



ATMEA

Contribution to the Nuclear Reactor Design Description to
the IAEA ARIS database

May 11, 2015

Contents

1. Introduction	4
1.1. Generalities	4
1.2. Proven Technology: ATMEA1 designed from currently operating PWRs.....	4
1.3. Design features	6
2. Description of the Nuclear Systems.....	7
2.1. Main characteristics of the primary circuit.....	7
2.2. Reactor core and fuel design.....	7
2.2.1. Core design.....	7
2.2.2. Fuel Assemblies (FAs)	7
2.2.3. Reactor Cluster Control Assemblies (RCCAs) and Reactivity Control	7
2.3. Fuel handling systems.....	7
2.4. Primary circuit component description	8
2.4.1. Reactor Pressure Vessel.....	8
2.4.2. Reactor internals	8
2.4.3. Steam generators (SGs)	8
2.4.4. Pressurizer (PZR)	8
2.4.5. Reactor coolant pumps (RCPs).....	9
2.4.6. Main coolant lines	9
2.5. Reactor auxiliary systems.....	9
2.5.1. Chemical and volume control system (CVCS).....	9
2.5.2. Component cooling water system (CCWS)	9
2.5.3. Essential service water system (ESWS).....	10
2.6. Operating modes.....	10
2.7. Alternative fuel options	11
3. Description of safety concept.....	12
3.1. Safety concept, design philosophy and licensing approach	12
3.1.1. Design philosophy	12
3.1.2. Licensing status	12
3.2. Provision for simplicity and robustness of the design.....	13
3.3. Safety systems.....	13
3.4. Defense-in-depth description.....	13
3.5. Core damage frequency, large early release frequency and Autonomy.....	14
3.5.1. Probabilistic targets and safety assessment.....	14
3.5.2. Probabilistic safety objectives.....	14
3.5.3. Autonomy.....	15
3.6. Safety systems to cope with Design Basis Event (DBE)	15
3.6.1. Safety system configuration	15
3.6.2. Safety injection system (SIS).....	15
3.6.3. In-containment refueling water storage pit (IRWSP)	16

3.6.4.	Emergency feedwater system (EFWS).....	16
3.6.5.	Containment spray /Residual heat removal system (CS/RHRS).....	16
3.6.6.	Extra borating system (EBS)	17
3.7.	Systems to mitigate severe accidents.....	17
3.8.	Provisions for safety under seismic conditions	17
3.9.	Probabilistic risk assessment (PSA)	18
3.10.	Extreme external hazards	18
3.11.	Emergency planning measures.....	19
4.	Proliferation resistance	20
4.1.	Intrinsic barriers.....	20
4.1.1.	Fresh fuel	20
4.1.2.	Fuel under irradiation	20
4.1.3.	Spent fuel handling and storage in fuel pool.....	20
4.1.4.	Misuse of the reactor for producing weapon grade plutonium.....	20
4.2.	Safeguard ability	21
5.	Safety and Security (physical protection).....	22
6.	Description of turbine-generator systems	23
7.	Electrical and I&C systems.....	24
7.1.	Power supply system	24
7.2.	Human-machine interface facilities and control rooms.....	24
7.2.1.	Human System Interface (HSI).....	24
7.2.2.	Main Control Room (MCR).....	24
7.2.3.	Remote Shutdown Station (RSS)	25
7.3.	I&C Systems	25
7.3.1.	General Description and Design Basis	25
7.3.2.	Diversity and Defense in Depth Concept.....	25
8.	Spent fuel and waste management	27
8.1.	General	27
8.2.	Spent fuel.....	27
8.3.	Radioactive wastes	27
9.	Plant layout	28
9.1.	Buildings and structures	28
9.1.1.	External and internal hazards.....	28
9.1.2.	Radiation protection	28
9.2.	Containment.....	29
10.	Plant performance	30
10.1.	Plant operation	30
10.2.	Reliability	30
10.3.	Availability targets.....	30

10.4.	Provision for design simplification, reduced capital and construction costs	30
10.5.	Construction schedule	31
10.6.	Provisions for low fuel reload costs	31
11.	Deployment status of technologies relevant to the NPP	32
Figure 1:	Steam generator.....	37
Figure 2:	General arrangement of the safety systems	38
Figure 3:	General arrangement of the cooling chain systems	38
Figure 4:	Containment - General features.....	39
Figure 5:	Defense levels and instrumentation and control systems.....	39

1. INTRODUCTION

1.1. Generalities

ATMEA 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 is ATMEA1.

ATMEA1 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 has a basic set of common design features adaptable to regulatory and commercial requirements of any country in which it will be proposed. ATMEA1 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 designed from currently operating PWRs

The ATMEA1 is built up on evolution of operating pressurized water reactor using existing proven components/equipment and addressing the latest safety requirements of a generation III+ plant.

ATMEA1 NPP is a design proven in the field through operation experiences of the operating PWRs; the plant overall configuration is similar to operating PWRs designed, built by AREVA and MHI and operated with good records. ATMEA1 NPP is considered as an evolutionary design as opposed to a revolutionary design. This strategy minimizes risks of the FOAK and allows ATMEA company and operators to utilize all experience feedbacks from ~130 PWRs operating worldwide.

ATMEA1 reactor integrates evolutionary features into its design to improve safety and economy compared to currently operating PWRs designed decades ago. Redundancy and diversification have been systematically implemented, and systems have been simplified using advanced equipment, etc.

Most of ATMEA1 items, i.e., systems, components, structures, design and analysis techniques, maintainability and operability features and construction techniques have several years of operation in existing nuclear power plants. The list below shows several examples of important components:

- CRDM (control rod drive mechanism) and related equipment like the control rod guide assembly: The same are used in German KONVOI reactors with good operating records.
- Control rods with Ag-In-Cd and B4C as absorber material: The same material and design are widely used in French operating PWRs.
- SG (steam generator) with axial economizer: ATMEA1 SG design is the same as EPRs', and both are close to French operating N4 ones which have good operating records and performance.
- RCP (reactor coolant pump): The reference design of ATMEA1 RCP is the French operating N4 plants'. This design is used in ATMEA1 with minor adaptation.

