able to travel several meters undisturbed before hitting the vessel head, as confirmed by the example of one of the BERDA tests, which proves the instability of a slug accelerated during a gas expansion.

- computer codes development:

  - o significant progress has been made in the development and validation of computer codes, such as MATTINA (FZK) and MC3D (developed by CEA and IRSN), using in particular the results of the FARO pre-mixing tests [**Struve (1999)**]. Considering realistic reactor conditions, a low superheating of the corium with the possibility of partial solidification and relocation in a completely depressurised loop could lead to highly voided pre-mixing conditions, which naturally limits the probability and potential for a high-energy phenomenon. The results of the computer code performance analysis are presented in the framework of the OECD SERENA program [**Mag (2005)**]: as far as dynamic loading is concerned, confirmation that the no-load rate is correctly calculated would practically exclude the risk of an in-vessel steam explosion.

  - o The most recent computer codes show that any corium/water contact leads to depletion of the water available, which limits the amount of pre-mixing. The anticipated mass of mixed corium, taking explosivity limits into consideration, always remains in the neighbourhood of a few tons, regardless of the relocation path of the corium. According to the BERDA tests, the corresponding energy which could be transferred to a hypothetical slug remains, to a large extent, lower than the energy needed to initiate deformation of the pressure-vessel head.

Therefore, it may be concluded that an in-vessel steam explosion can be practically eliminated as a containment failure mode. Specific design measures to prevent in-vessel steam explosion are therefore not justified.

In the event of a non-explosive corium coolant interaction limited to melt quenching, the pressure increase in the reactor vessel still remains low enough not to challenge the structures, due to the venting provided by the high bleed capacity of the system designed to depressurise the primary loop.

1.2.3. Ex-vessel phenomena

Design features guarantee a dry reactor pit and a dry spreading area (see Chapter F.2.6). Based on this, an ex-vessel corium coolant interaction is only considered during corium flooding. The time delay in flooding associated with the strategy described in Chapter F.2.6 ensures the formation of a crust or viscous layer on the corium surface.

The experimental data (derived, for example, from the MACE tests) show that flooding of the corium under these conditions, at a low flow rate, does not lead to high-energy interactions capable of challenging the containment integrity. Moreover, flooding of the dispersed corium, with a realistic assumption of 20% of the corium surface area fragmented, results in containment pressures much lower than the design pressure.

1.3. EX-VESSEL MOLTEN CORE STABILISATION

Ex-vessel stabilisation of the corium is addressed in a specific section (see Chapter S.2.2.4).

To prevent foundation raft melt-through resulting in a large release of fission products, including groundwater contamination, a specific design is provided to stabilise the corium inside the containment. The EPR concept of corium retention is based on spreading the corium over a large surface area, followed by flooding and cooling above and below using water drained passively from the IRWST. The details of the technical solution are provided in the section related to the containment (see Chapter F.2.7), and a demonstration (including the thermal-hydraulic calculations) is presented in Chapter S.2.2.4. The corresponding boundary conditions (masses, initial temperatures) are obtained from the in-vessel degradation calculations in Chapter S.2.2.1.

The following engineering systems and solutions have been implemented:

- **Temporary retention of the corium in the reactor pit:** To promote spreading of the corium, the EPR design uses a preliminary phase of temporary melt retention in the reactor pit. This is achieved by applying a layer of sacrificial concrete on the inside of the pit which the corium has to penetrate, as well as by eliminating potential routes for melt leakage other than through the transfer channel located at the bottom. The duration of the retention phase is determined by the time needed for ablation of the sacrificial layer and thermal destruction of the melt gate which opens the transfer channel between the pit and the spreading area.

- **Sacrificial layer in the reactor pit:** the concrete addition equalises the spectrum of corium characteristics, and makes the spreading process and the subsequent stabilisation process independent of the uncertainties associated with the in-vessel melt pool formation and the vessel failure mode. The strategy chosen for the temporary retention, i.e. ablation of a sacrificial concrete layer by the corium, has a beneficial self-regulating feature. A lower initial mass of corium released from the vessel or a lower level of decay heat results in a correspondingly longer retention time and vice versa. This guarantees an effective accumulation independent of the melt release scenario and the time of initial vessel failure. Adding a relatively well-defined amount of sacrificial concrete leads to more homogeneous corium properties and to a smoothing of the possible melt characteristics spectrum at the end of the retention process. The sacrificial concrete used in the reactor pit is composed of iron oxide and siliceous concrete in approximately equal proportions, the binding agent being ordinary Portland cement.

- **Protective layer in the reactor pit:** The sacrificial concrete is surrounded by a protective layer of ZrO_2 refractory material which separates the corium from the structural concrete in the reactor pit during the temporary retention phase and thus prevents ablation of the load-bearing structures.

- **Melt gate and corium transfer channel:** To obtain a complete and homogeneous distribution of the corium in the spreading area, a single spreading event is ensured by the installation of a gate beneath the sacrificial concrete, which constitutes a well-defined weak point and which fails upon contact with the corium, resulting in a sufficiently large cross-section for flow.

