Status report 99 - ATMEA1 (ATMEA1)

Overview

Full name	ATMEA1
Acronym	ATMEA1
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	3150.00 MWth
Electrical capacity	1150.00 MWe
Design status	Basic Design
Designers	ATMEA
Last update	03-02-2011

Description

Introduction

1.1. Generalities

ATMEA^{TM} is the joint venture created in July 2007 between AREVA NP (AREVA) and Mitsubishi Heavy Industries, Ltd (MHI). The purpose of the Joint Venture is to design, market and sell worldwide, **a 1100 MWe class evolutionary PWR that** encompasses innovative and proven nuclear technologies from AREVA and MHI, including top-level safety systems, high-thermal efficiency, and a flexible 12- to 24-month operational cycle, leading to less waste and minimized impact to the environment. The primary system design, loop configuration, and main components are similar to those of currently operating PWRs, thus forming a proven foundation for the design.

The 1100 MWe class reactor developed by ATMEA^{TM} is ATMEA1^{TM}.

ATMEA1^{TM} is an 1100 MWe class evolutionary Pressurized Water Reactor (PWR). It is a three-loop plant with a rated thermal power of 3,150 MWth designed for a 60-year operating life.

ATMEA1^{TM} has a basic set of common design features adaptable to regulatory and commercial requirements of any country in which it will be proposed. ATMEA1^{TM} has a basic set of safety features, such as three redundant trains of emergency core cooling systems, shielded containment and shielded buildings against airplane crashes and a core-melt retention system for the mitigation of severe accidents. It is designed to meet major current sets of regulatory and commercial requirements.

1.2. Proven Technology: ATMEA1^{TM} designed from currently operating PWRs

ATMEA1^{TM} is composed of operated, licensed, or certified systems or components, integrating the most modern, proven technologies developed by AREVA and MHI

The following features of ATMEA1^{TM} provide examples of this integration of the most modern, proven technologies:

- Steam generators with axial economizer and TT690 tubes
- Advanced accumulators
- Reactor internals with Heavy Neutron Reflector
- Digital Instrumentation and Control (I&C)
- Top Mounted Instrumentation

1.3. Design features

ATMEA1^{TM} has three-loop configuration constituting the Reactor Coolant System (RCS).

The RCS is equipped with:

- A Reactor Pressure Vessel (RPV) that contains the Fuel Assemblies,
- A Pressurizer (PZR) including control systems to maintain system pressure,
- One Reactor Coolant Pump (RCP) per loop,
- One Steam Generator (SG) per loop,
- Associated piping and related control and protection systems.

The Reactor Building (RB) containment structure is of a pre-stressed concrete type with a steel liner inside. An annular space, surrounding the lower part of the RB, collects possible leakages from all penetrations. The RB is surrounded by the Safeguard Building (SB) and the Fuel Building (FB)

Redundant 100% capacity safety systems are strictly separated into three Divisions. This divisional segregation is provided for electrical and mechanical safety systems. One separated backup train for Essential Service Water System (ESWS), Component Cooling Water System (CCWS) and Emergency Power Supply system (EPS) constitutes the fourth train provided for on-power maintenance of the systems of any of the other train. This fourth train is also a diverse mean for beyond Design Basis Events (DBE).

Water storage for safety injection is provided by the In-containment Refueling Water Storage Pit (IRWSP) located at the bottom of the RB. Also, inside the containment and below the RPV is a dedicated spreading area called "Core Catcher" for molten core material following a postulated worst-case severe accident.

The fuel pool is located outside the RB in a dedicated building (FB) to simplify access for fuel handling during plant operation and handling of fuel casks. Fuel pool cooling is assured by three redundant cooling trains.

All the safety functions are ensured against aircraft hazard and explosions by means of hardened protection shells or geographical segregation.

Description of the nuclear systems

2.1. Main characteristics of the primary circuit

The primary circuit of ATMEA1^{TM} is constituted by the main components and systems as described in § 1.3

2.2. Reactor core and fuel design

2.2.1. Core design

The core consists of 157 Fuel Assemblies (FAs) as used in the conventional 3-loop cores designed by AREVA and/or MHI. The radial heavy neutron reflector is designed

- to improve neutron utilization, thus reducing the fuel consumption,
- to reduce the RPV irradiation
- to reduce the inspection work by removing all bolts from high neutron fluence area

The Nuclear Steam Supply System (NSSS) nominal thermal power is 3150 MWth.

Fuel cycle lengths can be set to be from 12 to 24 months. Fuel management variations in ATMEA1^{TM} can go from a full uranium core to a mix with MOX fuel up to 1/3 of the core for the standard design, and up to 100% without any major design modification.

2.2.2. Fuel Assemblies

Each FA consists of fuel rods arranged in a 17x17 square array, together with 24 control rod guide thimbles.

The fuel rods have slightly enriched uranium dioxide pellets or MOX pellets. Some of the fuel rods may contain uranium dioxide pellets blended with gadolinia.

2.2.3. Reactor Cluster Control Assemblies (RCCAs) and Reactivity Control

The RCCAs of ATMEA1^{TM} control the relatively fast reactivity changes mainly due to reactor thermal power changes and coolant temperature variations. At any reactor operation state, the RCCAs have the ability to rapidly shut the reactor down even with full MOX cores.

The control rod has a neutron absorber encased in a stainless steel tube. In order to reduce boron concentration and to improve the primary water chemistry control, ^{10}B enriched boric acid is used as a soluble absorber in the coolant.

2.3. Fuel handling systems

The fueling and refueling system handles fresh and spent fuel.

New fuel brought into the NPP is stored in the fuel storage rack in the fuel handling building. After reactor shutdown, spent fuel is transferred to the spent fuel pit.

2.4. Primary circuit component description

2.4.1. Reactor Pressure Vessel

The RPV is subject to low neutron fluence thanks to a heavy neutron reflector (see 2.4.2), which enabled to design it for 60 years of operation.

The RPV consists of three inlet nozzles and three outlet nozzles, which are located between the RPV flange and the top of the core, so as to allow a volume of coolant within the RPV to be maintained in the event of leakage in the reactor coolant loop.

The RPV closure head dome and its flange are integral and made from a one-piece forging.

The RPV closure head is equipped with nozzles to which the pressure housings for the CRDM are connected. The top closure head is also equipped with in-core instrumentation nozzles.

The main parts of the RPV are made of low alloy steel with stainless steel cladding on all internal surfaces exposed to the reactor coolant. All other parts exposed to the coolant are made of either stainless steel or nickel base alloys.

2.4.2. Reactor internals

The RPV internals consist of upper internals and lower internals. Upper and lower internals are suspended from the RPV flange. A hold-down annular spring, compressed between the upper and lower internal flanges, ensures the internals' vertical stability and prevents their lifting when subjected to hydraulic forces.

The heavy neutron reflector, placed inside the core barrel, is the designed to ensure low neutron fluence to the RPV and to minimize inspection work and the risk of SCC by removing all bolts from high neutron fluence areas. It rests on the lower support plate and extends almost all the way up the core cavity

It consists of austenitic stainless steel forged slabs, without bolts or welds (welded or bolted connections close to the core are thus avoided). These slabs are positioned one above the other, and axially restrained by keys and tie rods.

2.4.3. Steam generators

The SGs are vertical shell, natural circulation, U-tube heat exchangers with integral moisture separating equipment. They are fitted with an axial economizer to provide increased steam pressure. The axial economizer directs all the feedwater (FW) to the cold leg side of the tube bundle and the re-circulated water to the hot leg thanks to the double wrapper in the cold leg of the downcomer.

This design enhances the heat exchange efficiency between the primary side and the secondary side and increases the outlet steam pressure compared to a boiler type SG with the same tube surface.

The tube material is alloy 690 TT which is widely used in SGs throughout the world, and is highly resistant to corrosion.