- Fuel (17x17, 14ft, with M5 material): This fuel design is widely used in France especially in N4 reactors and in other countries
- Reactor vessel and its internal structures including hydraulics: the reactor vessel design is the same as operating PWRs' with appropriate size adaptation. The upper internals design is close to KONVOI design and the lower internals design is close to N4 design with minor adaptation of their sizes to ATMEA1 core size. Internals design is the scale down version of the EPR internals. In addition, the hydraulic tests specific to ATMEA1 design (full scale tests or part scale tests) are performed in order to obtain and confirm necessary hydraulic performances and characteristics for licensing, like inlet flow distribution for each fuel assembly, flow induced vibration, etc.
- Cell coolers as diversified second ultimate heat sink: This evolutionary feature providing an additional level of defence-in-depth for severe accident and/or extreme external hazards has already been implemented in some of German KONVOI reactors (e.g., GKN II)

Only a few components were newly developed and are not currently installed in operating PWRs i.e. the advanced accumulator with vortex dumper. The design is identical to the one developed for APWR. It had been extensively tested and validated. Due to the similar dimensions, materials, capacity and operating conditions, the tests are fully applicable to ATMEA1 reactor.

Regarding systems, the major improvement from operating PWRs is an increase of redundancy and diversity, e.g., 3-train safety system instead of 2-train system and diversified division-X train. Each redundant and diversified train has a similar system configuration as operating reactors. Each train is directly connected to each primary loop without cross-connections to improve independency of the safety trains while maintaining the proven in the field design.

To adapt to more stringent requirements from Authorities, some systems were newly developed and do not exist in the operating PWRs (e.g., severe accident heat removal system, extra borating system). All these systems were developed, fully tested, validated through extensive tests for EPR or APWR reactors, already licensed by several safety authorities, and under constructions. In addition, all components (pumps, valves, tanks, ...) of such systems are proven and also fully tested and validated for use for their purposes and conditions.

Civil structures, support structures of equipment, and other structures are also experienced design mostly in operating PWRs. Building configuration and related layout strategy referenced to Japanese experienced design especially to make full use of Japanese seismic design experiences and construction techniques. Regarding protection against "air plane crash" including large commercial airplane and military airplane, the ATMEA1 reactor makes use of a typical and common bunker-type protection of all buildings housing necessary equipment. This protection is done by an important thickness of highly reinforced concrete. The design addresses not only the impact to buildings and structures but also the induced vibrations and fires, and was satisfactorily assessed by the French ASN and the Canadian CNSC.

As for the design and analysis techniques, the latest methodology and computer codes already fully tested and validated through the experienced or on-going licensing process of the operating PWRs, EPRs, and APWRs are applied.

Maintainability and operability are strongly related with the layout design, system configurations, and equipment design. As already shown above, ATMEA1 design fully benefits from the proven in the field equipment and also from the latest and advanced requirements and guidelines for operation and maintenance through the maintenance activities carried out by AREVA and MHI on operating reactors worldwide.

ATMEA1 construction techniques like pre-installed steel frames, modularization, etc. are also experienced techniques in France and in Japan.

To minimize construction, operation and maintenance risks, the ATMEA1 reactor has been designed taking full benefits of the operating experience of structures, systems and components and methodologies of AREVA's and MHI's existing reactors. The latest requirements of generation III+ reactor and the evolution of techniques have been implemented in the design after an extensive and validated test program performed by AREVA and MHI for their larger size reactors.

1.3. Design features

ATMEA1 Reactor Coolant System (RCS) is a three-loop configuration.

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 a pre-stressed concrete type with a steel liner on the inner surface. 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 both 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 two redundant cooling trains and by a diversified cooling train.

A protection against large aircraft crash and explosions consisting of both a reinforced concrete shell and segregation is provided to guarantee the accomplishment of safety functions.

2. DESCRIPTION OF THE NUCLEAR SYSTEMS

2.1. Main characteristics of the primary circuit

The primary circuit of ATMEA1 is composed of the main components and systems 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 operating 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 minimize neutron irradiation embrittlement by reducing the RPV irradiation;
- to avoid stress corrosion cracking of baffle bolts and 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 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 (FAs)

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 to control excess reactivity.

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

The RCCAs of ATMEA1 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. The control rod has a neutron absorber encased in a stainless steel tube.

Soluble absorber is used to control the relatively slow reactivity changes mainly due to fuel depletion. In order to reduce boron concentration and to improve the primary water chemistry control, 10B 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) and a wider downcomer ensuring 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 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 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 alloy.

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 designed to ensure low neutron fluence to the RPV and to minimize inspection work and the risk of Stress Corrosion Cracking (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. These slabs are positioned one above the other, and axially restrained by keys and tie rods.

2.4.3. Steam generators (SGs)

The SGs are natural circulation vertical U-tube heat exchangers with integrated 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 side 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.

2.4.4. Pressurizer (PZR)

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 valve limits reactor coolant system pressure rises following less severe transients to prevent undesirable opening of the PZR safety valves.

2.4.5. Reactor coolant pumps (RCPs)

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 valves. 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.

For the primary system, the leak Before Break concept (LBB) is only applied to the design of the main coolant lines themselves.

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 (HP) charging pumps, the volume control tank (VCT), the regenerative heat exchanger, two high pressure coolers in parallel, two high pressure reducing stations 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 (CCWS)

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

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

- Containment Spray System (CSS)/Residual Heat Removal System (RHRS) heat exchangers and pump motors;
- Spent Fuel Pool (SFP) heat exchangers;
- RCP thermal barriers;
- HP coolers
-

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 ESWS from either UHS1 or UHS2, are of a different type. UHS 1 and UHS 2 are also of a different type.

2.5.3. Essential service water system (ESWS)

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.

ESWS trains 1, 2 and 3 take the essential service water from UHS 1, supply each component to be cooled, and discharge the essential service water back to UHS 1. Train X of the ESWS takes the essential service water from either UHS 1 or UHS 2 to supply each component being cooled, and discharges the essential service water back to either UHS 1 or 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 control modes and control channels allow for high flexibility in normal operation.

