EPR

AREVA

> Readers accustomed to British units can use the following table to convert the main units from the International Metric System.

1 meter (m)	=	3.2808 feet
	=	39.370 inches
1 square meter (m^2)	=	10.764 square feet
1 cubic meter (m^3)	=	264.17 US gallons
1 kilogram (kg)	=	2.2046 pounds
1 tonne (t)	=	1.1023 short ton
1 bar	=	14.5 psi

> Conversion of temperature (°C into °F)

Temp. °C x 9/5 + 32	=	Temp. °F

> All pressures are expressed in absolute bar.

> FOREWORD

Security of energy supply and energy cost stability in the long term, plus the efforts to combat the greenhouse effect and potential global warming, argue in favor of a greater diversity in sources of energy supplies. Against this background nuclear power, which is more and more economically competitive, safe, reliable and environment friendly, has a vital role to play.

A world expert in energy, AREVA creates and offers solutions to generate, transmit and distribute electricity; its businesses cover on a long-term basis every sector in the use of nuclear power to support electricity needs: front end (Uranium ore mining and conversion, Uranium enrichment, fuel fabrication), reactor design and construction, reactor services, back end of the fuel cycle, transmission and distribution from the generator to the large end-users.

The EPR is a large advanced evolutionary reactor of the Pressurized Water Reactor (PWR) type offered by AREVA to satisfy electricity companies' needs for a new generation of nuclear power plants even more competitive and safer while contributing to sustainable development.

The EPR's key assets to support a strategic choice

An evolutionary, safe and innovative design

The EPR is a 1,600 MWe class PWR. Its evolutionary design is based on experience from several thousand reactor - years of operation of Light Water Reactors worldwide, primarily those incorporating the most recent technologies: the N4 and KONVOI reactors currently in operation in France and Germany respectively. The EPR design integrates the results of decades of research and development programs, in particular those carried out by the CEA (French Atomic Energy Commission) and the German Karlsruhe research center. Through its N4 and KONVOI filiation, the EPR totally benefits from the uninterrupted evolutionary and innovation process which has continuously supported the development of the PWR since its introduction in the Western marketplace in the mid-fifties.

Offering a significantly enhanced level of safety, the EPR features major innovations, especially in further preventing core meltdown and mitigating its potential consequences. The EPR design also benefits from outstanding resistance to external hazards, including military or large commercial airplane crash and earthquake. Together, the EPR operating and safety systems provide progressive responses commensurate with any abnormal occurrences.

Thanks to a number of technological advances, the EPR is at the forefront of nuclear power plants design. Significant progress has been incorporated into its main features:

- the reactor core and its flexibility in terms of fuel management,
- the reactor protection system,
- the instrumentation and control (I & C) system, the operator friendly man-machine interface and fully computerized control room of the plant,
- the large components such as the reactor pressure vessel and its internal structures, steam generators and primary coolant pumps.

These innovations contribute to the high level of performance, efficiency, operability and therefore economic competitiveness offered by the EPR to fully satisfy customers' expectations for their future nuclear power plants.

The straightforward answer to utilities' and safety authorities' requirements for new nuclear power plants

The French-German cooperation set up to develop the EPR brought together, from the start of the project:

- power plant vendors, Framatome and Siemens KWU (whose nuclear activities have since been merged to form Framatome ANP, now an AREVA and Siemens company),
- EDF (Electricité de France), and the major German utilities now merged to become E.ON, EnBW and RWE Power,
- the safety authorities from both countries to harmonize safety regulations.

The EPR design takes into account the expectations of utilities as stated by the "European Utility Requirements" (EUR) and the "Utility Requirements Document" (URD) issued by the US Electric Power Research Institute (EPRI). It complies with the recommendations (1993) and positions on major issues (1995) that the French and German safety authorities jointly set up. The technical guidelines covering the EPR design were validated in October 2000 by the French standing group of experts in charge of reactor safety (*"Groupe Permanent Réacteurs"* which is the advisory committee for reactor safety to the French safety authority) supported by German experts.

On September 28, 2004, the French safety authority, on behalf of the French government, officially stated that the EPR safety options comply with the safety enhancement objectives established for new nuclear reactors.