Figure 1 illustrates the principle of the axial economizer.

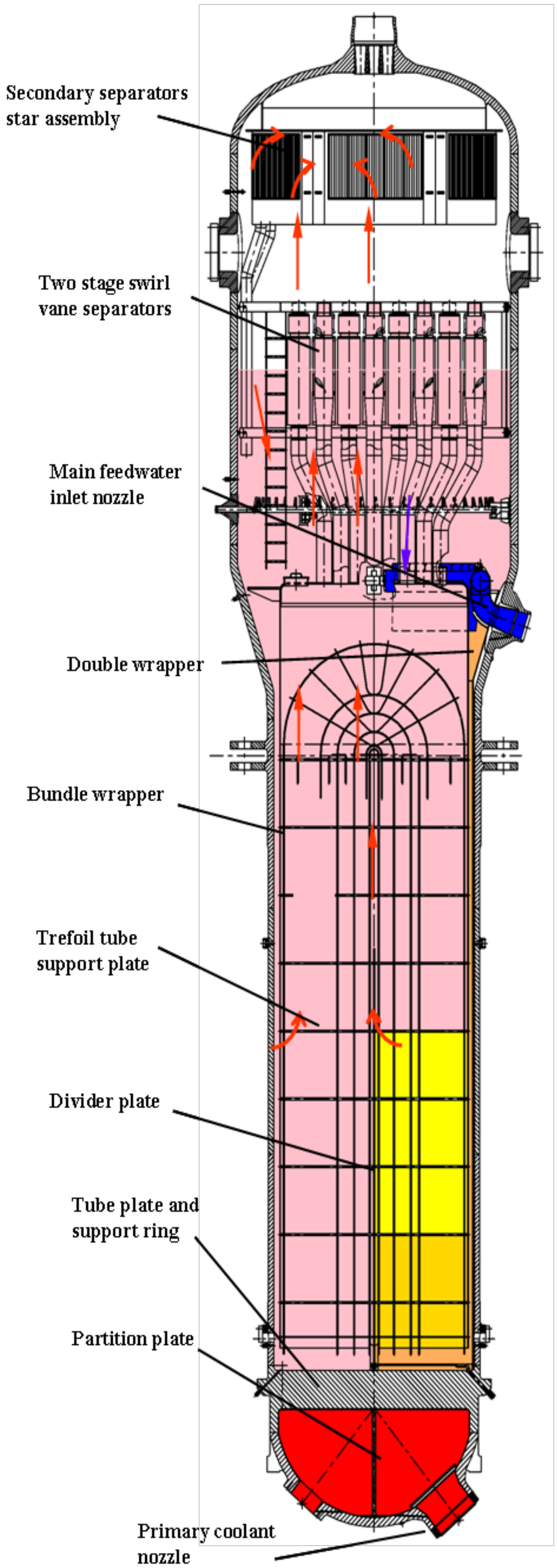
Secondary separators star assembly
Two stage swirl vane separators
Main feedwater inlet nozzle
Double wrapper
Bundle wrapper
Trefoil tube support plate
Divider plate
Tube plate and support ring
Partition plate
Primary coolant nozzle

Fig. 1. Steam Generator

2.4.4. Pressurizer

The PZR is designed to accommodate positive and negative volume surges caused by load transients. The surge line, which is attached to the bottom of the PZR, connects to the hot leg of a reactor coolant loop.

PZR safety valves provide overpressure protection for the reactor coolant system. The spray valves limit reactor coolant system pressure rises following less severe transients to prevent undesirable opening of the PZR safety valves.

2.4.5. Reactor coolant pumps

The RCPs are vertical, single-stage, shaft seal units, driven by air-cooled, three-phase induction motors. The complete unit is a vertical assembly consisting of (from top to bottom) a motor, a seal assembly, and a hydraulic unit.

The shaft seal system is made up of a series of three seals and a standstill seal.

2.4.6. Main coolant lines

The reactor coolant pipework consists of all the pipes connecting the RPV, the SGs, the RCPs, and the PZR, together with the various branches of the main piping up to the appropriate isolating valve. It also includes instrumentation connections to the reactor coolant system that enable flow, temperature, pressure, chemical composition and radiation to be monitored.

The reactor coolant pipes and fittings are made of austenitic stainless steel. All smaller diameters piping that is part of the reactor coolant system, such as the PZR surge line, spray line, and connecting lines to other systems, are also made of austenitic stainless steel.

The main coolant line is designed according to the Leak Before Break concept (LBB).

2.5. Reactor auxiliary systems

2.5.1. Chemical and volume control system (CVCS)

The CVCS is the interface between the high pressure RCS and the low pressure coolant treatment and supply systems. The CVCS is an operational system and has the following functions:

- Control the RCS water inventory via PZR level control;
- Provide make-up to the RCS in the event of small leakages;
- Adjust the boron concentration in the reactor coolant during start-up and shutdown and normal operation ;

The system operates continuously. The main components of the CVCS are the high pressure charging pumps, the volume control tank (VCT), the regenerative heat exchanger, two high pressure coolers in parallel, two high pressure reducing stations, also in parallel, and a low pressure reducing station.

The two HP charging pumps take suction from the volume control tank. The volume control tank acts as a surge tank to permit smooth control of variations in charging and letdown flow rates. The HP charging pumps can also take suction from the IRWSP in the event of a low level in the VCT, in the event of a failure of the reactor boron and water make-up system, or if a boron dilution is detected.

2.5.2. Component cooling water system

The CCWS transfers heat from safety-related systems and operational cooling loads to the heat sink, via the ESWS, under all operating conditions.

The CCWS supplies cooling water to the safety and operational systems such as:

- CSS/Residual Heat Removal System (RHRS) heat exchangers and pump motors;
- Spent Fuel Pit (SFP) heat exchangers;
- RCP thermal barriers;
- HP coolers
- and so on.

The CCWS consist of three trains (Trains 1, 2 and 3) plus one additional train (Train X) for On-Power Maintenance (OPM) and also for diversity. The heat exchangers of Trains 1, 2 and 3, cooled by the ESWS from Ultimate Heat Sink (UHS) 1 and the heat exchanger of Train X, cooled by the ESW from UHS 2, are of a different type. UHS 1 and UHS 2 are also of a different type.

2.5.3. Essential service water system

The ESWS consists of three separate safety-related trains that provide cooling of the CCWS heat exchangers with water from the UHS during all normal plant operating conditions, transients, and accidents. The ESWS also includes a diverse train, Train X, for OPM.

ESWS transports heat to the UHS after removing heat from the following consumers under all operating conditions including normal operation, residual heat removal and accidents:

- CCWS heat exchangers,
- EPS generators,
- Others.

Trains 1, 2 and 3 of the ESWS take the essential service water from UHS 1, supply each component being cooled, and discharge the essential service water back to UHS 1. Train X of the ESWS takes the essential service water from UHS 2 to supply each component being cooled, and discharges the essential service water back to UHS 2.

Each of Trains 1, 2 and 3 consists of one pump, two pump discharge strainers, one plate type heat exchanger with inlet strainer, and the associated piping, valves and instrumentation. Train X consists of one pump, two pump discharge strainers, and one shell & tube type heat exchanger including cleaning unit, the associated piping, valves and instrumentation.

2.6. Operating modes

ATMEA1^{TM} control modes and control channels allow for high flexibility in normal operation.

- For frequency control, variations of 15% in power (85%-100%) throughout cycle life time,
- Load-follow requirements: Daily fluctuations in the energy demand characterized by a plateau at a minimum power level of 25%. ATMEA1^{TM} is capable of even greater flexibility with instantaneous return to full power without notice (when selected) at a rate of 5% per minute,
- ATMEA1^{TM} is also capable of fast power reductions shifting to house load operation when necessary ,

2.7. Alternative fuel options

The ATMEA1^{TM} core design allows various fuel management schemes to be accommodated, from uranium to full MOX cores.