- For frequency control, variations of 15% in power (e.g. 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 is capable of even greater flexibility with instantaneous return to full power without notice (when selected) at a rate of 5% per minute. Load-follow operation can be superimposed with frequency control operation;
- ATMEA1 is also capable of fast power reductions shifting to house load operation when necessary.

2.7. Alternative fuel options

The ATMEA1 core design allows various fuel management schemes to be accommodated, from uranium to 1/3 MOX cores as a standard design, and to full MOX cores with minor design changes.

3. DESCRIPTION OF SAFETY CONCEPT

3.1. Safety concept, design philosophy and licensing approach

3.1.1. Design philosophy

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

IAEA Review

Under the auspices of the IAEA, an international team of senior experts reviewed the ATMEA1 design and evaluated it against applicable IAEA Safety Standards.

This review aimed at assessing whether the Conceptual Safety Features address IAEA requirements in a complete and comprehensive manner. The report concluded in the summer of 2008 that the ATMEA1 conceptual design addresses the IAEA's Fundamental Safety Principles as well as key design and safety assessment requirements. These safety features are the basis for the basic design work. The report also highlighted that the conceptual design of ATMEA1 showed proven design derived from the good experiences gained from operation of existing plants.

ASN Review

In mid-2010, the ATMEA Company initiated the review of ATMEA1 safety options and of the detailed design with the French Nuclear Safety Authority (ASN) in order to obtain ASN's position about their compliance with:

- the safety objectives as set out in France for the new generation of pressurized water reactors, as expressed in section A.1.1 of the "Technical Guidelines for the design and construction of the new generation of pressurized water nuclear reactors";
- the statutory and para-regulatory texts applicable to basic nuclear installations in France.

The ASN issued its final position on January 2012, which considered that "safety objectives adopted by the ATMEA company for the ATMEA1 reactor are satisfactory" and that "safety options and design choices adopted for the main equipment of the ATMEA1 reactor are globally satisfactory".

Diversifications implemented in the design (4th train, the 2nd ultimate heat sink, alternative power source, water sources, etc.) were also highly considered including in their role to address extreme external hazards.

CNSC Review

In mid-2011, ATMEA started a pre-licensing review with the Canadian Nuclear Safety Commission (CNSC). CNSC pre-licensing review of a Vendor Reactor Design is a prerequisite for selection by Canadian utilities showing that there is no outstanding issue for a subsequent licensing.

CNSC reviewed the compliance of ATMEA1 reactor design against CNSC regulatory document RD-337 "Design of New Nuclear Power Plants" and other applicable regulatory documents.

The CNSC issued its final position on June 2013, which concluded that, "ATMEA understood the CNSC regulatory requirements and expectations for the design of new NPPs in Canada" and that "ATMEA1 design intent is compliant with the CNSC regulatory requirements and meets the expectations for new NPPs in Canada".

CNSC also evaluated that ATMEA1 strategy on extreme external hazards and severe accidents are fully in line with Canadian expectations for a Generation III reactor.

3.2. Provision for simplicity and robustness of the design

Simplifications have been performed in all areas of design, construction, operation and maintenance to achieve a cost-effective design and to improve 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

The use of active and passive systems has been optimized for reliability and economic purposes.

ATMEA1 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 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 strongly limit the need for it.

For all safety analysis, the long term safe state is retained as the 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 deterministic and probabilistic approaches are used to assess the specific measures available for their management.

Level 3 of defense-in-depth has been reinforced by addressing large commercial airplane crash and extreme external hazards.

Even though the probability of severe accidents has been highly reduced by improving defense-in-depth's 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. Core damage frequency, large early release frequency and Autonomy

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 design. Probabilistic assessment figures shall be considered as indicators as it was recognized by Safety Authorities.

The Nuclear Island safety objectives are :

- 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 $< 10^{-5}/\text{r.y.}$

and

- The integral Frequency of “Large Releases” (LRF) is $< 10^{-6}/\text{r.y.}$

3.5.3. Autonomy

For the limitation of the consequences of accidents, design provisions ensure that no operator action will be required before a given period. This minimizes the risk of operator error during diagnostics or in operator actions, in accordance with general safety objectives.

In addition, achieving and maintaining safe shutdown state requires a heat sink and uninterruptible power sources. Design provisions are implemented to, on the one hand, ensure the adequate autonomy of these sources, by delaying the need to use mobile equipment as much as possible.

In view of improving the prevention of human errors and making the facility less vulnerable to these errors, reaction times are systematically considered in the safety demonstration. Operator action cannot be credited during this period. Any action required during this period is automated. Operator actions are not needed for at least 30 minutes from the main control room and 1 hour from the plant. Beyond this period, all operator action, including the use of mobile equipment, may be credited, within the limits of the periods of autonomy mentioned below.

The emergency feedwater tanks for the steam generators are adequate to maintain hot shutdown for a period more than or equal to 24 hours. Safeguard electrical supply and SBO supply each have an autonomy of more than or equal to 7 days.

Safety-class batteries have an autonomy of more than or equal to 2 hours. Dedicated severe accident batteries have an autonomy of more than or equal to 12 hours.

The autonomy of the main heat sink UHS1 is 30 days, in line with the requirements of RG 1.27. Standard ATMEA1 location is assumed to be near sea coast, in which case the UHS1 would be the sea which can be considered as an infinite cooling source. In addition, beyond the requirements of RG 1.27 on UHS1 autonomy, the autonomy of the diversified ultimate heat sink UHS 2 must be adequate to restore the main ultimate heat sink if the latter fails.

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

3.6.1. Safety system configuration

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

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

Advanced accumulators achieve functions of high flow injection during the blowdown phase just after a Loss of Coolant Accident (LOCA) and low flow 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 In-containment refueling water storage pit (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:

- to avoid switching 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 around 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 EFWS 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 a diversified Alternative AC Power Source (AAC).

One EFWS train is located in each division of the Safeguard Buildings (SAB), 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 /Residual heat removal system (CS/RHRS)

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

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

All three CS/RHRS trains are powered from separate emergency buses, each backed up by an EPS. Each CS/RHRS train is located in the associated division in SAB, 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 (beyond design basis event-BDBE).