Continuity in technology

The N4 and KONVOI reactors are children of the earlier Framatome and Siemens KWU generation reactors which are themselves derivative of standard US type PWRs, first implemented in the US, then refined and expanded upon by Framatome and Siemens KWU. The EPR is the direct descendant of the well proven N4 and KONVOI reactors, guaranteeing a fully mastered technology. As a result, risks linked to design, licensing, construction and operation of the EPR are minimized, providing a unique certainty to EPR customers.

Operator expertise acquired through the operation of nuclear power plants using the same technology as the EPR is maintained and its value is increased.

Another major advantage is that the existing industrial capacities for design, engineering, equipment manufacturing, nuclear power plant construction and maintenance – including capacities resulting from previous technology transfers – can be easily deployed and utilized to carry out new nuclear plant projects based on EPR technology.

- ➡ **The EPR relies on a sound and proven technology.**
- ➡ **It complies with safety authorities requirements for new nuclear plants.**
- ➡ **Design and licensing, construction and commissioning, operability and maintainability of EPR units benefit from Framatome ANP long lasting and worldwide experience and expertise. Therefore, EPR customers uniquely minimize their technical risks and associated financial impacts.**

Enhanced economic competitiveness

The next generation of nuclear power plants will have to be even more competitive to successfully cope with deregulated electricity markets.

Thanks to an early focus on economic competitiveness during its design process, the EPR offers significantly reduced power generation costs. They are estimated to be 10% lower than those of the most modern nuclear units currently in operation, and more than 20% less than those of the largest high-efficiency advanced combined-cycle gas plants currently under development (taking into account a gas price in the US$* 3.5 per MBtu range). The advantage over fossil plants is even more pronounced when the "external costs" (such as costs related to the damage to environment and human health) are taken into account.

* In 2001 US$.

This high level of competitiveness is achieved through:

- ➡ **a unit power in the 1,600 MWe range (the highest unit power to date), providing an attractive cost of the installed kWe,**
- ➡ **a 36-37% overall efficiency depending on site conditions (presently the highest value ever for water reactors),**
- ➡ **a shortened construction time relying on experience feedback and continuous improvement of construction methodology and tasks sequencing,**
- ➡ **a design for a 60-year service life,**
- ➡ **an enhanced and more flexible fuel utilization,**
- ➡ **an availability factor up to 92%, on average, during the entire service life of the plant, obtained through long irradiation cycles, shorter refueling outages and in-operation maintenance.**

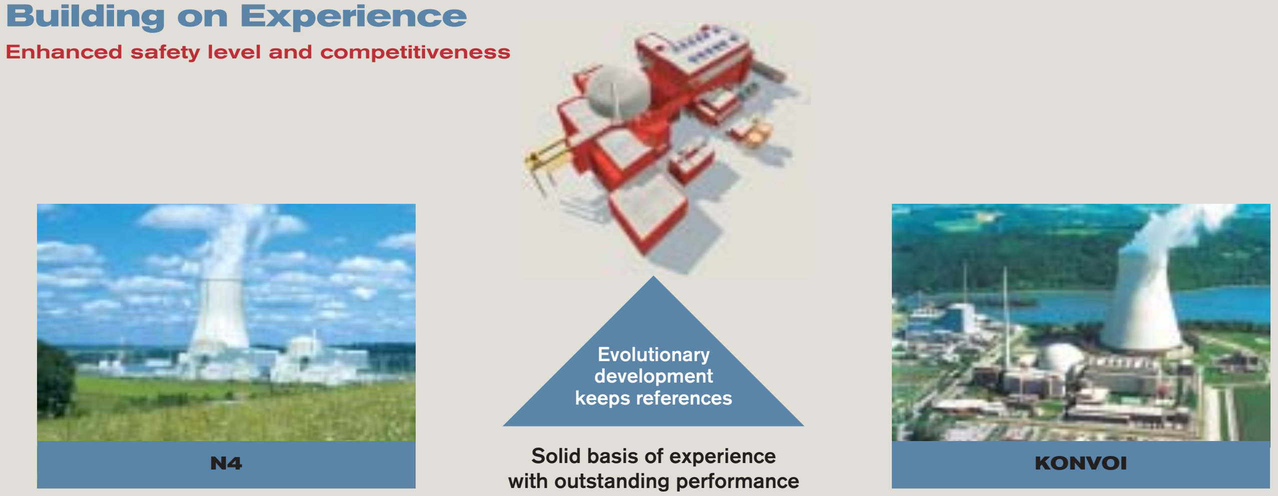
Significant advances for sustainable development