Description of safety concept

3.1. Safety concept, design philosophy and licensing approach

3.1.1. Design philosophy

ATMEA1^{TM} design philosophy is based on the following objectives relative to the current Generation III+ PWRs:

- Increased redundancy and segregation,
- Reduced core damage frequency (CDF),
- Reduced large release frequency (LRF),
- Mitigation of severe accidents,
- Protection of critical systems from external events (e.g. airplane crash),
- Improved human-system interface,
- Extended response times for operator actions.

3.1.2. Licensing status

ATMEA1^{TM} is based on technologies developed by AREVA and MHI, which are licensed or under licensing.

ATMEA1^{TM}'s conceptual design has been reviewed by IAEA and found compliant with the main requirements. The French Safety Authority, ASN, is reviewing ATMEA1^{TM}'s safety options according to the French safety legislation and French technical requirements.

3.2. Provision for simplicity and robustness of the design

Simplifications have been performed in all areas of design, construction, operation and maintenance with a view to achieving cost-effective design and to improved safety.

The advanced accumulator provides a good example of this approach. The advanced accumulator is the passive system which provides for reactor vessel refill at a high injection flow rate and for core re-flooding at a lower flow rate. As a result, low-head injection pumps are no longer necessary.

3.3. Safety systems

For ATMEA1^{TM,} active and passive systems have been optimized for reliability and economic purposes.

ATMEA1^{TM} integrates all top-level safety systems with three independent active safety trains. These trains are totally separated from operational systems and are protected against external hazards.

One of the examples of the passive safety systems is the advanced accumulator as described in 3.2.

3.4. Defense-in-depth description

The safety design of ATMEA1^{TM} is based primarily on deterministic analyses complemented by probabilistic analyses. The deterministic approach is based on the “defense-in-depth” concept which comprises five levels:

1. A combination of conservative design, quality assurance, and surveillance activities to prevent departures from normal operation,

2. Detection of deviations from normal operation and protection devices and control systems to cope with them (This level of protection is provided to ensure the integrity of the fuel cladding and of the reactor coolant pressure boundary in order to prevent accidents.),

3. Engineered safety features and protective systems that are provided to mitigate accidents and consequently to prevent their evolution into severe accidents,

4. Measures to preserve the integrity of the containment and enable control of severe accidents,

5. Off-site emergency response is not per se part of the safety design although safety objectives set forth for ATMEA1^{TM} strongly limit the need for it.

For all safety analysis, the long term safe state is retained as safe shutdown state.

List of design basis events is based primarily on the USNRC regulation completed by events from the designers’ experience.

Low probability events with multiple failures and coincident occurrences up to the total loss of safety-grade systems are considered in addition to the deterministic design basis (i.e. beyond design basis). Representative scenarios are defined for preventing both core melt and large releases. A probabilistic approach is used to define these events and both a deterministic and probabilistic approaches are used to assess the specific measures available for their management.

Even though the probability of severe accidents has been highly reduced by improving defense-in-depth, levels 1 to 3 innovative features practically eliminate energetic scenarios that could lead to containment failure. Design provisions for a further reduction of the residual risk, core melt mitigation, and the prevention of large releases are implemented.

External events such as large commercial or military aircrafts hazards, explosion pressure wave, seismic events, missiles, tornado and fire, have been considered in the design of the shielded safeguard buildings, containment and fuel building.

3.5. Safety goals (core damage frequency, large early release frequency and operator grace period)

3.5.1. Probabilistic targets and safety assessment

The aim of a probabilistic safety analysis is to determine all significant contributors to the radiation risk from a facility or activity and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety objectives. In the area of reactor safety, the probabilistic safety analysis has been carried out using a comprehensive, structured approach to identify failure scenarios, constituting a conceptual and mathematical tool for deriving numerical estimates of risk. The probabilistic approach used realistic assumptions and provided a framework for explicitly addressing many of the uncertainties and insights into system performance, reliability, interactions and weaknesses in the design, defense in depth and risk that may not be possible to be derived from a deterministic approach.

3.5.2. Probabilistic safety objectives

The proposed figures for probabilistic safety objectives follow the international trend (e.g. the INSAG12 recommendations).

Safety objectives shall not be interpreted as safety limits, but they are specified in order to ensure sufficient safety provisions in the ATMEA1^{TM} design. Probabilistic assessment figures shall be considered as indicators as it was recognized by Safety Authorities.

The Nuclear Island, safety objectives are such that:

- The integral Core Damage Frequency (CDF), considering all plant states and all types

of events (internal events, internal hazards and external hazards except sabotage) is $< 10^{-6}$/r.y.

and

- The integral Frequency of “Large Releases” (LRF) is $< 10^{-7}$/r.y.

3.5.3. Operator action response time

In conjunction with decreasing the frequency of initiators and providing features to manage and deal with hypothetical accidents, it is of the utmost importance to facilitate the management of accidents, and to provide operators and emergency teams with increased response time.

ATMEA1^{TM} provides a large autonomy for safety systems and an increased operator action response time. Water capacities are provided to maintain hot shutdown conditions for about 24 hours, storage capacity of engine fuel for EPS are available on-site for at least 7 days, no site based mobile light equipment and no offsite or onsite mobile heavy equipment are needed for respectively 6 hours and 72 hours.

Operator actions are not needed for at least 30 minutes from the main control room and 1 hour in the plant.

3.6. Safety systems to cope with Design Basis Event (DBE)

3.6.1. Safety system configuration

The main features of the ATMEA1^{TM} safety systems are three independent trains which are respectively fitted to the three reactor loops. See Figure 2