3.7. Systems to mitigate severe accidents

A severe accident results in fuel rod failure, degradation of the structural integrity of the reactor core, and release of radioactive fission products into the reactor coolant system. Such an event can only occur as a result of a sustained loss of adequate core cooling, which leads to elevated core temperatures due to the sustained production of decay heat. Coupled with multiple failures of a plant's safety-grade emergency cooling systems, the resulting consequence of melting the reactor core (and internals) may lead to the breaching of the reactor vessel. The transfer of molten core material in the containment may potentially compromise the ability of the containment to perform its radionuclide retention function.

The ATMEA1 safety concept responds to advanced regulatory requirements, which require that the plant is designed to cope with a severe accident in such a way that the consequences are limited to the immediate vicinity and render stringent countermeasures unnecessary, such as evacuation or the relocation of the neighboring population as well as long term restrictions in the consumption of food.

This goal is achieved by a severe accident mitigation design which deliberately closely follows the design of the EPR by the following design measures:

- Depressurization of the RCS to reliably prevent high pressure vessel failure and induced steam generator piping rupture
- Avoidance of ex-vessel steam explosion and recriticality.
- Hydrogen control to prevent early containment failure and large fission product release.
- Melt retention to prevent basemat penetration.
- Containment heat removal to prevent late containment failure and increased fission product release. Figure 7 shows the redundant trains of the severe accident heat removal system.
- Monitoring of overall plant behavior (leak tightness), activity distribution within the containment and releases to the environment.

3.8. Provisions for safety under seismic conditions

For the standard 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 Asian countries. In the USA, this value is described in the top tier of the URD.

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 standard 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. Extreme external hazards

For a new NPP design, as a countermeasure against such extreme events, ATMEA1 strategy is to use installed equipment and resources to maintain or restore core cooling, containment cooling, and spent fuel cooling for a prolonged period of time.

The objective is, for extreme scenarios considering extreme external hazard conditions, including potential consequential long-term LUHS and loss of EPS power, as well as their relevant combinations:

- To prevent core melt in reactor vessel and to limit radiological consequences should a core melt occur in reactor vessel (defense-in-depth approach)
- To exclude fuel melt in fuel building

The extreme external hazards means are organized in two different levels of Defense in Depth:

- A Preventive Level of Defense, whose objective is to avoid Core Melt in the reactor and spent fuel uncovering and criticality in the Spent Fuel pool in case of extreme external hazards. This level of defense is ensured by preventive Extreme Hazards Protected (EHP) Equipment;
- A Mitigation Level of Defense, whose objective is to limit radiological consequences in the event of Core Melt in the reactor, and to avoid spent fuel uncovering and criticality in the Spent Fuel pool in case of extreme external hazards combined with the failure of Preventive line of Defense for SFP. This level of defense is ensured by mitigation EHP Equipment.

Most safety functions of NPPs depend on alternating current (AC) power, hence high reliability of AC power supply is essential. This high reliability is achieved through an adequate combination of redundancy and diversity. Ensuring adequate protection of the AC power supply against site-specific extreme external hazards is a lesson from the Fukushima Dai-ichi accident. Regarding emergency power supply, diverse, electrically adequately isolated AC power sources (AAC) is implemented as a part of defense-in-Depth concept of the plant.

The Defense-in-Depth approach needs to be applied also to the ultimate heat sink. The design of new nuclear power plants needs to provide diverse means to provide reactor and spent fuel cooling. The use of a secondary ultimate cooling water system (UHS2) is an acceptable diverse means to provide reactor and spent fuel cooling for decay heat removal in case of unavailability of the primary ultimate heat sink.

For both lines of defense, ATMEA1 baseline strategy is to credit the AAC, the UHS2 and EHP equipment and to protect them against site-specific extreme external hazards. Short, mid and long terms definition and additional means are adjusted to the site specificities, to country's requirements or to address investment protection purposes.

3.11. Emergency planning measures

ATMEA1 adopts stringent radiological consequences criteria in line with those used for reactors currently under construction in Europe.

These criteria ensure limited emergency response plan 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.

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 are country-specific. The following thresholds apply in France for actions to protect populations:

- an effective dose of 10 mSv: protect the population,
- an effective dose of 50 mSv: evacuate the population,
- an equivalent thyroid dose of 50 mSv: administer iodine tablets.

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) 50 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.

4. PROLIFERATION RESISTANCE

The ATMEA1 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 ATMEA1 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 less than 5% ²³⁵U 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 ²³⁵U 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.

5. 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.

6. DESCRIPTION OF TURBINE-GENERATOR SYSTEMS

ATMEA designs the ATMEA1NI 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.

7. 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 under voltage 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. In the event of an SBO, the AAC is manually started within 1 hour. AAC is then manually connected to the train-X bus bar to supply 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.

In order to ensure the continuity of the capabilities to bring the plant to safe state and maintain it, safety batteries supply the needed loads until AC power is restored.

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 errors and to be less demanding on the operator.

Emphasis is put during the design to secure the most ergonomic solution in terms of level of automations. An ergonomic analysis is performed at every stage of the design to ensure that the operator has the appropriate amount of information. Ergonomic design principles are also implemented in order to give operators the necessary time to properly manage all plant situations.

Operators have all the necessary information to manage all plant situations on the Human Machine Interface displays: Normal operation, Design Basis Events (DBEs) and Beyond Design Basis Events (BDBEs)

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 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. I&C Systems

7.3.1. General Description and Design Basis

The I&C systems:

- Control and monitor plant system functions during normal operations;
- Control plant functions that initiate corrective measures in the event of deviation from LCO;
- Control limitation functions that initiate corrective measures;
- Control protection functions that mitigate the consequences of a DBE up to the controlled state;
- Control and monitor post-accident functions that mitigate the consequences of a DBE, and to bring and maintain the plant from the controlled state to the safe shutdown state;
- Control functions that mitigate the consequences of BDBEs, including severe accidents.

7.3.2. Diversity and Defense in Depth Concept

The general I&C architecture is designed in accordance with the concept of diversity and defense-in-depth.