The EPR, due to its optimized core design and higher overall efficiency compared to the reactors in operation today, also offers many significant advantages in favor of sustainable development, typically:

- **17% saving on Uranium consumption per produced MWh,**
- **15% reduction on long-lived actinides generation per MWh,**
- **14% gain on the "electricity generation" versus "thermal release" ratio (compared to 1,000 MWe-class reactors),**
- **great flexibility to use MOX (mixed UO_2-PuO_2) fuel.**

> Building on Experience

Enhanced safety level and competitiveness



Solid basis of experience with outstanding performance

> INTRODUCTION

In a nuclear power plant, the reactor is the part of the facility in which the heat, necessary to produce steam, is generated by fission of atom nuclei. The produced steam drives a turbine generator, which generates electricity. The nuclear steam supply system is therefore the counterpart of coal, gas or oil-fired boilers of fossil-fuelled plants.

In a Pressurized Water Reactor (PWR) like the EPR, ordinary water is utilized to remove the heat formed inside the reactor core by the nuclear fission phenomenon. This water also slows down (or moderates) neutrons (constituents of atom nuclei that are released in the nuclear fission process). Slowing down neutrons is necessary to keep the chain reaction going (neutrons have to be moderated to be able to break down the fissile atom nuclei).

The heat produced inside the reactor core is transferred to the turbine through the steam generators. From the reactor core coolant circuit (primary circuit) to the steam circuit used to feed the turbine (secondary circuit), only heat is transferred and there is no water exchange.

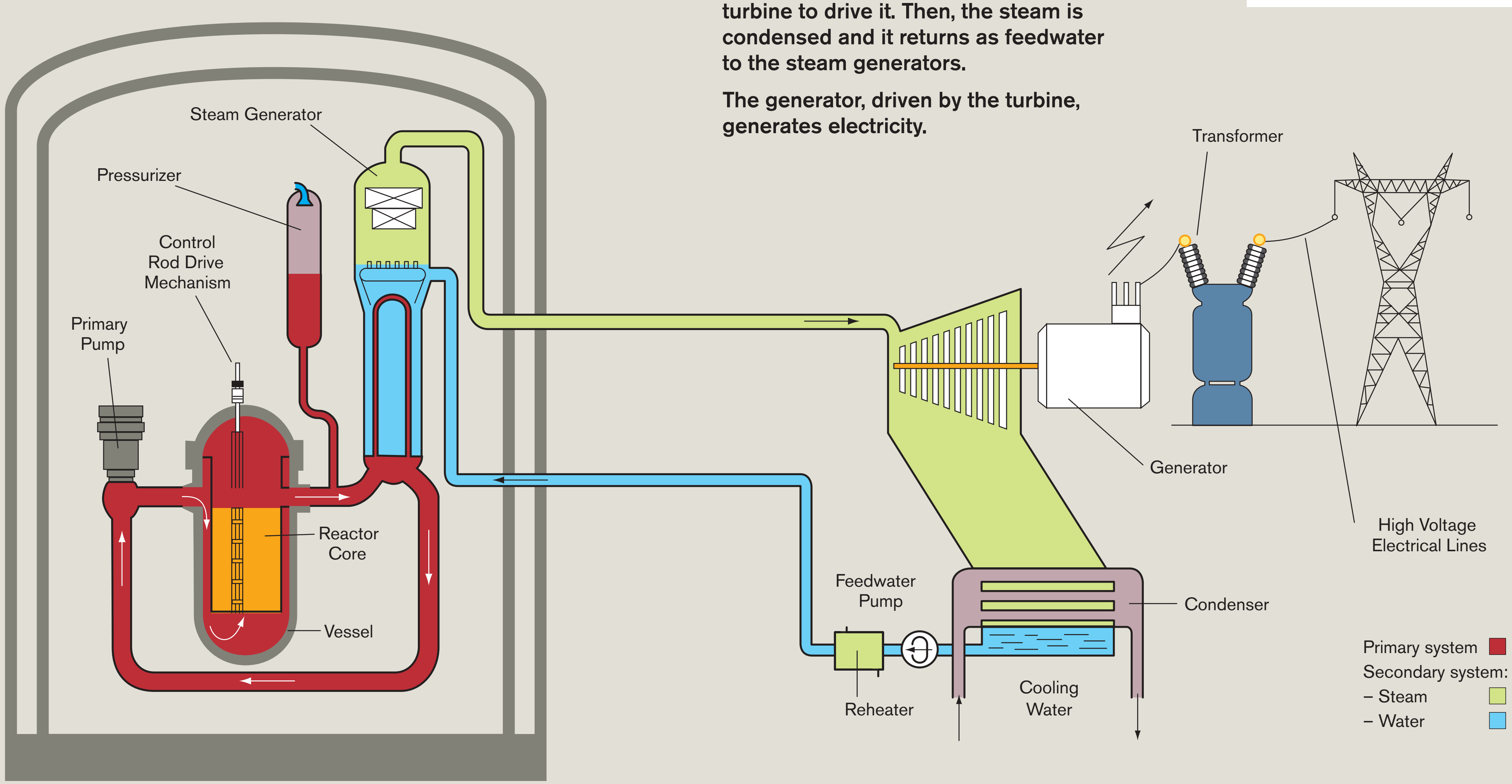
The primary water is pumped through the reactor core and the primary side of the steam generators, in four parallel closed loops, by electric motor-powered coolant pumps. Each loop is equipped with a steam generator and a coolant pump.

