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REVISION HISTORY

Issue	Description	Date
00	First issue for INSA information	04-12-2007
01	Integration of technical and co-applicant comments	29-04-2008
02	PCSR June 2009 update including: - text clarification - addition of references - Technical update to account for December 2008 Design Freeze including ARE [MFWS] design (section 1), and EVU [CHRS] interface with SED and several changes in tanks and valves on EVU [CHRS] (section 7)	29-06-2009
03	Consolidated Step 4 PCSR update: - Minor editorial changes - Text clarification - Clarification of cross references - References added or updated - Addition of clarification regarding recombiners (§4.2.2.1) - Addition of information regarding design of the IRWST filtration system	31-03-2011
04	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes - SDM references updated in line with design reference (UKEPR-I-002) - Annulus Ventilation System: Addition of pre-filter in the list of equipment and modification of functional flow diagram – CMF020 (§2)	12-10-2012

Continued on next page

REVISION HISTORY (Cont'd)

Issue	Description	Date
04 Cont'd	Consolidated PCSR update (cont'd): - Containment Heat Removal System: the stand-by position of the IRWST suction valves is now closed (rather than open) (§7.4.1) - Containment Heat Removal System: passive flooding of the corium – correction of text (§7.4.2.1) - Containment Heat Removal System: Section 6.2.7 - Figures 1 and 2 correction to remove analogue level meter to the leak collection tank	

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SUB-CHAPTER 6.2 – CONTAINMENT SYSTEMS

1. CONTAINMENT FUNCTIONAL REQUIREMENTS AND FUNCTIONAL DESIGN

1.1. INTRODUCTION

The EPR containment systems include the containment, the containment isolation system, and the containment combustible gas control system. These systems contain any radionuclides released from the fuel during postulated accidents preventing further release to the balance of the plant and to the environment. They also limit the accumulation of combustible gases generated during the accident.

The EPR reactor building consists of a cylindrical reinforced concrete outer shield building, a cylindrical pre-stressed concrete inner containment building with a steel liner, and an annular space between the two buildings. The shield building protects the containment building from external hazards. The reactor shield building functions as a secondary containment to prevent the uncontrolled release of radioactivity to the environment following a postulated design basis accident. The reactor shield building and annulus ventilation system EDE [AVS] are designed to provide the secondary containment function under the environmental conditions of normal operation, maintenance, testing, and postulated accidents, including protection against the dynamic effects associated with a design basis accident. The EDE [AVS] maintains the annulus at a sub-atmospheric pressure during normal operations and following postulated design basis accidents, establishing an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment.

The containment is designed to meet the functional requirements. The design bases for these functional requirements are described in more detail below. The containment is required to withstand the environmental and dynamic effects associated with both normal plant operation and postulated accidents. The containment is required to be a leak-tight barrier against the release of radioactivity to the environment. It is also required to accommodate the temperatures and pressures resulting from loss of coolant accidents (LOCA) and main steam line breaks (MSLB).

Lines that penetrate containment or directly connect to the containment atmosphere have one of the following isolation valve configurations:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment.

2. One automatic isolation valve inside and one locked closed isolation valve outside containment.

3. One locked closed isolation valve inside and one automatic isolation valve outside containment.

4. One automatic isolation valve inside and one automatic isolation valve outside containment.

This section describes the containment concept and the design requirements for the containment function provided by the reactor building and the peripheral buildings (safeguard buildings, nuclear auxiliary building, fuel building and the effluent treatment building).

The containment function is based on specific characteristics and provisions to limit activity releases in PCC-1 to PCC-4, RRC-A and RRC-B events and accidents. Calculations to determine the corresponding radiological consequences are described in Chapter 14 for PCC events and Chapter 16 for RRC-A and RRC-B accidents.

The containment function is provided by:

- The reactor building.
- Maintaining dynamic containment using ventilation and filtering equipment for the peripheral buildings surrounding the reactor building.
- Static containment characteristics which improve the leak-tightness of the buildings or specific rooms in the event of loss of ventilation systems. Improving leak-tightness reduces the flow rate necessary to provide an effective dynamic containment and thus makes the dynamic containment easier to maintain.

1.2. SAFETY REQUIREMENTS

1.2.1. Safety objectives

The purpose of the containment function is to prevent radioactive releases into the environment, particularly during an accident, whilst maintaining the general safety objectives. In contrast to previous generations of nuclear power plants, the EPR containment design addresses the possibility of core meltdown accidents involving low-pressure vessel rupture, and aims to prevent leakage from the reactor building into the environment in such accidents.

1.2.2. Statutory framework

Paragraphs A.1.1, A.1.2, A.1.3, A.1.4, B.1.4, B.1.4.1, B.1.4.2, B.2.1, B.2.3.5, D.2.4, E.2.2.5, E.2.2.6, G.1, G.2 and G.4 of the Technical Guidelines are applicable to the containment design (Sub-chapter 3.1).

1.2.3. Safety requirements

1.2.3.1. Safety requirements for PCC-1 to PCC-4 and RRC-A events and accidents

For accidents without core meltdown, the containment function must ensure that the radiological objectives defined in Sub-chapter 14.6 are met.

1.2.3.2. Safety requirements RRC-B accidents

For severe accidents (RRC-B), the containment function must ensure that the radiological objectives defined in Sub-chapter 16.2 are met.

For core meltdown accident sequences which may occur during shutdown states with the reactor building open, the reactor building must be capable of being closed with an acceptable reliability before the occurrence of a large radioactive release in the containment.

A large corium spreading area, where the corium is cooled, is provided to mitigate the consequences of core meltdown with low-pressure vessel rupture.

1.2.3.3. Safety requirements applicable to structures, systems and equipment contributing to the containment function

For the reactor building:

- The design parameters must be consistent with the values used to calculate the PCC and RRC scenarios which are taken into account in the plant design (including hydrogen combustion).

- The design must allow a grace period of at least 12 hours before the need to activate heat removal systems in the event of severe accidents (RRC-B).

- There must be no direct leakage paths from the reactor building into the environment.

- Water contamination must be avoided.

- It must be possible to remove decay heat from the reactor building without requiring the building to be depressurised.

- Accident sequences with core meltdown involving containment bypass (via steam generators or via circuits connected to the primary cooling system which exit the reactor building) should be practically eliminated, either by design, or by the use of operating provisions that ensure reliable isolation and prevent failures.

- Periodic tests on the containment leak-tightness must be performed.

For the peripheral buildings:

- A maximum leakage rate must be defined for the peripheral buildings that have a containment function, i.e. the nuclear auxiliary building, the safeguard buildings and the fuel building.

- Suitable means must be provided to re-establish the leak-tightness of the safeguard buildings after rupture of the safety injection system outside containment when it is being used in the residual heat removal mode.

- Containment must be ensured for accidents in the fuel building resulting in fuel pool boiling.

For the reactor building and the peripheral buildings:

The maximum leak rate for the buildings contributing to the containment function must be such that the radiological consequences satisfy the safety objectives.

The safety requirements for systems or functions contributing to the containment function, described above, are described in detail in the corresponding sections as follows:

- Containment isolation contributes to the containment function by minimising the radioactive releases into the atmosphere in the event of an accident with fission product release (section 3 of this sub-chapter).
- The combustible gas control system reduces the risk of hydrogen combustion (section 4 of this sub-chapter).
- The containment heat removal system EVU [CHRS] ensures control of the conditions inside the reactor building and the integrity of the foundation raft in the event of a RRC accident (section 7 of this sub-chapter).
- The safety injection system RIS [SIS] ensures decay heat removal in the event of a PCC event (Sub-chapter 6.3).
- The annulus ventilation system EDE [AVS] contributes to collecting and filtering potential leaks (section 2 of this sub-chapter).
- The leak rate control and testing system (EPP) contributes to collecting potential leaks associated with certain penetrations (section 5 of this sub-chapter).
- The ventilation systems which ensure dynamic containment for all the peripheral buildings contributing to the containment function and which maintain an air transfer direction that ensures potential radioactive releases to be avoided in shutdown states, taking into account the location of the fuel during these states (Sub-chapter 9.4).

1.3. CONTAINMENT FUNCTION

The EPR has a double-wall containment concept with a leak-tight steel liner. The structures and systems contributing to the containment function are as follows:

- The reinforced concrete foundation raft.
- The internal and external containment walls (also referenced as internal and external containments), and the space between these walls, called the inter-containment space or annulus. This space is maintained at a negative gauge pressure to collect leaks from the internal containment and to filter them before release into the environment via the stack.
- The systems required for isolating, retaining and monitoring leaks.
- The systems required for maintaining the pressure and temperature conditions inside the containment within the limits that are compatible with requirements for containment leak-tightness and structural integrity.

During shutdown states, the equipment hatch may be opened during part of state C or during states D, E and F. When the equipment hatch is open, containment is ensured by the ventilation systems for the global volume formed by the reactor building and the fuel building hall in front of the equipment hatch.

Depending on the plant state, systems and equipment contributing to the containment function include:

(1) The ventilation systems for filtering discharges.

(2) The containment isolation valves.

(3) The thermal inertia of the internal walls, structures and equipment inside the containment.

(4) The RIS [SIS] ensuring heat removal from the RCP [RCS] and IRWST.

(5) The containment hydrogen control system inside the containment.

(6) Actions for collecting potential leaks from the containment.

(7) The leak-tight steel liner.

(8) The EDE [AVS] annulus ventilation system.

(9) The reactor building iodine extraction/filtering equipment used during shutdown (carried out by the containment sweeping ventilation system EBA [CSVS]).

(10) The EVU [CHRS] system ensuring heat removal.

(11) The corium spreading area and its cooling system that protects the structural concrete against high temperatures that may occur in postulated core meltdown accidents.

(12) Actions preventing a high pressure reactor vessel rupture and large steam explosions outside the vessel.

For PCC-1 events, item (1) is required.

For PCC-2 transients, items (2) and (4) are required.

For PCC-3 and PCC-4 incidents and accidents, items (2), (3), (4), (6), (7), (8) and in some special situations (10) are required, combined with other accident control actions such as the isolation of possible radioactive releases in the containment and RCP [RCS] cool down to cold shutdown state. Item (9) is used to maintain dynamic containment in the reactor building when the equipment hatch is open.

For RRC-A accidents, item (10) is also used.

For RRC-B accidents, items (5), (10), (11) and (12) are also used.

1.3.1. General assumptions about the containment function

The internal containment with its leak-tight steel liner together with its internal walls provides the main component of the containment function.

The presence of the leak-tight steel liner on the internal containment wall enables the containment leak-tightness function to be differentiated from the pressure withstand function. Containment leak-tightness is ensured by the liner, whilst the resistance to pressure is provided by the pre-stressed concrete containment structure.

The containment design limits are: 5.5 bar and 170°C [Ref-1] [Ref-2].

Results obtained for PCC and RRC scenarios taken into account in the design (Sub-chapter 3.4) are consistent with these design limits.

The containment analysis results are described in this section for PCC sequences and other specific studies. Corresponding results for RRC-B sequences and for studies of the hydrogen combustion risk are presented in Sub-chapter 16.2.

The maximum leak rate from the internal containment is 0.3% vol/day at design pressure and temperature (see Sub-chapter 3.3).

As a result of design provisions, there is no direct leakage into the environment (section 5 of this sub-chapter).

Leaks that bypass the EDE [AVS] system are discussed in section 5 of this sub-chapter.

Leaks collected in the annulus are filtered by EDE [AVS] filters before being released into the atmosphere via the stack. The EDE [AVS] safety filter retention effectiveness (before release from the stack) is [Ref-3]:

- 99.9% for elemental iodine,
- 99.9% for aerosols, including iodine in aerosol form,
- 99% for organic iodine.

In addition, the activity released into the atmosphere in the event of accidents is reduced, because:

- The EDE [AVS] flow rate is low.
- Fission products entering the annulus decay and are deposited within the structure before they are released into the stack.

These provisions are such that the radiological consequences in the case of PCC and RRC events and accidents are within corresponding radiological limits. The results of the radiological consequence assessments are presented in Chapters 14, and 16, respectively for PCC, RRC-A and RRC-B events and accidents.

For severe accidents RRC-B, the design assumes a grace period of at least 12 hours before the EVU [CHRS] is brought into operation (Sub-chapter 16.2).

The negative pressure maintained in the annulus in normal operation is such that in the event of an accident, a significant grace period is available before the negative pressure is lost, even in the complete absence of ventilation (section 2 of this sub-chapter).

The probability of accident sequences with core meltdown and containment bypass is reduced to an insignificant level, as described in Chapter 16.

1.3.2. Design of the containment penetrations

Containment isolation valves are provided for all fluid system penetrations. These isolation valves are either locked closed or are closed automatically in the event of an accident except where systems are necessary for accident management, e.g. the RIS [SIS]. Generally, duplicate isolation valves are installed in ventilation systems and systems containing fluids; one on the inside and one on the outside of the double containment.

A single isolation valve is provided if the following apply (section 3 of this sub-chapter):

- The fluid system does not contain primary coolant and includes a sealed sheath outside the containment which is designed to withstand the containment design pressure.

- The fluid system does not contain primary coolant and includes a sealed sheath inside the containment which is protected against hazards.

1.3.3. Prevention of containment bypass

The various potential leakage paths from the reactor building into the environment are shown in Figure B.

1.3.3.1. Potential leaks via the internal containment wall

The leak-tight steel liner ensures acceptable leak-tightness of the internal containment.

Potential leakage paths are mainly associated with specific features of the liner, i.e. welds between the steel liner plates and the penetrations. After passing through the internal containment concrete, any leaks that are discharged into the annulus are then released into the stack via the filters. The corresponding leakage path is shown as path 1 in Figure B and is also shown by the arrow in the annulus in Figure A.

1.3.3.2. Penetrations

Potential leakage paths through the penetrations are shown in Figure A.

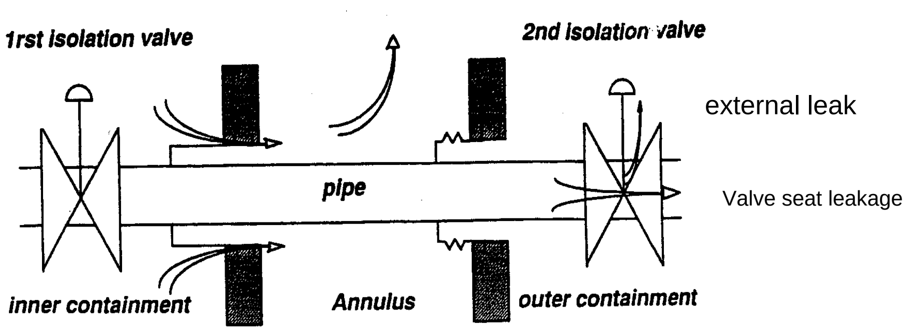


FIGURE A: IDENTIFICATION OF POTENTIAL LEAKAGE PATHS THROUGH A PENETRATION

There is no leakage from the annulus into the environment through the external containment because the annulus is maintained at a negative pressure.

In general, two isolation valves are provided for penetrations containing fluids (section 3 of this sub-chapter). Failure of the first (internal) isolation valve could potentially result in two leakage paths via the second (external) isolation valve: a leakage via the stuffing box, if the valve has one (an "external" valve leak), and a leakage via the valve seat (an "internal" leak).

Potential leaks through penetrations communicating with the peripheral buildings

There are penetrations where the internal valve is directly open to the containment atmosphere (e.g. hatches, heating ventilation and air conditioning (HVAC) penetrations, transfer tube). For these penetrations, if a potential leak via the internal valve is judged to have a potentially significant radiological impact, the external valve is equipped with a device that collects any potential leak from the gland and routes it to the annulus where it is filtered before being released via the stack. The corresponding leakage path is shown as path 1 in Figure B.

For similar penetrations that have zero or limited radiological impact, containment by the pipe connected to the penetration external valve is considered sufficient to prevent a significant radiological release into the peripheral buildings (in most cases the pipe will be full of water).

In all cases, and whether or not a device is installed to collect a leakage from the external isolation valve, the peripheral buildings provide an additional containment barrier.

Potential leaks via the equipment hatch and air locks

The equipment hatch and air locks that provide access for equipment and staff have a leak prevention system for collecting possible leaks (section 5 of this sub-chapter).

Penetrations that do not communicate with peripheral buildings

The only penetrations in this category are the main steam line and main feed line penetrations.

Provided the integrity of the SG tubes is maintained, the steam and feed penetrations do not perform a containment function. Under these circumstances, containment is provided by the secondary cooling system boundary. In the event of SGTR, the isolation valves in the secondary systems contribute to the long-term containment function.

All reasonably practicable measures have been applied to avoid potential direct leakages into the environment.

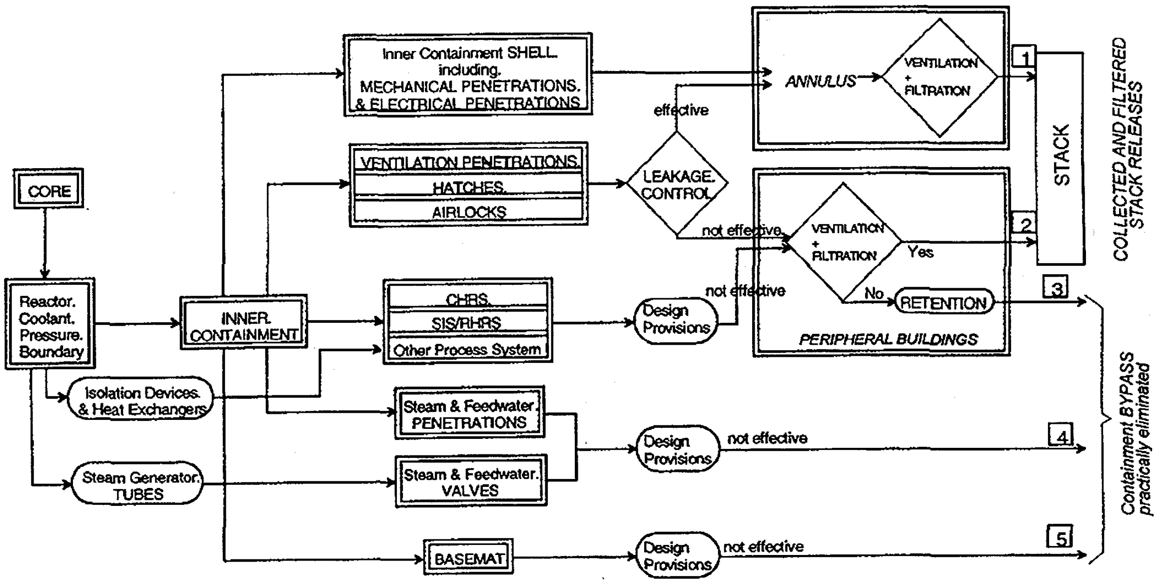


FIGURE B: SCHEMATIC DIAGRAM OF POTENTIAL LEAKAGE PATHS FROM THE REACTOR BUILDING TO THE ENVIRONMENT

1.3.3.3. RIS [SIS] and EVU [CHRS]

In certain accidents, the RIS [SIS] and EVU [CHRS] systems transport contaminated water outside the containment. This is discussed below.

1.3.3.4. Foundation raft

All reasonably practicable measures have been taken to exclude potential direct leaks into the environment via the foundation raft. In the event of accidents, specific measures are provided to ensure leak-tightness with respect to potential liquid leaks. Most of the lower part of the reactor building is occupied by the IRWST, which has a leak-tight liner, and the corium spreading area, which is also leak-tight. The leak-tight steel liner covering the internal walls of the inner containment extends to the interface between the foundation raft and the internal structure of the reactor building. Also, the presence of water on the foundation raft forms a barrier, preventing radionuclides from being transferred into the environment. The spreading area that collects the corium in the event of core melt accident leading to vessel rupture comprises specific layers for corium stabilisation. A corium cooling system beneath the spreading area protects the reactor building foundation raft from high temperatures.

These provisions enable foundation raft integrity to be maintained (i.e. significant cracking is prevented) even in the event of a severe accident (section 6 of this sub-chapter).

1.3.3.5. Conclusion

The containment is designed to prevent direct leakage from the containment to the environment throughout the life of the plant. Any leakage that does occur is collected and filtered before release to the environment. Hence the probability of direct release into the environment is insignificant.

1.4. CONTAINMENT FUNCTION OF THE PERIPHERAL BUILDINGS AND THE EFFLUENT TREATMENT BUILDING

1.4.1. Interface with the reactor building following activity release inside the reactor building

1.4.1.1. Circulation of contaminated water outside the containment

During certain accidents (e.g. severe accidents) the RIS [SIS] and EVU [CHRS] transport contaminated water outside the containment. Therefore, specific measures are provided for these systems to maintain the containment function.

1.4.1.1.1. *Specific measures with regard to leakages*

Specific measures are applied to parts of the systems that are likely to be potential sources of leakage outside the containment e.g.:

- The functional characteristics of the EVU [CHRS] cooling system are such that leaks from heat exchangers are precluded (section 7 of this sub-chapter).
- Specific leak detection provisions are made to isolate each RIS [SIS] and EVU [CHRS] train if a leak is detected (section 7 of this sub-chapter for the EVU [CHRS] and Sub-chapter 6.3 for the RIS [SIS]).
- A specific isolation device is fitted to the pipes between the IRWST and the RIS [SIS] and EVU [CHRS] pumps. The pipework between the IRWST and the isolation valves outside the containment is contained in a leak-tight protective sheath (section 7of this sub-chapter for the EVU [CHRS] and Sub-chapter 6.3 for the RIS [SIS]).

The sections of these systems outside the reactor building are located in specific rooms in the peripheral buildings. If one of the systems fails, the containment function is provided by peripheral buildings.

1.4.1.1.2. *Long-term reparability*

To ensure containment of the radioactive substances over the long term, it is possible to repair:

- the RIS [SIS] providing core cooling
- the EVU [CHRS] providing the containment heat removal in the event of a severe accident.

Despite the presence of highly contaminated water, (when the EVU [CHRS] is operating) or moderately contaminated water (when the RIS [SIS] is operating), it is possible to repair the pumps after isolating the main system by closing the corresponding suction valve in the IRWST and draining and rinsing the connected pipework sections. Similarly, access to the dedicated EVU [CHRS] intermediate cooling system is possible without risk of contamination by pressurising the system and closing the power-operated valve.

A preliminary list of requirements for accessibility to repair the systems used to maintain the final state in the long term is provided in Sub-chapter 12.5.

1.4.1.1.3. *Re-injection of contaminated effluents into the reactor building after an accident*

Systems are provided to enable highly contaminated effluents to be re-injected into the reactor building following an RIS [SIS] leak in a safeguard building post LOCA.

In addition, although an EVU [CHRS] leak is not expected to occur after a severe accident, means are provided to re-inject contaminated effluent back into the reactor building.

The effluent re-injection function into the reactor building is provided by the RPE [NVDS] system during PCC and RRC-A events (Sub-chapter 11.4).

For long term maintenance of the EVU [CHRS] system following a severe accident, the contents of the circuits are pumped back into the reactor building using the EVU [CHRS] circuits, and the circuits are then rinsed.

1.4.1.2. Leaks through containment penetrations

Most leakage is collected in the annulus, which is kept at sub-atmospheric pressure by the annulus ventilation system (EDE [AVS]). This system is also used to route any leakage through HEPA and iodine filters in series to the stack, from which they are released to the environment in a controlled manner.

Leakages which are not collected in the annulus enter the peripheral buildings and are filtered before being released to the environment.

Thus the peripheral buildings (safeguard buildings, nuclear auxiliary building and fuel building) contribute to the containment function.

A static global leak-tightness criterion for each peripheral building (safeguard buildings, nuclear auxiliary building and fuel building) is defined at 0.3% vol/day (see Sub-chapter 3.3). Provisions implemented to ensure that this criterion is met, are as follows:

- Semi-leak-tight doors are installed on the boundary of the controlled area (opening to the outside or located between the controlled and non-controlled areas). These semi-leak-tight doors are equipped with leak-tight seals on three sides and a friction brush at the floor.

- Placing dampers on the air supply side in the nuclear auxiliary building.

These provisions give significant margins and therefore a visual inspection of the equipment is sufficient to maintaining its efficiency.

1.4.2. LHSI rooms in safeguard buildings 1 and 4

In the event of a postulated RIS/RRA [SIS/RHRS] break in safeguard building divisions 1 and 4 then, because the primary circuit temperature could exceed 100°C (only these two trains are operable above 100°C), the corresponding rooms (e.g. LHSI pump and heat exchanger rooms) could become pressurised. This could lead to the loss leak-tightness and release of radioactive material into the atmosphere without the possibility of containment within the rooms in the long-term. To avoid this, dedicated devices are provided:

- Protection of the LHSI pump and heat exchanger rooms in safeguard buildings 1 and 4 is provided by reinforced doors which are designed to withstand a differential pressure of 120 mbar. See Sub-chapter 3.3 for a description of the safeguard buildings.

- Limitation of the pressure increase using a dedicated device: In the event of a break, the released steam is routed to an opening in an area that is normally isolated. A device is provided that releases the pressure when the pressure difference exceeds 50 mbar. This device is designed so that containment isolation can be restored after a period of time following the break isolation and to restore the dynamic containment using the ventilation systems.

1.4.3. Fuel building

The following measures are applied to ensure the containment function provided by the Fuel Building:

- The ventilation systems provide dynamic containment following a fuel handling accident, regardless of whether it occurs in the fuel building or in the reactor building, when the equipment hatch is open (Sub-chapter 9.4).

- The fuel building provides static containment in the event of the atmosphere becoming saturated as a result of high temperatures in the spent fuel pool following loss of the two main PTR [FPCS] trains.

- The design ensures that melting of spent fuel is practically eliminated (Sub-chapter 16.3) and the probability of spent fuel pool boiling is sufficiently low (Sub-chapter 16.3) that the fuel building does not need to provide a containment function.

1.4.4. Nuclear auxiliary building

Because the nuclear auxiliary building ventilation system is not F1 classified, then post accident it is assumed that the air from the rooms is ventilated through a High Efficiency Particulate Air (HEPA) filter before being discharged into the stack. Aerosol retention in these filters is conservatively ignored. Nevertheless, the analysis of bounding accidents in the nuclear auxiliary building (i.e. rupture of a TEG [GWPS] or RCV [CVCS] line), shows that the resulting discharges are much lower than the allowable limits for each category of accident (Chapter 14).

For the special case of an earthquake causing multiple pipe ruptures in the nuclear auxiliary building, containment of the radiological releases is provided statically by the isolation of the air inlet dampers and closure of the dampers in the extraction lines located at the building exhausts. The radiological consequences of an earthquake affecting the nuclear auxiliary building are presented in Sub-chapter 14.6 and are shown to meet the dose objectives even assuming the overall leak rate being an order of magnitude greater than assumed for non-seismic events (i.e. 5% vol/day) taking into account the inventory of contaminated liquid and gas discharged into the building.

1.4.5. Effluent treatment building

The containment function extends to the Effluent Treatment Building, which is equipped with an F2 classified Waste Building Treatment Ventilation system (Sub-chapter 9.4). In normal operation, extraction is via a HEPA filter. If activity is detected, extraction is automatically switched to the iodine traps. If the ventilation system is unavailable, static containment is applied.

1.5. DEFINITION AND ANALYSIS OF THE CONTAINMENT LOAD COMBINATIONS

1.5.1. Definition of the containment load combinations

The load combinations considered for containment design are derived from accident sequences taken into account during the design process.

Containment load combinations are calculated for the following conditions:

- LB LOCA Loss of Coolant Accident - Large Break
- 2A-LOCA Loss of Coolant Accident - Guillotine Break
- SLB Steam Pipe Rupture
- 2A-SLB Steam Pipe Rupture - Guillotine Break
- Severe accident

Pressure and temperature calculations associated with the first four cases are presented in this section. The calculation of pressures and temperatures associated with severe accidents is presented in Sub-chapter 16.2.

As discussed in this section, (P, T) decoupling curves have been used for the containment design. The combination of loads considered for the containment design (steel liner and concrete internal containment) is presented in Sub-chapter 3.4.

1.5.2. Definition of Mass and Energy Release in Incidents and Accidents as Design Basis for Containment

The consequences of the Mass and Energy Release (MER) on the behaviour of the containment wall are examined for the PCC Large Break Loss of Coolant Accident, 2A-LOCA, PCC Steam Line Break, 2A-SLB, and severe accidents.

1.5.2.1. Large break LOCA (PCC)

1.5.2.1.1. *Introduction*

This section summarises results with respect to mass and energy release (MER) into containment in the event of a complete rupture of the pressuriser surge line (largest LOCA break size in PCC).

1.5.2.1.2. *Calculation model*

The calculation is performed using the CATHARE SB LOCA/IB LOCA methodology. The input model uses the same nodalisation as used in the Chapter 14 PCC studies: the evolution of the accident is also described there. The containment code CONPATE 4 is coupled with CATHARE to achieve a more precise result with respect to the MER. In this way, both the RCP [RCS] response and the containment response are calculated simultaneously. The CONPATE 4 results (containment backpressure and MHSI / LHSI temperature) are used as a boundary condition for CATHARE.

The CATHARE and CONPATE 4 codes are described in Appendix 14A.