Instrumentation and control systems are combined into three defense levels:

- the level corresponding to normal operation and to incident-prevention functions (including the RCML and the PAS),
- main level of defense (including RPS and SAS),

- the risk reduction level (including the DAS and the SA I&C dedicated to manage severe accidents).

The application of this concept requires independence between the different levels.

Figure 5 shows the positioning of each of the instrumentation and control systems in each of the defense levels.

The I&C system used as human-machine interface are designed to allow the operator to monitor and control the unit in all circumstances, and are assigned to all defense levels.

The functions used in normal operating conditions are guaranteed by I&C systems assigned to preventive line of defense.

The Main Line of Defense can be used to achieve and maintain safe shutdown state after a DBE 2 – 4 condition event.

The Risk Reduction Line can be used to provide additional resources in order to:

- limit the consequences of credible common cause failures for main level I&C systems, when they are combined with a DBE 2 – 4 condition (DEC-A sequence),
- limit the consequences of a severe accident with core meltdown (DEC-B sequence).

The safety I&C system of the Main Line of Defense are located in dedicated cabinets in the nuclear island. The RPS system has four logic channels. Depending on the system design and requirement the protective each function is implemented using 3 or 4 channels logics. Each of Trains A, B, C and X is installed in a dedicated independent room associated to different fire zones and with independent safety HVACs. Each of them has its own safety-related power source. Additionally, each cabinet has redundant power supplies.

8. 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 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 provides for an on-site storage of the spent fuel at the spent fuel pool for 10 years of full power operation. Beyond this period, plant operators decide whether to store the spent fuel assemblies on site or to ship them to an interim storage facility or 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

General safety objectives have been defined relative to the protection of the public and the environment. These objectives lead to the minimization of liquid and gaseous effluent discharges and to the minimization of the volume of solid waste produced, as compared with the nuclear facilities currently in use.

The process to minimize effluents and waste, leads to:

- the identification of the sources of production of these effluents and waste,
- the implementation of design provisions aiming to limit the production and discharge of effluents (collection, processing, storage and decay equipment before discharge) and the production of waste.
- the implementation of resources to monitor the discharge of effluents (periodic and continuous measuring equipment in the environment).

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.

9. 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 Essential Service Water related buildings and galleries.

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

9.1.1. External and internal hazards

Internal hazards are events originating on the plant site with the potential to cause adverse conditions or even damages inside or on safety classified buildings (e.g., internal fires, flooding, missiles or explosions). Such effects can potentially lead to common cause failures within systems needed to reach or maintain the plant in a safe state.

Within buildings containing safety systems, two main cases are considered depending on the layout type of building where the internal hazard occurs. These are:

- buildings which are split into divisions, e.g. safeguard buildings, diesel buildings,
- buildings or parts of them which are not split into divisions, e.g. containment.

The protection against internal hazards is taken into account in the design of the plant through layout requirements (mainly spatial and/or structural separation of individual redundancies of safety related systems from each other and from non safety related system as far as reasonable achievable) and/or design against loads. These requirements are adapted to the two buildings of different type as described above.

External hazards are considered as load cases. Protection relies upon both the deterministic dimensioning of the structures and equipment with respect to reference load cases and, for some hazards affecting only part of the plant, on segregation of systems and components.

The robustness of the design against hazards is a combination of the seismic design with a large common basemat with specific layout in the inner structures of the buildings, and of the large commercial airplane crash protections.

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 auxiliary systems. The RB consists of the Containment Building (also called the Pre-stressed Concrete Containment Vessel (PCCV)) and its internal structure surrounded, in the lower part, by the annulus space and an outer wall of reinforced concrete. See Figure 4.

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 basement. 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 retains all radioactive material and withstands 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.

10. PLANT PERFORMANCE

10.1. Plant operation

The reference configuration (nominal power) for ATMEA1 is:

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

The standard ATMEA1 core design allows various fuel management schemes from full uranium to 1/3 MOX cores with 12 to 24 month operational cycles.

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

10.2. Reliability

ATMEA1 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, MHI's APWR and some currently operating reactors. The fact that the ATMEA1 design is derived from these 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'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 and a high thermal efficiency of up to 37% leads to lower fuel consumption and waste generation. 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's target schedule is 48 months from first concrete to a fuel loading.

10.6. Provisions for low fuel reload costs

The ATMEA1'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 design allows various fuel management schemes to be accommodated, from uranium to 1/3 MOX cores and from 12 to 24 month operational cycles with optimized fuel cost.

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

11. DEPLOYMENT STATUS OF TECHNOLOGIES RELEVANT TO THE NPP

The development of the ATMEA1 model as the standard design includes various design phases and engineering activities under several assumptions as the standard conditions. This process is broken down into three phases:

- "conceptual design" phase,
- "basic design" phase,
- "generic detailed design" phase.

The development of a reactor generally continues with a "site specific detailed design" phase, which contains all site specific design and design adaptations.

During the standard design development, design codes and standards remain the NRC regulation complemented by US codes and standards, provided that European Regulations and code requirements will be evaluated to ensure that ATMEA1 reactor design can simply and easily be adapted to the European requirements if a customer mandates so.

Conceptual design phase

The objectives of the conceptual design phase are described below.

- Define the global concept behind the model in terms of:
 - safety and regulatory or similar requirements,
 - cost,
 - performance and availability,
 - operating limitations,
 - construction limitations,
 - interfaces with the electrical system.
- Define the main characteristics of the model and margins to be taken into consideration.

This includes all design bases, that is the definition of safety requirements and structural technical decisions, such as the general configuration of systems, the characteristics of the main items of equipment, the block plan, the organization of the main nuclear island buildings, the general architecture of electrical and instrumentation and control systems, the organization of engineering activities, the definition of project procedures and methods and tools.

- Define the structure and organization of the basic design phase.

The main characteristics of the model, as defined following the conceptual design phase, are:

- A net electrical capacity of 1100 MWe class, and high levels of availability and efficiency,
- Highest level of safety as a Generation III+ reactor also considering the latest regulatory trends like APC Protection, severe accident mitigation, etc.
- extensive operating flexibility: 12 - 24 month cycles and load following functions,
- use of active and passive safety systems with three trains,
- digital instrumentation and control systems equipped with an advanced human-machine interface,

- adaptability to a wide range of sites, including areas with high levels of seismic activity and adaptability to 50 or 60 Hz grids.