The reactor operating pressure and temperature are such that the cooling water does not evaporate and remains in the liquid state, which intensifies its cooling efficiency. A pressurizer controls the pressure; it is connected to one of the loops.

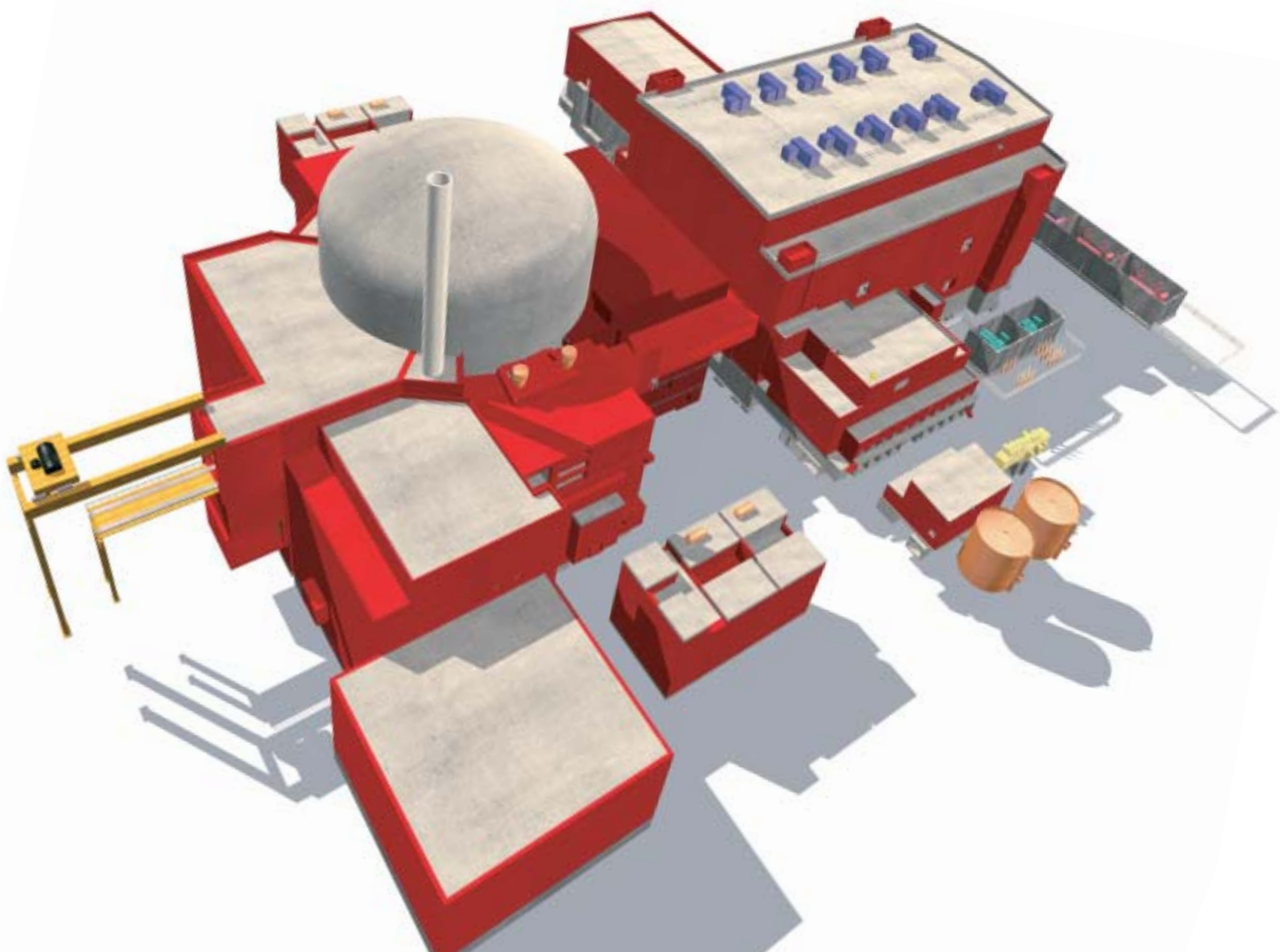
The feedwater entering the secondary side of the steam generators absorbs the heat transferred from the primary side and evaporates to produce saturated steam. The steam is dried in the steam generators then routed to the turbine to drive it. Then, the steam is condensed and it returns as feedwater to the steam generators.