1.5.2.1.3. *Initial and boundary conditions*

CATHARE modelling of primary/secondary systems

Conservative assumptions with regard to the MER are used. The most important are (see Sub-chapter 14.5, Section 14.5.6 – Table 1):

- Maximum decay heat (ORIGEN/S results + uncertainties).
- Maximum initial reactor power (102% nominal power).
- Loss of off-site power with RT signal.
- Loss of 1 diesel due to single failure (loss of 1 MHSI + 1 LHSI + 1 ASG [EFWS]).
- Loss of 1 diesel due to preventive maintenance (loss of 1 MHSI + 1 LHSI + ASG [EFWS]).
- Protection and safeguard actions limited to F1 systems.
- Uncertainties assumed for F1 I&C functions credited (as in PCC-4 core calculations, Sub-chapter 14.5).
- Minimum effectiveness of F1 fluid systems assumed (as in PCC-4 core calculations, Sub-chapter 14.5).
- No manual actions credited.

CONPATE 4 modelling of the containment

Conservative assumptions are used for maximum containment pressure and maximum IRWST temperature. Specifically the containment volume, IRWST volume, and the heat transfer coefficient of atmosphere/water are minimised.

1.5.2.1.4. *Analysis results*

The thermal hydraulic behaviour of the RCP [RCS] is described in the section covering the PCC Large Break LOCA (Sub-chapter 14.5).

The MER related to the LB LOCA "rupture of the PZR surge line" have not been updated for the UK EPR PCSR. Instead the MER calculated in the BDR-99 analysis are used to assess the containment pressure and temperature behaviour.

The MER related to the EPR_{4500} characteristics given in the PCSR, are bounded by the MER calculated for the EPR_{4900} characteristics given in the BDR-99 (PCSR Appendix 6, section 6.2.1.5.2.1) as the:

- Core power level is lower (-8% in the EPR_{4500}).
- Initial RCP [RCS] water inventory and temperature are similar.
- Initial SG water inventory is smaller (-8% in the EPR_{4500}), initial SG water temperature is slightly higher, resulting in a lower initial SG energy.
- RIS [SIS] injection flow rate is identical at low pressure level, is slightly increased at high pressure level (MHSI with higher delivery pressure, accumulators and LHSI unchanged, in the EPR_{4500}).
- IRWST initial water content and temperature are unchanged, providing a larger heat sink reserve in the EPR_{4500} with respect to the core power level.

Immediately after the break, during the RCP [RCS] depressurisation/blowdown phase, the MER of the EPR_{4500} is similar to that of the EPR_{4900} (as they have similar initial mass and energy content).

In the long term when the RCP [RCS] pressure is close to the containment pressure, after the end of RCP [RCS] blowdown, the MER of the EPR_{4500} are less limiting for the containment pressure and temperature behaviour than the MER of the EPR_{4900} (lower core power, larger LHSI heat removal capacity, larger IRWST heat sink).

1.5.2.2. 2A-LOCA

1.5.2.2.1. *Introduction*

This section summarises results with respect to the MER into containment in the event of a double-ended (2A) guillotine break of a main coolant line cold leg. This break is considered despite the RCP [RCS] break preclusion concept as a defence-in-depth measure to check the capability of the containment to withstand the largest RCP [RCS] pipe break under realistic conditions.

1.5.2.2.2. *Calculation model*

The analysis is performed using the CATHARE computer code. Containment pressure is provided by the computer code CONPATE 4, which runs interactively with CATHARE. In this way, both the RCP [RCS] response and the containment response are calculated simultaneously.

CATHARE and CONPATE 4 codes are described in Appendix 14A.

1.5.2.2.3. *Initial and boundary conditions*

CATHARE modelling of primary/secondary systems

Best-estimate assumptions are used; the most important are [Ref-1]:

- Initial rated reactor power 100%.

- Decay heat corresponding to ORIGEN/S results (without uncertainties).
- All trains of safety injection system available (no single failure, no preventive maintenance).
- Off-site power available.

CONPATE 4 modelling of the containment

In general, best-estimate initial and boundary conditions are used [Ref-1], i.e. containment volume and heat transfer areas in containment correspond to nominal values. Assumptions, which maximise the MER are made for those parameters which do not have a clear best-estimate value: e.g. heat transfer coefficient between atmosphere and IRWST, LHSI heat exchanger characteristics.

1.5.2.2.4. *Analysis results*

The MER related to the 2A cold leg break have not been updated for the UK EPR PCSR. Instead the MER calculated in the BDR-99 analysis are used to assess the containment pressure and temperature behaviour.

The MER related to the EPR_{4500} characteristics given in the PCSR, are bounded by the MER calculated for the EPR_{4900} characteristics given in the BDR-99 (PCSR Appendix 6, section 6.2.1.5.2.2): for the same reasons as given for the PCC LB LOCA above.

1.5.2.3. Steam line break (PCC)

1.5.2.3.1. *Introduction*

Application of the break preclusion concept for the main steam lines (MSL) inside the containment building, means that the PCC steam line break (SLB) is limited to rupture of a pipe connected to the MSL (sensor or venting line). The associated break size is far below the MSL cross section and the SG-outlet flow limiter area. The resulting MER are, therefore, much lower than the MER which could result from the rupture of the MSL.

In accordance with the approach used in the PCSR for the PCC SLB-analyses related to the core behaviour (Sub-chapter 14.5), the PCC SLB assumed in the PCSR for the containment behaviour is the double-ended guillotine break of a MSL located at SG outlet. This conservative approach does not take account of the break preclusion concept for the MSL design inside the containment, which excludes the rupture of the MSL as a PCC accident (and as a RRC-A one). This conservative approach is used to simplify the calculation:

- Consideration of the MSLB accident bounds all of the "MSL-connected pipe break" PCC events and the "feed line break" PCC event.
- The BDR-99 PCC calculation of the 2A SLB, performed without claiming the break preclusion concept (not implemented in the EPR_{4900} design covered by the BDR-99), is used directly to determine a bounding MER for the EPR_{4500} design covered by the PCSR, without the need for a specific EPR_{4500} calculation.

As a result, for the PCSR, the double-ended guillotine break of a main steam line inside the containment building is selected as the bounding break case for the containment loads for all PCC events affecting the secondary system. The general description of the accident is given in the section covering the PCC SLB (see Sub-chapter 14.5).

An MSLB leads to a MER that depends on initial and boundary conditions, and on the single failure postulated for the safety grade systems. Sensitivity studies are then performed to find the most onerous case:

- Sensitivity to the fuel management (UO2, MOX).
- Sensitivity to the initial level of power (from 0% to 102% of full power).
- Sensitivity to one of the following single failures:
    - Failure to drop the highest worth RCCA; stuck in its fully withdrawn position.
    - Failure to close a main feed water isolation valve on the affected SG.
    - Failure to close the main steam isolation valve (VIV [MSIV]) on the affected SG.

1.5.2.3.2. *Calculation model*

The computer code used is THEMIS (Appendix 14A). This code calculates the MER through the break from the affected SG, and from the three non-affected SG through the main steam header, up to time of VIV [MSIV] closure. In the event of a failure to close the VIV [MSIV] on the affected SG, the blowdown of the main steam header is taken into account.

The analysis methodology for the core behaviour calculation is described in the PCC SLB section (Sub-chapter 14.5), except that the core neutronics is directly calculated using a conservative "point kinetics model". Actually, only the total core power is of interest for the MER calculations. In this methodology, calculation assumptions are adapted to ensure conservative MER results, as indicated below.

1.5.2.3.3. *Initial and boundary conditions*

a) General assumptions (see Sub-chapter 14.5, section 2)

These assumptions correspond to a conservative methodology for the calculation of the MER for a SLB, aimed at maximising the MER results:

- Maximisation of the initial energy in the primary side:

Uncertainties are applied as follows:

- + 2% on core power.
- + 2.5 bar on pressuriser pressure.
- + 2.5°C on average primary temperature.

Sensitive heat from primary metal structures and power from the primary pumps are taken into account.

- Maximisation of the heat transferred from the primary circuit to the secondary circuit:
    - The SG tubes are clean and none are plugged.
    - The mechanical primary flow rate is considered.
    - The RCP [RCS] pumps are kept in operation.
    - Conservative neutronics data is used with respect to the core return to power.
    - The maximum decay heat curve is used.
- Maximisation of the mass and energy release from the secondary side:
    - +10% uncertainty is applied to the initial SG water inventory, which includes uncertainties in the SG level control and in the SG water temperature.
    - The ARE [MFWS] temperature is maximised: nominal + 5°C.
    - The flashing effect[1] in the main feedwater line is taken into account.
    - A constant back pressure of 1 bar is assumed in the containment atmosphere.
    - Perfect moisture separation is assumed at the steam generator outlet, leading to a break flow quality of 1 (pure steam flow at the break maximises the total energy released).
    - Blowdown of the main steam header is taken into account until VIV [MSIV] closure is achieved.

b) Specific assumptions (see Sub-chapter 14.5, section 2)

The I&C protection signals are described in the SLB section (see Sub-chapter 14.5). The effective protection actions are taken into account as follows, (with maximum occurrence times considered conservatively):

- Reactor trip (for SLB at power):
    - Performed by RCCA drop, F1A qualified.
    - On F1A signal “SG pressure drop > MAX 1", with 0.9 seconds signal delay, 0.3 seconds RCCA gripper release, and 5 seconds RCCA dropping time.
- MS isolation:
    - Performed by closing the four VIV [MSIV], F1A qualified.
    - On F1A signal “SG pressure drop > MAX 1", with 0.9 seconds signal delay and 5 seconds VIV [MSIV] closing time (step wise).

[1] After Main Feed Water isolation, a volume of water remains in the ARE [MFWS] pipes at the initial ARE [MFWS] temperature and SG pressure. As the SG pressure decreases below the saturation pressure of the ARE [MFWS], this water flashes to steam in the ARE [MFWS] pipes and flows into the affected SG.

- ARE [MFWS] isolation:
  - o Performed by closing the three ARE [MFWS] isolation valves[2] on each SG, all F1A qualified:
    - the high load isolation valve (MFIV-HL) closes the high load line,
    - the low load isolation valve (MFIV-LL) closes the low load line,
    - the main isolation valve (MFIV) closes the main FW line.
  - o The MFIV-HL is closed on F1A signal "SG pressure drop > MAX 1", with 0.9 seconds signal delay and 15 seconds valve closing time (step wise),
  - o The MFIV-LL and the MFIV are closed on F1A signal "SG pressure drop > MAX 2", with 0.9 seconds signal delay and 15 seconds valve closing time (step wise).

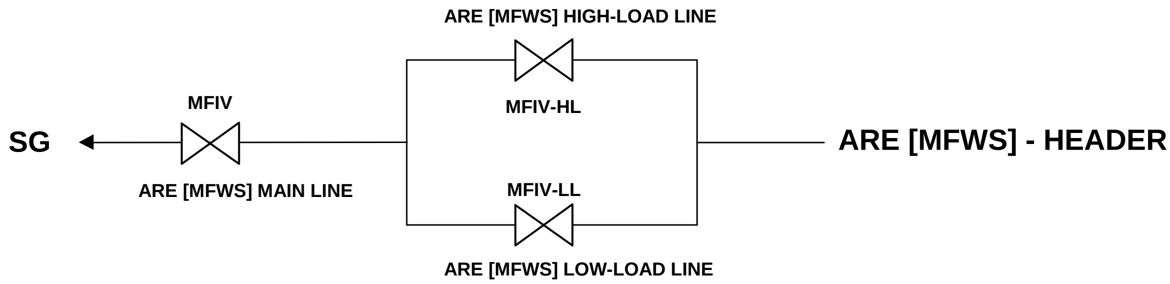


Figure C: ARE [MFWS] isolation valves

The ARE [MFWS], AAD [SSS] and ASG [EFWS] flows entering the affected SG prior to their isolation are bounding values:

ARE [MFWS] flow rate[3]

For an initial power state higher than 20% Nominal Power (NP), the ARE [MFWS] flow rate entering the affected SG is:

- o ~160% of nominal ARE [MFWS] flow before main steam header isolation.
- o ~240% of nominal ARE [MFWS] flow after main steam header isolation, and before ARE [MFWS] high-load line isolation.
- o ~130% of nominal ARE [MFWS] flow after main steam header isolation, and ARE [MFWS] high-load line isolation, 0 kg/s after ARE [MFWS] main line and low-load line isolation.

[2] This ARE [MFWS] isolation valve configuration is claimed independently of the EPR final configuration. With respect to MER, this is the most limiting case because only one isolation valve closes on the occurrence of the first isolation signal (MAX 1), which results in the non-isolation of the high-load line until the occurrence of the second isolation signal (MAX 2) in the event of a single failure when the high-load isolation valve attempts to close.

[3] Following a more precise definition of the Conventional Island, these values will be redefined

For an initial power state lower than 20% NP, the high-load line is closed in normal operation, and the ARE [MFWS] flow rate entering the affected SG is:

- ~130% of nominal ARE [MFWS] flow before ARE [MFWS] main line and low-load line isolations, 0 kg/s after ARE [MFWS] main line and low-load line isolation

AAD [SSS] flow rate: The start-up system is not considered in this study because the ARE [MFWS] system remains in operation until total ARE [MFWS] isolation. The ARE [MFWS] flow rate bounds the AAD [SSS] flow rate.

ASG [EFWS] flow rate: The emergency feed water flow rate into the affected SG is taken at the constant value of 200 m^3/h, with a maximum temperature of 50°C, from the onset of the accident. This is cancelled by the operator 0.5 hours after reactor trip. The value of 200 m^3/h does not currently take credit for the ASG [EFWS] flow active limitation; this is excessively conservative since this failure is assumed to be coincident with the already assumed single failure.

Note: The design of the ARE [MFWS] system, given in Figure C above, has changed since the Basic Design. The modifications are as follows:

- The current ARE [MFWS] design is presented in Sub-chapter 10.6; there is redundancy on the isolation valves (MFIVs). This prevents any failure of isolation of the main feedwater system.
- The MFIV closure time is longer on the current EPR studies.

Considering the BDR-99 analysis was carried out at 4900 MW, with a larger volume of water in the steam generator and a failure of the ARE [MFWS] MFIV, the study presented in the PCSR is the most conservative case, both in terms of mass and energy released and meeting the safety criteria.

c) Single Failure

Sensitivity to the following single failures is performed, in order to identify the most limiting:

- Single failure to close of the MFIV-HL valve of the affected SG:

As a consequence, the isolation of the ARE [MFWS] high-load line fails. The ARE [MFWS] flow entering the affected SG remains at ~240% NF (instead of ~130% NF) until the isolation of the ARE [MFWS] main line and ARE [MFWS] low-load line is achieved.

- Single failure of the highest worth RCCA:

The reactivity is calculated considering all RCCA inserted in the core except the highest worth RCCA, which is assumed to be stuck in its fully withdrawn position.

- Single failure to close of the VIV [MSIV] of the affected SG:

The VIV [MSIV] of the affected SG remains open. Consequently, the steam contained in the main steam header flows into the containment. The main steam header includes the main steam lines up to the main steam bypass and the turbine inlet pipes up to the turbine stop valves. The total main steam header volume is assumed to be 300 m^3 [Ref-1].

d) Preventive maintenance

Preventive maintenance has no consequence for the MER analysis in the event of a SLB, as the protective actions are limited to reactor trip and valve actuations, (isolation functions) which are not impacted by the preventive maintenance.

e) Break assumption

The break area on the SG side corresponds to the flow area of the SG flow-limiter located at the SG outlet, i.e. 1,300 cm². The break area on the opposite side corresponds to the flow area of the fully open VIV [MSIV], i.e. 3,200 cm² (this kind of valve presents an area corresponding to a diameter between 19” and 23”. However, a bounded section value is taken into account with regard to pressure and temperature in the containment. It corresponds to a diameter of 25”, i.e. an area of 3,200 cm²).

1.5.2.3.4. *Analysis results*

Sensitivity studies have shown that the most limiting configuration corresponds to the following conditions:

- 0% for initial power (+2% considered, including decay heat at hot shutdown).
- Failure to close the VIV [MSIV] on the affected SG.

The MER for the 2A SLB under PCC analysis conditions have not been updated for the UK EPR PCSR. Instead the MER calculated in the BDR-99 analysis are used to assess the containment pressure and temperature behaviour.

The MER related to the EPR_{4500} characteristics given in the PCSR, are bounded by the MER calculated for the EPR_{4900} characteristics given in the BDR-99 (PCSR Appendix 6, section 6.2.1.5.2.3):

- Initial SG water mass is smaller (-8% in the EPR_{4500}), initial SG water temperature is slightly higher (+3°C in the EPR_{4500}), resulting in a lower initial SG energy (-8% in the EPR_{4500}). This is the major contributor to the containment overpressure and over-temperature peaks.
- The longer closing time of the ARE [MFWS] isolation valves in the EPR_{4500} (15 seconds instead of 10 seconds), leads to 3 te more water being injected into the affected SG. However this does not change the assessment.
- Other parameters are similar or less onerous: initial RCP [RCS] water inventory and temperature are similar, ARE [MFWS] flow rate is lower (due to lower core power level), ASG [EFWS] flow rate is lower, core return to power is expected to be similar or lower (higher shutdown margin credited, similar neutronics coefficients).

1.5.2.4. 2A-SLB

Even though the break preclusion concept is applied to the MSL inside the containment, its failure is considered as a defence-in-depth measure to check the capability of the containment to withstand the largest MSS pipe break under realistic conditions. In the PCSR, this 2A-SLB case under realistic conditions is bounded by the 2A-SLB case treated under PCC conditions above (conservative case retained for PCC SLB).

1.5.2.5. Severe accidents

The containment pressures and temperatures were calculated assuming total core rupture following a large LOCA based on a new reference model of the containment called the "Basic Containment Model".[Ref-1]

1.5.3. List of Inputs for P and T Calculations under LB LOCA, 2A-LOCA, SLB and 2A-SLB Conditions

P and T calculation code

The PAREO 9 computer code has been used. It uses the same physical models as PAREO 8 code which is described in Appendix 16A.

<u>Containment data [Ref-1]</u>

The free volume of the containment is 80,235 m^3. The volume of water in the IRWST is assumed to be 1,920 m^3 for the maximum containment pressure calculation.

The following material properties have been assumed:

Material	Specific Mass (kg/m^3)	Thermal Conductivity (W/m °C)	Specific Heat (J/kg °C)
Steel	7,770	45	502
Concrete	2,306	1.73	880

The steel properties are usually taken into account.

The concrete properties are taken into account for conservative studies.

The following characteristics are assumed for the steel liner:

- Coating thickness: 0.4 mm
- Steel liner thickness: 7.4 mm
- Generalised gap (air): 3 mm (this assumption is strongly conservative).

The impact of the steel liner on containment pressure and temperature calculations has only been analysed for the most limiting case, i.e. the steam line break.

<u>Initial conditions</u>

Uniform initial conditions have been assumed:

- Wall structures, atmosphere and IRWST temperature, T = 42°C.
- Containment pressure, P_{gaz} = 1.1 bar (conservative value).

Other constant conditions have been assumed:

- Outside temperature: $T_{outside}$ = 30°C.
- Ground temperature: T_{ground} = 20°C.

Heat transfer coefficient

A conservative heat transfer coefficient has been used between the containment atmosphere and the structures. This is normal for LOCA calculations.

1.5.4. Results of Analyses

1.5.4.1. P and T in the event of LB LOCA

The event analysed is the total rupture of the pressuriser surge line at its connection to the hot leg. This scenario, which is used as part of the containment design, is a PCC-4 LOCA.

Pressure and temperature results

Pressure and temperature in the containment are presented in Appendix 6 (BDR 99 extract - 6.2.1.5.4 - Figures 1 and 2) with:

- P_{gaz}: total containment pressure.
- P_{vap}: partial pressure of steam.
- T_{liq}: IRWST temperature.
- T_{gaz}: atmosphere temperature.
- $T_{sat}P_{vap}$: saturation temperature for the partial pressure of steam.

The maximum pressures and temperatures are 4.5 bar and 177°C, occurring at 150 seconds.

1.5.4.2. P and T in the case of 2A-LOCA

The event analysed is the guillotine break of a primary cooling system cold leg. This bounding accident is neither a PCC-4 nor a RRC-A sequence; it is a specific study which is taken into account in the containment design.

The MER were calculated in accordance with the 2A LOCA guidelines with realistic assumptions. The pressure and temperature calculations are performed with conservative assumptions defined below.

Pressure and temperature results

Pressure and temperature in the containment are presented in Appendix 6 (BDR 99 extract - 6.2.1.5.4 - Figures 3 and 4) with:

- P_{gaz}: total containment pressure.
- P_{vap}: partial pressure of steam.
- T_{liq}: IRWST temperature.
- T_{gaz}: atmosphere temperature.
- $T_{sat}P_{vap}$: saturation temperature for the partial pressure of steam.

The maximum pressures and temperatures are 4.3 bar and 182°C, occurring at 20 seconds.

1.5.4.3. P and T in the case of SLB

The event analysed is the 2A break of the main steam line. This case bounds the SLB PCC-4 accidents. The calculation relies on main feed water line isolation occurring at 16.1 seconds (signal + valve closure) after the steam line break [Ref-1].

This accident is the most limiting one identified previously in the BDR and has been selected to evaluate the effects of the steel liner on containment pressure and temperature.

Pressure and temperature results

Pressure and temperature in the containment are presented in Section 6.2.1 - Figures 1 and 2 (24 hour period) and in Section 6.2.1 - Figures 3 and 4 (2000 second period) with:

- P_{gaz}: total containment pressure
- P_{vap}: partial pressure of steam.
- T_{liq}: IRWST temperature.
- T_{gaz}: atmosphere temperature.
- $T_{sat}P_{vap}$: saturation temperature for the partial pressure of steam.

The P and T curves show two peaks. The maximum values obtained are:

- 5.2 bar and 183°C, occurring at 324 seconds for the first peak corresponding to depressurisation of the affected SG.
- 4.8 bar and 138.5°C, occurring at 1,800 seconds for the second peak, limited by isolation of the ASG [EFWS] of the affected SG.

Compared with the results in the BDR, the impact of the steel liner is limited to the second peak which is 300 mbar and 3.5°C greater. This is due to the very conservatively assumed generalised gap which “insulates” the concrete from the liner.

After 30 minutes, there is no further significant release into the containment and the energy present in the containment is removed by the structures. Pressure and temperature decrease to 1.3 bar and 62.5°C 24 hours after the start of the accident.

The impact of the steel liner on the P and T containment loads is very limited even assuming a total air gap between the steel liner and the concrete. This confirms that it is not necessary to re-perform the calculations for all cases.

The maximum temperature for containment walls is lower than the saturation temperature. This indicates that a film of condensed water will form on the containment walls, where the temperature does not exceed 140°C.

1.5.4.4. P and T in the event of 2A-SLB

Even though the break preclusion concept is applied to the main steam line inside the containment, this event is considered in order to ensure the containment can withstand the largest main steam pipe break assuming realistic conditions, in accordance with the defence-in-depth methodology.

The P and T loads in the containment calculated for a 2A-SLB with realistic modelling conditions are bounded by the PCC-4 2A-SLB P and T loads presented above.

1.5.4.5. P and T in the event of a severe accident

The severe accident sequences taken into account are:

- LB LOCA: guillotine break of the surge line
- SB LOCA: 50mm break in cold leg
- SBO

The containment response in severe accidents is examined based on the most unfavourable scenario (355 mm break of the surge line at the level of the hot leg without safety injection). The calculations are performed using the COCOSYS code which includes modelling of the EPR double containment. The calculations are described in detail in Sub-chapter 16.2.

The following table presents a summary of the maximum values predicted for the severe accident scenarios LB LOCA, SBLOCA and SBO.

Severe accident Sequences	Peak Pressure (bar)	Peak Liner Temperature (^{o}C)	Quench Peak Time (s)
LB LOCA	4.8	130	19,300
SB LOCA	3.6	125	181,800
SBO	4.7	125	40,100

In addition, global combustion of the maximum quantity of hydrogen produced in a severe accident results in pressures lower than the containment design pressure (Sub-chapter 16.2). This simulation was carried out for all of the relevant severe accident scenarios.

1.5.4.6. Comparison with the acceptance curves

Summary of the in-containment pressure and temperature load combinations

The maximum values are given in the following table for all of the scenarios considered in the containment design.

Scenario	Pressure peak (bar)	Temperature peak (°C)	
LB LOCA	4.5	177	$T_{containment}$
2A-LOCA	4.3	182	
SLB	5.2	183	
Severe accident: LB LOCA	4.7	137	T_{liner}

Calculations for non-core melt accidents provide the bounding containment temperatures whereas those for severe accidents provide the bounding temperature for the steel liner.

P and T acceptance values selected for the containment design

The pressure and temperature acceptance curves used for the design of the steel liner and the internal containment structure are presented in Sub-chapter 3.4.

Adequacy of the P and T acceptance curves with respect to calculated limits

The (P, T) loads obtained for the MSLB bound those obtained for the LB LOCA, 2A-LOCA and 2A-SLB accidents. The MSLB load combinations are therefore compared to the acceptance curves (Section 6.2.1 - Figures 5 and 6).

The pressure is bounded by the acceptance curve but the temperature curve is not bounded by the acceptance curve for a very short period (approximately 250 seconds) when the containment temperature exceeds 170°C. However, considering the power level of 4900 MWth, the calculation is conservative. In addition, the temperature of the walls is predicted to remain lower than the saturation temperature, thus ensuring the formation of a film of condensed water on the walls (Section 6.2.1 - Figure 7). As the maximum saturation temperature during the transient is 140°C, which is considerably lower than the 170°C limit, it is judged that the (P, T) load combinations for the containment are practically bounded by the acceptance curves.

Similarly, the pressure and temperature curves for the steel liner obtained for a severe accident are completely bounded (Section 6.2.1 - Figures 8 and 9).

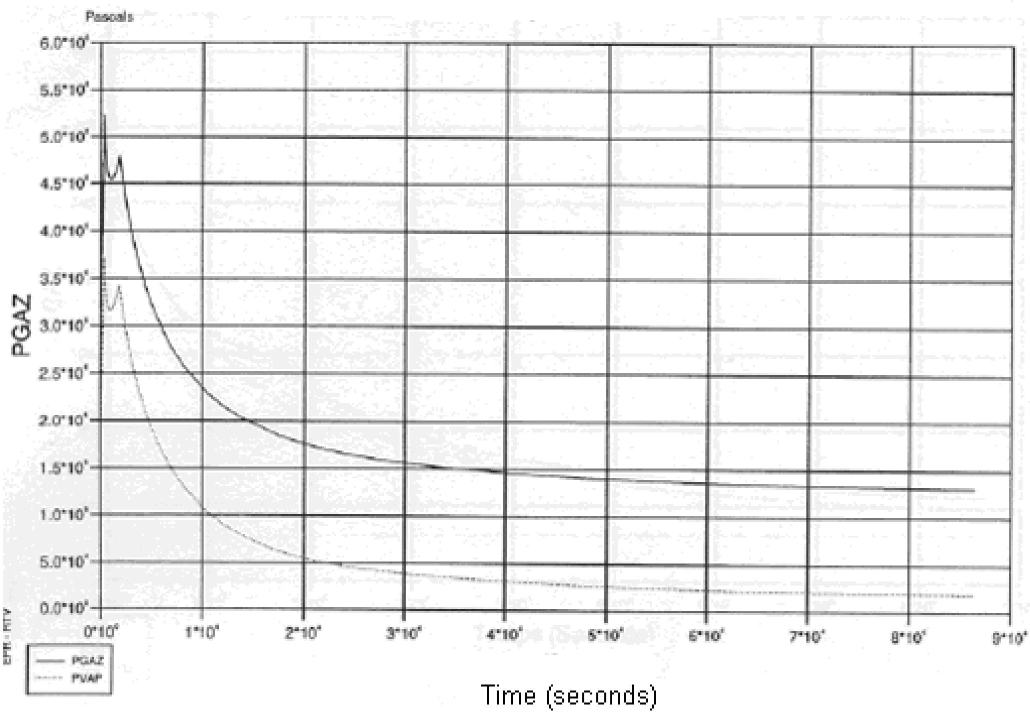
1.6. SYSTEM SIZING

Containment Sizing

Among other events, the large break loss of coolant accident (LB LOCA) and the steam line break (SLB) are considered in the containment sizing calculations.