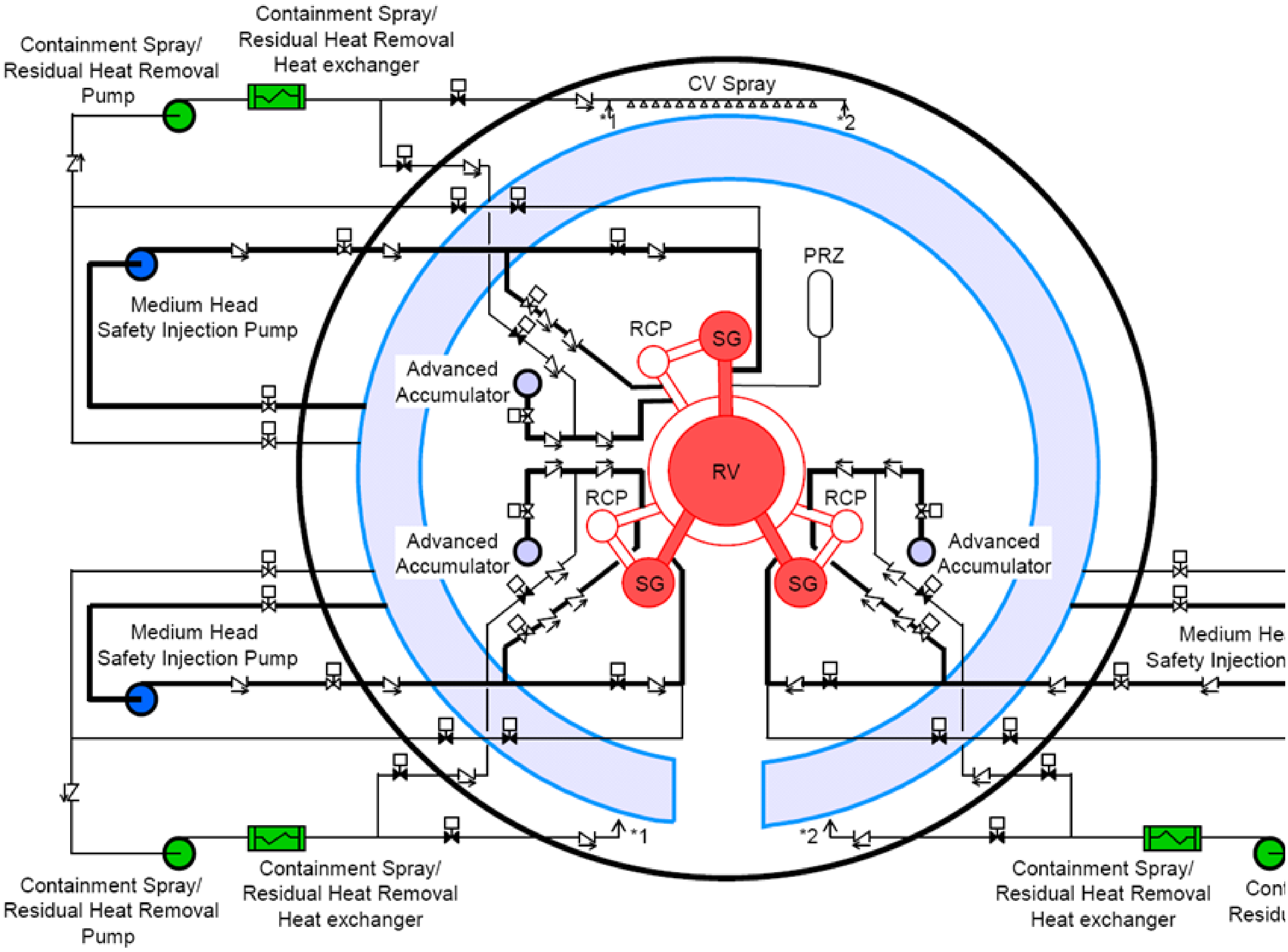

Fig. 2. General arrangement of the safety systems

The additional fourth train is installed for cooling chain systems, which provides both an On-Power Maintenance (OPM) capability and diversity. See Figure 3

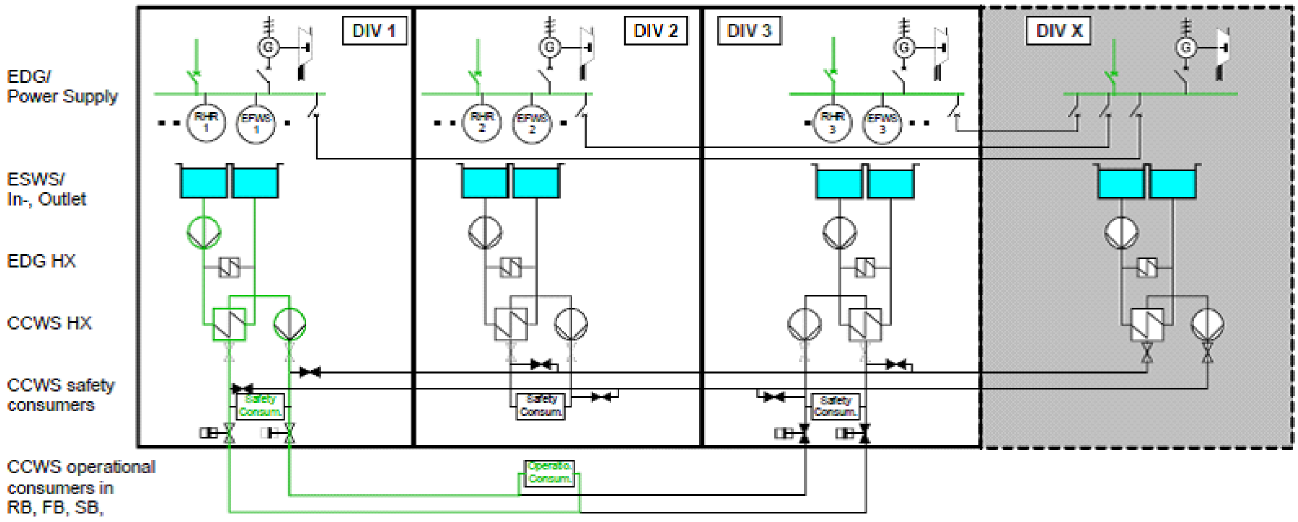

Fig. 3. General arrangement of the cooling chain systems

Advanced accumulators achieve functions of high flow injection during the blowdown phase just after a Loss of Coolant Accident (LOCA) and low head injection during subsequent core re-flooding phase.

3.6.2. Safety injection system (SIS)

SIS injects and re-circulates medium head emergency coolant to maintain the reactor core's coolant inventory following a LOCA, or following a Main Steam Line Break (MSLB). SIS consists of three independent identical trains. Each train has a Medium Head Safety Injection (MHSI) pump and an advanced accumulator pressurized by nitrogen. Each train has its own suction connection to the IRWSP via a series of screens, thus protecting the MHSI pumps against debris being entrained with the IRWSP fluid.

3.6.3. In-containment refueling water storage pit (IRWSP)

The IRWSP is located inside the containment for the following reasons :

- This avoids the requirement to switch over from injection mode to recirculation mode after the tank(s) are empty in the event of a LOCA or a steam line break;
- To provide water for corium cooling in the event of a core melt.

The IRWSP is situated at the bottom of the containment and srrounding the reactor pit. The area between the IRWSP and containment wall is filled with concrete in order to avoid non-recoverable water losses during accident conditions. Other areas without concrete filling, relevant for water losses, are taken into account in the IRWSP volume balance.

3.6.4. Emergency feedwater system (EFWS)

The EFWS feeds water to the SGs to maintain their water level and to remove decay heat following the loss of normal FW supplies due to Anticipated Operational Occurrence (AOO) and DBE conditions. This ensures the removal of the heat from the RCS, which is transferred to the secondary side via the SGs and via the SG Main Steam Relief Valves (MSRV) to the atmosphere.

The EFWS has sufficient capacity and independence to perform its required safety functions following DBEs, assuming a single failure in one EFW train.

The EFWS has three separate and independent trains, each consisting of a water storage pool, pump, control valves, isolation valves, piping and instrumentation and powered by a separate electrical train. In the event of a common mode failure of all EPSs, one of the motor-driven EFW pumps can be powered by an Alternative AC Power Source (AAC).

One EFWS train is located in each division of the Safeguard Buildings (SB), providing separation and/or physical protection to cope with external and internal hazards. The storage pools are made of concrete with internal liner and are a structural part of SB.

3.6.5. Containment spray system (CSS)/Residual heat removal system (RHRS)

The CSS/RHRS performs normal shutdown cooling, as well as containment spray injection to maintain RB conditions within design limits in the event of a LOCA, Main Steam Line Break (MSLB) or Main Feedwater Line Break (MFLB).

The CSS/RHRS has safety functions such as the reduction of pressure and temperature inside the containment, down to acceptable levels in the event of a LOCA, MSLB or MFLB as well as the normal operational functions such as reaching and maintaining safe shutdown state and refueling conditions.

The CSS/RHRS has sufficient capacity, and independence to perform its required safety functions following design basis transients or accidents, assuming a single failure in one train.

The CSS/RHRS consists of three independent trains, with a separate suction connection to the IRWSP; the sumps have a series of screens, ensuring protection of the containment spray pumps against debris being entrained in the IRWSP fluid.

All three CSS/RHRS trains are powered from separate emergency buses, each backed up by an EPS. Each CSS/RHRS train is located in the associated division in SB, thereby providing separation and/or physical protection from external and internal hazards.

3.6.6. Extra borating system (EBS)

The EBS injects borated water into the RCS in the event of DBEs in order to maintain the core subcritical for safe shutdown.

The EBS is composed of two identical trains. Each EBS train is composed of its own boron tank, a high pressure 100% capacity pump, a test line and injection lines to the RCS.

The EBS is also used to cope with ATWS which is BDBE –beyond design basis event.

3.7. Safety system to cope with severe accidents

For advanced Gen III+ nuclear power plants, severe accidents must be addressed during the design stage. Full benefit is taken from insights gained from such inputs as probabilistic safety assessments, operating experience, severe accident research, and accident analysis.