The generator, driven by the turbine, generates electricity.

➡ The following chapters will provide detailed explanation about the description and operation of PWR nuclear power stations based on the EPR reactor.



> TABLE OF CONTENTS



EPR NUCLEAR ISLAND

Civaux nuclear power plant, France
(N4, 1,500 MWe)

EPR LAYOUT

1 Reactor Building

The Reactor Building located in the center of the Nuclear Island houses the main equipment of the Nuclear Steam Supply System (NSSS) and the In-Containment Refueling Water Storage Tank (IRWST). Its main function is to ensure protection of the environment against internal and external hazards consequences under all circumstances. It consists of a cylindrical pre-stressed inner containment with a metallic liner surrounded by an outer reinforced concrete shell.

The main steam and feedwater valves are housed in dedicated reinforced concrete compartments adjacent to the Reactor Building.

The primary system arrangement is characterized by:

- pressurizer located in a separate area,
- concrete walls between the loops and between the hot and cold legs of each loop,
- concrete wall (secondary shield wall) around the primary system to protect the containment from missiles and to reduce the spread of radiation from the primary system to the surrounding areas.

2 Fuel Building

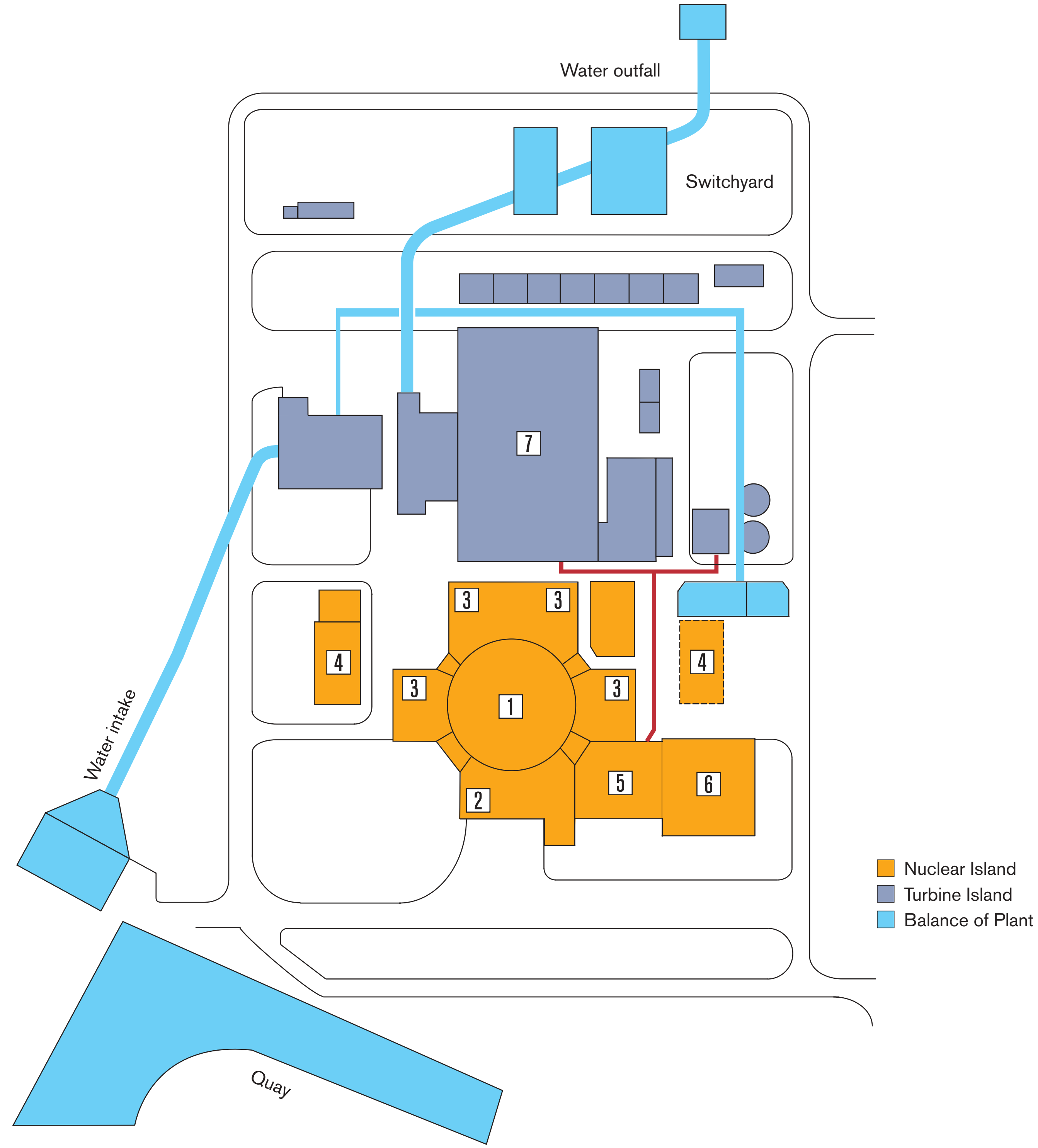
The Fuel Building, located on the same common basemat as the Reactor Building and the Safeguard Buildings, houses the fresh fuel, the spent fuel in an interim fuel storage pool and associated handling equipment. Operating compartments and passageways, equipment compartments, valve compartments and the connecting pipe ducts are separated within the building. Areas of high activity are separated from areas of low activity by means of shielding facilities. The mechanical floor houses the fuel pool cooling system, the emergency boration system, and the chemical and volume control system. The redundant trains of these systems are physically separated by a wall into two building parts.

3 The Safeguard Buildings

The four Safeguard Buildings house the safeguard systems such as the Safety Injection System and the Emergency Feedwater System, and their support systems. The four different trains of these safeguard systems are housed in four separate divisions, each located in one of the four Safeguard Buildings.
The Low Head Safety Injection System is combined with the Residual Heat Removal System. They are arranged at the inner areas in the radiologically controlled areas, whereas the corresponding Component Cooling and Emergency Feedwater Systems are installed at the outer areas in the classified non-controlled areas. The Main Control Room is located in one of the Safeguard Buildings.

4 Diesel Buildings

The two Diesel Buildings shelter the four emergency Diesel generators and their support systems, and supply electricity to the safeguard trains in the event of a complete loss of electrical power. The physical separation of these two buildings provides additional protection.



5 Nuclear Auxiliary Building

Part of the Nuclear Auxiliary Building (NAB) is designed as a radiological non-controlled area in which parts of the Operational Chilled Water System are located. Special laboratories for sampling systems are located at the lowest level. The maintenance area and some setdown areas used during the refueling phase are arranged on the highest level. All air-exhausts from the radiological controlled areas are routed, collected and controlled within the Nuclear Auxiliary Building prior to release through the stack.