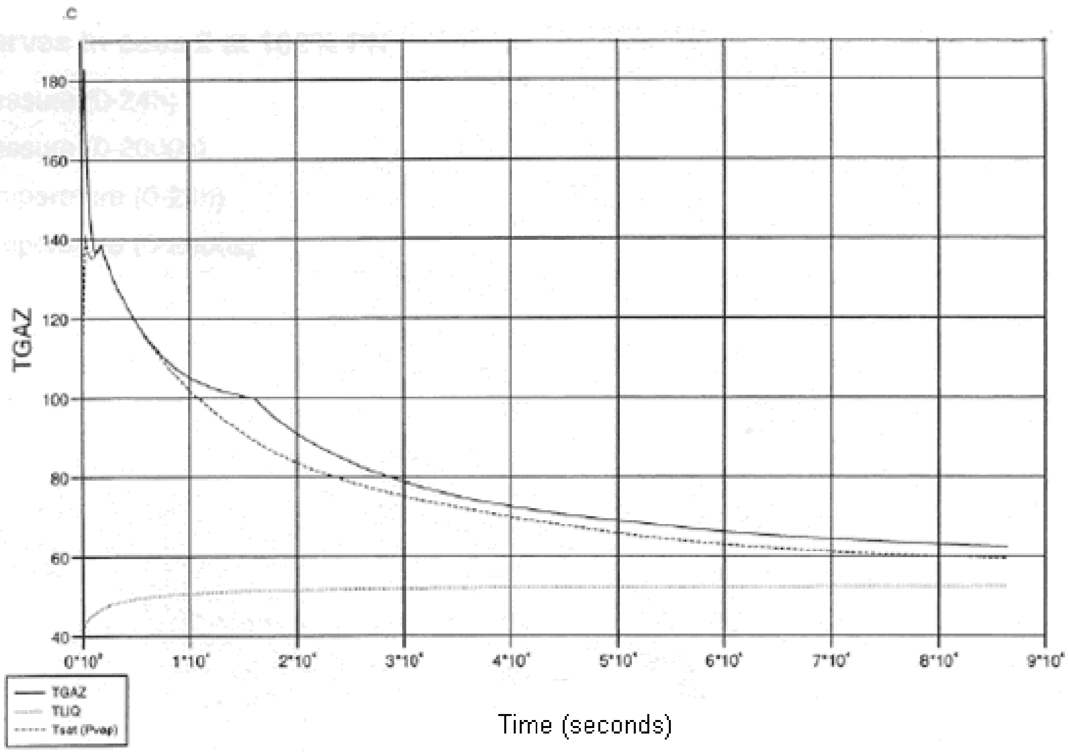
SECTION 6.2.1 - FIGURE 1

Steam Line Break - Containment Total and Steam Pressure (24 hours) [Ref-1]



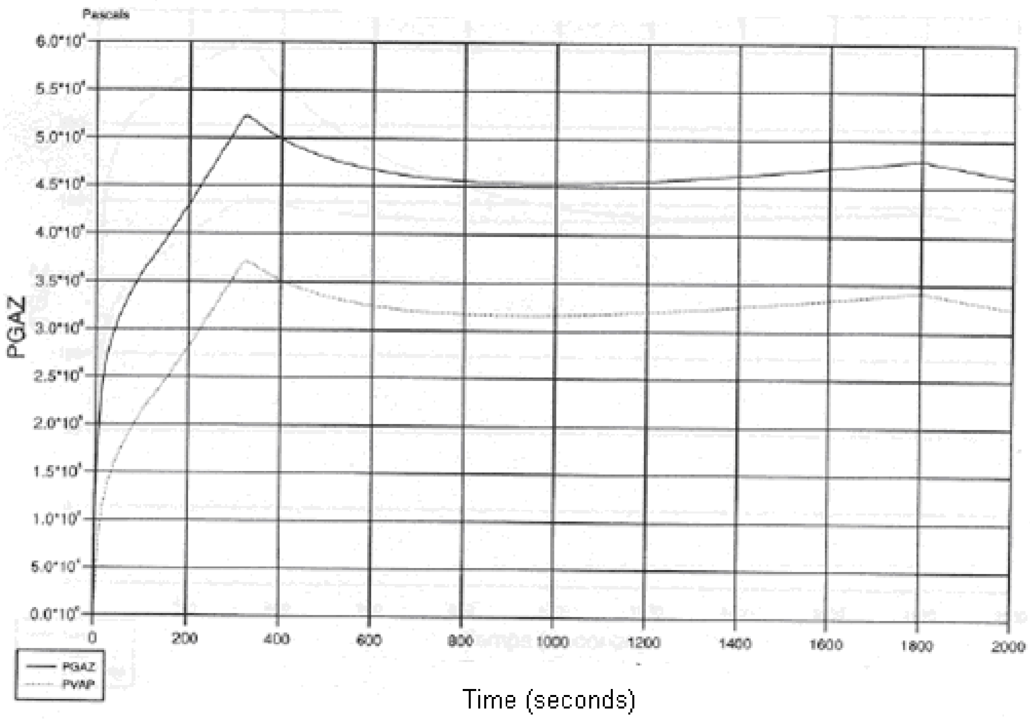
SECTION 6.2.1 - FIGURE 2

Steam Line Break - Atmosphere and IRWST Temperature (24 hours) [Ref-1]



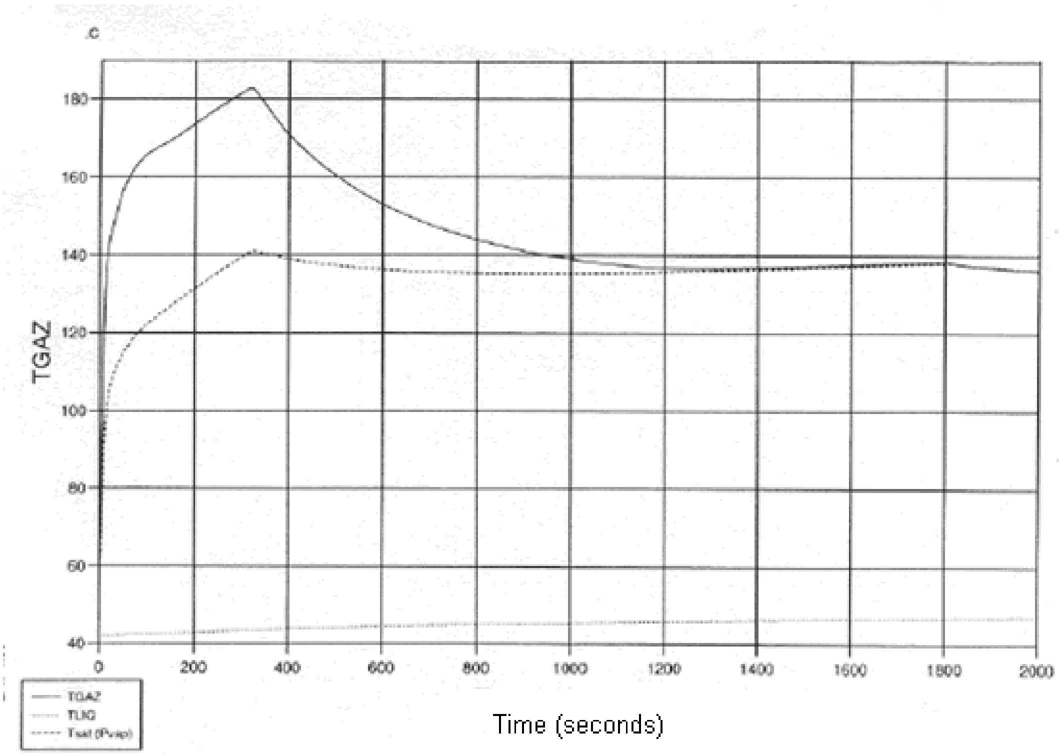
SECTION 6.2.1 - FIGURE 3

Steam Line Break - Containment Total and Steam Pressure (2000 seconds) [Ref-1]



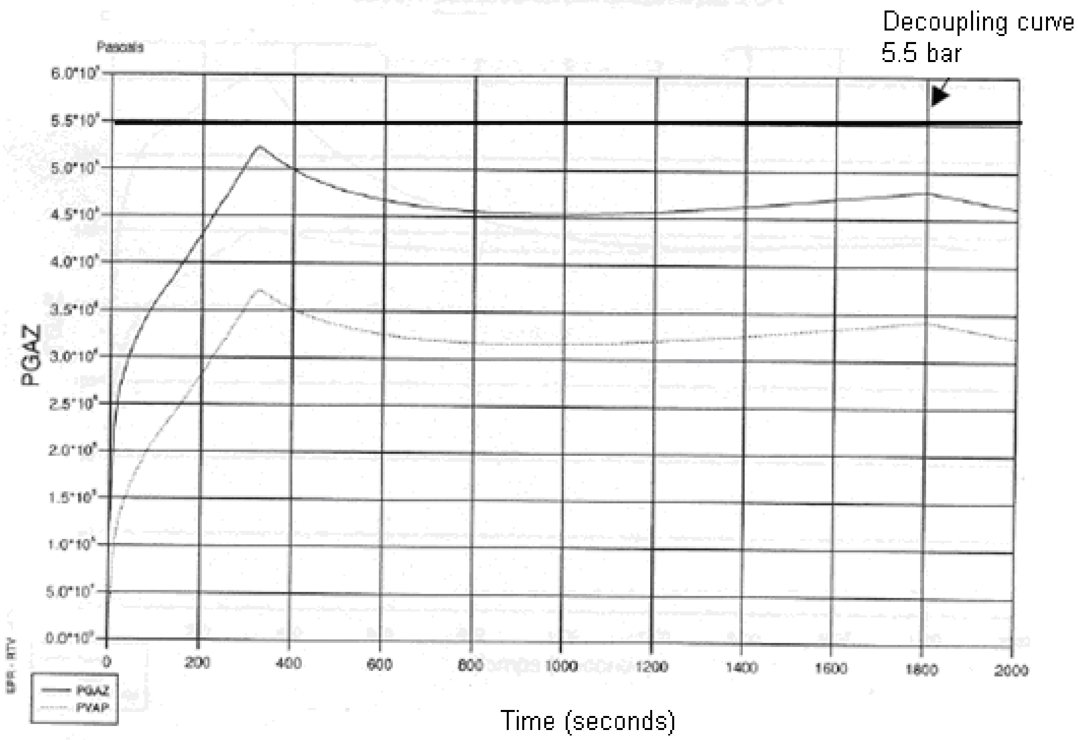
SECTION 6.2.1 - FIGURE 4

Steam Line Break - Atmosphere and IRWST Temperature (2000 seconds) [Ref-1]



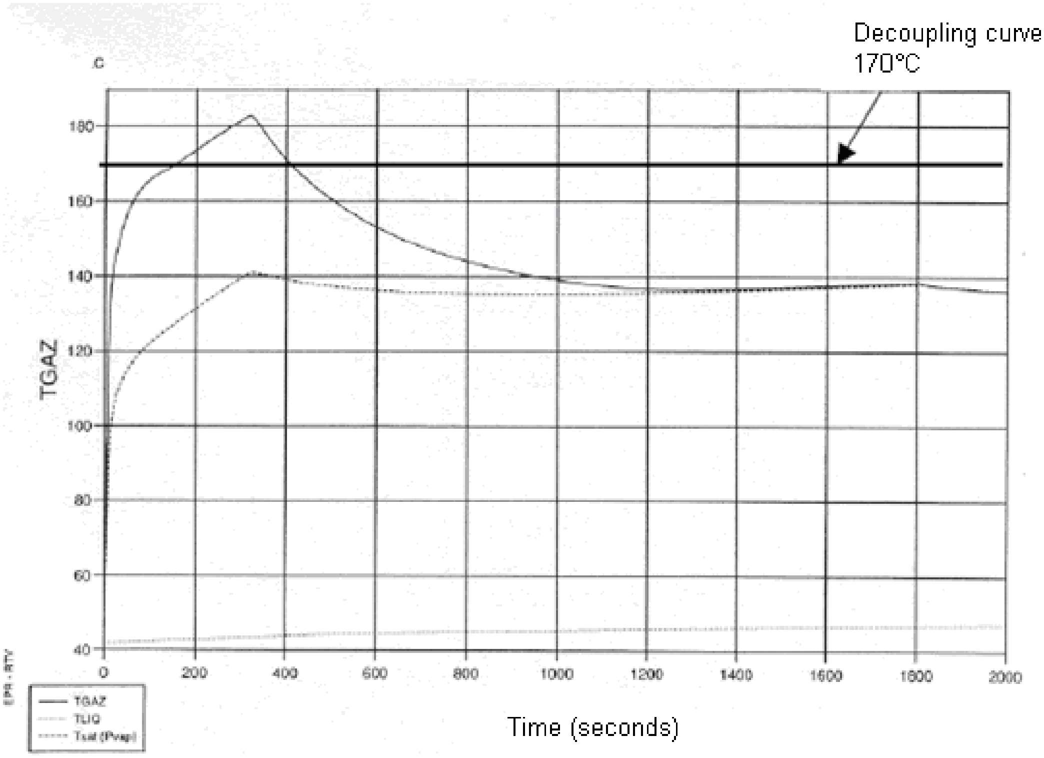
SECTION 6.2.1 - FIGURE 5

Steam Line Break - Containment Pressure compared with the Decoupling Value [Ref-1]



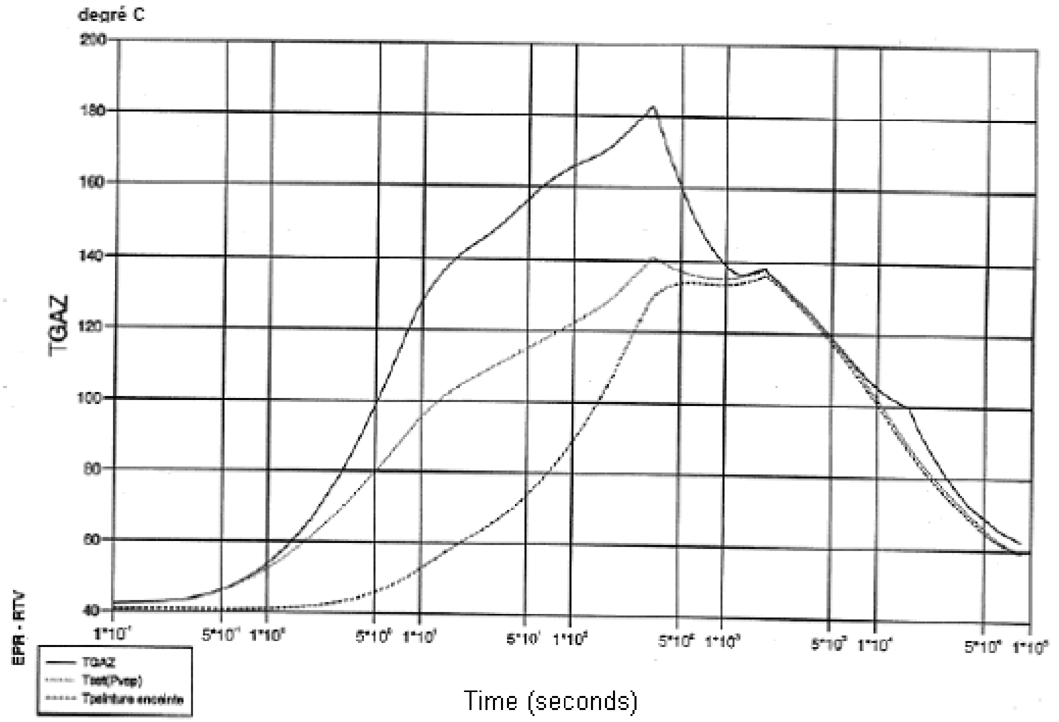
SECTION 6.2.1 - FIGURE 6

Steam Line Break - Containment Temperature compared with the Decoupling Value [Ref-1]



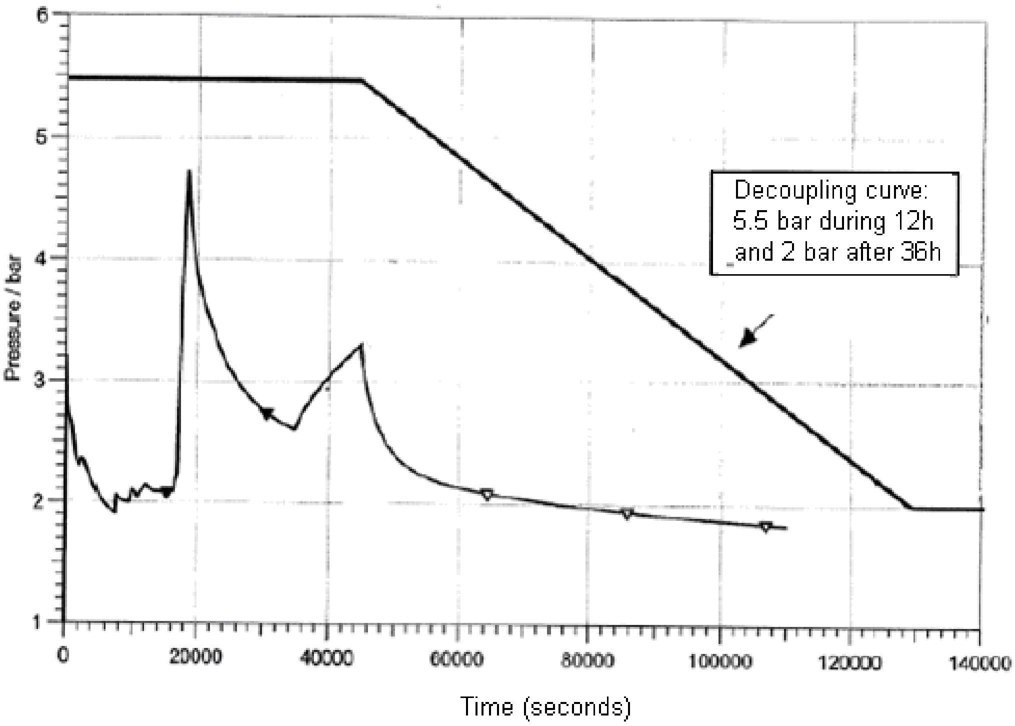
SECTION 6.2.1 - FIGURE 7

Steam Line Break - Containment and Steel Liner Temperatures [Ref-1]



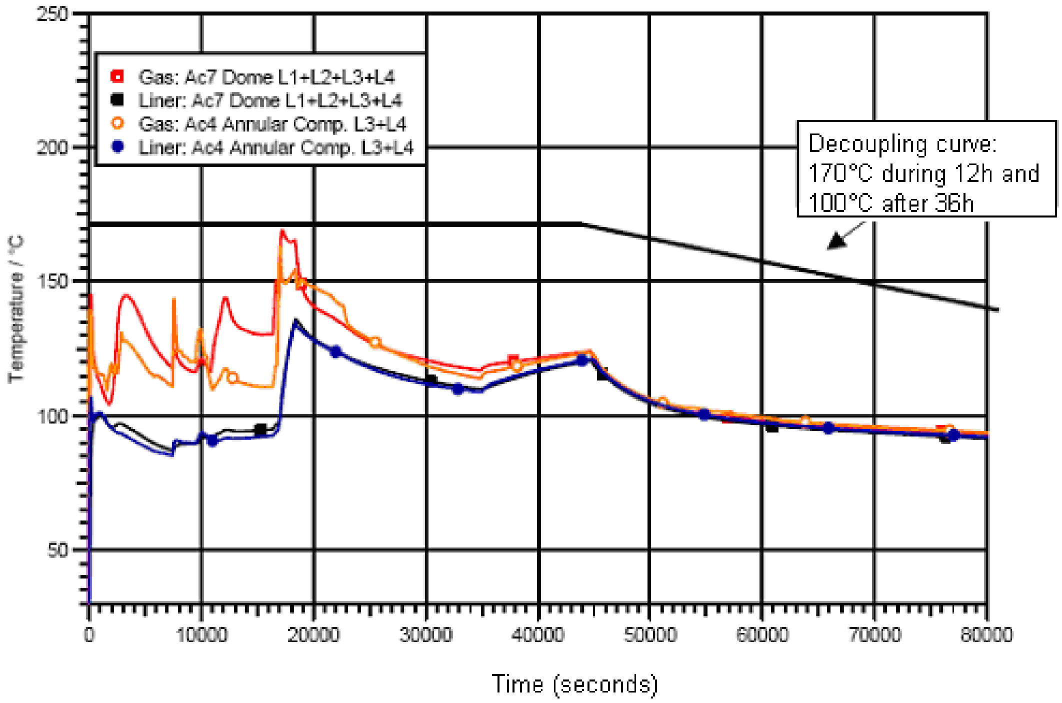
SECTION 6.2.1 - FIGURE 8

Severe Accident (LB LOCA) – Containment Pressure compared with the Decoupling Curve [Ref-1]



SECTION 6.2.1 - FIGURE 9

Severe Accident (LB LOCA) – Containment Temperature compared with the Decoupling Curve [Ref-1]



2. ANNULUS VENTILATION SYSTEM (EDE [AVS])

The section describes the Annulus Ventilation System [Ref-1] to [Ref-7] and its operation in fault conditions.

2.0. SAFETY REQUIREMENTS

2.0.1. Safety functions

The annulus ventilation system (EDE [AVS]) reduces the radioactive discharges into the environment for PCC (F1 function), RRC-A or RRC-B (F2 function) events and accidents that could cause a radioactive release into the containment.

In addition, the annulus contains heaters to maintain the minimum required temperature.

2.0.2. Functional criteria

The EDE [AVS]:

- Maintains the annulus at a negative pressure to collect any leaks from inside the containment following an accident. This includes leaks collected by the (EPP) containment leak rate control and testing system (see section 5 of this sub-chapter).

- Discharges these leaks to the vent stack after filtering using High Efficiency Particulate Air (HEPA) and iodine filters; the efficiencies required are [Ref-1] [Ref-2]:

    - 1000 for the HEPA filters,

    - 100 (methyl iodide) for the iodine filters.

The period for which the annulus is maintained at a negative pressure after the EDE [AVS] has shutdown must be specified and justified.

The minimum temperature to be maintained in the Bo3 boron room (2000 ppm, safety classified systems) is 7°C (5°C without margin) [Ref-3].

2.0.3. Design requirements

2.0.3.1. Requirements from the safety classification

- Classification: The annulus ventilation system is safety classified in accordance with Sub-chapter 3.2.

- F1 parts of the system must meet the single failure criterion.

- F1 parts of the system must be supplied by the emergency switchboards.

- The annulus ventilation system components must be qualified for PCC, RRC-A and RRC-B conditions: the unit heaters must be qualified for PCC conditions.

- Required mechanical, electrical and instrumentation and control classifications are given in Sub-chapter 3.2.
- Seismic classification requirements are given in Sub-chapter 3.2.
- Safety classified systems require periodic testing and inspection to confirm their availability with a sufficient degree of confidence.

2.0.3.2. Others statutory requirements

Paragraph G2 of the Technical Guidelines (see Sub-chapter 3.1) is applicable to the EDE [AVS] system.

2.0.3.3. Hazards

The list of hazards taken into account in the EDE [AVS] design is presented in Chapter 13 for external hazards and for internal hazards.

2.1. SYSTEM ROLE

In addition to the safety functions described in section 2.0.1 of this sub-chapter, the EDE [AVS] system is designed to maintain the annulus ambient conditions within acceptable limits (see Sub-chapter 9.4).

2.2. DESIGN BASIS

NB. Wind speeds and corresponding pressures are French specific references and are given for information only. These references will be replaced by site-specific figures when they are known.

<u>Maintain negative pressure in the annulus (Δp)</u>

- During a PCC-2 to PCC-4 or an RRC-A event, the minimum absolute negative pressure value to preserve (6.2 mbar) corresponds to the negative pressure produced by a 22 m/s wind on the external wall of the reactor building. Thus, when the wind speed is lower than 22 m/s, the entire annulus will be at a negative pressure (22 m/s is considered to be a decoupling value corresponding to a high wind frequency of exceedence of less than 10^{-3}/yr at all potential sites in France). The selected wind speed is lower than the maximum design wind speed because if the selected wind was very strong (the probability of which is very low), this would increase the global radioactive discharges calculated for weaker and more frequent winds.
- During a severe accident, the minimum negative pressure to maintain corresponds to the negative pressure produced by a 12 m/s wind on the external wall of the reactor building. (12 m/s is a value for all potential sites in France which corresponds to a wind speed V0 where the probability (V < V0) = 10 x probability (V > V0).). The grace period of 1 hour 45 minutes (time during which the pressure changes from the initial value to 1.85 mbar in the event of shutdown of all of the fans) ensures that discharges of radioactive substances are delayed beyond the time when the leak is highly active. However, this is not taken into account when calculating the radiological consequences (see section 3 of Sub-chapter 16.2).

- In normal operating conditions the absolute value is greater than the value required during an accident because, during an accident, leakage from the internal containment is collected. The duration of the grace period depends on the initial negative pressure.

Leakage from the containment

The maximum leak rate from the internal containment is 0.3% per day of the mass of gas contained in the free volume of the internal containment in the conditions specified in section 1 of this sub-chapter.

The EDE [AVS] system design assumption is for a maximum leak rate from the external containment toward the annulus of 1.5% of the total volume of the containment per day at an annulus negative pressure of 6.2 mbar.

In normal operating conditions, the system flow matches the leakage from the external containment, as leakage from the internal containment is negligible.

In the event of design basis accidents and severe accidents, the system is able to accommodate the increased leakage from internal containment due to overpressure, and also the normal in-leakage from the external containment.

Containment leak rate control and testing system (EPP)

Leakages collected by the containment leak rate control and testing system (EPP) are discharged into the annulus and therefore into the EDE [AVS] (gaseous phase). They form part of the total leakage rate of 0.3% vol/day (see section 5 of this sub-chapter).

EDE [AVS] flow rate

The system design accommodates a maximum flow rate of 300 m^3/hr. This value used in radiological consequence calculations.

2.3. EQUIPMENT DESCRIPTION AND CHARACTERISTICS

2.3.1. Description of the system

The system consists of:

- two complete safety trains with a HEPA filter and iodine filter. The two trains are physically separated. A metallic pre-filter, located upstream of each iodine line, minimises radiological releases in the event of a severe accident. For other situations, this equipment is by-passed.

- a fully operational train with a HEPA filter but without an iodine filter.

The operational train functions continuously during normal plant operation so that a negative pressure in the annulus exists at the start of any accident.

During an accident, one train with HEPA and iodine filters is initiated and the operational train is isolated.

A flow diagram is given in Section 6.2.2 - Figure 1.

2.3.2. Equipment characteristics

The efficiency of the HEPA filters and iodine filters is such that radioactive discharges into the environment are below limits prescribed for initiating event and hazard categories.

The EDE [AVS] fan is designed to:

- maintain negative pressure in the annulus during any type of accident considered in the design,
- extract air from the annulus (including leakage from the containment) and discharge it to the environment via filters.

The fan is designed according to the required flow rate and the pressure loss. The fan speed depends on:

- the leak rate from the internal containment wall: Qi (from the Reactor Building to the annulus),
- the leak rate from the external containment wall: Qe (from the outside to the annulus).

The pressure loss depends on:

- the negative pressure in the annulus : Δp,
- the overall pressure drop within the system: ΔP.

The fan capacity is such that a negative pressure is maintained in the annulus.

2.4. OPERATING CONDITIONS

Reactor in normal operation

The system operates continuously using the operational train so that there is a negative pressure at initiation of any accident. All the exhaust air is treated by a HEPA filter before being discharged to the stack.

Following a PCC-2 to PCC-4 or RRC-A accident

The operational train is automatically isolated by power-operated dampers and the exhaust function switches over to one of the safety trains to treat any leakage from the reactor building internal containment wall. The safety function is automatically started with the fans in the safety trains maintaining the negative pressure in the annulus. If the first safety train is unavailable then the second train is started automatically.

Following an RRC-B accident

The power-operated dampers isolate the operational train and the fan is stopped. An iodine filtering train is started on a minimum negative pressure criterion. The first iodine extraction line is preferentially started by aligning the metallic pre-filter.

The unit heaters start and stop automatically depending on the annulus temperature.

2.5. PRELIMINARY SAFETY ANALYSIS

2.5.1. Compliance with regulations

The system complies with general regulations in force (see Sub-chapter 1.4).

2.5.2. Compliance with functional criteria

The EDE system [AVS] is designed to limit radioactive discharges during accidents, in particular:

- by maintaining negative pressure in the annulus,
- by achieving adequate HEPA and iodine filtering efficiency.

The unit heaters maintain the annulus temperature above the minimum required

2.5.3. Compliance with the design requirements

2.5.3.1. Safety classification

Compliance of the design, construction, materials and equipment with the classification rules is described in detail in Sub-chapter 3.2.

2.5.3.2. Single failure criterion or redundancy

The iodine filtering trains used during accidents, and the bypass line isolation dampers, have 2 x 100% redundancy.

The annulus temperature remains above the minimum required temperature in case of failure of one unit heater.

2.5.3.3. Qualification

The equipment is qualified in accordance with the requirements presented in Sub-chapter 3.6.

2.5.3.4. Instrumentation and control

Compliance of the design and construction of instrumentation and control and equipment with the classification rules is described in detail in Sub-chapter 3.2.

2.5.3.5. Emergency power supplies

The first iodine extraction line is supplied by division 1 electrical train and the second iodine extraction line is supplied by division 4 electrical train. Also, one bypass isolation damper is supplied by division 1 electrical train and one by division 4 electrical train.

To allow maintenance of the electrical supply system, a cross connection is made between the division 1 switchboard and the division 2 switchboard (respectively between the division 4 switchboard and the division 3 switchboard).

Loss of Off-Site Power

The bypass line isolation dampers, two iodine extraction lines and unit heaters are powered by the emergency switchboards dedicated to the nuclear island.

Station Blackout

The EDE [AVS] iodine lines are backed-up in the event of a station blackout by the emergency diesel generator sets and by the emergency power supply dedicated to severe accidents (see Sub-chapter 8.3). The grace period enables connection to the severe accident battery and switches on an iodine line.

2.5.3.6. Hazards

Protection against hazards is presented in Section 6.2.2 - Table 1.

2.6. TESTS, INSPECTION AND MAINTENANCE

2.6.1. Periodic tests

The safety functions are subject to periodic tests.

2.6.2. Inspection and maintenance

The design and installation of EDE [AVS] components provide easy access enabling maintenance and in-service inspections to be carried out.