- **Cooling structure in the spreading area:** The core catcher in the EPR, in which the corium spreads out in a dry environment, is a shallow crucible whose surface area is about 170 m^2, which is made of cast iron. The lower part is made up of hundreds of individual elements covered with a layer of sacrificial concrete. The arrival of corium in the cooling structure initiates the opening of spring-actuated valves, which activate the gravity-driven flow of water from the IRWST. The water first fills a central feed line below the spreading room. From there, water penetrates into a system of parallel, horizontal cooling channels formed by the fins which form the base of the cooling elements. It then fills the space behind the side walls of the cooling structure. The water finally spreads over the surface of the corium starting from the outside edges. The water front which moves along causes local quenching and fragmentation of the corium, as well as a strong steam release into the containment. By the time the process has ended, a water pool will accumulate above the encrusted corium. The corium is expected to be completely flooded after approximately 15 min. The flow of water continues until the hydrostatic pressure in the spreading area and the IRWST are equalized. The link to the IRWST ensures fully passive heat extraction from the corium. Due to the high surface-area-to-volume ratio created by the spreading and because the corium is completely surrounded by cooled surfaces, confinement of the corium within stable crusts is achieved.

- **Sacrificial layer above the cooling structure:** The cooling elements which make up the bottom and side walls of the EPR core catcher are covered with a layer of standard siliceous concrete. As the cooling lines have a slope of 1°, the thickness of the layer situated above the lower part of the cooling structure is not uniform but increases from the edge toward the centre. The average thickness in this geometry then is approximately 14 cm. The thickness of the concrete layer on the side walls of the cooling structure is constant and is about 10 cm.

  Ablation of the sacrificial concrete layer by the corium delays corium contact with the cooling structure and in addition reduces the corium temperature at the time of initial contact. Furthermore, the density of the oxide portion of the corium relative to the density of metallic corium diminishes significantly until the end of the corium/concrete interaction, so that a stable configuration results in the corium pool, in which the molten oxides lie above the molten metals. Such a configuration in the corium melt prevents any focussing effect and ensures flooding of the oxide layer. Flooding the oxide pool during corium/concrete interaction is beneficial to cooling, given that the interaction encourages fragmentation and consequently solidification of the corium. The risk of a steam explosion resulting from flooding the free surface of the corium is negligible; experiments on this topic, such as the MACE tests (see Chapter S.2.2.4), have revealed no risk of explosion.

- **Integration into the EVU [CHRS]:** Although the EVU [CHRS] is not needed for cooling the corium itself, it can be used in the long-term to feed water directly into the core catcher, instead of operating in spray mode. Consequently, the water flowing in the cooling channels and onto the corium surface remains or becomes subcooled. The decay heat of the dispersed corium is then removed by single-phase flow instead of steam release in the containment atmosphere. Atmospheric pressure can thus be reached in the containment, which terminates any further radiological release due to potential leakage.

1.4. PREVENTION OF CONTAINMENT OVERPRESSURISATION

The prevention of containment over pressurisation is addressed in Chapter S.2.2.5.

The volume and design pressure of the containment allow a grace period of 12 hr before it is necessary to activate the EVU [CHRS] to safely prevent overpressurisation. The containment heat removal system is described in another section (see Chapter F.2.7).

A containment spray system using external circulation has been chosen for heat removal, taking into account the type of containment selected for the EPR (a double-walled concrete containment with a layer of steel). The selection was based on the following criteria:

- the possibility of reducing pressure and temperature in a reasonably short time (in order to reduce leakage and thus the radiological source term)
- the capacity for returning containment pressure to a value close to atmospheric
- low sensitivity to conditions resulting from severe accidents inside the containment, particularly due to the absence of any active component inside the containment
- no R&D need to develop a spray system
- low operational constraints during the normal service life of the facility (testing, maintenance) and in the event of an accident.

The selected two-train system has the following characteristics and performances, taking into consideration the overall characteristics of the containment:

- actuation of two EVU [CHRS] trains after a grace period of 12 hours maintains the pressure in the containment at a value lower than its absolute design value (5.5 bar), reduces the pressure below 2 bars in less than 24 hours, and then keeps the pressure at a value lower than 2 bars
- one train is sufficient to maintain the containment pressure below the design value at every stage of an accident, even in the short term
- one train is capable of reducing the containment pressure to 2 bars after 10 days, if the core is a UO_2 core, and after 15 days if the core is MOX.

The scenario adopted to define the pressurisation of the containment after core melt, to determine the grace period, and to design the EVU [CHRS] is a large-break LOCA (rupture of the surge line), as it results in the fastest pressurisation.

The principal contributors to mass and energy transfers in the containment for the scenario are the following:

- loss of primary coolant, with masses and energies released by the RCP [RCS]
- combustion or recombination of hydrogen produced during the accident
- production of non-condensable gas, H_2, CO, and CO_2 during the phase of the accident outside the vessel, due to corium-concrete interaction.
- production of steam during the quenching phase after onset of molten pool flooding.