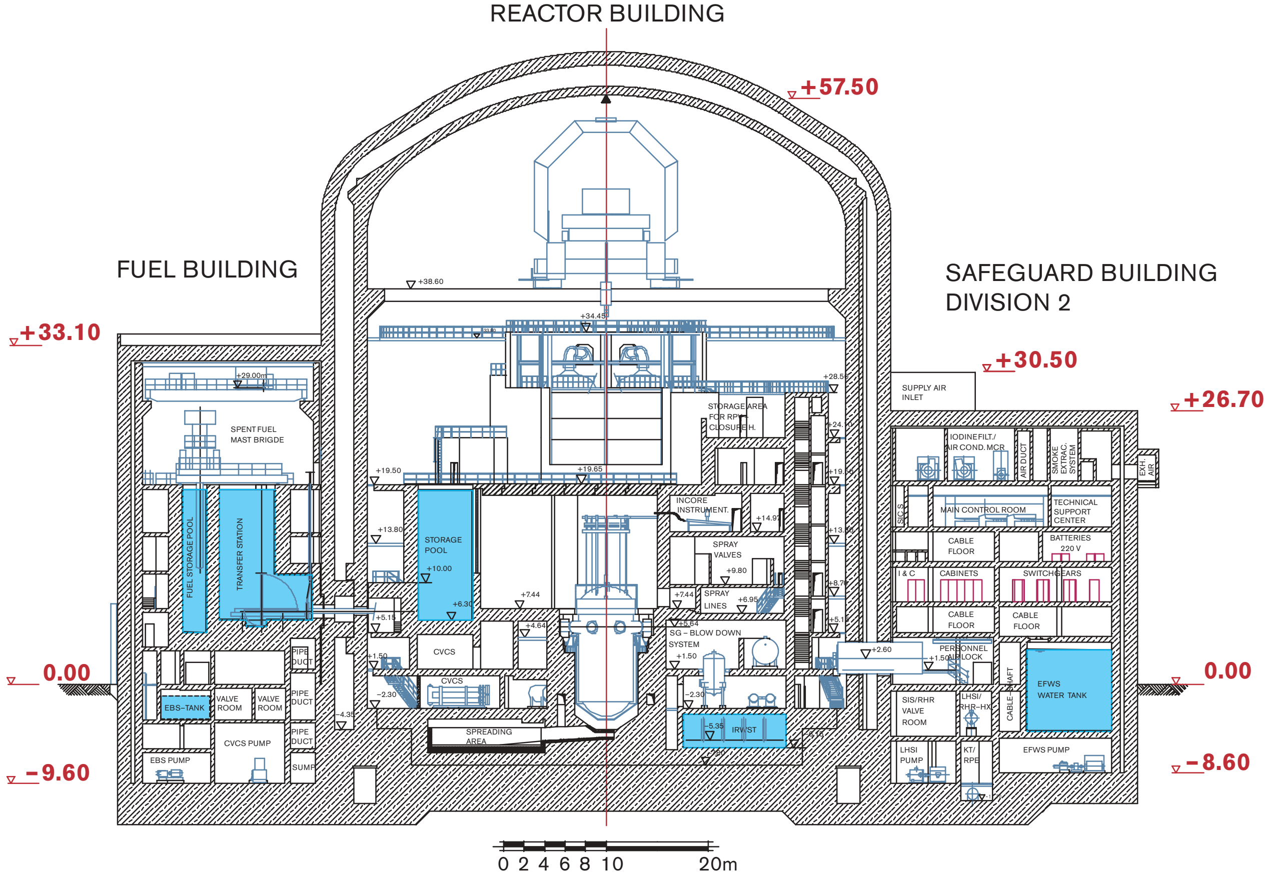
6 Waste Building

The Waste Building is used to collect, store and treat liquid and solid radioactive waste.