The approach is primarily based on the prevention of accident situations with core melt which would lead to early loss of the containment and consequently to large early releases. Specifically, the approach aims at preventing high pressure core melt and direct containment heating, hydrogen detonation, steam explosion with risk of containment failure and core melt sequences with containment bypass and hence significant radiological impact.

The implementation of designed mitigation features and severe accident procedures based on severe accident monitoring aims at achieving safety goals:

- Prevention of high pressure core melt by high reliability of decay heat removal systems complemented by dedicated primary system overpressure protection;

- Primary system discharge into the containment in the event of a total loss of secondary side cooling;

- Features for corium spreading and cooling - the spreading area is coated with a sacrificial protective material and has a cooling system to protect the basemat;

- Prevention of hydrogen detonation by reducing the hydrogen concentration in the containment at an early stage with catalytic hydrogen recombiners;

- Control of the containment pressure increase by a dedicated Severe Accident Heat Removal System (SAHRS) consisting of a spray system with recirculation through the cooling structure of the melt retention device;

- Collection of all leaks and prevention of bypass of the confinement, in an annulus.

3.8. Provisions for safety under seismic conditions

For the reference design, the SSE is taken equal to 0.3g (horizontal acceleration free field level) and the US NRC ground response spectra, as defined in RG 1.60, is used.

This value is an envelope of European countries, most of the USA and some of the Asian countries. In the USA, this value is described in the top tier of the URD and most new reactors are designed to withstand the same seismic level.

If site conditions exceed this value, the seismic relief devices will be adopted so as not to affect the layout or support structures. To minimize the impact of adopting the seismic relief devices, layout considerations, such as a flat common base mat, were already taken into account in the Basic Design.

3.9. Probabilistic risk assessment (PSA)

The aim of a PSA is to determine all the significant contributors to the risk from the plant and to evaluate the extent to which the overall design is well balanced and meets the probabilistic safety objectives. The combined Level 1 and level 2 PSA were carried out with a comprehensive, structured approach to identify failure scenarios, constituting a conceptual and mathematical tool for deriving numerical estimates of risk. The probabilistic approach uses realistic assumptions whenever possible and provides a framework for addressing many uncertainties explicitly. Probabilistic approaches were used to provide insights into system performance, reliability, interactions and weaknesses in the design, defense-in-depth and risk that cannot be derived from a purely deterministic approach.

3.10. Emergency planning measures

ATMEA1^{TM} adopts stringent 21st century reactor criteria consistent with those used for reactors currently under construction in Europe. There criteria also covers the requirement of US NRC regulations.

These criteria ensure limited emergency response plans needs and limited areas to deal with. This applies in both the short and the long terms after an accident, long-term referring to tens of years. In countries where land is scarce, where the density of population is high, where public acceptance needs to be gained, such capabilities are much needed.

For infrequent and limiting faults, the following overall criteria apply:

- No short term countermeasures (shelters, evacuation, distribution of iodine tablets)

- No need for long-term relocation

- Food restrictions limited to the immediate vicinity of the affected site

Dose limits corresponding to objectives associated with radiological consequences are:

- Effective dose 10 mSv

- Dose equivalent to the thyroid 100 mSv

- Food consumptions: European Union limits

Distances are 500m (typical limit of site) for the short term phase and 2 km from the release location for the long term.

Although severe accidents are beyond design basis events, ATMEA1™ criteria are also extremely stringent.

The severe accident overall approach is consistent with objectives established for radiological consequences that can be interpreted as “maintain its role as a reliable, leak-tight barrier throughout the accident and in the long term for the relevant severe accident sequences and phenomena”.

- No need for short-term countermeasures over the first 24 hours. Criterion: an effective dose of 50 mSv;

Short-term countermeasures (shelters, evacuation) are required only in the immediate vicinity of the site. Distances are defined above. Criterion: Sheltering (effective dose) 10 mSv ; Evacuation (effective dose) 50mSv ; Iodine distribution (Dose equivalent to the thyroid) 100 mSv; Long and short term measures (Relocation: 10 mSV/month for long exposure (irradiation dose rate from the ground) or 1 Sv (Effective dose))

- No long-term relocation is necessary, except for the population residing in the immediate vicinity of the site. Criterion: vicinity of the site: 1000 mSv and 10mSv/month at the vicinity of the site;
- Restricted consumption applies only to the first harvest following the event. Criterion: European Union limits.

Proliferation resistance

The ATMEA1TM renders the diversion or undeclared production of nuclear materials or misuse of technology, by host state, very difficult due to inherent technical impediments. On the one hand the declared inventory is not appealing and diversion of either fresh or spent fuel elements is made difficult by design; on the other hand undeclared production of weapon grade materials has a very significant cost and is easily detectable.

4.1. Intrinsic barriers

The intrinsic barriers that reduce the risk of diversion or misuse pathways are summarized here below.

4.1.1. Fresh fuel

The ATMEA1TM is designed for operation with fissile material that has a poor attractiveness from the proliferation point of view.

Fresh fuel assemblies made use

- either of low enriched uranium fuel (LEU), the enrichment is no higher than 5% ^{235}U thus far below "weapon-grade",
- or of uranium-plutonium mixture known as Mixed Oxide fuel (MOX), the reactor grade plutonium coming from the reprocessing of LWR fuel.

Diversion of LEU fuel assemblies for use as feed in enrichment devices and/or diversion of fresh MOX assemblies for processing to separate plutonium are made extremely difficult. A fuel assembly weights more than half a metric ton and requires specific equipment to be lifted. Furthermore the fuel assemblies are handled and stored in the fuel building which is protected by a heavy shielding and has limited and controlled access.

4.1.2. Fuel under irradiation

When the reactor is operating the reactor pressure vessel is closed and the fuel is inaccessible. Refueling operation with an open vessel and fuel transfers occur under transparent water shielding permitting direct visual observation for safeguards purposes.

4.1.3. Spent fuel handling and storage in fuel pool

LEU fuel which has gone through a normal operating life (e.g. 3 fuel cycles) will reach a high burn-up, so it is of limited interest for proliferation purpose: the ^{235}U content is below 1% and the poor isotopic quality of the plutonium leads to high neutron emission rate, high heat emission and high level of radiation.

In the spent MOX fuel assembly, the remaining plutonium content has an even worst isotopic quality, and thus a further reduced attractiveness.

In both cases, the spent fuel is highly radioactive and would require a heavily shielded cask to be moved, therefore theft is unrealistic.

4.1.4. Misuse of the reactor for producing weapon grade plutonium

The design of the reactor vessel internal structure does not leave room to allow irradiation of specific uranium target. Moreover, producing plutonium for proliferation purpose will need irradiated fuel with low burnup in order for its quality to be of practical interest.

The normal operating fuel cycle for the reactor being between 1 and 2 years before shutdown for refueling, short cycle durations would obviously be detected by safeguard inspectors.

4.2. Safeguard ability

The design of the plant facilitates the implementation of safeguard inspections control and accounting measures that are the extrinsic barriers enforcing the institutional agreements and policies.

Refueling operation and associated fuel movements are conducted at a low frequency and take place in only two building that may easily be placed under surveillance. The integrity of these building boundaries is ensured by the structure designed against external hazards. The few access points allow monitoring and surveillance of all passages.

Safety and security (physical protection)

Security provisions are integrated in the design to deal with malevolent actions by protecting sensitive structures, systems and components and by allowing the implementation of security procedures during operation and maintenance activities.

The objectives are the integrity of nuclear materials and NPP safety features, to avoid the threat of radiation for the public or any theft of nuclear fuel.

Stipulations on physical protection are mostly classified information, and only some general principles are given here below.

The plant arrangement and the design of the buildings allow implementing different levels of security areas accessible only after passing access control points.