System maintenance is carried out when the plant is shutdown and when negative pressure is not required in the annulus.

2.7. FLOW DIAGRAM

See Section 6.2.2 - Figure 1.

SECTION 6.2.2 - TABLE 1

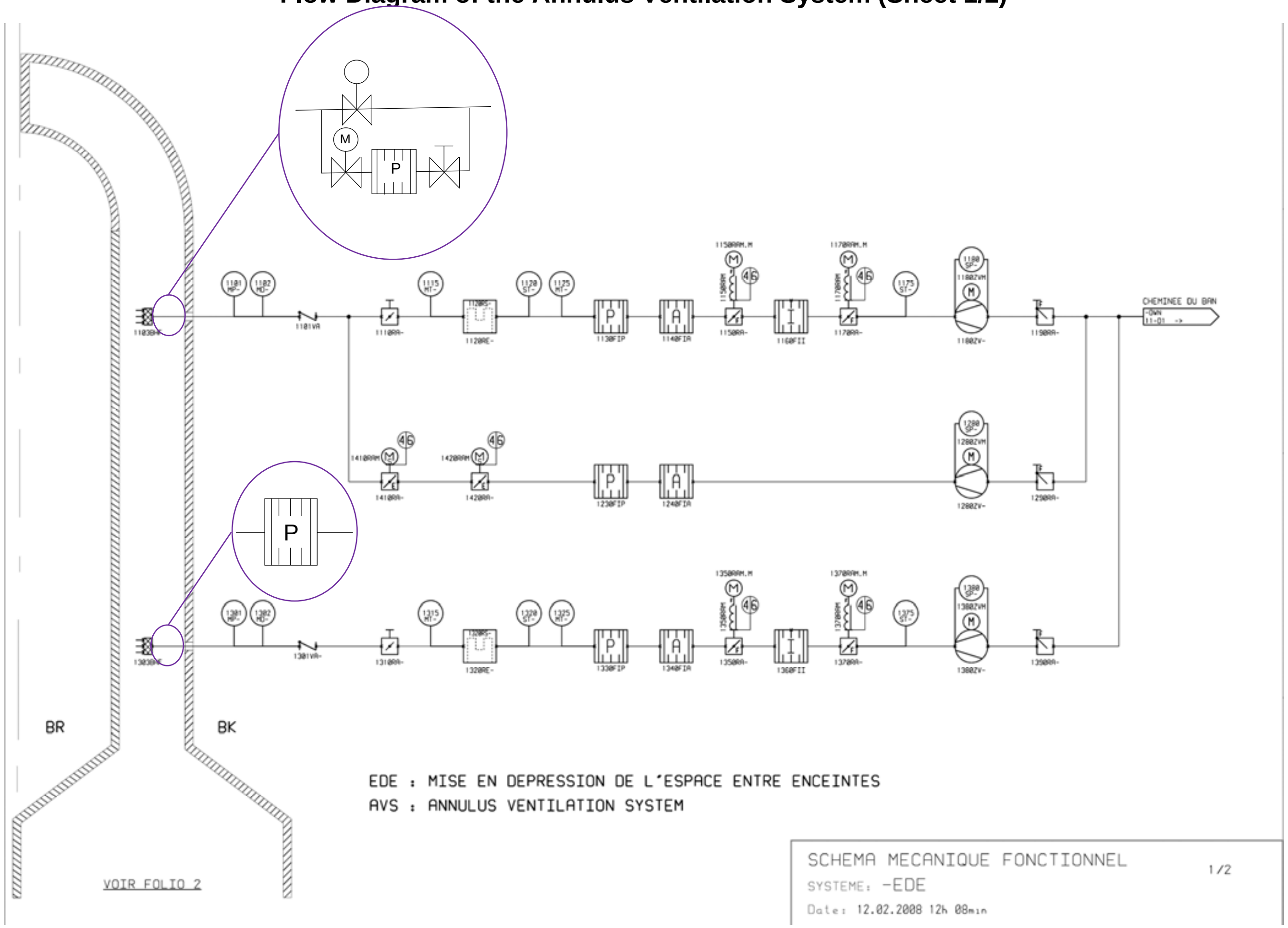
Summary table of the inclusion of hazards for the EDE [AVS] System

Internal hazards	**Protection required in principle**	**General protection**	**Specific protection introduced in the system design**
Pipe ruptures	No loss of more than one iodine extraction train and one bypass system isolation damper Minimum temperature maintained in the annulus	Physical separation of redundant equipment	-
Tank, pump and valve ruptures		Physical separation of redundant equipment	-
Internal projectiles		Physical separation of redundant equipment	-
Dropped load		Physical separation of redundant equipment	-
Internal explosion		Physical separation of redundant equipment	-
Fire		Fire sectors in the fuel building and annulus	Fire dampers around the iodine filters (to limit the spread of the fire)
Internal flooding		Physical separation of redundant equipment	-

External hazards	**Protection required in principle**	**General protection**	**Specific protection introduced in the system design**
Earthquake	No loss of more than one iodine extraction train and one bypass system isolation damper Minimum temperature maintained in the annulus	Location in the fuel building and annulus	SC1 for the classified parts
Aircraft crash		Location in the fuel building and annulus	-
External explosion		Location in the fuel building and annulus	No (no air intake)
External flooding		Location in the fuel building and annulus	-
Snow and wind		Location in the fuel building and annulus	-
Extreme cold		Location in the fuel building and annulus	Unit heaters in the annulus to protect the other annulus systems
Electromagnetic interference		Location in the fuel building and annulus	-

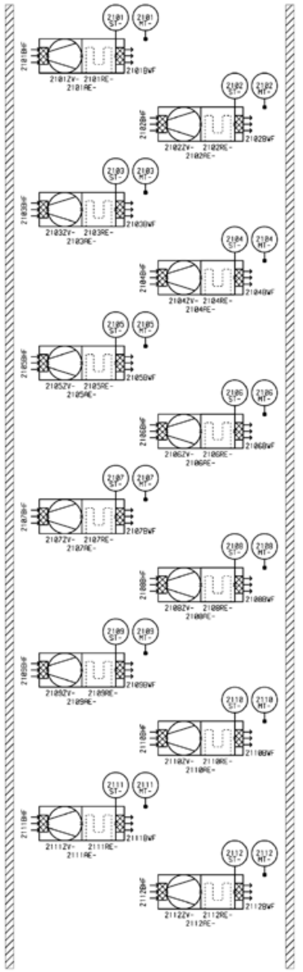
SECTION 6.2.2 - FIGURE 1

Flow Diagram of the Annulus Ventilation System (Sheet 1/2)



SECTION 6.2.2 - FIGURE 1 (CONT'D)

Flow Diagram of the Annulus Ventilation System (Sheet 2/2)



EDE : MISE EN DEPRESSION DE L'ESPACE ENTRE ENCEINTES
AVS : ANNULUS VENTILATION SYSTEM

SCHEMA MECANIQUE FONCTIONNEL 2/2
SYSTEME: -EDE
Date: 12.02.2008 12h 08min

3. CONTAINMENT ISOLATION

This section describes the concept and the design requirements for the containment isolation [Ref-1] function, which contributes to the containment function described in section 1 of this sub-chapter.

There are several types of penetrations allowing systems (fluid and electrical), equipment and personnel to cross the containment shell [Ref-2] [Ref-3]:

- fluid penetrations:
  - pipe penetrations transporting high-energy fluid,
  - main steam line or feedwater penetrations,
  - standard fluid pipe penetrations,
  - sump suction penetrations (RIS [SIS]),
  - ventilation penetrations.
- spare penetrations,
- electrical penetrations:
  - medium voltage,
  - low voltage.
- equipment hatch,
- air locks,
- fuel transfer tube.

The containment isolation concept only applies to fluid penetrations. The containment function for the other types of penetration (equipment hatch, air lock, transfer tube, electrical penetrations), is described in section 1 of this sub-chapter.

A detailed description of the mechanical components of the fluid penetrations, electrical penetrations, equipment hatch, air locks and transfer tube, together with the provisions taken to ensure the leak-tightness of these penetrations, is presented in Sub-chapter 3.4.

3.0. SAFETY REQUIREMENTS

3.0.1. Safety functions

The containment isolation function contributes to the containment of radioactive substances (see section 1 of this sub-chapter). It ensures that, during accidents with fission product release, the radioactive release through the fluid penetrations towards the environment is minimised.

3.0.2. Functional criteria [Ref-1]

The isolation function is provided by means of isolation valves (motorised and/or check valves). Containment isolation must be ensured if demanded.

Specific function

Depending on their normal operating position and their role in case of an accident, the various containment isolation valves may have to:

- close at the beginning of the accident,
- remain leak tight during the post-accident phase,
- remain operable during the post-accident phase.

The containment isolation function must ensure these specific functions.

3.0.3. Requirements relating to the design [Ref-1] [Ref-2]

3.0.3.1. Requirements from safety classifications

3.0.3.1.1. Safety classification

The containment isolation function is safety classified in accordance with the classification defined in Sub-chapter 3.2.

3.0.3.1.2. Single failure criterion (active and passive)

The single failure criterion applies to the design of F1 classified systems (see Sub-chapter 3.2).

The active single failure criterion is not considered for the main steam isolation valves VIV [MSIV] in case of a steam generator tube rupture (SGTR) (see Sub-chapter 3.1).

A passive single failure is not considered in the containment penetration area when the containment function is demanded (see Sub-chapter 3.1).

3.0.3.1.3. Emergency power supplies

The power supplies to the containment isolation valves are backed-up by emergency power supplies.

3.0.3.1.4. Qualification for operating conditions

The penetrations and isolation valves in the reactor building are capable of fulfilling their specific function in the radiation, temperature, pressure and humidity conditions expected following a PCC or RRC event.

The containment external isolation valve is qualified for the ambient conditions of the transported fluid following failure of the containment internal isolation valve.

3.0.3.1.5. Mechanical, electrical and instrumentation and control classifications

The containment isolation valves are subject to mechanical, electrical and instrumentation and control classification in accordance with the classification specified in Sub-chapter 3.2.

3.0.3.1.6. Seismic classification

The containment isolation valves are subject to seismic classification in accordance with the classification specified in Sub-chapter 3.2.

3.0.3.1.7. Periodic tests

Periodic tests are carried out on the containment isolation valves to ensure their availability.

3.0.3.2. Other statutory requirements

Applicable Technical Guidelines are listed in Sub-chapter 3.1.

3.0.3.3. Hazards [Ref-1]

3.0.3.3.1. Protection against external hazards

The containment isolation valves are designed against external hazards in accordance with the protection against hazards rules and criteria presented in Sub-chapters 3.1 and 13.1.

3.0.3.3.2. Protection against internal hazards

The containment isolation valves are designed against internal hazards in accordance with the protection against hazards rules and criteria presented in Sub-chapters 3.1 and 13.2.

The isolation function is protected against the effects of a loss of coolant accident or a secondary coolant pipe rupture since it may be required to operate after these accidents.

3.1. DESIGN BASIS [REF-1]

3.1.1. Basic requirements and design bases

See section 1 of this sub-chapter and Sub-chapter 3.1.

3.1.2. General system design principles

Reactor in normal operating conditions or at shutdown:

The isolation function is not required during normal operation. The position of the isolation valves depends on the operation of the system to which they belong.

During accidents:

All fluid penetrations, except those belonging to safety systems used for post-accident management, are isolated (section 1 of this sub-chapter).

3.1.3. Assumptions used in creating flow diagrams

The penetration isolation concept (number and type of isolation valves) is defined in section 3.2 below.

The containment isolation function is automatically activated by signals from the reactor protection system. The instrumentation and control principles for isolation of penetrations are presented in section 3.3 below.

3.2. FLUID PENETRATION ISOLATION PRINCIPLES

For fluid penetrations, isolation is generally carried out by two independent devices in the pipes crossing the containment boundary.

The following are considered as effective barriers provided they are protected from internal missiles:

- metallic pipe walls, tanks and valve bodies, whose design pressures and temperatures are higher or equal to the bounding conditions for PCC and RRC accidents that may occur in the reactor building,
- an isolation valve which is closed automatically or by operator action from the Main Control Room,
- an isolation valve with local manual control, normally closed; the position of this valve is subject to administrative control or is controlled by a mechanical locking device,
- a single check valve on a pipe transporting a liquid entering the containment, provided it is located inside the containment and there is an additional isolation valve outside the containment.

3.2.1. Lines passing through the containment internal and external walls [Ref-1] [Ref-2]

The isolation valves are defined in accordance with the following rules:

3.2.1.1. Lines connected to the Primary Cooling System or lines directly connected to the containment atmosphere

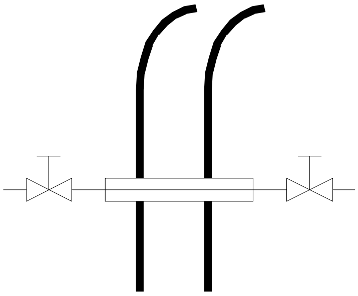
These lines are equipped with one or two isolation valves (one outside the containment and the other, if necessary, inside) which constitute, with the corresponding pipe walls, a double isolation barrier at the containment penetration.

This configuration enables the single failure criterion to be met.

Generally, the design requirement for the isolation valves is as follows:

3.2.1.1.1. For systems which are only used during shutdown

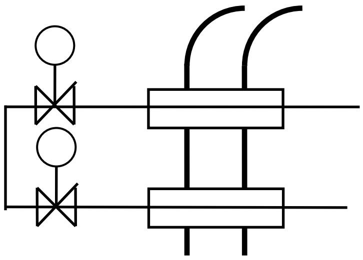
Two manually operated valves, or two power-operated valves actuated from the Main Control Room may be used.



3.2.1.1.2. For systems used in normal operating or accident conditions and whose part outside the containment constitutes a closed boundary

The closed pipe section outside the containment is considered as forming part of the containment and is designed to withstand the containment design pressure.

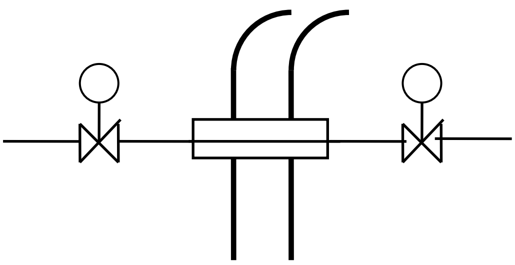
An automatic isolation valve located outside the containment, at each penetration provides the isolation function.



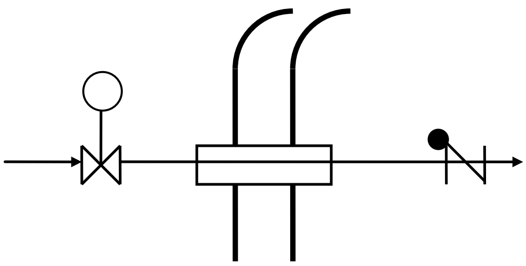
3.2.1.1.3. For other systems used in normal or accident conditions

Two automatic containment isolation valves, one inside and the other outside the containment (1), or for lines transporting fluid towards the containment (2), an automatic valve and a check valve, with the check valve located inside the containment.

(1)

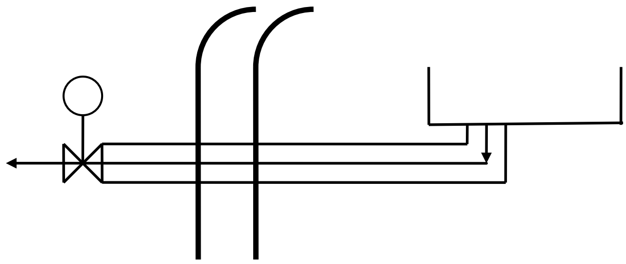


(2)



3.2.1.1.4. Special provisions for the penetrations in lines from the IRWST to the RIS [SIS] and EVU [CHRS] system pumps

Each of these lines only contains one isolation valve located outside the containment. There is no automatic isolation, valve closure being controlled by manual action from the Main Control Room. The section of piping between the sump and the valve is enclosed in a leak tight casing (guard pipe) thus providing a double leak tight penetration barrier.



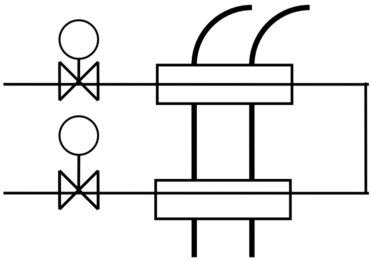
To ensure external leak-tightness of the isolation valves, bellows-sealed globe valves are used where possible. If other valves are used, they are equipped with a leak-off recovery system.

3.2.1.2. Lines not directly connected to the containment atmosphere which are not within the secondary cooling system

Lines entering the containment that are connected neither to the containment atmosphere, nor to the primary cooling system, and do not belong to the secondary cooling system, (see figure) are subject to the following requirements:

- their integrity must be maintained after events requiring isolation of the containment,
- they must be safety and earthquake classified.

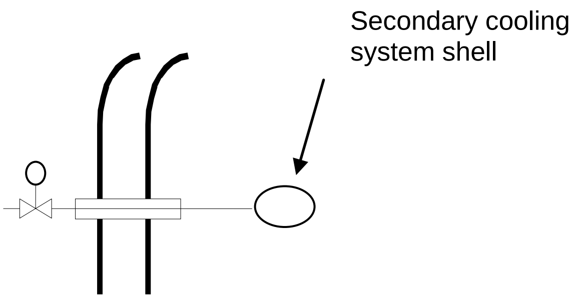
These lines are equipped with an isolation valve outside each penetration, thus providing a double barrier at the penetration (closed system on the inside and isolation valve on the outside).



In general, for systems which are used in normal operation or in accidents, the isolation valves are automatic or remotely actuated from the Main Control Room. For other systems the valves are manually operated and manually locked.

3.2.1.3. Lines forming part of the secondary cooling system

As long as the SG tubes are intact, the steam line and feedwater penetrations do not perform a containment function; this is provided by the secondary cooling system boundary.



The following secondary cooling system isolation valves contribute to the containment function in the long-term following SGTR:

- the main steam isolation valves and steam generator safety relief valves,

  Note: With reference to the main steam isolation valves, in the event of SGTR, the single failure criterion does not apply to the mechanical parts of the main valves themselves, but it does apply to the pilot devices.

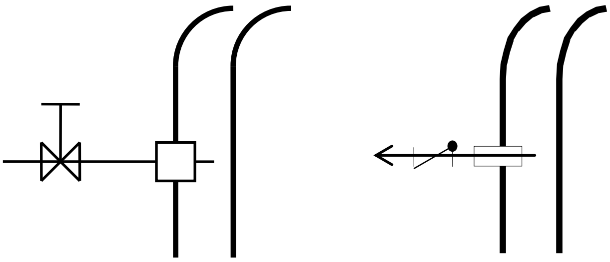
- the normal feedwater and emergency feedwater isolation valves outside the containment, and the normal feedwater check valve inside the containment,

- the secondary isolation valves from the steam generator blowdown system (APG [SGBS]).

3.2.2. Lines passing through the external containment wall only

3.2.2.1. Lines passing through the external containment wall only, which are open to the containment annulus atmosphere

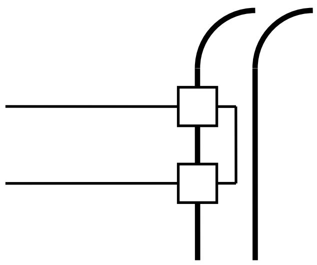
These lines are equipped with an isolation valve or a check valve outside the containment.



N.B.: The check valve configuration is acceptable as long as the inter-containment space is at negative pressure.

3.2.2.2. Lines passing through the external containment wall only, which are closed to the containment annulus atmosphere

No isolation valves are required for these lines.



The classification requirement for the pipe section in the inter-containment space is the same as for the penetrations.

3.3. ISOLATION INSTRUMENTATION AND CONTROL LOGIC

3.3.1. Principles

The automatic containment isolation valves close when an isolation signal is received from the reactor protection system. This does not apply to certain safety system valves that are required for post-accident management.

The isolation valves may be reopened only after isolation is complete and the isolation signal is cleared.

It is possible for the operators in the main control room to manually isolate the containment.

The automatic containment isolation valves are power-operated with power supplies backed-up by emergency power supplies, or are designed to move to a fail-safe position in the event of power failure (in general, the fail-safe position is closed, except for the safety systems required for the post-accident management).

Position indication and power supply status information is provided in the main control room for each isolation valve.

3.3.2. Valves that do not receive automatic closing signals

Valves in safety system required for the post-accident management do not receive automatic closing signals.

3.4. PRELIMINARY SAFETY ANALYSIS

3.4.1. Compliance with the design requirements

3.4.1.1. Safety classification

Compliance of the design, construction, materials and equipment with the requirements of the classification rules is described in detail in Sub-chapter 3.2.

3.4.1.2. Single failure criterion

The single failure criterion applies to all penetrations, with the following exceptions:

The active single failure criterion does not apply to the main steam isolation valves in the event of SGTR (Sub-chapter 3.1). However, it does apply to the pilot devices for these valves.

The passive single failure criterion does not apply to containment penetrations when the isolation function is required (Sub-chapter 3.1).

3.4.1.3. Emergency power supplies

The design of the isolation valve power supplies is described in Sub-chapter 8.3.

The power-operated valves inside the containment are backed-up by 2 hour batteries and by the main diesel generator sets.

The power-operated valves outside the containment are backed-up by the main diesel generator sets.

The air-operated valves close if the power or air supply is lost.

To ensure containment isolation in severe accidents with total loss of external power supply together with loss of the six diesel generator sets, the isolation valves outside the containment are powered by 12 hour batteries.

3.4.1.4. Qualification in service conditions

The penetrations and isolation valves inside the reactor building are designed to remain intact and operable in the radiation, temperature, pressure and humidity conditions expected following PCC and RRC events.

The isolation valves outside the containment are qualified for the ambient conditions of the fluid transported following PCC and RRC events. This ensures that the external valves will function following failure of any of the valves inside the reactor building.

3.4.1.5. Protection against hazards

<u>Protection against external hazards</u>

The containment isolation function is protected against external hazards. This protection is mainly provided by installing the isolation valves in the reactor building, fuel building and safeguard buildings.

The containment isolation function is designed to withstand the design basis earthquake.

Protection against internal hazards

The containment isolation function is protected against the following internal hazards: failure of pipes carrying high energy fluid, fire, internal flooding and missiles.

Redundant isolation valves are generally placed in two different buildings and are therefore protected against common cause failure resulting from a hazard.

The isolation function which may be required following a loss of coolant accident or rupture of the secondary coolant pipework is protected against the effects of pipework failure.

3.4.2. Other provisions relating to safety

Layout

The penetration pipes are short, with isolation valves placed as close to the containment as possible. Pipe supports for the penetrations are designed to absorb the loads produced by failure of lines close to the penetrations, without creating large stresses on the penetrations themselves.

Temperature effect

In the event of overpressure caused by expansion of the fluid between two isolation valves, design provisions are implemented to limit the overpressure (protection against the effect of fluid boiling).

3.4.3. Requirements applying to specific tests

In order to confirm their operational performance, components of the containment isolation function are tested during the pre-startup test programme.

After startup, some valves are subject to type C periodic leak tightness tests. This applies to isolation valves from systems meeting the following three criteria:

- the system crosses the containment internal and external walls,
- the system is connected to the containment atmosphere,
- the system transports gas.

4. COMBUSTIBLE GAS CONTROL SYSTEM (ETY [CGCS]) [REF-1] TO [REF-5]

4.0. SAFETY REQUIREMENTS

4.0.1. Safety functions

The Combustible Gas Control System ETY [CGCS] contributes to the safety function “containment of radioactive substances” by ensuring:

- Limitation and reduction of loads on containment structures caused by hydrogen combustion.
- In PCC-4 events, reduction and limitation in hydrogen mole fraction during loss of coolant accidents (LOCA), as well as to prevent any risk of combustion in the containment.
- Reduction in mean and local hydrogen concentration (mole fraction) during severe accidents (RRC-B) to ensure containment integrity in the event of a global deflagration and to reduce the risk of dynamic phenomena related to flame acceleration and deflagration-to-detonation transition (DDT).
- Limitation and reduction of loads on containment inner structures caused by breaks of high energetic pipes.

4.0.2. Functional criteria

The functional criteria governing the ETY [CGCS] safety role are related to hydrogen concentration:

- The quantity of hydrogen (mainly resulting from oxidation of the zirconium contained in the core) must be distributed throughout the containment and reduced in order to ensure that the total hydrogen mole fraction is below the ignition limit within approximately 12 hours.
- The system has to be designed to limit and to reduce the combustible gas concentration such that the containment can withstand the pressure loads resulting from complete adiabatic and isochoric combustion of the quantity of hydrogen which may be contained in the building, regardless of the scenario selected.

4.0.3. Design requirements

4.0.3.1. Requirements of safety classifications

- Point 1: Safety classification

The ETY [CGCS] system is classified in accordance with the classification defined in Sub-chapter 3.2.

- Point 2: Single failure criterion

The single failure criterion is applied to the ETY [CGCS] system.

- Point 3: Emergency power supply

The passive auto-catalytic recombiners (PARs) of the combustible gas control system do not need electrical power supplies. Additionally, the rupture panels and convection dampers in the ceiling above the steam generators are of passive design and do not need electrical power supplies.

The mixing dampers for enhancing hydrogen distribution in the containment do not require emergency power supplies to achieve their safety function. Opening of the hydrogen mixing dampers is ensured through the fail-safe principle.

- Point 4: Qualification for operating conditions

The ETY [CGCS] system equipment is qualified depending on its safety role under ambient conditions to which it is subjected when fulfilling its task.

- Point 5: Mechanical, electrical and instrumentation and control classifications

The ETY [CGCS] system is classified in accordance with Sub-chapter 3.2.

- Point 6: Seismic classification

The ETY [CGCS] system is classified in accordance with Sub-chapter 3.2.

- Point 7: Surveillance testing

Surveillance tests will be carried out on the ETY [CGCS] safety classified components to ensure their availability with sufficient confidence.

4.0.3.2. Other regulatory requirements

- Basic Safety Rules:

There are no Basic Safety Rules to be applied.

- Technical Guidelines:

The ETY [CGCS] system fulfils the requirements in § A1.3, B1.4.1, E2.2.4, E2.3.3 and E2.4 of the Technical Guidelines (see Sub-chapter 3.1, General Principles).

- EPR specific documents:

Not applicable.

4.0.3.3. Hazards

Measures described in Sub-chapter 3.1 for requirements relating to external and internal hazards are applicable.

4.1. SYSTEM ROLE

4.1.1. PCC-1 events

During normal operation, the PARs of the combustible gas control system are in stand-by mode and the hydrogen mixing equipment contributes to the separation into two room compartment by its closed (intact) position (state) and leak tightness.

4.1.2. PCC-2, PCC-3, PCC-4 and RRC events

The hydrogen released into the containment atmosphere following a severe accident with core degradation, or a loss-of coolant accident, must not be allowed to form an unacceptable concentration.

The combustible gas control system promotes mixing of the containment atmosphere and ensures hydrogen reduction to prevent hydrogen concentrations forming which could be critical to the integrity of the containment.

In the event of an accident, the two zone containment will be transformed into a single zone containment configuration very early in the accident following pressure build-up in the equipment cells:

- Opening of the hydrogen mixing dampers, which are equipped with spring loaded actuators following the fail-safe principle.
- Possible opening of some rupture panels.

Depending on the accident scenario the rupture panels and convection dampers at the top of SG cells will open. Rupture panels open passively due to the pressure difference across the foils. Convection dampers have a temperature sensitive opening mechanism and open passively either due to the pressure difference or to exceeding a predefined temperature (80-85°C).

Hydrogen mixing dampers together with, at least, the convection dampers provide good atmospheric mixing in the containment and thus quickly and efficiently reduces locally high concentrations and concentration gradients. In combination with the inerting effect of steam and a sufficient containment volume, the hydrogen mixing system will efficiently prevent fast combustion which could be critical to the integrity of the containment.

The passive autocatalytic recombiners (PARs) of the combustible gas control system, installed in various parts of the containment, reduce the hydrogen concentration below flammability limits even under inert conditions and thus prevents the possibility of hydrogen combustion in the long-term.

To check the function of ETY1 – Hydrogen Reduction – a total of six temperature sensors are provided downstream of the recombiners.

The oxidation of hydrogen is an exothermic reaction. Hence, the temperatures measured downstream of the recombiners are compared to temperatures measured elsewhere in the same area to give a qualitative indication that the PARs are operating.

4.2. DESIGN BASIS

The combustible gas control system ETY [CGCS] also has the following sub-functions:

- ETY1: Hydrogen Reduction
- ETY2 and ETY3: Hydrogen Mixing and Distribution (CONVECT)

4.2.1. System design

4.2.1.1. General system design rules

Analyses have been performed with validated computer codes for postulated severe accident scenarios to quantify H_2 release rates, time histories, hydrogen distribution in the containment and efficiency of PAR systems (see Sub-chapter 16.2).

The efficiency of PARs in single and multiple room configurations has been investigated experimentally.

The PAR arrangement is supported by various lumped-parameter and 3D-code analyses for severe accident scenarios in an iterative way:

- Starting with an arrangement based on engineering judgement.
- Assessing modification to the arrangement (distribution between equipment cells, dome and annular rooms).
- Selecting the best arrangement.
- Performing calculations, both with lumped parameter codes and CFD codes for this arrangement to justify the hydrogen mitigation system (see Sub-chapter 16.2).