The conceptual design of ATMEA1 has been completed in 2007.

Basic design phase

The scope of the basic design phase covered the nuclear island. It mainly covered the following systems:

- the nuclear steam supply system (the secondary systems (feedwater and steam production) are limited to the geographical limits of the nuclear island buildings),
- the main nuclear auxiliaries,
- the main nuclear island ventilation systems,
- the panels of the operational and safety electrical systems for the nuclear island, including batteries,
- the instrumentation and control systems of the nuclear island.

The main structures covered were:

- the reactor building,
- the fuel building,
- the safeguard auxiliary building,
- the nuclear auxiliary building,
- the emergency power supply buildings,
- the waste building.

The following systems were excluded from the scope of the basic design phase:

- the turbine hall and the rest of the conventional island, including the sections of secondary systems (feedwater and steam production) located outside of the buildings of the nuclear island,
- the solid waste treatment systems:
- the normal electrical supply systems,
- Ultimate heat sinks (UHS, main and diversified),
- instrumentation and control systems not connected to nuclear island systems.

The basic design of ATMEA1 had been completed at the end of 2009 with the issuance of the Preliminary Safety Analysis Report.

Generic Detailed Design Phase

The Generic Detailed Design (GDD) is a detailed design phase of the standard model in continuation of the Basic Design Phase. As an example, all the small bore piping is implemented in the 3D Mockup in GDD phase. The objective of the Generic Detailed Design, launched in January 2011, was to complete the design and to complete some of the confirmatory analysis to reduce uncertainties. In addition, the objective of the Generic Detailed Design was to minimize Specific Detailed Design and construction schedule risk.

The purpose of the Generic Detailed Design phase was therefore to:

- allow the marketing and bidding of ATMEA1 Nuclear Island product based on detailed enough design information with a limited level of uncertainty;

- study assurance of the licensibility of the product in the targeted markets by ensuring its consistency with worldwide known requirements.

The Generic Detailed Design was planned without specifying some of the major technologies such as the manufacturer of the primary equipments, the fuel manufacturer nor the I&C platform. In these areas, design has been detailed as much as possible without these choices. This will provide the flexibility to select those technologies at a later date after their further analysis.

The same design assumptions used in the Basic Design Phase continued to apply for the Generic Detailed Design with some additional design scope excluded in Basic Design, e.g. standard site waste treatment concept, standard UHS..

The Generic Detailed Design of ATMEA1 has been completed in April 2015 with the issuance of the revision of the Preliminary Safety Analysis Report, the component data sheet and the documentation for procurement.

APPENDIX: Summarized ATMEA1 Technical Data (ATMEA1)

General plant data		
Reactor thermal output	3150	MWth
Power plant output, gross		MWe
Power plant output, net	1100-1150	MWe
Power plant efficiency, net	35-37	%
Mode of operation	Baseload and load follow + frequency control	
Plant design life	60	Years
Plant availability target	> 92	%
Seismic design, SSE	0.3g	
Primary Coolant material	Light water	
Moderator material	Light water	
Safety goals		
Core damage frequency	< 10 ⁻⁶	/RY
Large early release frequency	< 10 ⁻⁷	/RY
Occupational radiation exposure	< 0.5	Person-Sv/Ry
Nuclear steam supply system		
Steam pressure	7.2 (BE at SG outlet)	MPa(a)
Reactor coolant system		
Primary coolant flow rate	24800 (BE)	m ³ /h/loop
Reactor operating pressure	15.5	MPa(a)
Reactor vessel inlet temperature	291 (BE)	°C
Reactor vessel outlet temperature	326 (BE)	°C
Reactor core		
Fuel material	UO ₂ or MOX	
Cladding tube material	Zirconium based alloy	
Rod array of a fuel assembly	17x17	
Number of fuel assemblies	157	
Enrichment of reload fuel at equilibrium core	< 5.0	Wt%
Fuel cycle length	12 to 24	months
Burnable absorber (strategy/material)	Gd ₂ O ₃	kg
Control rod absorber material	AIC/B ₄ C	
Soluble neutron absorber	Enriched H ₃ BO ₃	
Reactor pressure vessel		
Inner diameter of cylindrical shell	4250	mm
Base material	Low alloy steel	
Steam generator		
Type	U tubes with axial economizer	
Number	3	
Total tube outside surface area	8000	m ²
Tube outside diameter	19	mm
Tube material	Alloy 690	

Reactor coolant pump		
Type	Shaft seals	
Number	3	
Flow at rated conditions	24800	m ³ /h
Pressurizer		
Total volume	65	m ³
Primary containment		
Type	Pre-stressed concrete with steel liner	
Overall form (spherical/cylindrical)	Cylindrical	
Residual heat removal systems		
Active/passive systems	Active	
Safety injection systems		
Active/passive systems	Active pumps and passive advanced accumulators	