The plant area is surrounded with the site fence, including the gatehouse, the vehicle barrier…These security features are usually within the plant owner responsibility. Resistance of walls and closure (e.g. doors, grids) is required for the protected area as well as the surveillance of all passage. Within the protected area, the vital area contains all the systems and equipment important to plant safety or security and the storage of the nuclear material.

Boundaries between areas with different security level are structural barriers designed against unauthorized access. Specific features allow monitoring, surveillance and recording of all passages through the boundary between the different areas.

Physical protection of the vital areas against destructive acts from outside is typically based on:

- provision against external hazards, such as the physical separation of redundant systems and the implementation of the air plane crash protection structure,

- design features implemented for coping with station blackout; they provide grace periods in case of destructive acts from outside, this time allowing to restore water inventories and/or to recover damaged plant equipment.

- security provision made to prevent and to detect incorrect inputs in the I&C systems and equipment.

Physical protection against unauthorized manipulations of an authorized person within the vital area takes benefit from the segregation of the redundant trains of the mechanical, electrical and I&C safety systems.

To ensure the plant security of any project, features addressing the same general principles as those considered in the reference design are implemented. For every project, the detailed definition and design of the security provisions have to be achieved independently of the other projects in order to ensure the confidentiality of the information.

Moreover, requirements for safeguarding and measures connected are usually ruled by national regulation and that may lead to specific alteration of the scope.

Description of turbine-generator systems

ATMEA designs the ATMEA1™ NI and the design is open to turbine-generator designs. The design interfaces from the NI to the CI are defined but this does not limit turbine-generator suppliers to any specific ones.

Electrical and I&C systems

7.1. Power supply system

Onsite electrical power is supplied by offsite power systems which consist of two offsite transmission systems connected to the grid, either via unit auxiliary transformers or reserve auxiliary transformers.

The onsite electrical power system has safety-related electric power system and non-safety-related electrical power systems

The safety electrical power system has three redundant trains (Train A, B, and C). Each safety train has an Emergency Power Source (EPS) which is capable of supplying the electrical power necessary to bring the plant to a safe state and maintain it there.

For the purpose of OPM of specified equipment and the EPS, there is an additional safety electrical power system (Train X).

When Loss of Offsite Power (LOOP) occurs, the EPS starts automatically and reaches its rated voltage within a specified time after receiving an undervoltage signal.

Each EPS circuit breaker is closed automatically, and the circuit breakers of the safety-related loads are closed automatically, in sequence.

To cope with Station Black Out (SBO), which is the condition when all the EPSs are inoperable, an Alternative AC source (AAC) provides the power. It is not safety-related and of a different type from the EPS. The AAC is normally isolated from the safety electrical power systems. When LOOP occurs, just as for the EPS, the AACs start automatically and then supplies electrical power to specified non-safety-related loads. In the event of an SBO, the AACs supplies electrical power to those safety-related loads of one train that are needed to bring the plant to a safe state and maintain it there, by manual operation.

Until AC power is restored, batteries supply power directly to the DC loads or, through converters, to the AC loads.

7.2. Human-machine interface facilities and control rooms

7.2.1. Human System Interface (HSI)

Human factors are taken into account at the design stage, covering operation, testing, and maintenance requirements. The general aim is to minimize the likelihood of operator error and lowering the demands on the operator. For this purpose, appropriate ergonomic design principles are implemented and operators are given enough time to think through and plan their actions. The time necessary depends on the complexity of the situation to be diagnosed and the actions to be taken.

All the information supplied by the I&C systems enable plant status during normal operation to be assessed, including Design Base events (DBEs), and Beyond DBEs, and to evaluate the effects of any actions taken.

7.2.2. Main Control Room (MCR)

For all plant conditions (except if the MCR becomes inaccessible) the plant is supervised and controlled from the MCR. The MCR is equipped with identical operator workstations consisting of Process Information and Control System (PICS) driven screens and soft controls. These operator workstations are used as follows:

- Two of the operator workstations are staffed during normal plant operations;
- A third operator workstation is used by the shift supervisor;
- A fourth operator workstation is staffed during plant states requiring increased operating and monitoring tasks (e.g., refueling period, post accident conditions).

7.2.3. Remote Shutdown Station (RSS)

If the MCR becomes inaccessible the operators supervise and control the plant from the RSS.

The RSS is equipped with:

- Manually-actuated switches for disconnecting all the MCR equipment that may send spurious signals to Level 1 systems and switching the RSS workstations to control mode. Technical and administrative precautions prevent spurious or unauthorized actuation of this function;
- Operator workstations consisting of PICS-driven screens and soft controls which are of the same type and provide the same functionality as those in the MCR. The operators can bring the plant to a safe shutdown state and monitor plant conditions from these operator workstations;
- Communication equipment for maintaining communications with other plant personnel;
- Fire fighting controls.

7.3. Reactor Protection System (RPS) and other safety systems

The RPS is the main I&C line of defense and performs any automatic safety-related functions that are needed to bring the plant to a controlled state if a DBE occurs. The RPS performs the following functions:

- Monitoring the safety parameters of the process, detecting plant conditions that indicate the occurrence of a DBE 2, 3 or 4, and initiating automatic protection functions to mitigate the event. This purpose is fulfilled through the automatic actuation of the reactor trip and Engineered Safety Feature (ESF) systems;
- Generating signals enabling or disabling certain protective actions in line with the current plant state;
- Providing the capability for manually controlling functions that are vital for safety;
- Providing information on the status of safety parameters for post-accident management to PICS and Safety Information and Control System (SICS);
- Managing the automatic control of safety-related support systems.

The RPS is a safety-related digital I&C system, located in dedicated cabinets in the nuclear island. The system has three main trains of redundancy: Trains A, B and C. In addition a fourth train named Train X exists. Train X processes only some of the protective functions for the purpose of increasing the reliability of the essential ESF actuation functions. Each of Trains A, B, C and X is mounted in a room located in the corresponding division. Each of them has its own safety-related power source. Additionally, each RPS cabinet has redundant power supplies for its electronics. The RPS is functionally independent of all other I&C automation systems, thereby ensuring that the failure of one of those systems does not prevent the RPS from performing its safety functions. The system can process safety-related (and non-safety-related) functions.

Spent fuel and waste management

8.1. General

With steam generator axial economizer technology and optimized operating parameters of the primary loops, the thermal efficiency of ATMEA1^{TM} is 37% (the actual value depending however on site conditions), which is around 10% more than the current operating plants, leading to less fuel consumption and less waste generation for the same amount of energy produced.

8.2. Spent fuel

The standard design of ATMEA1^{TM} provides for an on-site storage of the spent fuel at the spent fuel pit for 10 years of full power operation. Beyond this period, plant operators decide whether to store the spent fuel assemblies on site or to reprocess and ship them to a reprocessing plant where they will be recycled. Materials, such as plutonium and uranium, will be re-used.

The plutonium originating from reprocessed spent fuel is recycled in the form of mixed uranium and plutonium oxides fuel (MOX).

8.3. Radioactive wastes

The waste building stores radioactive waste in a controlled area which is readily accessible for reception and dispatch and has sufficient capacity to store waste for an operationally appropriate period.

Plant layout

9.1. Buildings and structures

The ATMEA1™ Nuclear Island (NI) consists of:

- The Reactor Building (RB), the Safeguard Auxiliary Building (SAB) and the Fuel Building (FB), which are located on a common basemat;
- The Nuclear Auxiliary Building (NAB), two EPS Buildings, the Radioactive Waste Processing Building, and the Access Control Building, which are located on individual basemats.

The Turbine Building (TB) is independent from the Nuclear Island (NI).

9.1.1. External and internal hazards

Each safety system train is protected against the propagation of internal hazards (e.g., fire, high energy line break, flooding) from one train to any other. This requirement leads to placing each train in a specific area or division that is separated from the others.