The combustible gas control system is designed so that the pressure and temperature created inside the containment will not result in containment failure as a consequence of a hydrogen release during a severe accident.

The possibility of creating mixtures of gases that could burn or explode in a way which endangers containment integrity shall be small in all accidents.

To minimise combustion loads, hydrogen must be removed effectively, preferably under elevated or inerted (> 55 vol%) steam conditions.

The global hydrogen concentration shall be maintained below 10 vol%.

The hydrogen concentration shall be reduced below 4 vol% within approximately 12 hours of the start of a severe accident.

This overall goal is reached by requiring containment conditions such that:

- Diffusion flames, that may develop when melt contacts the sacrificial concrete in the reactor pit and the spreading compartment, do not endanger the integrity of the containment during any phase of the accident.

- Deflagrations do not result in pressures exceeding the failure pressure of the containment. The adiabatic isochoric complete combustion (AICC) pressure defines an upper limit for global laminar combustion pressures.

- The risk of significant flame acceleration is small. The local gas mixture composition shall be such that flame acceleration can be excluded with high probability. Practical exclusion of flame acceleration can be demonstrated e.g. if the local sigma index of the gas mixture is maintained below 1.

- The possibility of detonations is practically excluded. The local gas mixture composition shall be such that deflagration to detonation transition (DDT) is extremely unlikely. The possibility of the flame front reaching sonic velocities and initiating DDT is prevented if either:

    - the sigma index of the gas mixture is below 1

  or,

    - if the sigma index is higher than 1, the characteristic dimension of the mixture cloud is less than seven times the detonation cell size (the lambda criterion).

If the criteria for the requirements shown in bullet points 3 and 4 are significantly exceeded, then more advanced methods, such as performing 3D-combustion calculations to determine the combustion loads on the containment structures must be applied. Practically, the 3D-combustion calculation will be performed if requirement 3 (sigma index) is violated because the “characteristic length” is difficult to define for the complex geometrics existing in a real plant.

Definition of the criteria:

- Sigma criterion to ensure flame acceleration cannot reach “sonic” speed.

  The sigma value is defined as the ratio of the density of the unburned gas mixture, to the density of the burned gas mixture at constant pressure. The sigma index is the ratio between the sigma value and an experimentally found limit value for sigma.

  Fast flame acceleration can be excluded if the value of the sigma index is less than 1. The sigma index (which may be further corrected to account for venting effects) depends on gas temperature and the composition of the gas mixture in the various compartments within the containment.

- Lambda criterion for the exclusion of amplification and transition to a stable detonation (DDT).

  The lambda criterion states that transition to detonation is only possible for lambda index values of D/(7*lambda) > 1. D is defined as the “characteristic length” of the mixture cloud or compartment. Lambda is an experimental value called the detonation cell size and depends on the composition of the gas and also, weakly, on the temperature.

  Moreover, the lambda criterion is linked to the sigma criterion. The potential for transition to detonation is indicated by lambda index > 1 but only for gas clouds which also satisfy the sigma index > 1 condition.

4.2.1.2. Design aspects of the PAR system

The PAR system design and layout in the different containment areas is intended to limit and reduce local and global hydrogen concentrations.

Therefore the location of the recombiners considers:

- hydrogen release areas,
- containment compartmentalisation,
- enhancement of containment atmosphere mixing,
- taking credit for global convection paths,
- efficient reduction of hydrogen concentration and minimisation of peak concentrations.

The PAR arrangement is optimised for relevant RRC scenarios. The developed design is also capable of reducing hydrogen concentrations in PCC-2 to PCC-4 events. This is expected because of the distribution of PARs in equipment and operation areas and because of the high hydrogen reduction capacity of the PAR system compared to the low quantities of hydrogen released in PCC-2 to PCC-4 events.

The system design will also minimise containment loads during PAR system and spray system operation. It should consider the atmosphere homogenisation effect as well as the removal of hydrogen under elevated steam concentrations or inerted atmospheric conditions, e.g. > 55 vol% steam, particularly for RRC events.

4.2.1.3. Design aspects for hydrogen distribution and mixing system - CONVECT

This subsystem, designed to create global and local convection loops to promote atmosphere mixing, consists of:

- the section of an equipment cell wall equipped with
    - an atmosphere inlet section, located at lower building level, and
    - an equipment cell atmosphere release section, located at high building location, approximately 30 m above the inlet section.

Atmosphere inlet section equipment cells:

Eight fail-safe opening hydrogen mixing dampers with a free flow cross-sectional area of approximately 5 m² are installed at the lower boundary of the equipment cells between the annular rooms and the IRWST gas space.

They are activated by absolute pressure or pressure difference (measured at the SG-pressure equalisation ceiling). They open fail-safe if power supplies are lost and can be activated manually.

Atmosphere release section equipment cells:

At the upper boundary of the equipment cells, i.e. the SG pressure equalisation ceiling, the following components are installed:

- approximately 100 passive pressure sensitive rupture panels (of different sizes) and

- approximately 120 passive temperature and pressure sensitive convection dampers.

Rupture panels and convection dampers above the SGs, installed in a structural steelwork ceiling, are used to equalise the pressure particularly following a LB(LOCA) within the containment, to prevent unacceptable loads on civil structures.

The rupture panels (including convection dampers) cover a relatively large area (approximately 76 m² in total). These panels open passively when the pressure difference across them reaches approximately 50 mbar.

The convection dampers, covering approximately 10 m^2 above each SG, are also equipped with temperature activated opening mechanisms to ensure the minimum required flow cross-sectional area also for SB(LOCA) conditions where the differential pressure is insufficient to open enough rupture panels.

The design requirements differ for PCC-2 to PCC-4 and RRC events. For PCC-2 to PCC-4 events, the requirement for pressure equalisation must be fulfilled while for RRC scenarios there are additional requirements to ensure appropriate mixing of the containment atmosphere.

The LOCA scenarios are divided into SB(LOCA) and LB(LOCA). For SB(LOCA) even a small open flow cross sectional area enables pressure equalisation while the LB(LOCA) requires the maximum pressure equalisation area to open.

In the event of SB(LOCA) only one or a few rupture panels in the affected loop would open. Under these circumstances hydrogen mixing is improved by convection dampers located above the SGs and hydrogen mixing dampers at the bottom of annular rooms towards the IRWST. At the top of each SG room an opening / interconnection surface consisting of convection dampers with an area of 10 m² is provided and on the bottom of the containment towards the IRWST air space there are eight openings of approximately 5 m². The local and global mixing effects resulting from this cross section was calculated using the 3D-code GASFLOW.

4.2.2. Component Design Basis

4.2.2.1. Design basis for Passive Autocatalytic Recombiners (PAR)

The PARs are designed for pressure, temperature, humidity, atmosphere and radiation conditions anticipated following a severe accident. The qualification of components considers aspects such as poisoning by aerosols from a molten core and operation of the containment spray system containing boric acid.

The components are designed to withstand loads from accident temperatures and seismic events.

The PARs are self-starting at a hydrogen concentration of approximately 2 vol%, but once they operate they continue to deplete hydrogen down to a concentration of about 0.5% per volume (hysteresis effect) [Ref-1]. The hydrogen reduction capability of PARs is proven. The corresponding characteristic was implemented in the 3D-code GASFLOW as well as in the lumped parameter codes and hence was considered in the hydrogen reduction analyses.

4.2.2.2. Design basis for rupture panels

The rupture panels are designed to withstand normal operation air ventilation pressure differences and retain their leak tightness during normal plant operation.

The components are designed to withstand seismic loads. Rupture panels will open passively if the bursting pressure difference of approximately 50 mbar is exceeded [Ref-1].

4.2.2.3. Design basis for convection dampers

The convection dampers are designed to withstand normal operation air ventilation pressure differences, withstand ambient air temperatures and retain their leak tightness during normal plant operation.

The components are designed to withstand seismic loads. The convection dampers will open passively if the ambient gas temperature reaches 80 - 85°C or they will open passively if the bursting pressure difference of approximately 50 mbar is exceeded [Ref-1].

4.2.2.4. Design basis for hydrogen mixing dampers

The hydrogen mixing dampers are designed to withstand temperature, pressure, humidity, radiation and seismic loads anticipated during accidents.

The opening pressures are defined to ensure the function of the hydrogen mixing dampers under the design basis conditions.

The mechanical design pressures are defined to avoid any hazard to nearby safety related equipment in the event of failures.

4.2.3. Fluid characteristics

The fluids are mainly air, steam, hydrogen and the various products released during accidents.

4.3. SYSTEM DESCRIPTION – EQUIPMENT CHARACTERISTICS

4.3.1. System description

The hydrogen control concept is based on the use of 47 catalytic recombiners distributed throughout the containment. The PARs are installed in the accessible part (main operation floor and annular rooms) as well as in the non-accessible part (SG-compartments, RCP [RCS], PZR, PRT, etc.). The arrangement takes account of the containment compartmentalisation and supports global convection, homogenises the atmosphere and reduces the average global hydrogen concentration and local peak hydrogen concentrations.

The lowest level where PARs are installed is the floor above main coolant pipes and the highest level is at the polar crane. The arrangement of PARs is able to cope with different situations during an accident, e.g. high hydrogen concentrations at release sites, stratification caused by steam, depletion after homogenisation, promotion of global convection, etc.

The arrangement (location and number) of recombiners takes into account PCC-2 to PCC-4 events and RRC scenarios.

The hydrogen reduction capacity is based on 41 PARs type FR1-1500T and six PARs type FR1-380T or equivalent types, and corresponds to a nominal hydrogen depletion rate of approximately 220 kg/h (corresponding to 1.5 bar abs and 4 vol% hydrogen concentration). The PARs have to fulfil the requirements mentioned in Sub-chapter 4.3. The PAR characteristics in hydrogen mixing, distribution and reduction shall be consistent with those used in the H_2 -analysis.

The results of the analyses, performed for the justification of the hydrogen control system are presented in Sub-chapter 16.2.

The sub-functions of hydrogen mixing and distribution (ETY2 and ETY3) complete the combustible gas control system.

4.3.2. General system description

The combustible gas control system is divided into subsystems corresponding to their operational functions:

- Hydrogen reduction system.
- Hydrogen mixing and distribution / pressure equalisation system CONVECT.

4.3.2.1. Hydrogen reduction system

The subsystem consists of several passive autocatalytic recombiners (PARs) installed in various parts of the containment.

The PARs reduce the hydrogen concentration in order to maintain containment integrity and leak tightness during severe accidents and in the long-term post-LOCA phase.

For PCC-3 and PCC-4 events small quantities of hydrogen are formed mainly by radiolysis of primary coolant, whilst during RRC large quantities of hydrogen are formed by a zirconium-steam reaction inside the reactor vessel.

4.3.2.2. Hydrogen mixing and distribution system

In the event of an accident, the two zone containment must be transformed into a single-zone containment to ensure hydrogen distribution and mixing in the containment atmosphere, preferably in a passive way.

Therefore, the hydrogen distribution and mixing system for the creation of global convection loops inside the containment atmosphere consists of three different component types:

- mixing dampers in the lower part of containment between IRWST and annular areas and
- convection dampers and rupture panels above each of the four SGs.

During normal plant operation this subsystem ensures the separation of ventilation between accessible and inaccessible parts of containment. The accessible part is mainly the dome area with the main operations floor and the annular areas, whilst the inaccessible part is characterised by equipment cells (SG, main coolant pumps, pressuriser, pressuriser relief tank, IRWST, etc.).

In the event of LOCA or severe accident, steam and hydrogen is released from the primary circuit. All release locations are inside the equipment cells. For LOCAs the hydrogen release occurs at the break location. In the event of reactor depressurisation, the hydrogen and steam would be released via the pressuriser relief tank into the lower SG cells.

During abnormal plant operation (PCC-2 to PCC-4 and RRC events), rupture panels and convection dampers will open completely or partially. To ensure adequate mixing of the atmosphere throughout the containment, the hydrogen mixing dampers must open. The objective is to create a global convection loop within the containment. The mixing results in a more homogeneous distribution of the hydrogen and a lower local maximum concentration.

4.3.3. Equipment characteristics

4.3.3.1. Recombiners

The recombiner comprises a metal housing designed to promote natural convection with a gas inlet at the bottom and a lateral gas outlet at the top. The horizontal cover of the housing at the top of the recombiner protects the catalyst from direct water spray and aerosol deposition. Numerous parallel plates with catalytically active coating zones are arranged vertically in the bottom of the housing. Good accessibility to the catalytic plates is guaranteed through the provision of a removable inspection drawer. This ensures easy in-service inspection and maintenance of the recombiner.

Gas mixtures containing hydrogen are recombined upon contact with the catalytic zones in the lower part of the housing. The thin plate technology catalyst heats up quickly as a result of this reaction. The reduction of gas density in this area causes a buoyancy effect thereby ensuring a continuous supply of large volumes of hydrogen bearing gases thus ensuring high recombination efficiency.

The flat compact design of the modules minimises the obstruction of escape routes, of outage-related storage areas and of space provided for the maintenance of valves and components. The supporting steel structures are designed to allow flexibility with regard to exact installation location e.g. mounting on walls, floor, existing steel structures, etc. (see Section 6.2.4 – Figure 1), whilst ensuring that, in principle, the hot gas exhausts to the free volume of the compartments.

Section 6.2.4 - Table 1 shows the performance (nominal hydrogen reduction capacity) of similar PARs (equivalent to types FR1-1500T and FR1-380T). The hydrogen reduction capacity is mainly dependent on hydrogen concentration and containment pressure. These characteristics are used to develop multi-compartment computer model using codes such as COCOSYS [Ref-1] and WAVCO [Ref-2] and CFD codes such as GASFLOW. The GASFLOW [Ref-3] [Ref-4] code is used for calculations presented in Sub-chapter 16.2 which deals with the hydrogen risk.

Qualification of the recombiners also addresses:

- thermal ageing,
- radiation effects (radiation during normal operating conditions and radiation resulting from a core meltdown accident),
- effects of chemical impurities (catalytic poisons, solvents and weld gases and smoke coming from oil and cable fires),
- endurance testing, and
- vibration testing.

Between 1993 and 1998 the CEA, the IRSN and EDF undertook a large-scale experimental programme at CEA Cadarache (KALI-H2) [Ref-5] [Ref-7] to measure the performance of recombiners under varying conditions (100°C – 140°C, hydrogen concentration corresponding to 10% under dry conditions with and without spray) in a 15.6 m^3 vessel. Some of the tests were conducted in cooperation with EPRI.

An additional experimental programme “H_2 PAR” [Ref-6] was launched at IRSN Cadarache by EDF and the IRSN. This programme was dedicated to the investigation of the impact of catalytic poisons (e.g. iodine and other aerosols) on recombiner efficiency, and to the potential for recombiner ignition when the hydrogen concentration at the PAR intake exceeds a certain level.

The final test on enveloping recombiner catalyst poisoning under real severe accident atmospheric conditions, was a part of the PHEBUS programme (FPT3) [Ref-8]. The FPT3 test was the final experiment within the International PHEBUS FP Programme. A particular objective of this test was to identify the potential poisoning effects of representative containment environment conditions at scaled down hydrogen recombiner samples, exposed to containment environment conditions representative of a realistic severe accident scenario. The purpose of these tests was to determine the effect of fission product poisoning on the catalyst plates from each manufacturer. The obtained results complete the PAR severe accident qualification programme.

The main parameters investigated during the tests are described below:

KALI-H2

Recombiner performance was investigated under spray conditions, including the use of chemicals such as sodium hydroxide and boric acid in water sprayed at different temperatures (100 to 140°C).

The following parameters were assessed in association with EPRI:

- initial hydrogen concentration,
- initial gas temperature and pressure,
- steam and nitrogen impact,
- low oxygen content in the atmosphere,
- poisons emanating from cable fires,
- effects of humidity and low ambient temperatures during start-up and
- recombiners igniting.

H_2 PAR Programme

An innovative installation and test system was developed to conduct experiments with hydrogen concentrations of up to 12% volume, under inerted conditions in a safe environment.

A total of 38 tests were conducted for qualification of the system, measurements, core meltdown simulation events and aerosols produced by this system.

20 experiments were conducted which resulted in the following conclusions:

- Recombination efficiency was confirmed in an air/steam environment up to a maximum steam concentration of 66% volume.
- Under relatively dry conditions (steam concentration < 10% volume) and with hydrogen concentration of approximately 6% volume, certain recombiners produced gas temperatures sufficient to cause ignition of the fuel mixtures and deflagration spreading through the entire building (7 m^3). However, under wet conditions (steam concentration of 25 to 30%) recombination without deflagration was observed at > 6 - 8% volume.
- The presence of aerosols did not cause either poisoning of catalytic reactions or reduction in recombination efficiency.

FPT3 Tests

The international PHEBUS FP (Fission Product) programme is currently the largest reactor safety programme. Several in-pile experiments are being used to investigate the fission products release from a degraded corium, their subsequent transport and deposition in the coolant circuit and the containment.

The PHEBUS FP programme was launched in the late 1980s and is supervised by CEA (French atomic energy commission) in cooperation with the European Commission (EC) and international partners (Germany, USA, Japan, Canada, South Korea, and Switzerland).

The experiments took place in the research reactor PHEBUS in Cadarache, France.

The research programme (FPT3) was successfully completed (including the evaluation of the results from this active core-melt-test).

The multiple precious metal zone catalyst, similar to the types FR1-1500T and FR1-380T, was successfully tested. This type demonstrates the shortest start-up time and the highest efficiency. The efficiency was not influenced, and there was also no degradation drift, during these challenging actual severe accident atmospheric conditions.

These positive results complete the extensive PAR severe accident qualification programme.

4.3.3.2. Hydrogen mixing dampers

The design of the hydrogen mixing dampers (HMD) is typical of components normally used in HVAC-systems and they are equipped with actuators following the “fail-safe open” principle and LOCA proofed position indicators (see Section 6.2.4 - Figure 4).

The damper is normally operated by the actuator. During closing of the damper a spring is compressed which is held in the loaded position by a solenoid brake. In the event of power failure to the solenoid, the spring will drive the actuator and damper without any external energy supply to the safety position “OPEN”.

4.3.3.3. Rupture panels

The rupture panels facilitate pressure equalisation in both directions. They open passively due to the pressure difference across them.

The two-way rupture panels combine protection against unacceptable overpressure and underpressure pressure in one component.

The rupture panels must open without generating fragmented particles (see Section 6.2.4 - Figure 2).

The panels are installed in a structural steelwork forming the SG-ceiling.

4.3.3.4. Convection dampers

The convection dampers consist of rupture panels combined with a temperature sensitive mechanism which opens at a specific temperature in the range 80-85°C [Ref-1]. They open passively either due to the pressure difference or by exceeding pre-defined temperature (see Section 6.2.4 - Figure 3).

The leak tightness of the steel frame is ensured by the use of sealant.

The design combines two standard parts which are separately proven by operation experience.

The convection dampers are installed in the structural steelwork forming the SG cell ceiling.

4.3.4. Instrumentation and control

The PARs, the convection dampers and the rupture panels operate in a totally passive manner and therefore there are no instrumentation or control requirements.

The position of hydrogen mixing dampers is monitored.

Opening of HMDs is initiated by signals indicating LOCA and / or severe accident. In addition, damper opening can be initiated manually from the main control room. In the event of station black-out (complete loss of power) the fail-safe actuator ensures that the dampers open.

The position of the rupture panels and convection dampers is not monitored. The status can be derived indirectly by evaluation of containment pressure difference signals between the various equipment and service areas.

The temperatures downstream of the recombiners are displayed in the main control room.

4.3.5. Interfaces with other systems

The ETY [CGCS] system has no direct interface with other systems.

An operational interface exists between the ETY [CGCS] system used in the event of hydrogen release, the reactor coolant depressurisation system and the spray system.

4.4. OPERATING CONDITIONS

The ETY [CGCS] system is not required in normal and incident operating conditions. It is in stand-by mode.

The ETY1 to ETY3 subsystems are operational in severe accident conditions and in long-term post-LOCA management. System start-up is mainly passive.

4.5. PRELIMINARY SAFETY ANALYSIS

4.5.1. Compliance with regulations

Compliance with general regulations is addressed in Sub-chapter 1.4.

4.5.2. Compliance with functional criteria

The results of analyses carried out to justify the hydrogen control system, presented in Sub-chapter 16.2, show that the design adopted for the ETY [CGCS] system ensures compliance with the safety requirements and functional criteria described in sections 4.0.1 and 4.0.2 of this sub-chapter.

4.5.3. Compliance with design requirements

4.5.3.1. Safety classification

For design and construction of materials and equipment, compliance with the requirements of the classification rules is described in detail in Sub-chapter 3.2.

4.5.3.2. Single failure criterion or redundancy

The ETY [CGCS] system design takes account of the single failure criterion.

4.5.3.3. Qualification

The equipment is qualified in accordance with the requirements described in Sub-chapter 3.6.

4.5.3.4. Instrumentation and control and emergency power supply

The Passive Autocatalytic Recombiners (PARs), the convection dampers and the rupture panels operate without any power supplies.

The HMD actuators are supplied by an emergency diesel generator. The HMDs are designed to “fail-safe open”. Hence in the event of loss of power / station black-out the HMDs will open passively. When the diesel generator starts the HMDs will be closed automatically.

4.5.3.5. Hazards

The general provisions relating to the consideration of external and internal hazards, described in Sub-chapter 3.1 are incorporated into the system design.

4.6. TESTING, INSPECTION AND MAINTENANCE

Test, inspection and maintenance rules for the ETY [CGCS] system will be defined during the detailed design phase.

SECTION 6.2.4 - TABLE 1 (1/3)

Main Equipment Data [Ref-1]

Code No.	Description	Technical Data
ETY [CGCS]	System Design Data Combustible Gas Control Ambient Pressure Max. Ambient Temperature	Technical Data 5.3 bar 156°C
Sub-function ETY1	Main Component Data Hydrogen Reduction	
	Allowable Working Temperature, Max./Min. Passive Autocatalytic Recombiners (PARs)	 500/50°C
101-108RV 111-118RV 121-136RV 139-147RV	Type: FR1-1500T Number of PARs Hydrogen Reduction Rate (nominal at 0.5 bar and 4 vol% hydrogen) Material Weight Catalyst Main Dimensions (length, height, depth)	 41 5.36 kg/h stainless steel max. 140 kg multiple precious metal zone catalyst 1550x1400x326 mm
109-110RV 119-120RV 137-138RV	Type: FR1-380T Number of PARs Hydrogen Reduction Rate (nominal at 0.5 bar and 4 vol% hydrogen) Material Weight Catalyst Main Dimensions (length, height, depth)	 6 1.2 kg/h stainless steel max. 50 kg multiple precious metal zone catalyst 430x1400x326 mm

SECTION 6.2.4 - TABLE 1 (2/3)

Main Equipment Data [Ref-1]

Code No.	Description	Technical Data
Sub-function ETY2	Main Component Data Hydrogen Mixing and Distribution Equipment Rooms Inlet Section Allowable Working Temperature, Max./Min.	 156/20 °C
	Hydrogen Mixing Dampers Material Number Opening cross section between annular compartments and IRWST Type Safe Open Function Opening at Differential pressure Containment pressure De-energise to trip Opening by operator action	 Stainless steel 8 in total approx. 5 m² Based on HVAC - equipment approx. > 35 mbar approx. 0.2 bar Yes Yes

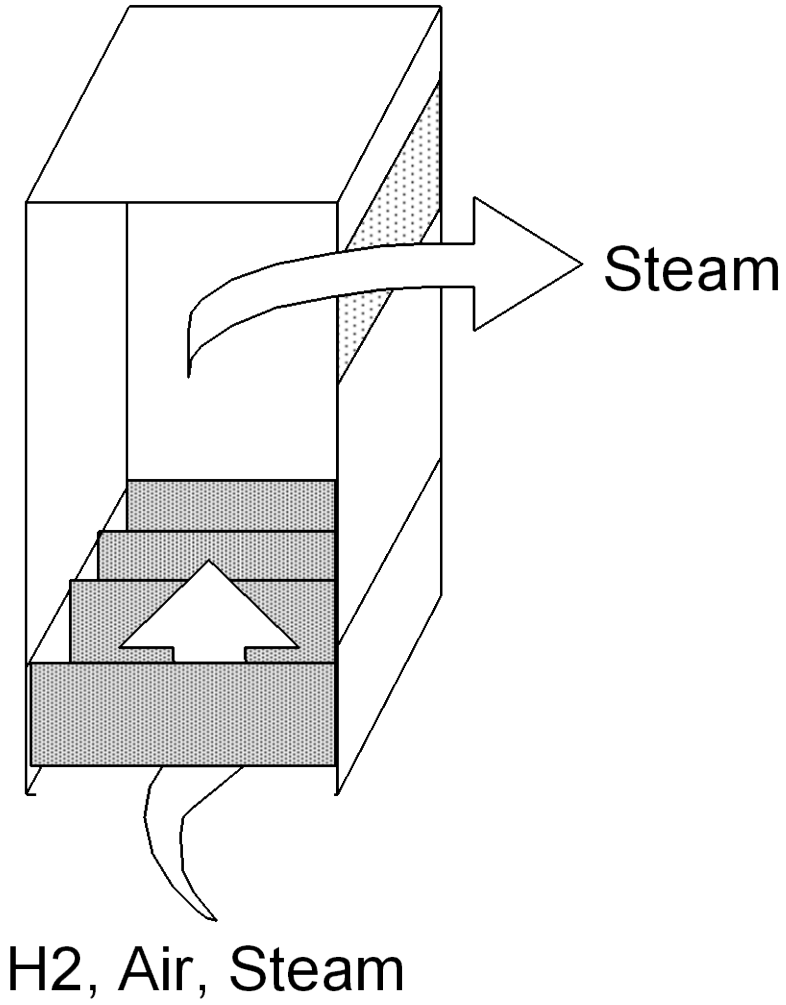
SECTION 6.2.4 - TABLE 1 (3/3)

Main Equipment Data [Ref-1]

Code No.	Description	Technical Data
Sub-function ETY3	Main Component Data Hydrogen Mixing and Distribution Equipment Rooms Release Section Allowable Working Temperature, Max./Min.	 156/20°C
	Rupture panels Material Number Opening cross section Type Opening at differential pressure	 Stainless steel plastic for sealing Approx. 100 (different sizes) Approx. 36 m² Two way rupture damper Approx. 50 mbar
	Convection dampers Material Number Opening cross section Type Opening at Differential pressure Ambient gas temperature	 Stainless steel plastic for sealing Approx. 120 Approx. 40 m² AREVA NP Approx. 50 mbar Approx. 80-85°C

SECTION 6.2.4 - FIGURE 1

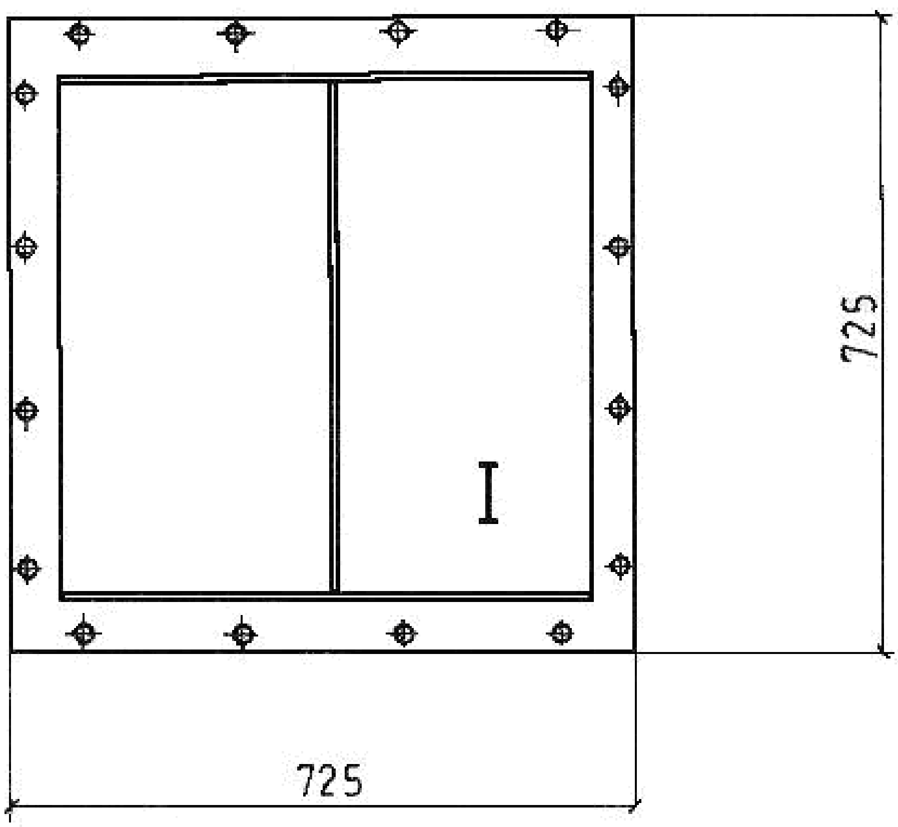
Function of Passive Autocatalytic Recombiner (PAR) Type AREVA FR1



SECTION 6.2.4 - FIGURE 2

Schematic Drawing of Rupture Panels for SG Cell Ceiling

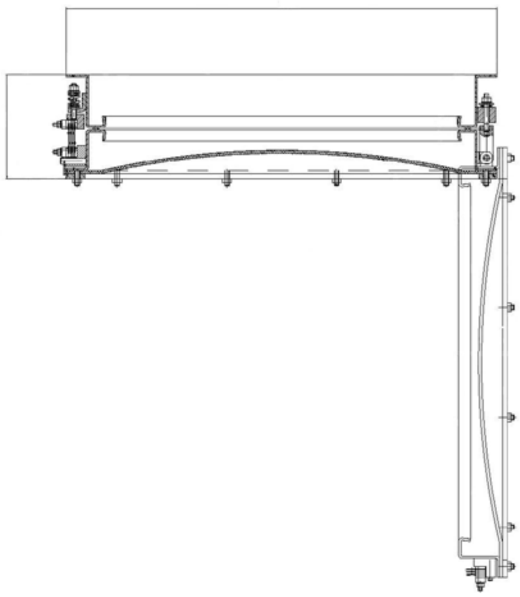
(Dimensions not Binding)



SECTION 6.2.4 - FIGURE 3

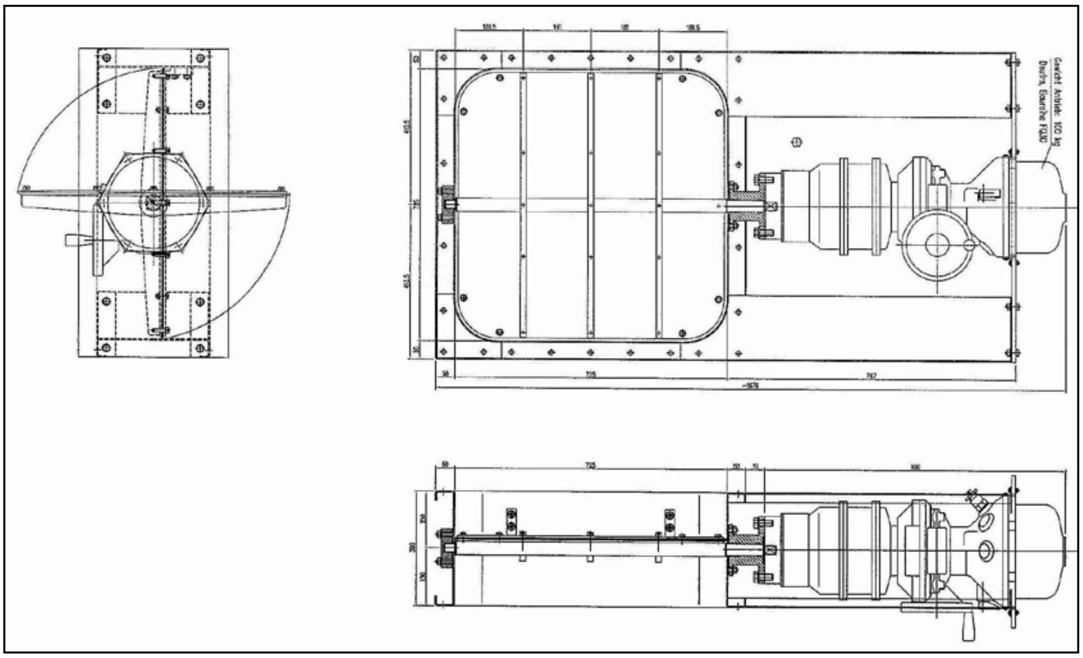
Schematic Drawing of Convection Dampers

(Dimensions not Binding)



SECTION 6.2.4 - FIGURE 4

Schematic Drawing of Hydrogen Mixing Damper
(Dimensions not Binding)



5. LEAK RATE CONTROL AND TESTING SYSTEM (EPP) [REF-1] TO [REF-4]

5.0. SAFETY REQUIREMENTS

5.0.1. Safety functions

The (EPP) system (for control and testing of containment leakage) contributes to the confinement of radioactive substances (see section 1 of this sub-chapter). The containment structure and associated systems are designed to limit the release of radionuclides to the environment, in accidents involving fission product release.