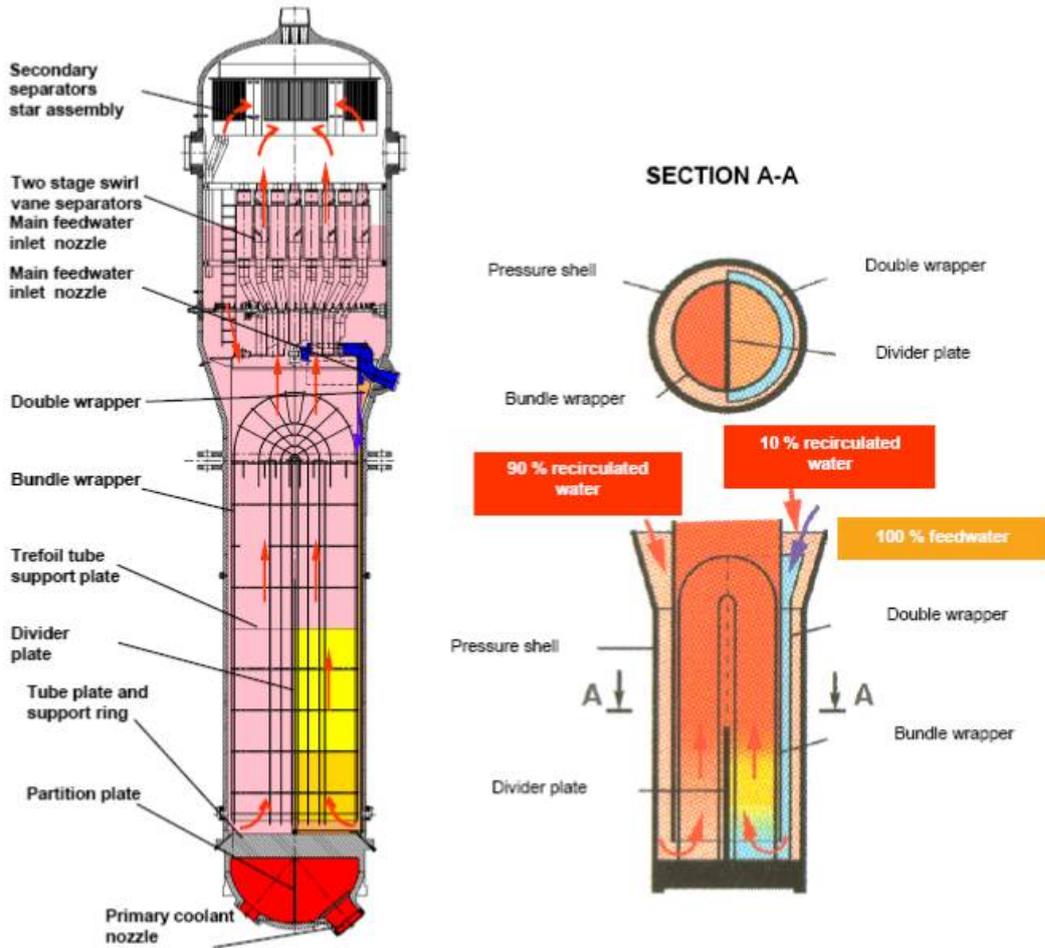


FIGURE 1: STEAM GENERATOR

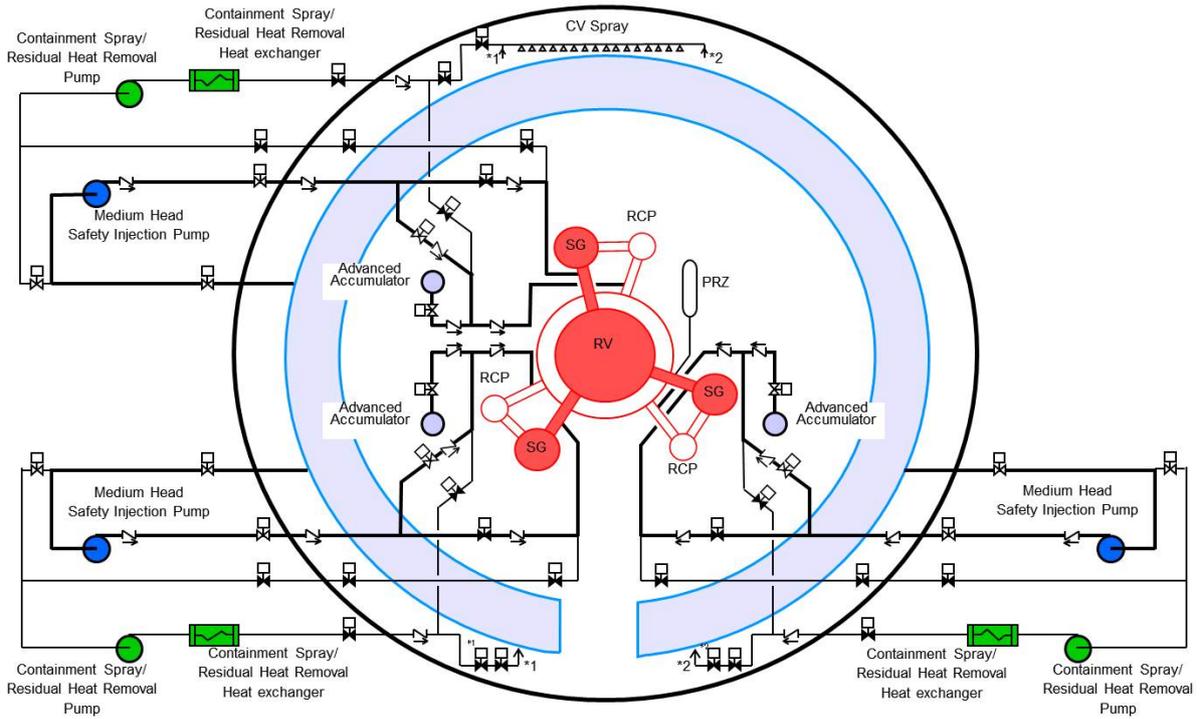


FIGURE 2: GENERAL ARRANGEMENT OF THE SAFETY SYSTEMS

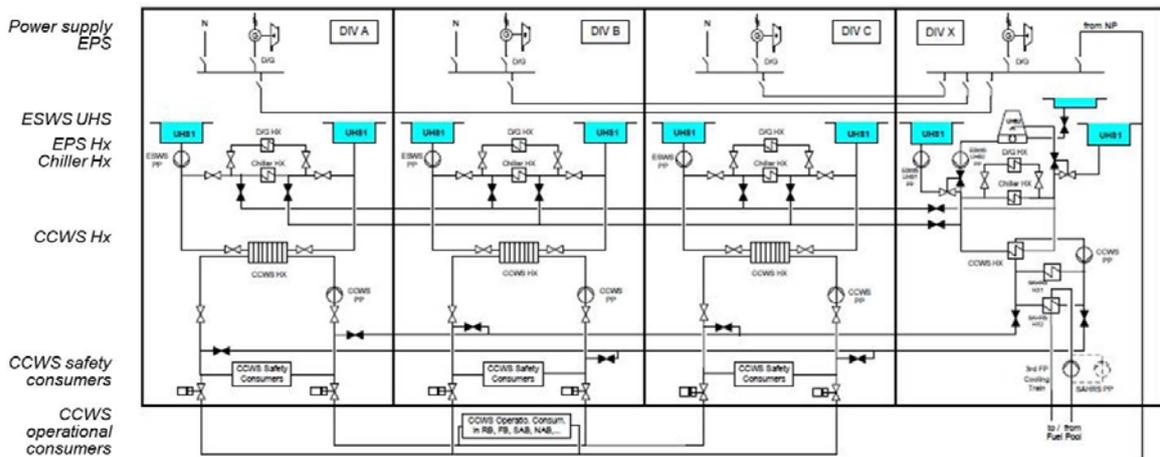