The structural design and physical arrangement of the buildings provide protection against both external and internal hazards. Additionally, all safety-related buildings are designed to withstand the effects of an SSE.

The ATMEA1^{TM} NI is designed to withstand a large commercial Air Plane Crash (APC) and an External Pressure Wave (EPW) due to an external explosion.

9.1.2. Radiation protection

Stringent radiation protection requirements limit personnel exposure during operation and while performing in-service inspection and maintenance to very low levels. The design target for the maximum collective dose exposure is less than 0.5 man-Sv per year.

The SAB is separated into radiologically controlled (“hot”) and non-controlled (“cold”) areas. Those systems that are radiologically “cold” under normal conditions, such as the CCWS and EFWS, are separated (including the dedicated access locations) from the systems that are radiologically “hot,” such as the RHRS. This arrangement minimizes the necessity for personnel to enter contaminated areas.

Primary layout design features providing radiation protection include the following:

- Equipment is located in separate compartments (tanks/heat exchangers, pumps and valves) according to access requirements and anticipated radiation levels;
- Access to radioactive components is provided via shielded service routes and transition from areas with lower radiation levels to areas with higher radiation levels.

9.2. Containment

The RB contains the major NSSS components and piping along with some associated safety-related auxiliary systems. The RB consists of the Containment Building (also called the Pre-stressed Concrete Containment Vessel (PCCV)) surrounded, in the lower part, by the annulus space and an outer wall of reinforced concrete. See Figure 4

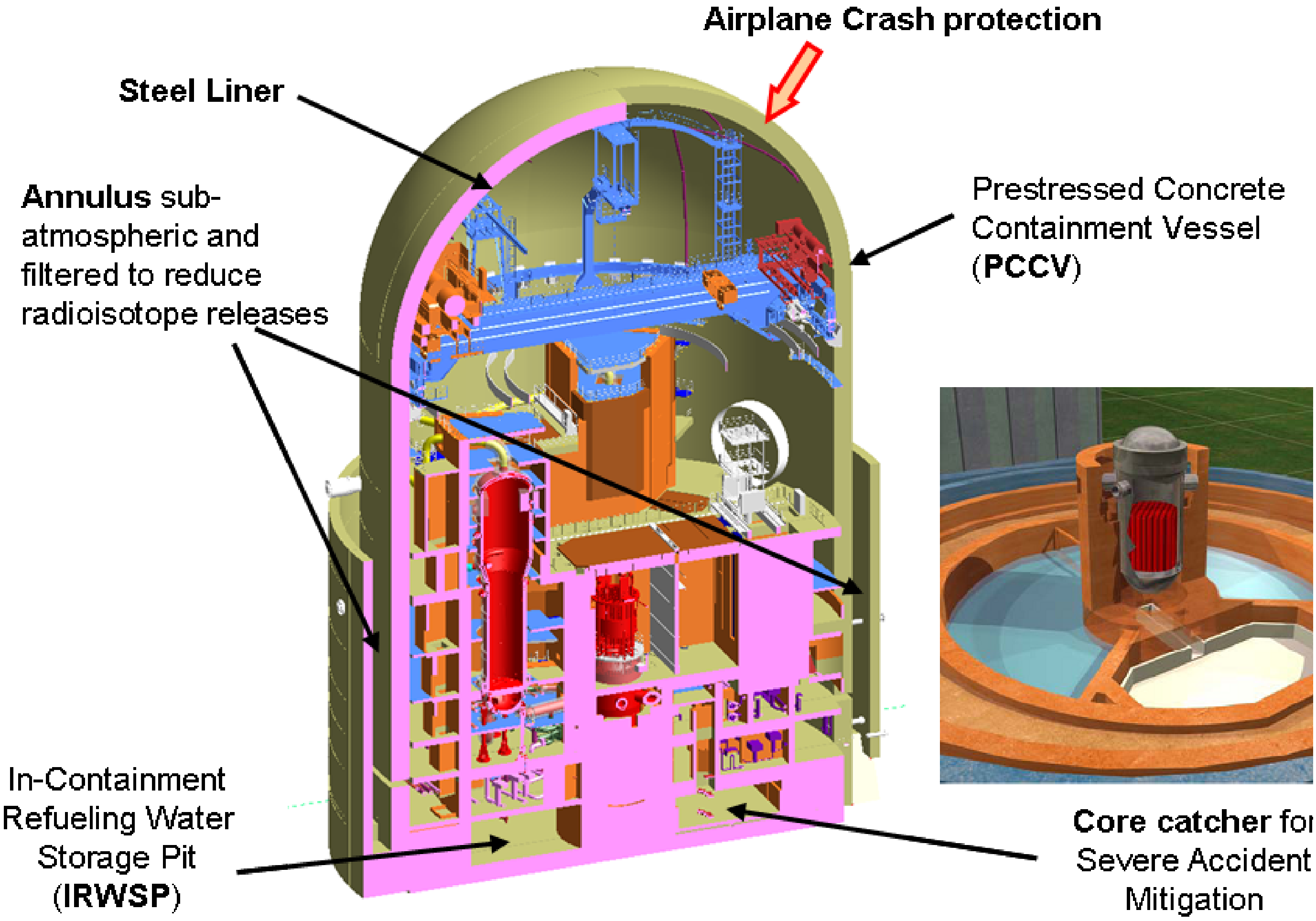

Fig. 4. Containment - General features

The RB has a steel liner covering the inner surface of the PCCV.

The RB is located at the center of the NI. It is surrounded by the SAB and FB which are on a same basemat. It is designed to withstand internal accidents as well as external hazards including the following: earthquake, APC, EPW, missiles, tornado, and fire.

In the event of a LOCA or severe accident, the RB serves to retain all radioactive material and to withstand the maximum pressure and temperature resulting from the release of stored energy.

The design pressure and temperature that the containment building must withstand are defined by the following events:

- Double-ended rupture of a reactor coolant pipe (Large Break LOCA (LBLOCA));
- Main steam line break;
- Severe accidents.

9.2.1. Containment description

The basis for the RB is an accessible containment which allows personnel access for planned maintenance and inspection activities several days before and after the reactor shutdown in order to keep the outage duration as short as possible.

To enable accessibility during power operation, a “two room concept” is provided. See Figure 5.