5.0.2. Functional criteria

The (EPP) system ensures that the maximum leak rate from the containment does not result in permissible radiological consequences being exceeded in any PCC (plant condition category) or RRC-A and B (risk reduction category) event or accident.

The (EPP) system ensures the collection and recovery of potential leaks when there is a risk of containment bypass.

5.0.3. Design bases

The (EPP) system is safety classified in accordance with the classification rules defined in Sub-chapter 3.2.

The safety classification requirements with respect to the single failure criterion, emergency power supply, qualification and periodic tests, and hazards considered in the system design differ depending on whether or not the system components contribute to the containment function.

5.1. FUNCTION OF THE EPP SYSTEM

The (EPP) system (for control and testing of containment leakage) contributes to the confinement of radioactive substances (see section 1 of this sub-chapter). The containment structure and associated systems are designed to limit the release of radionuclides to the environment, in accidents involving fission product release.

The (EPP) system fulfils the following functions:

- containment leak rate measurements during periodic containment tests (type A leak-tightness tests);
- containment pressure testing during pre-operational commissioning tests;
- containment pressurisation and depressurisation during periodic containment tests and pre-operational pressure tests;

- local measurement of the leak rate at penetrations (type B tests) and isolation valves (type C tests);
- monitoring of the containment leak-tightness and ambient conditions in the containment during operation (performed by the SEXTEN system [Ref-1]).

The (EPP) system supports access to the reactor building:

- via emergency and personnel airlocks for personnel and small equipment, during any operating state without affecting the containment function;
- via the equipment hatch for large equipment, when the reactor is shutdown.

During any PCC and RRC events the (EPP) system:

- contributes to ensuring that the maximum leak rate from the containment does not result in permissible radiological limits being exceeded. This applies to the equipment hatch, the site access area, emergency and personnel air locks, and the sleeves of any EPP system penetrations;
- ensures collection and recovery of potential leaks at containment penetrations, thus reducing the risk of containment bypass. [Ref-1]

5.2. DESIGN BASIS

Overall leak rate criterion:

The criterion defined for the maximum permissible leak rate under accident conditions is 0.3% per day of the gas mass inside the containment.

In severe accidents, the maximum design pressure and temperature for the containment are 5.5 bar absolute and 170°C.

The leak-tightness and pressure tests along with the associated criteria are defined in ETC-C Part III and are presented in sections 5.3.3 and 5.4 of this sub-chapter.

Leak collection:

In order to eliminate potential leaks to the environment, a leak recovery system is provided at the following penetrations:

- ventilation penetrations;
- equipment hatch, personnel and emergency airlocks and fuel transfer tube;
- fluid penetrations presenting radiological risks.

5.3. DESCRIPTION OF THE SYSTEM

5.3.1. Leak extraction system

In order to prevent containment bypass, a leak extraction system is provided at those containment penetrations where there is a potential risk of direct leakage into the atmosphere.

The leaks are collected by the (EPP) system and discharged into the containment annulus. The flow in the leak extraction system is driven by the negative pressure created and maintained in the containment annulus by the EDE [AVS] System (Containment Annulus Ventilation System).

The leak extraction system is provided at the following penetrations:

- Equipment hatch, personnel and emergency airlocks and fuel transfer tube.

  Potential leaks through the double seals are collected and transferred to the containment annulus by the leak extraction system.

- Ventilation penetrations.

  Potential leaks from the outer isolation valve are collected and transferred to the containment annulus by the leak extraction system.

- Pipe penetrations with potential radiological risks.

  Potential leaks from the outer isolation valve are collected and transferred to the containment annulus by the leak extraction system.

  This applies to those penetrations where the inner isolation valve is normally connected directly to the containment atmosphere, or which transport primary coolant at a temperature greater than 100°C.

5.3.2. Containment leak-tightness monitoring system

The SEXTEN system monitors containment leak-tightness and containment ambient conditions in normal operation. It comprises:

- temperature sensors,
- hygrometers,
- pressure sensors.

It should be noted that the SEXTEN monitoring and measuring system is a sub-system of the monitoring system used during Type A leak-tightness tests.

5.3.3. Leak-tightness test system

5.3.3.1. Definition of the different types of tests

Three types of tests are used to evaluate the leak-tightness of the containment with its steel liner and associated penetrations:

- Type A: overall test which determines the total leak rate from the internal containment,
- Type B: partial test which determines the local leak rate through specific containment penetrations equipped with seals,
- Type C: partial test which determines the local leak rates through the containment isolation valves.

Before plant commissioning, the internal containment is tested once at the maximum test pressure to demonstrate effective containment design in terms of leak-tightness (type A) and pressure resistance (see section 5.4.1 of this sub-chapter).

5.3.3.2. Description of the different types of tests

<u>Type A overall tests</u>

The Type A overall test is performed to assess the containment leak-tightness. This test involves gradually pressurising the containment in a succession of stages with measurements taken at each stage.

The leak rate is evaluated by the absolute method.

A monitoring system collects the required data to estimate the total leak rate (thermometers, hygrometers, pressure gauges, etc). The leak monitoring system used during plant operation is a sub-system of this monitoring system.

At the beginning of the tests, measurements are taken at atmospheric pressure to determine measurement system errors.

Additional measurements are taken at intermediate pressure levels between atmospheric and the maximum test pressure. The measurements are then repeated as the containment pressure is reduced to atmospheric pressure.

During Type A tests, containment behaviour is monitored using the Containment Instrumentation System (EAU).

<u>Type B and C partial tests</u>

The purpose of these tests is to detect and measure local leaks through the barrier formed by the reactor building containment and its penetrations. The leak rate is evaluated using the direct leak measuring method, the pressure blow-down method or any other suitable method.

The partial leak-tightness tests are divided into Type B and Type C tests:

- Type B tests determine local leaks through the leak-tight seals of specific containment penetrations:

- Equipment hatch,
- Personnel airlock,
- Emergency airlock,
- Fuel transfer tube, and the other penetrations equipped with closures using a removable plug and gasket seal,
- Electrical penetrations (canisters).

- Type C tests determine local leaks through the containment isolation valves (see section 3 of this sub-chapter addressing containment isolation).

  For systems that do not form part of a closed circuit outside the containment, Type C leak tightness tests are carried out if the following three criteria are met:

  - The system pipework crosses the internal and external containment walls.
  - The system is connected to the containment atmosphere.
  - The system transports gas.

  For other systems which form part of a closed circuit outside the containment (such as the RIS [SIS] Safety Injection System and the RCV [CVCS] Chemical Volume and Control System), no Type C leak tightness tests are carried out during outages. However, monitoring of physical parameters, such as pressure and temperature, enables an estimate of the leak-tightness to be made.

The design criterion is that direct leaks through penetrations measured in Type B and C tests must not account for more than 60% of the maximum permissible leakage from the containment. This criterion includes a margin to take account of ageing between two successive tests (factor of 0.75) [Ref-1].

The containment penetration seals requiring Type B tests, and the containment isolation valves requiring Type C tests, are tested after installation and periodically thereafter.

5.4. TESTS, MAINTENANCE, INSPECTION

5.4.1. Implementation tests

In addition to the various leak-tightness inspections carried out in the plant or on site when the containment is being built e.g.:

- inspection of welds in the steel liner and associated penetrations;
- inspection of leak-tightness of the personnel and emergency airlocks frames;
- inspection of leak-tightness of electrical penetrations (canisters) and containment isolation valves;

the following pre-operational tests are carried out (see section 5.3.3.2 of this sub-chapter):

- Type A overall test at design pressure;
- Type B partial tests at design pressure (local leak rate through specific containment penetrations equipped with seals);
- Type C partial tests at design pressure (local leak rates through the penetrations equipped with isolation valves);
- Resistance test at design pressure increased by 10% (see below);
- test to evaluate the leak rate through the external containment.

The use of pressure greater than the design pressure allows for the effect of temperatures reached during accident scenarios on the steel liner and the thrust that it exerts on the concrete structure. The containment instrumentation (EAU) is used to confirm the elastic behaviour of the containment and compliance with the design studies.

A primary containment leak rate of 0.3% per day of the gas mass in the containment free volume in severe accident conditions, corresponds to a test leak rate in the pre-operational leak test (5.5 bar absolute at 20°C) of 0.155% per day, assuming a margin is included for ageing (factor of 0.75) and allowing for the fact that in the test the containment contains air [Ref-1].

5.4.2. Periodic tests and maintenance

Containment leak-tightness tests

The following tests are undertaken (see section 5.3.3.2 of this sub-chapter):

- Type A overall test;
- Type B partial tests (local leak rates through specific containment penetrations equipped with seals);
- Type C partial tests (local leak rates through penetrations equipped with containment isolation valves).

Equipment hatch, personnel and emergency airlocks:

The leak-tightness of the following equipment is tested during refuelling outages:

- the equipment hatch and airlocks double seals and the leak recovery systems;
- the isolation valves and the mechanical parts that operate in the airlock opening;
- the electrical penetrations installed in the airlocks.

The following items are visually inspected during every refuelling outage:

- the double seals of airlocks and equipment hatch doors;
- the seals of the airlock pressure equalising system;
- the airlock view ports;

- the opening controls of airlock doors;
- the mechanical equipment used to open the airlock doors.

Containment penetration sleeves – leak extraction by the (EPP):

The following items are tested during refuelling outages:

- the partial leak rate of the penetrations;
- the partial leak rate of the isolation valves and the leak recovery system;
- the containment isolation procedure used during accidents.

6. BASEMAT PROTECTION

6.1. PRINCIPLE OF THE CORE MELT STABILISATION SYSTEM [REF-1]

The UK EPR is equipped with a core melt stabilisation system (CMSS) to prevent basemat attack and melt through by a core melt during a severe accident. A cross-sectional view showing the main components of the system is presented in Section 6.2.6 - Figure 1.

The system is designed to spread the molten core into a core catcher located laterally to the reactor pit. The system allows the core melt to spread into a coolable configuration by significantly increasing the surface/volume ratio of the melt. The decay heat is removed from the melt's upper surface by flooding and quenching from the top, and at the melt's underside and lateral boundaries by the cooling structures of the core catcher.

The overall approach includes temporary melt retention in the reactor pit in order to ensure long-term melt stabilisation in the core catcher regardless of the uncertainties involved in the late phases of in-vessel accident progression, notably the melt release sequence, melt composition and corresponding melt properties. In order to achieve the principal objective, the temporary melt retention is designed to ensure:

- Accumulation of the melt inventory in order to discharge it into the core catcher in a single event.
- Melt conditioning to achieve properties favourable for melt spreading at the same time as unifying the various melt compositions evolved during in-vessel accident progression.

Both objectives are achieved by a layer of sacrificial concrete, through which the melt has to erode before it contacts and destroys the melt gate and spreads into the core catcher. As shown in Section 6.2.6 - Figure 1, the sacrificial layer is backed by a protective layer of zirconia.

The coolant water necessary for heat removal from the melt and, thus for long-term stabilisation, is passively and continuously provided by the in-containment refuelling water storage tank (IRWST). In effect, the inclusion and stabilisation of the core melt within a permanently cooled cooling structure avoids the interaction of the molten core with the structural concrete and consequently also avoids:

- a weakening of load-bearing structures and penetration of the embedded liner
- a local melt through of the basemat and contamination of the subsoil
- a sustained release of non-condensable gases into the containment atmosphere
- a long-term heat-up of the residual basemat structure and the resulting mechanical deformation of the containment building with the risk of crack formation and leakage.

In addition to passive cooling, the concept for heat removal from the core catcher includes an active cooling mode using the EVU [CHRS], which can be applied in the long-term. In this mode, either one or both EVU [CHRS] trains are used to directly feed water into the core catcher. Active cooling is advantageous as it results in a sub-cooled water pool above the melt, thus terminating steam release into the containment and eventually leading to atmospheric pressure.

As a result of arranging the core catcher lateral to, instead of directly underneath the reactor pressure vessel (RPV), the core-catcher is protected against potentially critical loads resulting from failure of the RPV. At the same time, this arrangement ensures that the core catcher cannot impair plant normal operation or the operation of systems needed for control of design basis accidents.

6.2. DESIGN BASIS

The severe accident scenario underlying the design of the CMSS constitutes a set of representative scenarios including a Loss of Coolant Accident (LOCA) due to a break of the pressuriser surge-line combined with subsequent failure of all medium and low head safety injection systems.

Failure to re-establish sustained cooling of the core results in core dry-out and disintegration of fuel assemblies. Finally, a molten corium pool will form inside the core, which expands towards the heavy reflector and the lower core support plate. During this process, the remaining intact fuel elements will be destroyed.

Since the melt is primarily oxidic, its contact with the heavy reflector will not lead to instant failure of this structure but to a slow, crust-limited heat-up. Due to its high mass and correspondingly high thermal inertia, the heavy reflector together with the lower support plate will act as a temporary internal core catcher. Hence, it is expected that this intermediate molten pool will already contain a large fraction of the core. Melt-through of the heavy reflector is expected to occur in the upper region of the molten pool. During movement into the lower head the out-flowing melt will steadily widen the initial hole.

As a result of the contact with the residual water in the lower head, the released melt may either form a partially fragmented debris bed and/or an encrusted molten pool. After evaporation of the residual water, a secondary molten pool will form within the lower head. The lower support plate will then be heated from both sides, by convection from above and by thermal radiation from below.

Both pools will evolve independently. Within the upper pool, remaining fuel and solid debris will heat-up and newly created melt will exit through the existing hole in the heavy reflector and be incorporated into the lower pool. During this process the average temperature of the lower head will steadily increase, which leads to its deformation by thermal expansion and creep. Downward expansion of the lower head is ultimately limited by the concrete support structures provided at the bottom of the pit. These structures provide sufficient space for the outflow of melt and the later formation of a molten pool in the pit.

Eventually the RPV lower head will fail thermally. This is most likely to occur at a location in the upper part of the melt-contacted region. Therefore it is expected that only part of the contained melt will be released with the first pour. Subsequently, further outflow into the pit will depend on the development of the melt configuration within the RPV.

Under the expected dry conditions, the lower head will be subject to radiant heating from the surface of the Molten Core Concrete Interaction (MCCI) pool and/or the surrounding aerosol-rich gas. This heat flux will accelerate the failure of the lower head and lower internals of the RPV.

6.3. BOUNDARY CONDITIONS, REQUIREMENTS AND DESIGN FOR THE REACTOR PIT

6.3.1. Boundary Conditions and Requirements

The following requirements are considered in the design of the reactor pit:

- Prevention of energetic fuel coolant interactions including steam explosions by ensuring a dry reactor pit at RPV failure.

- Protection of the melt gate against the impact of the RPV lower head, e.g. by minimising the free fall height of the lower head and providing fins to bear the impact forces.

- Limitation of melt dispersal by establishing a tortuous way for the melt to escape from the pit for RPV failure pressures up to 20 bar.

- Provide the temporary melt retention function by a sacrificial concrete layer, which is eroded by and mixed with the melt thereby providing the time window necessary for melt accumulation and conditioning of the melt for spreading.

- Enable spreading of the melt inventory in the reactor pit in a single event after melt accumulation and conditioning is complete using a melt gate, which only opens upon contact with the melt.

- Limitation of the sideward concrete ablation to the depth of the sacrificial concrete layer by a refractory layer consisting of zirconia backing the sacrificial concrete.

- Exclusion of potential melt escape paths other than the melt discharge channel by positioning the intakes of the reactor pit ventilation above the maximum melt level in the reactor pit

- Transfer of the melt from the reactor pit to the core catcher via a melt discharge channel, which is protected by a refractory layer to minimise thermal loads on the structural concrete located underneath.

- Accessibility of the reactor pit during plant shutdown using a removable melt plug.

6.3.2. Design of the Reactor Pit

<u>Loads due to RPV Failure</u>

The reactor pit is designed to withstand the loads resulting from a failure of the RPV. These loads include melt jets and the potential mechanical impact of the detached lower head. The latter is borne by fins located around the centre of the bottom of the pit, which also protects the melt plug (see Section 6.2.6 - Figure 2).

Temporary Retention in the Reactor Pit

To enable the transfer of the core from the pit into the lateral core catcher, the molten core must first be accumulated and conditioned in the pit.

The provisions for melt accumulation below the RPV are designed to cope with the fact that the release of molten material from the RPV during a severe accident (SA) is unlikely to take place in one pour, but over a period. Without this prior accumulation, undefined and potentially unfavourable conditions could arise for melt spreading into the lateral core catcher.

Retention and accumulation is achieved by a layer of sacrificial concrete that has to be penetrated by the melt in the course of the MCCI. During temporary retention, the residual heat generated in the melt is partially consumed by the ablation and chemical decomposition of the sacrificial concrete. Another, almost equal fraction is transferred to the residual RPV. This results in a progressive heat-up of the lower head, which finally leads to its thermo-mechanical failure and, as a consequence, to the incorporation of the lower head and residual core into the corium pool in the reactor pit.

The composition of the sacrificial concrete used in the reactor pit is specifically selected to meet the requirements of temporary melt retention. The concrete aggregates mostly consist of Fe_2O_3 and SiO_2 in approximately equal proportion in wt%, while 15% wt of common cement in the dry concrete mixture is used as a binder.

Fe_2O_3 favourably contributes to oxidising the chemically aggressive metals Zr and U. The reaction by-product, Fe, does not affect the thermo-chemical characteristics of the melt. In addition, after the dissolved metal inventory in the oxide depletes, surplus Fe_2O_3 accumulates as FeO and Fe_3O_4 in the oxide melt, thus reducing the liquidus temperature and correspondingly, the temperature level at which the MCCI takes place. This effect is beneficial as, in combination with the formation of silicates, it contributes to reducing the fission product release from the MCCI pool.

Fe_2O_3 and SiO_2 are provided in their natural form as iron ore and siliceous pebbles. Hence, some impurities such as $MgCO_3$ (dolomite) and $CaCO_3$ (limestone) may be found in the concrete system. The concrete has similar mechanical characteristics as ordinary concrete and a water content of less than 5% wt.

The sacrificial concrete layer is backed by a refractory layer made of a ZrO_2-based ceramic (see Section 6.2.6 - Figure 2). This layer restricts the concrete ablation to the depth of the sacrificial concrete layer and thus protects the load-bearing concrete structure of the reactor pit against melt attack and excessive thermal loads.

Melt discharge from the reactor pit and melt spreading

The enclosure by the refractory layer has a predefined weak-point in the centre of the pit bottom which comprises a melt plug separating the pit from the melt discharge channel below. The plug is removable to provide access to the lower pit during plant shut-down. In its locked-in position the melt plug acts as an integral part of the surrounding sacrificial concrete layer. Therefore, in a core melt accident, the length of the melt accumulation period is determined by the time needed to erode the sacrificial concrete cover of the melt plug.

The 2.4 m^2 plug, see Section 6.2.6 - Figure 2, consists of a metal plate covered by a sacrificial concrete layer. The thickness of the complete plug is about 600 mm. The concrete has the same composition as the sacrificial concrete layer inside the reactor pit.

The metal plate at the bottom of the plug is made from an aluminium alloy. The plate is supported by steel beams, which transfer the forces resulting from the pressure loads at RPV failure into a lateral steel frame. The whole plug assembly can be removed through the melt discharge channel to give access to the reactor pit. Once the concrete cover of the plug is eroded, the melt will thermally attack and penetrate the lower metal plate of the melt plug, which results in a steady outflow and transfer of the melt into the core catcher. This takes place via the discharge channel that connects the reactor pit with the spreading compartment. This channel consists of a steel structure embedded within the concrete of the reactor pit. The bottom and side walls of this structure are layered with 200 mm protective material. The free inner dimensions of the channel are approximately 1200 mm in height and 1300 mm in width.

6.4. BOUNDARY CONDITIONS, REQUIREMENTS AND DESIGN OF THE CORE CATCHER

6.4.1. Boundary Conditions and Requirements

The relevant requirements are as follows:

- Protection of the containment basemat and the embedded liner against melt attack and unacceptable temperature loads by means of a core catcher.
- The inside of the core catcher, which later contains the melt, must be dry when the melt arrives.
- The surface of the cooling structure is protected against transient melt contact during melt spreading by a layer of sacrificial concrete. The thickness of this layer must be chosen so that it will not be penetrated during initial melt spreading.
- The core catcher and the melt must be flooded and cooled passively with water provided by the IRWST.

6.4.2. Mechanical Design of the Core Catcher

The EPR core catcher, into which the melt is transferred after its release from the pit, is a shallow crucible consisting of a cooling structure covered with a layer of sacrificial concrete. Bottom and side walls of the core catcher's cooling structure are composed of a large number of cast iron elements (see Section 6.2.6 - Figure 3, 4).

The flexible connection of these elements using a 'tongue and groove' technique is chosen to make the resulting structure largely insensitive to thermal expansion and deformation arising from temperature gradients. To enhance downward heat transfer, the bottom elements have fins that form rectangular, horizontal cooling channels (see Section 6.2.6 - Figure 5).

The lower sidewall elements have the same thickness as the bottom elements and are based on the same design concept. The top of the steam vent is covered by a lid-type structure, which prevents the entry of material that could have potentially been splashed by local fuel-coolant interactions. The inside of the cooling structure is lined with concrete.

The arrival of the melt in the core catcher triggers the opening devices that initiate the gravity-driven flow of water from the IRWST into the spreading compartment. The water first fills the central supply duct underneath the core-catcher. From there, it enters the horizontal cooling channels and then fills the space behind the sidewall cooling structure. The flow direction is indicated with arrows in Section 6.2.6 - Figure 4.

These lines are designed to yield an initial flooding rate of about 100 kg/s. Given this rate, the space behind the sidewall cooling structure will fill up in about 5 min and then water will pour onto the surface of the melt [Ref-1]. Overflow will continue until the hydrostatic pressure levels between IRWST and spreading room are balanced. This is shown in Section 6.2.6 - Figure 6.

In parallel with the continuous inflow of water, the spread melt interacts with the sacrificial concrete that covers the inner side of the horizontal and vertical cooling elements. The sacrificial concrete in the core catcher must:

- provide an easily accessible surface
- delay the contact between melt and cooling structure
- mechanically protect the structure during melt spreading
- reduce the temperature and density of the melt prior to its contact with the cooling structure
- promote melt fragmentation at the surface during flooding
- further improve fission product retention in the melt.

All of the above targets can be achieved with standard siliceous concrete with a minimum thickness of 10 cm [Ref-2].