The results of pressure and temperature calculations are provided in Chapter S.2.2.5. It has been demonstrated that the curves considered for the design of the containment are easily enveloped by the calculated curves, thus revealing comfortable margins.

Chapter F.2.7 describes the technical solution in detail; pressure and temperature calculations are presented in Chapter S.2.2.5.

1.5. PREVENTION OF A HIGH-PRESSURE REACTOR VESSEL FAILURE

A failure of the reactor vessel at high pressure is another significant potential contributor to early containment failure (missiles created by the pressure vessel movement, melt dispersal including Direct Containment Heating). Even though such a failure is physically unlikely, given that the primary loop is assumed to fail prematurely due to hot gases near the tubes, the Technical Directives require a design objective of transforming high-pressure core melt scenarios into low-pressure core melt scenarios with high reliability, so that reactor vessel failure under high pressure can be practically eliminated.

The design objective mentioned above is met by means of dedicated severe accident valves placed above the primary-loop pressuriser, with a discharge capacity sufficient to limit the pressure in the primary loop to a value well below 20 bars at the time of pressure-vessel failure (see Chapter S.2.2.2).

With regard to prevention of the consequences of high-pressure core melt scenarios and corium dispersal into the containment atmosphere, the EPR design has three categories of features:

a) Preventive measures:

Design provisions exist for preventing the occurrence of a core melt under high pressure (see Chapter S.2.2.1*)*: means to accomplish depressurisation of the secondary system and safety relief valves which allow the primary system to be depressurized.

b) Mitigation measures:

Design provisions exist for transforming high-pressure core melt scenarios to low-pressure core melt scenarios.

As indicated in Chapters S.2.2.1 and S.2.2.2, the relevant scenarios for specification of the operational requirements applicable to the pressuriser discharge correspond to the following initiator events:

- total loss of offsite power [LOOP] and unavailability of all Diesel generators
- total loss of feedwater combined with the failure of the feed-and-bleed cooling mode
- total loss of offsite power [LOOP], unavailability of all Diesel generators, and recovery of feedwater during the core melt phase.

The scenarios selected, as well as the results of calculations carried out with the MAAP computer code, are presented in Chapter S.2.2.2, which describes and justifies the operational requirements applicable to the method of depressurisation.

A capacity of about 900 tons/hr is sufficient to limit primary pressure to approximately 4 bars at pressure-vessel failure for the scenarios without late flooding. The existing pressuriser relief valves could therefore, in principle, be used for depressurisation. However, due to operating conditions specified for the EPR (elevated gas temperature, the need for the relief valves to remain open after a drop in primary pressure) and due to the design goal of "practically eliminating" high-pressure core melt, an additional diversified discharge valve, as well as an isolation valve, is implemented, whose capacity is equivalent to the existing pressuriser relief valves. The valves are opened manually by the operator in the event of elevated primary temperature (approximately 650°C at the core outlet).

c) Mitigation of the consequences of reactor pressure vessel failure:

To prevent dispersal of a large portion of the corium, the valve capacity is designed so that the RCP [RCS] pressure will be well below 20 bars at the moment of reactor-vessel failure. The support structures for the reactor pressure vessel and the pit are designed to support loads corresponding to a vessel failure of 20 bars. In addition, the layout of the reactor pit is designed in such a way that only small openings exist between the pit and the primary-loop compartments to limit the risk of corium dispersal. Experiments are currently ongoing to verify the absence of significant corium dispersal (see Chapter B.5). In all events, with the depressurisation system described (see Chapter E.4.8), the pressure in the RCP [RCS] at the time of pressure-vessel failure is below 4 bar, if there is no late reflooding; otherwise the pressure at failure could be between 1 and 20 bar.

The layout of the reactor pit is described in Chapter F.2.6, where it is shown that there is no direct path between the pit and the containment.

Consequently the risk of significant early release associated with core melt situations at high pressure and direct heating of the containment are "practically eliminated".

1.6. BYPASS OF THE CONTAINMENT BY A STEAM GENERATOR TUBE RUPTURE

The strategy of depressurising the RCP [RCS] using a dedicated high-capacity system, together with other measures for preventing high-energy phenomena such as the violent deflagration of hydrogen, significantly reduces the likelihood of early containment failure. However other sequences exist, such as containment bypass sequences or accidents involving reactor shutdown with the containment open, which could lead to significant releases in the event of a core melt. In spite of measures taken for their practical elimination, they are taken into consideration because of their potentially significant consequences.

A large contributor to containment bypass is the SGTR, as an initiating or consequential event, when secondary cooling is lost and the operator does not activate the pressuriser dedicated severe accident valves.

As far as the SGTR initiating event is concerned, the major contribution to radiological release is associated with the overfilling of the secondary side of the GV [SG]; fission products leaving the containment have to pass through the secondary water thus leading to efficient retention.

On the other hand, an induced SGTR is unlikely because of the high reliability of RCP [RCS] depressurisation. The high degree of leaktightness of Main Steam Isolation Valves, with the control of the sealing surface, ensures that residual leaks in case of an SGTR core melt scenario are consistent with radiological targets.

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