FIGURE 3: GENERAL ARRANGEMENT OF THE COOLING CHAIN SYSTEMS

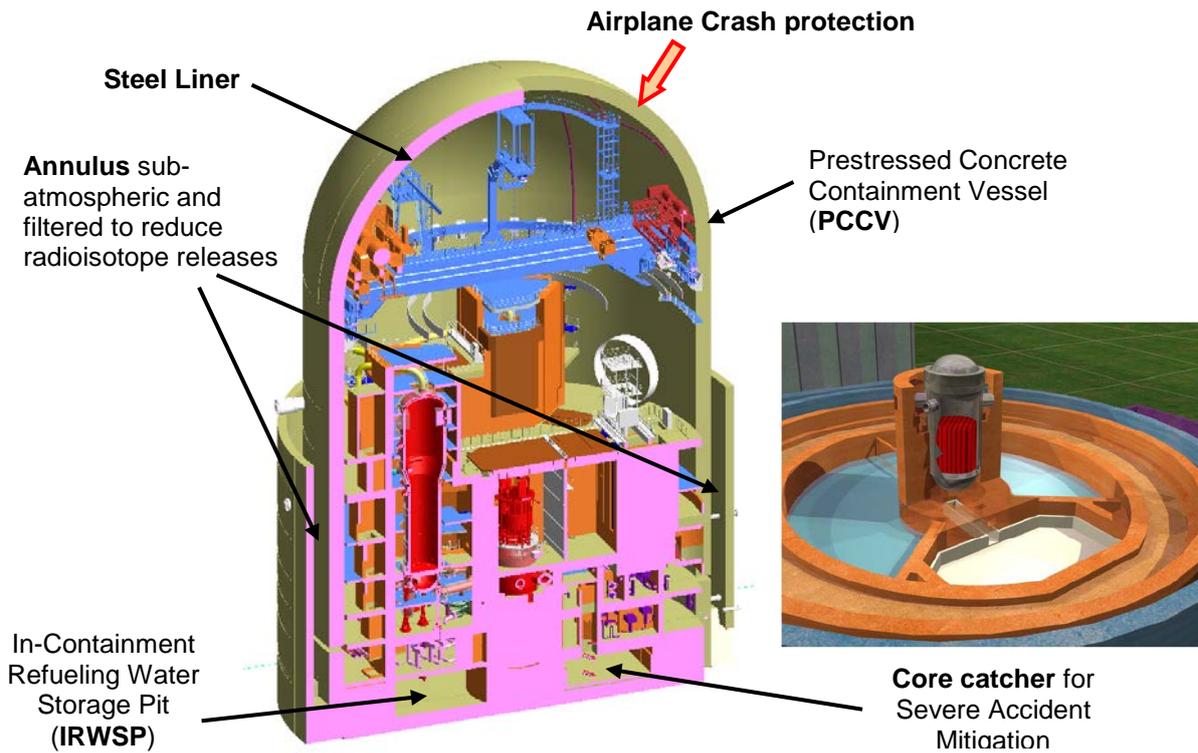


FIGURE 4: CONTAINMENT - GENERAL FEATURES

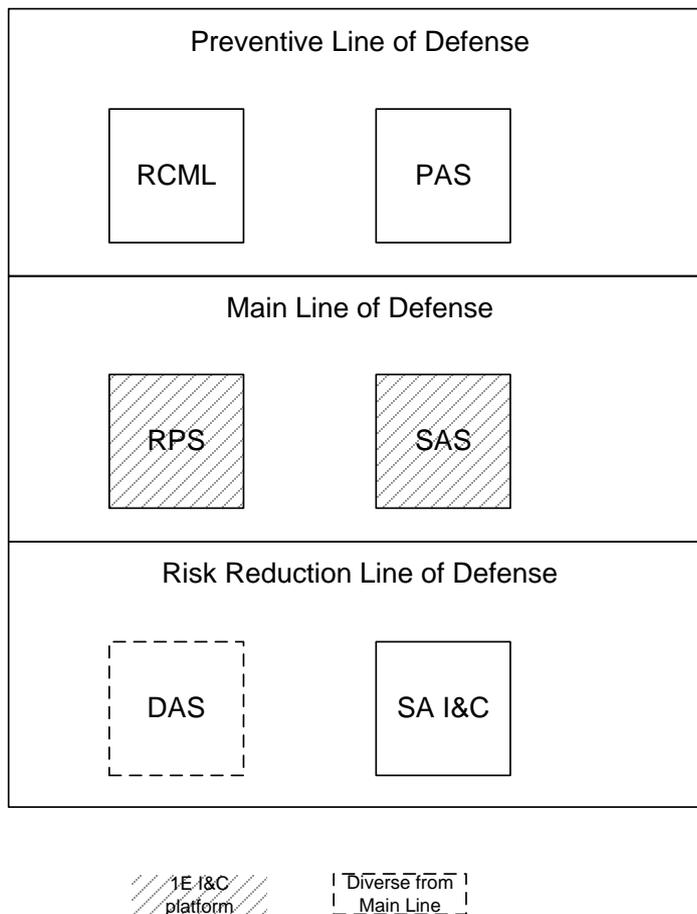


FIGURE 5: DEFENSE LEVELS AND INSTRUMENTATION AND CONTROL SYSTEMS