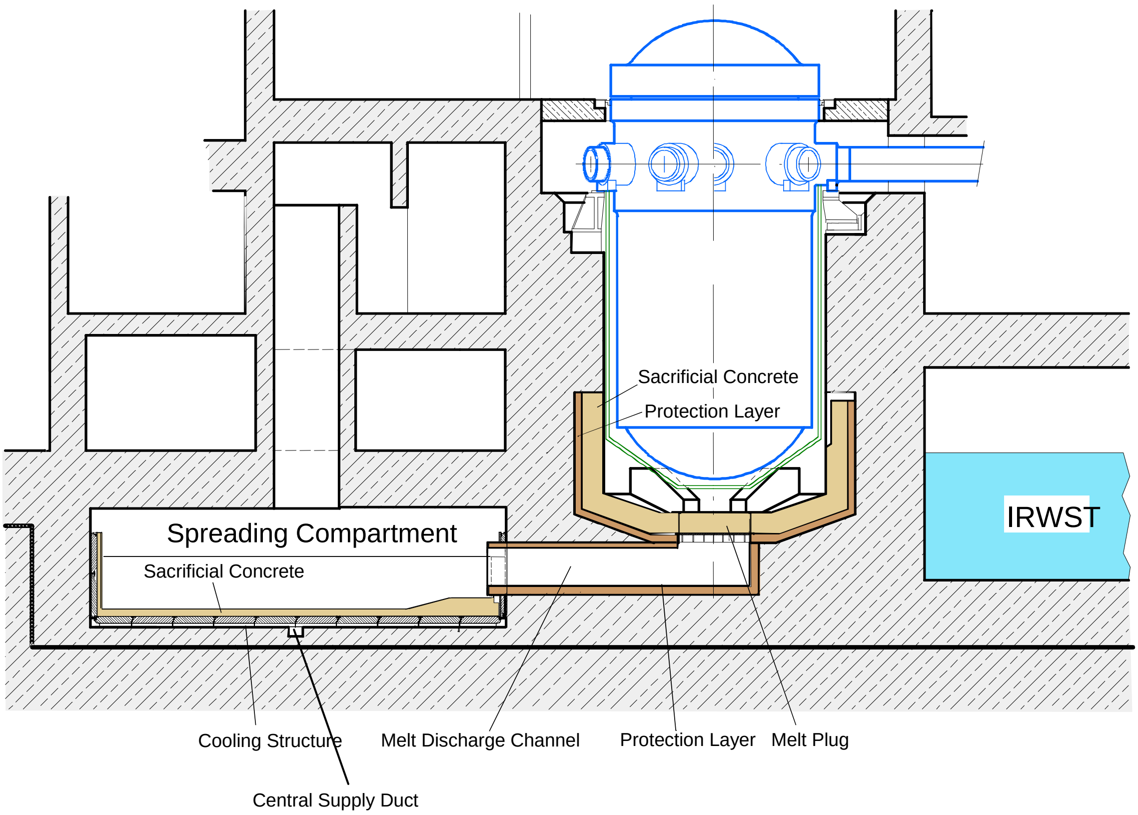
6.5. SAFETY CLASSIFICATION

The core melt stabilisation system must fulfil the following safety functions:

Description of function	**Safety class**
Protect the containment liner embedded in the basemat against attack and destruction by a core melt in RRC-A-events	F2
Cooling of, and decay heat removal from, the core melt	F2

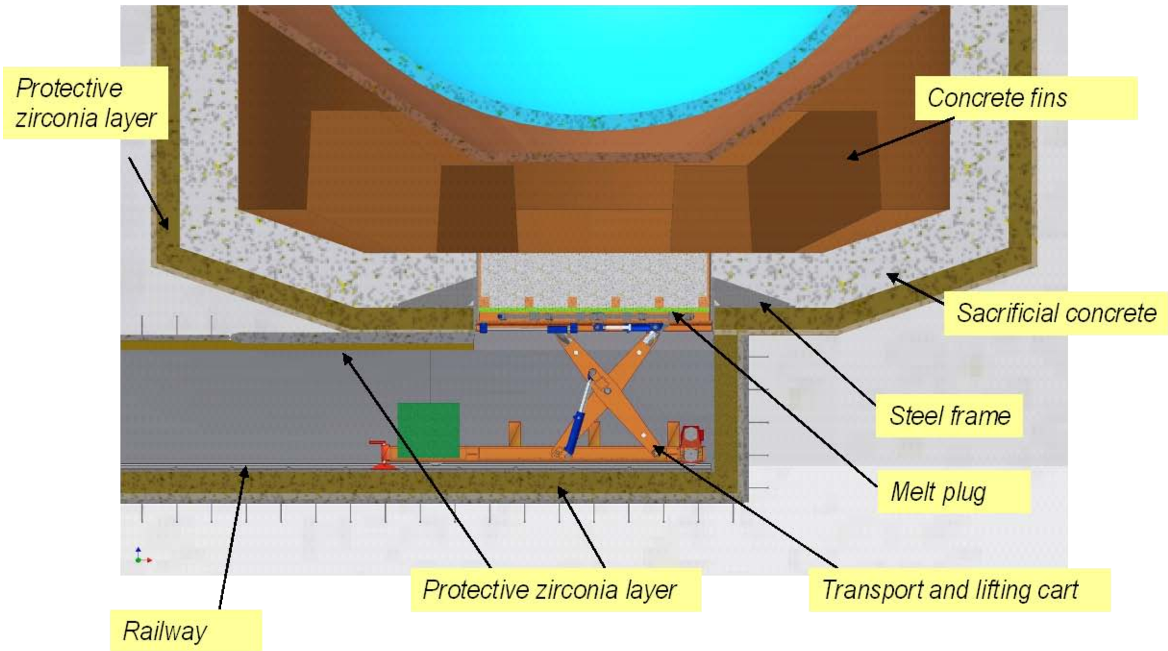
SECTION 6.2.6 - FIGURE 1

Main Components of the Core Melt Stabilisation System



SECTION 6.2.6 - FIGURE 2

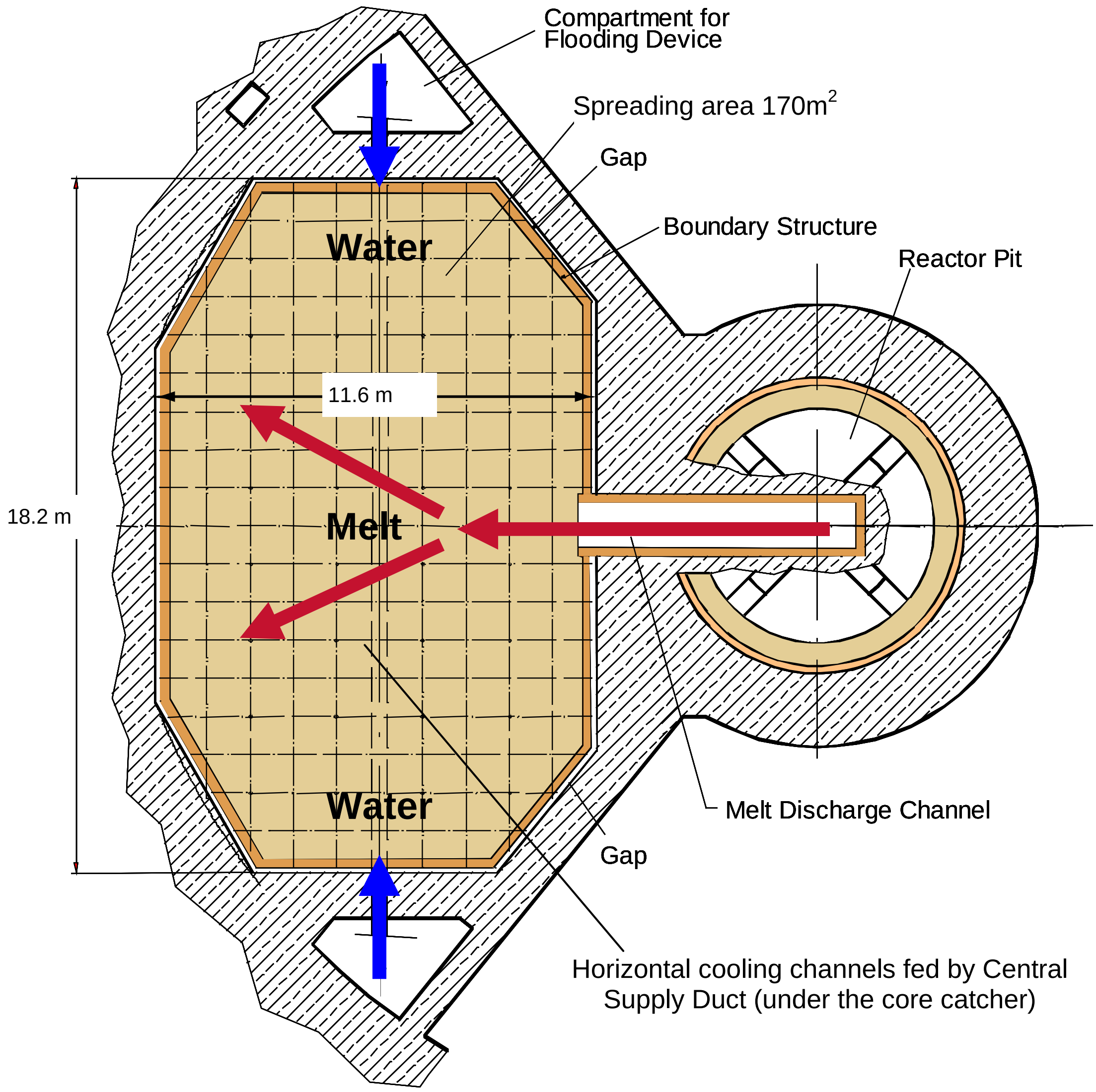
Cross-section through the lower pit, showing components of the CMSS in this area (dimensions are not binding for execution)



SECTION 6.2.6 - FIGURE 3

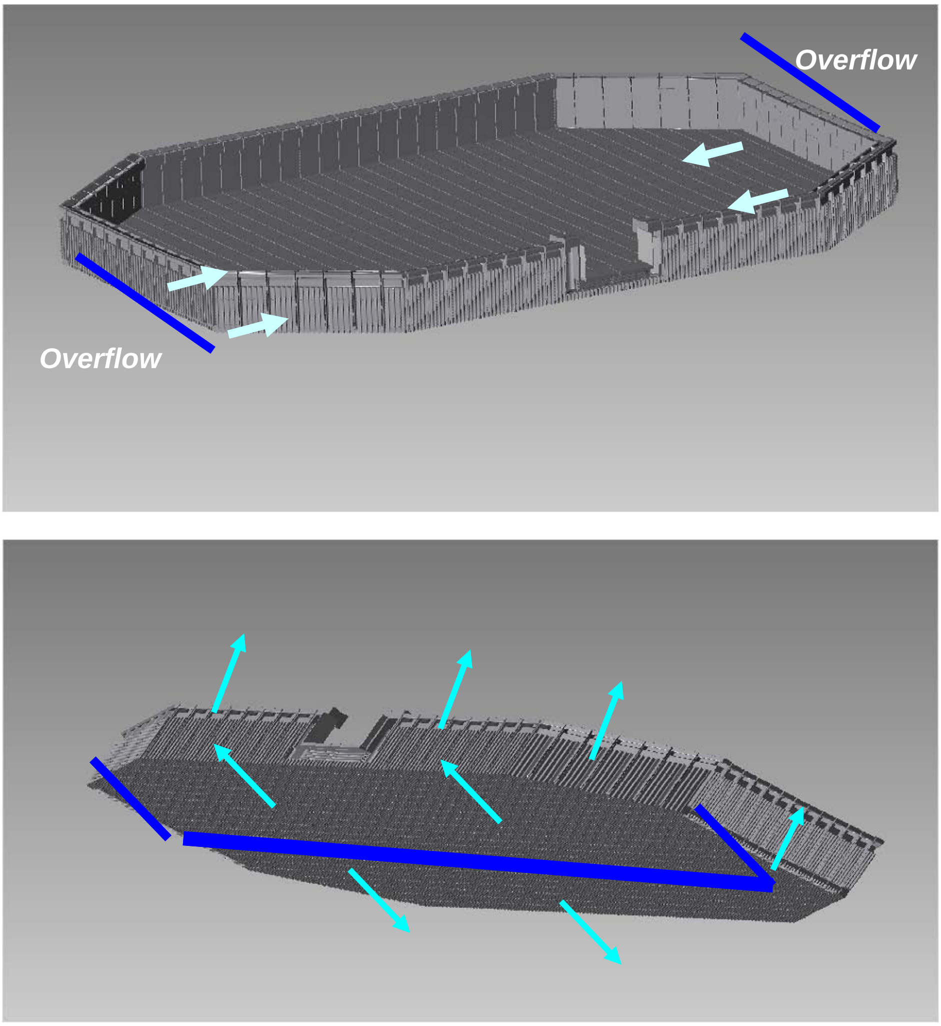
Top view of pit, transfer channel and core catcher, illustrating melt spreading (red) and water flow into the central supply duct (blue)

(dimensions are not binding for execution)



SECTION 6.2.6 - FIGURE 4

Cooling structure and water flow paths



SECTION 6.2.6 - FIGURE 5

Design example of rectangular bottom cooling element and sidewall cooling element

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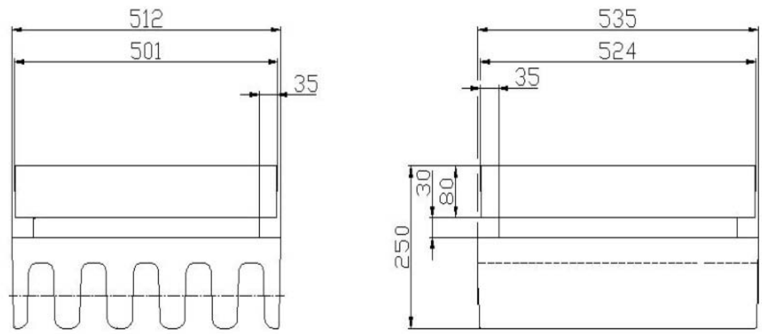
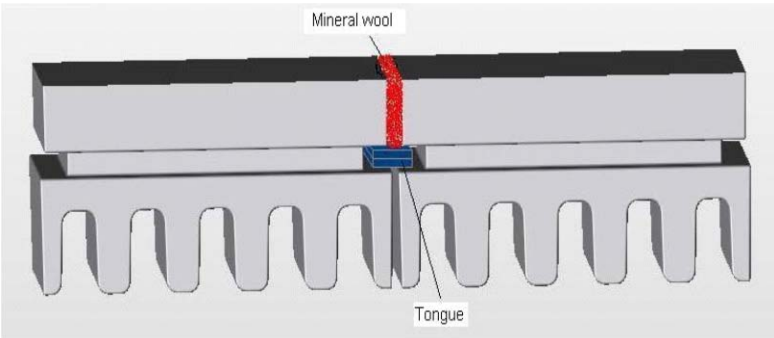
(dimensions are not binding for execution)

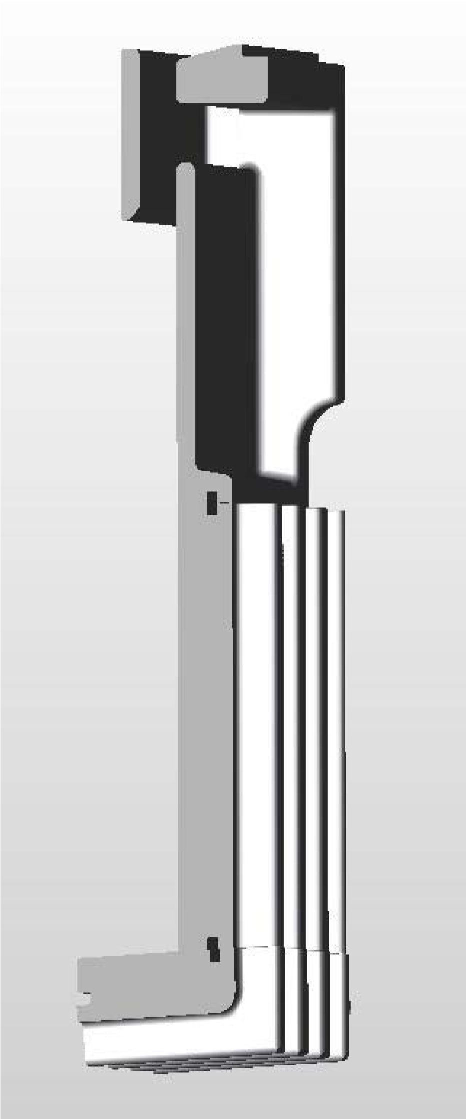




SECTION 6.2.6 - FIGURE 5 (CONT'D)

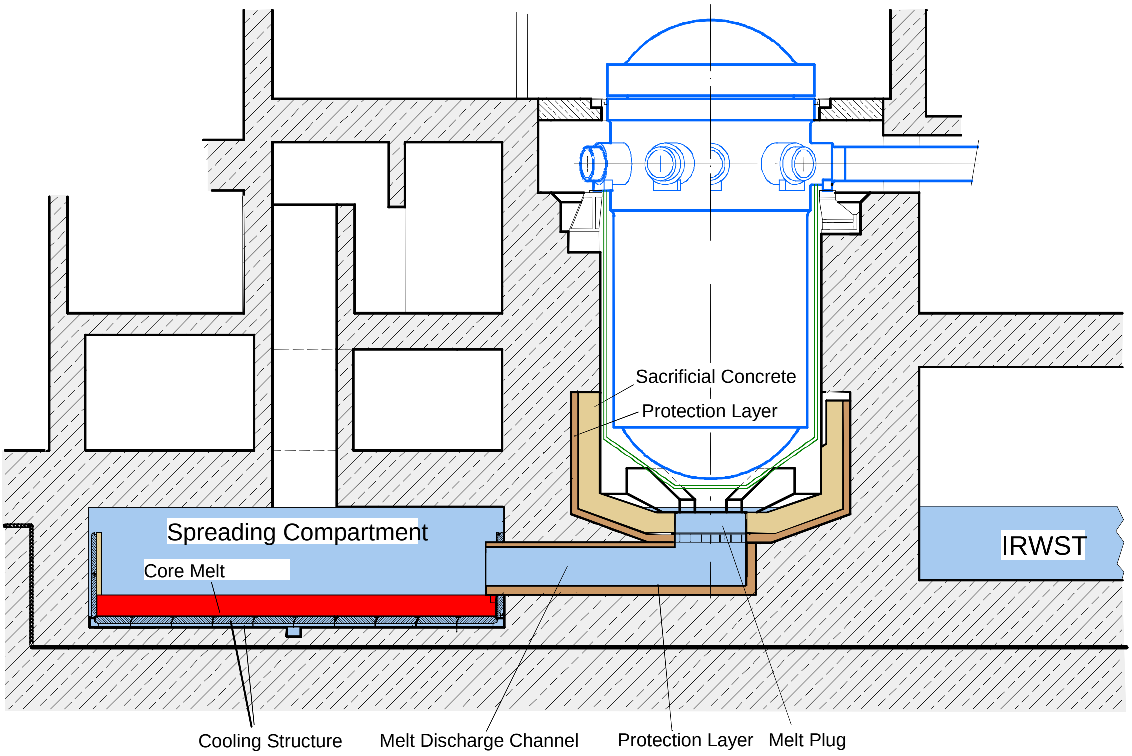
Design example of sidewall cooling element

(dimensions are not binding for execution)



SECTION 6.2.6 - FIGURE 6

Water level in the core catcher after passive flooding



7. CONTAINMENT HEAT REMOVAL SYSTEM (EVU [CHRS]) [REF-1] TO [REF-7]

7.0. SAFETY REQUIREMENTS

7.0.1. Safety functions

The main functions of the EVU [CHRS] system are to limit the pressure inside the containment and to ensure decay heat removal from the containment during severe accidents (Risk Reduction Category-B).

Control of reactivity

The system does not directly contribute to the reactivity control safety function.

Decay heat removal

a) The EVU [CHRS] system transfers decay heat from the in-containment refuelling water storage tank (IRWST) to the ultimate cooling water system using a dedicated cooling system:

- during severe accidents (Risk Reduction Category-B),
- in two RRC-A accidents:
  - pipe breaks with loss of the low head safety injection (LHSI)
  - total loss of RRI/SEC [CCWS/ESWS] cooling chains in state D
- transiently in certain PCC-4 pipe breaks outside the containment with the safety injection system (RIS) [SIS] in residual heat removal mode.

a') The intermediate cooling chain of the EVU [CHRS] train 1 also provides cooling of the third fuel pool cooling system (PTR) [FPCS] train when the latter is operating (Plant Condition Category or Risk Reduction Category-A).

Containment of radioactive substances

b) The EVU [CHRS] system transfers decay heat from the containment atmosphere to the IRWST during a severe accident (Risk Reduction Category-B) in order to maintain the containment pressure at values that are compatible with maintaining its integrity.

c) The EVU [CHRS] system floods the corium spreading compartment with water from the IRWST during a severe accident (Risk Reduction Category-B).

d) The EVU [CHRS] system ensures cooling of the core catcher during a severe accident (Risk Reduction Category-B).

e) The EVU [CHRS] system participates in containment isolation during accidents that do not require its operation.

f) The part of the EVU [CHRS] (main chain) located outside the containment constitutes a containment barrier (to maintain containment integrity and to contain radioactive substances) during accidents that require its operation.

g) The EVU [CHRS] contributes towards minimising the production of volatile iodine within the containment atmosphere from the liquid phase by means of sodium hydroxide injection in the IRWST during a Loss Of Coolant Accident (Plant Condition Category 4) via the RIS [SIS] or during a severe accident (Risk Reduction Category-B) via the EVU [CHRS]. The target is to obtain an alkaline pH of the IRWST [Ref-1].

7.0.2. Functional criteria

The EVU [CHRS] system meets the following functional criteria:

Control of reactivity

Not applicable.

Decay heat removal

a) The EVU [CHRS] system cooling capacity must be sufficient to ensure, in all system operating situations, decay heat transfer from the IRWST to the ultimate heat sink (SRU [UCWS]). The risk of EVU [CHRS] filter clogging must be considered.

a') The intermediate cooling chain of the EVU [CHRS] train 1 must be in service when the third PTR [FPCS] train is started.

Containment of radioactive substances

b) The EVU [CHRS] system suction capacity must be adequate to provide sufficient heat transfer from the containment atmosphere to the IRWST, in order to ensure that the containment pressure limits are met (see pressure curve).

c) The EVU [CHRS] system flooding capacity must be sufficient to ensure passive flooding function of the corium spreading compartment in order to cool the corium.

d) The EVU [CHRS] system cooling capacity must be sufficient to ensure the core catcher cooling function is met.

e) It must be possible to isolate the part of the EVU [CHRS] system located outside the containment from the containment in the event of an accident that does not require its operation.

f) Provisions must be made to prevent any leak in the main chain outside the containment.

g) The EVU [CHRS] system must ensure an alkaline pH of the IRWST water as soon as possible afterwards a loss of coolant accident or a severe accident (Risk Reduction Category-B).

Note 1: The EVU [CHRS] system must enable an RCV [CVCS] pump suction line to be connected to the IRWST, unless the accident requires the operation of the EVU [CHRS] system.

Given the containment characteristics (volume, design, thermal inertia of containment structures) and those of the IRWST, a grace period of at least 12 hours is available after the start of a severe accident; during this period, no system is to be required for maintaining the containment pressure below the design pressure; see note 2.

Note 2: This 12-hour time period is not an EVU [CHRS] system start-up criterion; it is a design value for designing system capacities.

Short-term functional criterion:

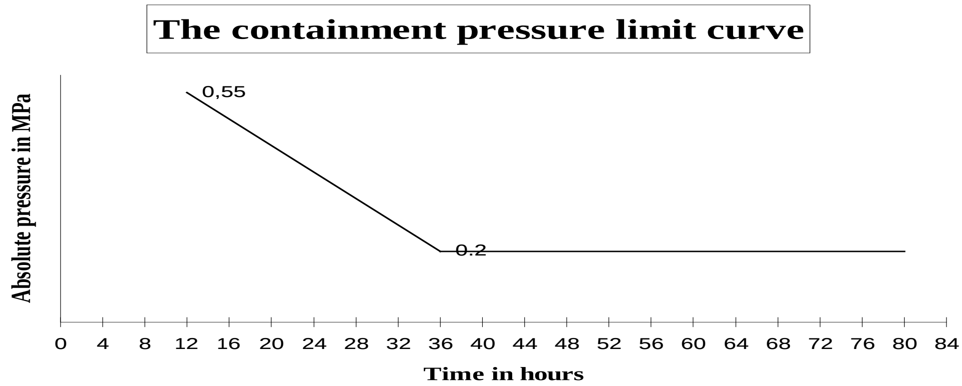
- Starting two EVU [CHRS] trains after a 12-hour grace period must be sufficient to reduce the containment pressure below 2 bar within 24 hours,

- Starting one EVU [CHRS] train after a 12-hour grace period must be sufficient to maintain the containment pressure below the design pressure (5.5 bar).

Long-term functional criterion:

Over the long term, one EVU [CHRS] train must enable the containment pressure to be maintained below 2 bar.

Pressure curve: If two trains are started up 12 hours after the severe accident.



7.0.3. Design-related requirements

7.0.3.1. Requirements from safety classification

Safety classification

The EVU [CHRS] is classified in accordance with the classification system presented in Sub-chapter 3.2.

Single failure criterion (active and passive)

For components that provide F1 functions (containment isolation), the single failure criterion is applied to ensure adequate redundancy.

Emergency power supplies

The EVU [CHRS] system has an emergency power supply so that its operation is ensured in the event of station blackout.

Qualification in operating conditions

The requirements relating to the EVU [CHRS] system qualification are presented in Sub-chapter 3.6.

Mechanical, electrical and instrumentation and control classifications

The EVU [CHRS] system mechanical, electrical and instrumentation and control classifications are defined in accordance with the classification system presented in Sub-chapter 3.2.

Seismic classification

The EVU [CHRS] system seismic classification is defined in accordance with the classification system presented in Sub-chapter 3.2.

Periodic tests

The EVU [CHRS] system is tested periodically to ensure its availability. The classified systems must be easily accessible to allow these periodic tests to be carried out.

7.0.3.2. Other regulatory requirements

Basic Safety Rules

Not applicable.

Technical guidelines

EPR Technical Guidelines are presented in Sub-chapter 3.1.

Specific requirements for the EVU [CHRS] system are:

- Section A.1.3: General strategy relating to severe accidents,
- Section B.1.4.2: Prevention of containment bypass,
- Section B.2.3.5: The containment heat removal function,
- Section E.2.3.1: Corium cooling capacity outside the vessel,
- Section E.2.3.2: Removal of containment heat without venting.

Specific EPR documents

Not applicable.

7.0.3.3. Hazards

The internal and external hazards considered in the EVU [CHRS] design are presented in Chapter 13.

7.1. SYSTEM FUNCTION

The EVU [CHRS] is not required during normal operation.

The main functions of the EVU [CHRS] system are to limit the pressure inside the containment and to ensure decay heat removal from the containment during severe accidents (Risk Reduction Category-B).

The EVU [CHRS] system also ensures cooling of the third PTR [FPCS] train and decay heat transfer from the IRWST to the ultimate cooling water using a dedicated cooling system in Risk Reduction Category-A pipe breaks with loss of the low head safety injection (LHSI). It also provides cooling transiently in certain Plant Condition Category-4 pipe breaks outside the containment with the RIS [SIS] in residual heat removal mode (see Sub-chapter 14.5, section 14).

The EVU [CHRS] system also has a sodium hydroxide injection circuit in the IRWST in order to limit iodine production and release during a loss of coolant accident (Plant Condition Category 4) via the RIS [SIS] or during a severe accident (Risk Reduction Category-B) via the EVU [CHRS].

7.2. DESIGN BASES

The EVU [CHRS] design is based on an assumed thermal power of 4500 MW_{th}.

The EVU [CHRS] is designed for fuel loading with conservative decay heat and IRWST temperature conditions irrespective of the system operating conditions (Risk Reduction Category-A, Risk Reduction Category-B and Plant Condition Category-4) and the SRU [UCWS].

<u>Redundancy</u>

Application of the single failure criterion is not required as the EVU [CHRS] is an F2 classified system. However, the EVU [CHRS] comprises two separate trains so that the long-term failure of a train does not prevent the system from fulfilling its function.

<u>Long-term repairability</u>

It is possible to carry out long-term maintenance operations on the main chain pumps and heat exchangers following a severe accident.

<u>Additional design requirements</u>

- Spray nozzles must be designed to prevent clogging,
- The main pumps must be able to operate with water containing small particles; they must remain operational over long periods and be leak tight,
- All sensitive parts (seals, flanges, etc.) must be able to withstand high irradiation (Sub-chapter 3.6).

7.3. EQUIPMENT DESCRIPTION AND CHARACTERISTICS

7.3.1. Description [Ref-1]

The EVU [CHRS] consists of two trains (see Section 6.2.7 - Figures 1 and 2), each including:

A main chain with:

- A suction line from the IRWST. According to the required EVU [CHRS] function, two configurations are possible:
  - o either the main chain pump suction is linked to a dedicated filter for the EVU [CHRS] belonging to the RIS [SIS],
  - o or the main chain pump takes its suction from the neighbouring RIS [SIS] filter via a cross connection EVU/RIS [CHRS/SIS]. This line up is used either, for the back-flushing of the RIS [SIS] filter dedicated to the EVU [CHRS] if one of the two EVU [CHRS] trains is unavailable, or for the IRWST cooling by EVU [CHRS] re-circulation. This latter is required to cover RRC-A and PCC-4 accidents.
- A pump and a heat exchanger. The heat exchanger, used to remove decay heat from the containment, is supplied by a dedicated intermediate cooling chain,
- Downstream of the pump and heat exchanger, there are three possible flow paths:
  - o to a dome spraying system consisting of a ring equipped with spray nozzles located in the dome of the containment, to rapidly reduce pressure and the temperature inside the containment. The condensate, which is generated by the sprayed coolant from the steam within the containment atmosphere, flows back to the IRWST.
  - o to a corium spreading compartment active flooding line (for long-term severe accident management),
  - o to a back-flushing line that is also used as a test line. This line is used to clean the RIS [SIS] filter dedicated to the EVU [CHRS] when too much debris has been accumulated on the filter. Each EVU [CHRS] main pump back-flushes both RIS [SIS] filters dedicated to the EVU [CHRS].
- A passive flooding system located in a compartment separate from the spreading compartment and the IRWST. This system includes a flooding valve held closed by a system of cables. During core meltdown, the corium melts the pre-stressed cable(s) and the valve opens under the pressure of the water.
- In normal plant operating conditions, the passive flooding line is filled up with IRWST water up to the flooding valve. So, in order to avoid the presence of water in the corium spreading area before the arrivals of corium, a leak-off recovery system and leak-off detection are implemented upstream of the passive flooding valve.
- A cooling system for the core catcher and the corium located below the spreading compartment layer of sacrificial concrete, ensuring that the water is poured into the spreading compartment. The cooling system for the core catcher structure and corium is connected to the IRWST via the flooding valves,

- A connection to the chemical and volume control system (RCV [CVCS]) on the suction line of the two trains, and to the discharge nozzle, or on the unclogging line, of train 2,

- A sodium hydroxide injection circuit. This circuit is mainly composed of sodium hydroxide stored in atmospheric pressure tanks, a venting line, a sodium hydroxide mixing device and three injection lines: two towards RIS [SIS] downstream of each RIS [SIS] LP pump and one towards EVU [CHRS] downstream of the EVU [CHRS] main pump (see Section 6.2.7 - Figure 3),

- A leak tight circuit of the seals main chain pump. This circuit is equipped with a pressurised water buffer tank, a SED (demineralised water distribution system) connection for the make-ups and heat exchanger for the cooling of the pump mechanical seals.

An intermediate cooling chain consisting of:

- A pump used to supply the EVU [CHRS] main chain heat exchanger and, for one train only, to supply the heat exchanger of the third PTR [FPCS] train,

- A heat exchanger supplied by the EVU [CHRS] dedicated cooling chain (SRU [UCWS]),

- An expansion tank maintained under pressure by means of a compressed air cushion. The make-ups to the tank are realised automatically by means of a SED (demineralised water distribution system) connection.