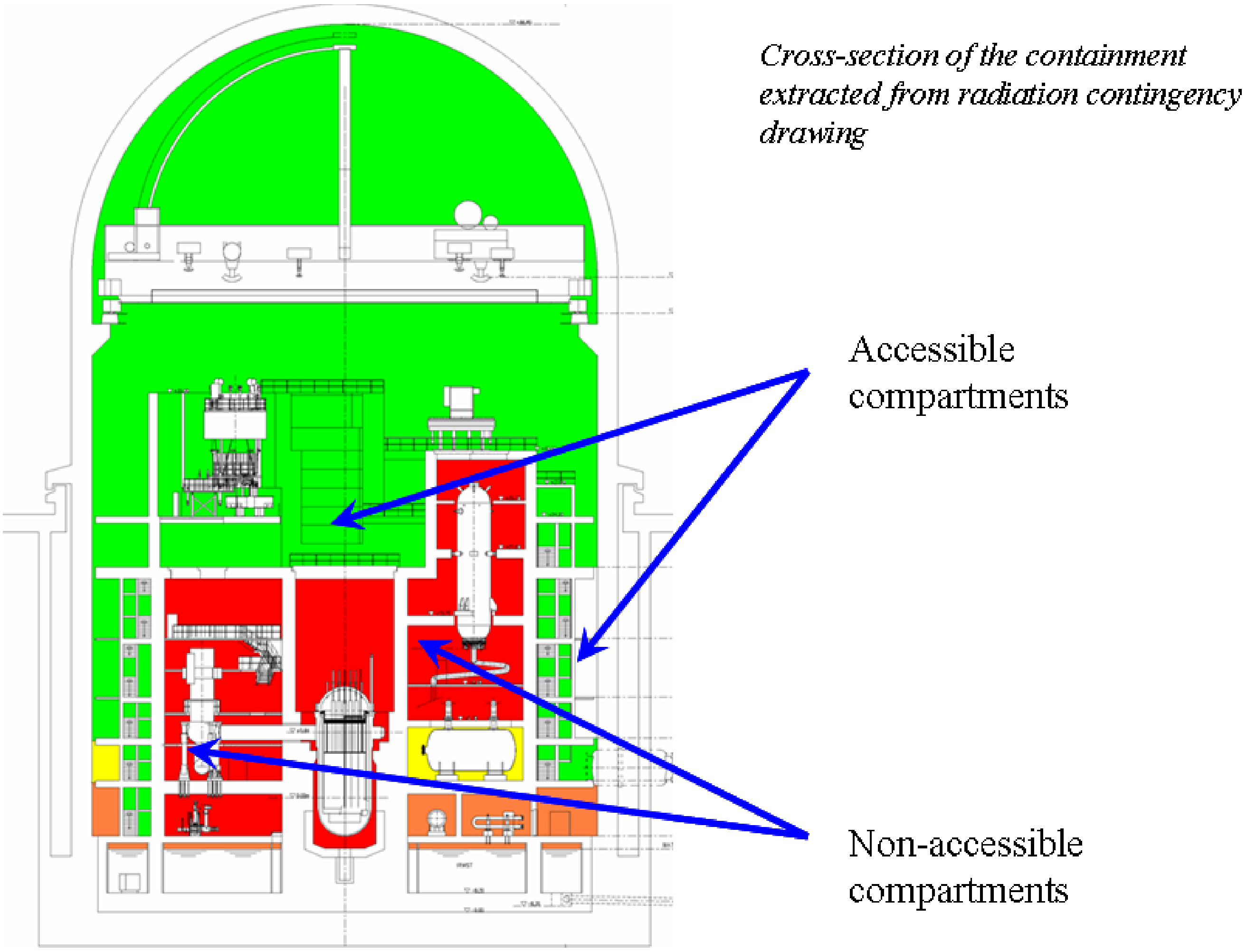

Fig. 5. Two room concept inside the containment

This “two room concept”, which divides the inner containment into two types of compartments:

- Inaccessible or limited-access compartments during reactor operation: the equipment compartments;

- Accessible compartments: the service compartments.

In the accesible compartment, the working environment (radiation level, temperature, etc.) is ensured by shielding walls and a dedicated HVAC system

Plant performance

10.1. Plant operation

The reference configuration (nominal power) for ATMEA1^{TM} is:

- Net electrical power:1100-1150 MWe with a 35-37% efficiency;
- NSSS output :3150MW;
- Fuel cycle lengths from 12 to 24 months.

The ATMEA1^{TM} core design allows various fuel management schemes from full uranium to full MOX cores with 12 to 24 month operational cycles.

In addition, ATMEA1^{TM} has a capability for frequency control and load-follow (see 2.6.).

10.2. Reliability

ATMEA1^{TM} is composed of either operated, licensed, or certified systems or components, integrating the most modern proven technologies developed by AREVA and MHI and implemented respectively into AREVA’s EPR^{TM} and MHI’s APWR. The fact that the ATMEA1™ design is derived from both reactors contributes to its safety and reliability.

10.3. Availability targets

Several design features contribute to achieve short outage and easy maintenance:

- On-power maintenance capability
- Accessible reactor building
- Layout provisions for maintenance
- High speed refueling machine
- High capacity coolant purification system

ATMEA1^{TM}'s design availability factor is expected to exceed 92% over 60 years of plant life. This assessment, based on the EUR methodology, has taken into consideration conservative and the most realistic values as for normal, specific or forced outage duration and outage extension for special maintenance operations.

10.4. Provision for design simplification, reduced capital and construction costs

Based on various assessments available, and in view of providing the most cost-effective design within the safety design approach, the thermal power level of 3150 MWth was retained for ATMEA1™. This thermal power level can provide a net electrical power between 1100 and 1150 MWe which is suitable for most site configurations.

Economic and environmental benefits have been extensively factored into the design. With design assumptions taken that covers a large variety of potential sites, a high thermal efficiency of up to 37% leads to lower fuel consumption and waste reduction. A design for a 60-year service life reduces the decommissioning burden for a given level of electricity generation.

The availability factor of 92% or more, during the entire service life of the plant, facilitated by long fuel cycles and short refueling outages, contributes to the economical aspect of ATMEA1™.

Simplification in all areas of design and the optimization of active and passive design features have been made for the best balance between capital costs and operational benefits resulting in an economical reactor while keeping the highest safety standards.

10.5. Construction schedule

The construction schedule largely depends on the site conditions. For a typical case, based on the parent companies' experience, ATMEA1^{TM}'s target schedule is 40 months from first concrete to a fuel loading, and 48 months from first concrete to the start of a commercial operation.

10.6. Provisions for low fuel reload costs

The ATMEA1^{TM}'s high thermal efficiency (typically 10% higher than currently operating reactors) reduces fuel consumption to generate the same amount of electricity and thus lowering the fuel cost.

In addition, due to relatively low linear heat rate of the core and high burn-up fuel, the ATMEA1^{TM} design allows various fuel management schemes to be accommodated, from uranium to full MOX cores and from 12 to 24 month operational cycles with optimized fuel cost.

Depending on operators' conditions, ATMEA1^{TM} can maximize the best balance of its fuel costs and overall plant availability. Both of them have an impact on the kWh generation costs

Development status of technologies relevant to the NPP

The ATMEA1^{TM} design is based on the experience of both AREVA and MHI companies, with about 130 nuclear power plant constructions and services provided to the operating plants.

The basic design of ATMEA1^{TM} had been completed at the end of 2009. The reactor is now ready to be deployed wherever 1100 MWe class reactors are needed.

Technical data

General plant data

Reactor thermal output	3150 MWth
Power plant output, net	1150 MWe

Power plant efficiency, net	36 %
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target	92 %
Seismic design, SSE	0.3
Primary coolant material	Light Water
Moderator material	Light water

Safety goals

Core damage frequency	10E-6 /Reactor-Year
Large early release frequency	10E-7 /Reactor-Year
Occupational radiation exposure	0.5 Person-Sv/RY

Nuclear steam supply system

Steam pressure	7.2 MPa(a)

Reactor coolant system

Primary coolant flow rate	6889 Kg/s
Reactor operating pressure	15.5 MPa(a)
Core coolant inlet temperature	291 °C
Core coolant outlet temperature at end of life	326 °C

Reactor core

Fuel material	UO_2 and MOX
Cladding material	Zirconium alloy
Rod array of a fuel assembly	17x17
Number of fuel assemblies	157
Enrichment of reload fuel at equilibrium core	5.0 Weight %
Fuel cycle length	24 Months
Burnable absorber (strategy/material)	Gd_2O_3
Control rod absorber material	Hybrid (AIC/B_4C)
Soluble neutron absorber	H_3BO_3

Reactor pressure vessel

Inner diameter of cylindrical shell	4250 mm
Base material	Low alloy steel

Steam generator or Heat Exchanger

Type	U tubes with axial economizer
Number	3
Total tube outside surface area	8000 m^2

Tube outside diameter	19 mm
Tube material	TT690 alloy

Reactor coolant pump (Primary circulation System)

Pump Type	Shaft seals
Number of pumps	3
Flow at rated conditions	6.89 m^3/s

Pressurizer

Total volume	65 m^3

Primary containment

Type	Sealed envelope of prestressed reinforced concrete with lining
Overall form (spherical/cylindrical)	Cylindrical

Residual heat removal systems

Active/passive systems	Active

Safety injection systems

Active/passive systems	Active and Passive