- A dedicated cooling system, the Ultimate Heat Sink (SRU [UCWS]),

- A trisodic phosphate injection line (SIR6, chemical reagents injection system). The goal of this chemical injection is to minimise the corrosion effect in the intermediate chain.

7.3.2. Main equipment data [Ref-1]

Main data for the EVU [CHRS] main chain pumps

- Type: centrifugal horizontal: pump with double mechanical seals
- Normal flow rate: approx. 387 m^3/h
- Normal flow head: approx. 150 mCW
- NPSH available at normal flow rate: approx. 2.5 mCW

Main data for the EVU [CHRS] main heat exchangers

Only the sizing situations are presented.

- The data below are given for one train.

Event	Trains operating	Functional requirements	Required EVU [CHRS] flow rate		
			Main chain	Intermediate chain	SRU [UCWS]
RRC-B	2 trains	P = 11.5 MW T_{IRWST} = 94°C	350 m^3/h	510 m^3/h	650 m^3/h
RRC-A and PCC- 4	2 trains	P = 18 MW T_{IRWST}=120°C	300 m^3/h	510 m^3/h	650 m^3/h

Main data for the EVU [CHRS] intermediate chain pumps

- Type: Centrifugal horizontal
- Normal flow rate: approx. 560 m^3/h
- Normal flow head: approx. 53 mCW
- NPSH available at normal flow rate: approx. 152 mCW

Main data for the EVU [CHRS] intermediate chain heat exchangers

Only the sizing situations are presented.

The below data are given for one train.

Event	Trains operating	Functional requirements	Required EVU [CHRS] flow rate		
			Main chain or PTR	Intermediate chain	SRU [UCWS]
RRC-B	2 trains	P = 11.5 MW	350 m^3/h	510 m^3/h	650 m^3/h
Cooling of the 3rd PTR train	Train 1 only	P = 20.23 MW T_{PTR}=95°C	550 m^3/h	520 m^3/h	650 m^3/h

Main data for the EVU [CHRS] expansion tanks

- Pressurisation: pressurised air blanket
- Volume: approx. 1.7 m^3
- Minimal pressure: approx. 18 bar abs

Main data for the EVU [CHRS] seal water buffer tanks

- Pressurisation: pressurised air blanket
- Volume: approx. 1 m^3
- Minimal pressure: approx. 13 bar abs

7.4. OPERATING CONDITIONS

7.4.1. Normal operation

The EVU [CHRS] is not required during normal operation. The system is used to mitigate the effects of severe accidents.

The EVU [CHRS] train will be filled with water during commissioning of the IRWST. As the pump level is lower than the water level of the IRWST, the water will flow to the pump by gravity.

When the unit is in normal operation, the EVU [CHRS] system may be stopped (in stand-by position) or actuated for short periods of time for testing purposes only. Its stand-by position is the following:

- The pumps are shut down and available, ready to start,
- The flooding valves are closed and the corresponding isolation valves are open,
- The back-flushing valves are open,
- The six other containment isolation valves (spray line, IRWST suction line and active flooding line) are closed,
- The manual isolation valves of the train are open,
- The intermediate cooling chains are pressurised and ready to start.

However, the third train of the PTR [FPCS] is started in the event of non-availability of a main PTR [FPCS] train during preventive maintenance of the PTR [FPCS] or its support systems. The ultimate heat sink (SRU [UCWS]) and the intermediate system for the EVU [CHRS] train 1 are in service when the third PTR [FPCS] train is started.

7.4.2. Operation during accidents

7.4.2.1. Severe accidents (RRC-B)

The system is designed to operate in the event of severe accidents with core meltdown (Risk Reduction Category-B).

The safety injection system is out of service.

Emergency plant cooldown diesel generator sets are available in the event of station blackout.

Other emergency plant auxiliaries (RRI [CCWS] and SEC [ESWS] systems, emergency power supply) are not necessary.

Passive flooding of the corium

When the corium reaches the spreading compartment, it melts the pre-stressed steel cables that maintain the counter weight of the flooding valves in position. The counter weight initiates the valve opening and water flows to the spreading compartment by force of gravity. The flooding valves themselves are located in a separate compartment from the spreading compartment and the IRWST. The water from the IRWST circulates through the discharge channel, which is now open, and into the cooling channels in the core catcher. When these channels are full, the water pours into the corium and the latter is flooded. The functional requirements associated to this function are described in section 6 of this sub-chapter.

Spraying

On contact with the corium, the water flowing from the IRWST boils and the corium is cooled. The steam produced causes an increase in containment pressure and temperature.

At the latest, within 12 hours of the start of a severe accident, if necessary, the operator starts one or both trains of EVU [CHRS] to maintain containment pressure and temperature within design limits.

Start-up is manual and the decision to start the EVU [CHRS] is based primarily on a containment pressure criterion.

Water is drawn from the IRWST pool and is cooled in the heat exchangers before being sprayed into the containment from the reactor building dome.

Once the pressure has dropped below its long-term nominal limit (2 bar) and after at least 15 days if MOX fuel is used (10 days with UO_2 fuel), a single train is sufficient to maintain the pressure below 2 bar.

Active flooding of the corium

When spray is no longer needed to maintain a low containment pressure, corium cooling may also be provided by the active flooding line: the water heated by the corium returns to the IRWST by overflowing from the corium spreading compartment.

Back-flushing

During a severe accident, the back-flushing line can be used to flush the RIS [SIS] filters dedicated to the EVU [CHRS].

7.4.2.2. RRC-A accident

The EVU [CHRS] is also required to operate in Risk Reduction Category-A pipe breaks with loss of the low head safety injection (LHSI) and in case of total loss of RRI/SEC [CCWS/ESWS] chain in state D. In these events, the EVU [CHRS] is used to ensure decay heat removal via the IRWST in EVU [CHRS] re-circulation mode.

These are the two only Risk Reduction Category-A accidents requiring the EVU [CHRS] to operate.

7.4.2.3. PCC-4 accident

The EVU [CHRS] system can be also used to operate during certain Plant Condition Category-4 pipe breaks with the RIS [SIS] in residual heat removal mode.

In this event, the EVU [CHRS] is used in EVU [CHRS] re-circulation mode to remove decay heat via the IRWST.

7.5. SAFETY ANALYSIS

7.5.1. Compliance with regulations

Basic Safety Rules

Not applicable.

Technical guidelines

To meet the requirements of Technical Guidelines listed in section E.2.3.2*: Removal of the containment heat without venting,* the EVU [CHRS] design features are described below:

Potential leaks from the system

During a severe accident, contaminated fluid circulates in the EVU [CHRS] main chains. The following precautions are taken to limit the doses outside containment:

1) The EVU [CHRS] main system components outside containment are classified in accordance with the classification system presented in Sub-chapter 3.2.

2) The main system equipment located outside containment is leak tight:

    - Bellows seal valves or leak-off connection valves,

    - Tubular heat exchangers,

    - Pump with double mechanical seals with a seal injection system to ensure leak tightness.

3) Sensors are implemented in the EVU [CHRS] main chain rooms to rapidly detect leaks and reduce the volume of leaks.

4) Appropriate instrumentation is installed (activity, water level).

5) The EVU [CHRS] main chain components located outside containment (valves, penetrations, pumps, heat exchangers, etc.) are located in dedicated compartments with specific protection (air lock, thick walls, etc.)

6) Filtered ventilation before discharge into the stack enables dedicated rooms to be isolated:
At the same time as starting the EVU [CHRS], the air supply to the EVU [CHRS] compartments is isolated by leak tight dampers. If the EVU [CHRS] is started, air from the EVU [CHRS] rooms is extracted and filtered via the DWL [CSBVS] iodine filtering system. This maintains a constant negative pressure and prevents contamination from spreading. The exhausted air passes through high efficiency filters and iodine traps before being released to the stack.

7) Each EVU [CHRS] train may be isolated if a leak is detected in its rooms by water level sensors.

8) The EVU [CHRS] dedicated intermediate cooling chain pressure is greater than the operating pressure of the EVU [CHRS] main chain. This ensures the absence of leaks from the main chain into the cooling chain in the event of a tube rupture in the main heat exchanger.

   This pressurisation is maintained during normal operation to ensure the availability of the EVU [CHRS] system if demanded. Also, a dedicated system enables the water inventory to be maintained (small leaks) in the cooling system without having to depressurise the system.

9) Detection means are provided to enable the dedicated intermediate cooling chain to be isolated in the event of a leak in the main heat exchanger (frequent water makeup, low level in the expansion tank).

10) A single isolation valve is provided in each of the lines between the IRWST and a main EVU [CHRS] pump. Each isolation valve (outside containment) is designed with a special leak tight device. The section of the pipe between the IRWST and the valve is contained in a leak tight sheath (suction guard pipe with double seals). This arrangement provides a double barrier.

Common causes of failure of the EVU [CHRS] with other systems

The common causes of potential failure for the EVU [CHRS] and RIS [SIS] functions may include:

- Loss of the RIS [SIS] IRWST (following a loss of water or blockage of the water intake channels),

- Loss of common auxiliary functions (cooling water, power supply).

The methods implemented to eliminate or limit the consequences of these common causes are described below:

<u>Loss of water from the IRWST</u>

The IRWST is lined to protect the concrete against the permanent presence of water during plant life. The liner is not required to ensure the leak tightness and, even if it is damaged, there are no consequences for the correct operation of the IRWST. To prevent the possible loss of water from a leak in a pipe connected to the IRWST, leak detection (water level and/or pressure measurement in the RIS [SIS] pump rooms) means that any train of the safety injection system in which water circulates outside the containment is isolated (with F1A classified isolation).

Clogging of the water intake channels in the IRWST

Provisions are made to prevent the sumps from blocking by:

- provision of suction protection via filters,
- use of suitable thermal insulation (type of components),
- use of suitable devices for retaining insulation and other materials.

Also, additional features are provided to prevent clogging, in particular:

- Separate suction lines are used for the RIS [SIS] and the EVU [CHRS],
- There is geographical separation of the sumps used by the RIS [SIS] and the EVU [CHRS],
- An EVU [CHRS] filter back-flushing system is provided. This system enables the filters to be unclogged if blocked. A dedicated unclogging line exists for each EVU [CHRS] train (see Sub-chapter 6.3).

Loss of common auxiliary functions:

Dedicated cooling systems and electrical power from emergency plant diesel generators has been provided to improve system efficiency and to meet the probabilistic safety analysis objectives.

Long-term corium stabilisation in the spreading compartment

After the flooding valves have passively opened, the heat extracted by passive flooding in the cooling channels is sufficient to ensure corium stabilisation, even over the long-term. In the long term, the EVU [CHRS] is also able to provide active cooling by supplying the core catcher with cold water.

Prevention of spray nozzle blockage

Screen filters with appropriate mesh diameters are located upstream of the IRWST suction lines.

Specific EPR documents

Not applicable.

7.5.2. Compliance with the functional criteria

Safety function a) is achieved by cooling the water from the IRWST via the EVU [CHRS] main heat exchanger, which is cooled by using a dedicated intermediate cooling chain. The intermediate cooling chain is cooled in turn by the ultimate heat sink SRU [UCWS]. A back-flushing system reduces the risk of the filters blocking.

Safety function a’) is achieved by using the PTR/EVU [FPCS/CHRS] heat exchanger which is cooled by the EVU [CHRS] intermediate cooling chain (train 1), which in turn is cooled by the ultimate heat sink SRU [UCWS]

Safety function b) is achieved by drawing cold water from the IRWST and spraying it through spray rings at the top of the containment. The spray water cools and condenses the containment atmospheric steam and the condensate falls by gravity into the IRWST. The condensed water is recirculated through heat exchangers and returned to the spray nozzles. The decay heat is transferred to the containment atmosphere by boiling in the corium recovery area. The evaporated water is replaced by water from the IRWST.

Safety function c) is achieved by opening the dedicated flooding valves connecting the IRWST and the corium spreading compartment through the core catcher cooling system. The flooding valves are opened passively when the corium melts the pre-stressed cables in the corium spreading compartment, and the water flows in under gravity. Over the long term, when spraying is no longer necessary, the EVU [CHRS] can continue to provide active corium cooling (sub-cooled water heat removal).

Safety function d) is achieved by the core catcher cooling system located below the spreading compartment layer of sacrificial concrete. The core catcher cooling system is supplied passively by water from the IRWST when the flooding valves are opened. It may be supplied actively during long-term accident management.

Safety function e) is achieved by the EVU [CHRS] containment isolation valves and the double seal valves, located on the main chain suction line, during accidents which do not require the EVU [CHRS] to operate.

Safety function f) is achieved by the robust mechanical design of the EVU [CHRS] main chain.

Safety function g) is achieved by the sodium hydroxide injection circuit.

During an accident requiring its operation, the containment function is provided by those parts of the EVU [CHRS] located outside the containment.

7.5.3. Compliance with design requirements

7.5.3.1. Safety classification

The EVU [CHRS] system is designed in accordance with the safety classification principles presented in Sub-chapter 3.2.

7.5.3.2. Single failure criterion

Two containment isolation valves are installed on each line crossing the containment boundary to meet the single failure criterion (for isolating the containment during an accident that does not require the EVU [CHRS] to be operated).

Although application of the single failure criterion is not required for the EVU [CHRS], the EVU [CHRS] system consists of two separate trains, so that long-term failure of one train does not prevent the system from operating.

7.5.3.3. Qualification

The part of the EVU [CHRS] that performs F1 or F2 classified functions is designed in accordance with the requirements presented in Sub-chapter 3.6.

7.5.3.4. Instrumentation and control

The EVU [CHRS] system mechanical, electrical and instrumentation and control classifications are defined in accordance with the classification principles presented in Sub-chapter 3.2.

7.5.3.5. Emergency power supplies

Although not required for non-F1 function elements, the EVU [CHRS] system can be powered by emergency power supplies (emergency plant cooldown diesel generator sets).

7.5.3.6. Hazards

Summary tables of the hazards taken into account

Internal hazards	Protection required in principle	General protection	Specific protection introduced in the system design
Pipe ruptures	For internal hazards that may result from an RRC-A accident (small break LOCA with LHSI loss) and from an RRC-B accident.	-	-
Tank, pump and valve ruptures		-	-
Internal projectiles		-	-
Dropped load		-	-
Internal explosion		-	-
Fire		-	-
Internal flooding		-	-

External hazards	Protection required in principle	General protection	Specific protection introduced in the system design
Earthquake	Yes	The protection is provided through the construction works of safeguard buildings (SBs) and reactor building (RB)	Seismic design for the entire system
Aircraft crash	No	This hazard does not lead to a demand for EVU [CHRS] operation	-
External explosion	No	The protection is provided through the construction works of safeguard buildings (SBs) and reactor building (RB)	-
External flooding	Yes	The protection is provided through the construction works of safeguard buildings (SBs) and reactor building (RB)	-
Snow and wind	Yes	The protection is provided through the construction works of safeguard buildings (SBs) and reactor building (RB)	-
Extreme cold	Yes	The protection is provided through the construction works of safeguard buildings (SBs) and reactor building (RB)	-

7.5.3.7. Other requirements

This system is taken into account in the demonstration of the practical elimination of the containment bypass risk (see Sub-chapter 16.3).

7.6. TESTS, INSPECTION AND MAINTENANCE

Tests

The EVU [CHRS] is designed to allow periodic testing to ensure:

- the structural integrity and leak tightness of equipment,
- the availability of the systems and active components,
- the availability of the entire system in conditions that are as close as possible to accident conditions; all the operational sequences to activate the system are carried out, including switching the normal power supplies to emergency power supplies and operation of the dedicated cooling system.

Preventive maintenance

Preventive maintenance is possible during plant operation.

Long-term maintenance

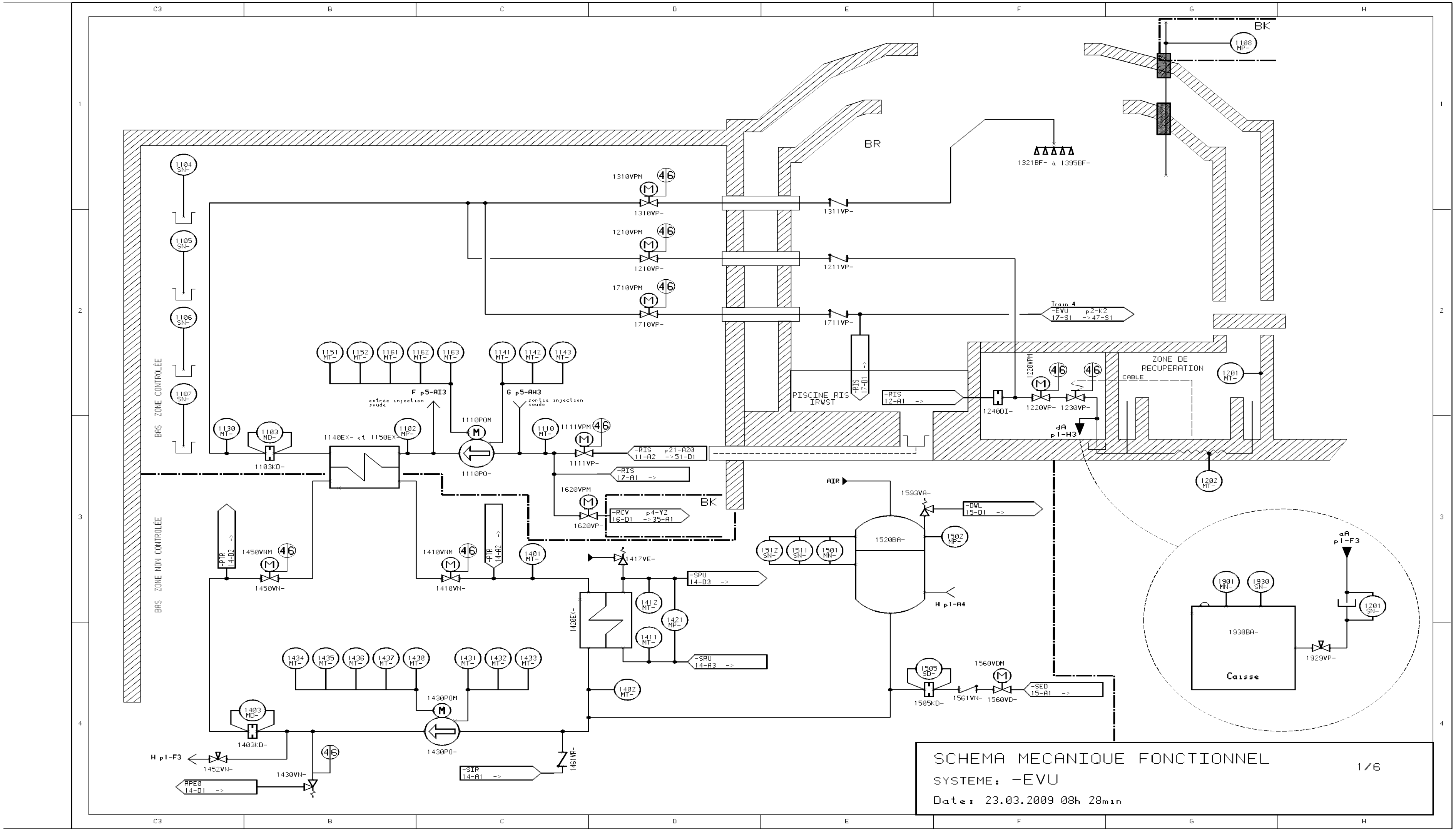
Long-term maintenance is possible following a severe accident. If maintenance takes place on the main chain pumps and heat exchangers, then additional equipment can be connected to the system. This allows water to be directly injected into the reactor building and also into the part of the system affected by the maintenance, for decontamination purposes.

7.7. FLOW DIAGRAMS

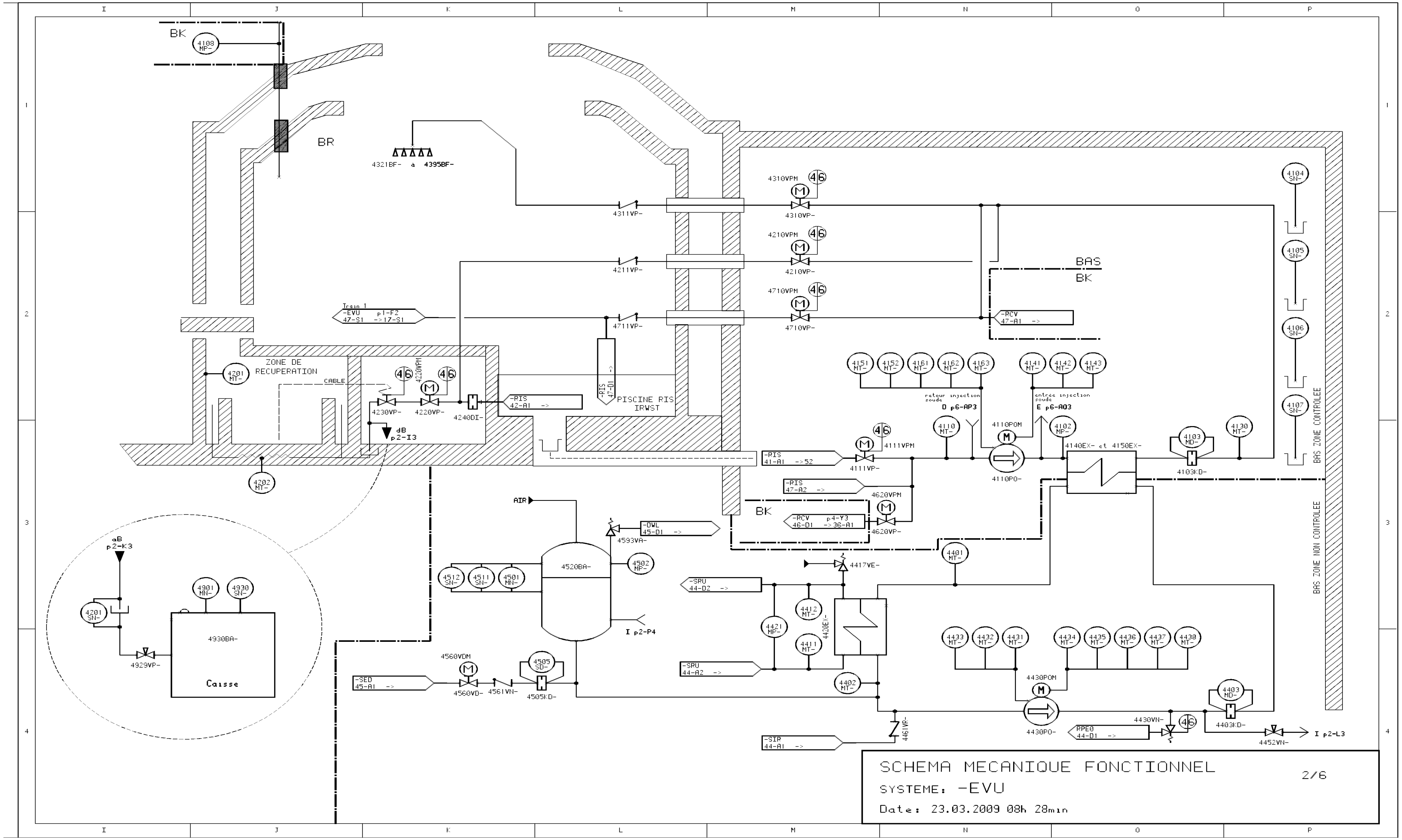
Functional flow diagram of the EVU [CHRS]: see Section 6.2.7 - Figures 1 and 2.

Conceptual sketch of the sodium hydroxide injection circuit in the EVU [CHRS]: see Section 6.2.7 - Figure 3.

SECTION 6.2.7 - FIGURE 1 : EVU [CHRS] Functional Flow Diagram – Train 1 [Ref-2]

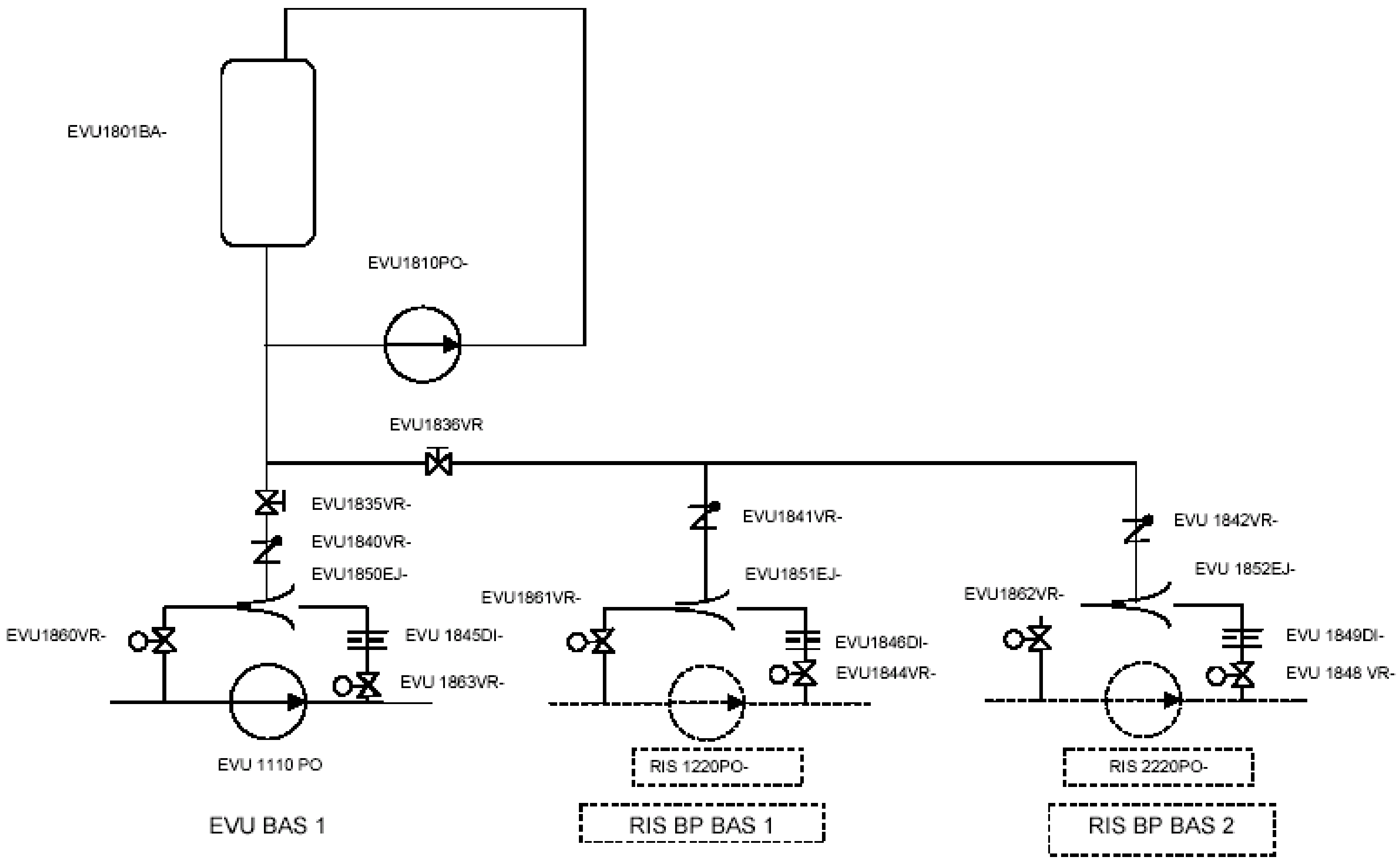


SECTION 6.2.7 - FIGURE 2 : EVU [CHRS] Functional Flow Diagram – Train 2 [Ref-2]



SECTION 6.2.7 - FIGURE 3

Conceptual Sketch of the Sodium Hydroxide Injection Circuit in the EVU [CHRS] – Train 1 [Ref-1]



SUB-CHAPTER 6.2 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1] [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

1. CONTAINMENT FUNCTIONAL REQUIREMENTS AND FUNCTIONAL DESIGN

1.3. CONTAINMENT FUNCTION

1.3.1. General assumptions about the containment function

[Ref-1] F. Lefevre. Mass and Energy Releases in the containment in case of Loss of Coolant Accident (LOCA). EPRR DC 1673 Revision C. Framatome ANP. December 2001. (E)

[Ref-2] Dr. Eyink. EPR Pressure and Temperature loads relevant for the steel liner. NGPS4/2003/en/0045 Revision D. July 2003. (E).

[Ref-3] Basic Design Report. Chapter 6.2.1.2.1.1, Leaktightness requirements for PCCs and RRC-A. Report jointly prepared by EDF, German utilities, Framatome, Siemens AG, Nuclear power international. February 1999. (E).

1.5. DEFINITION AND ANALYSIS OF THE CONTAINMENT LOAD COMBINATIONS

1.5.2. Definition of Mass and Energy Release in Incidents and Accidents as Design Basis for Containment

1.5.2.2. 2A LOCA

1.5.2.2.3. Initial and boundary conditions

[Ref-1] Colignon. Basic Design Report. Chapter 6.2.1.3.2.3, Initial and Boundary Conditions. Report jointly prepared by EDF, German utilities, Framatome, Siemens AG, Nuclear power international. February 1999. (E)

1.5.2.3 Steam Line Break (PCC)

1.5.2.3.3. Initial and boundary conditions

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