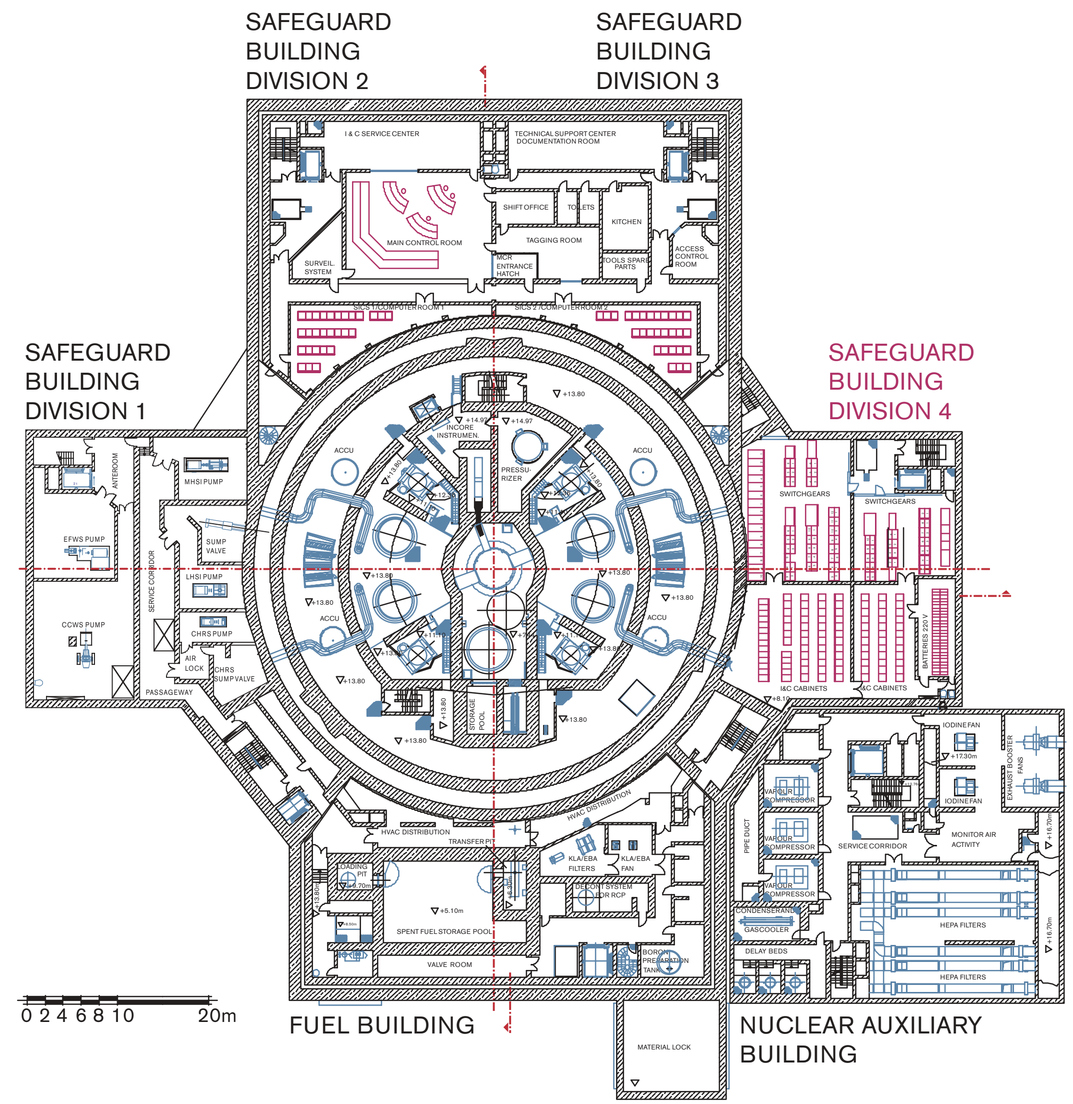
7 Turbine Building

The Turbine Building houses all the main components of the steam-condensate-feedwater cycle. It contains, in particular, the turbine, the generator set, the condenser and their auxiliary systems.

Nuclear Island building arrangement



Miscellaneous plan view



The outer shell (in blue in the image) protects the Reactor Building, the Spent Fuel Building and two of the four Safeguard Buildings including the control room.

➡ **The EPR layout offers exceptional and unique resistance to external hazards, especially earthquake and airplane crash.**

- **To withstand major earthquake, the entire Nuclear Island stands on a single thick reinforced concrete basemat. Building height has been minimized and heavy components and water tanks are located at the lowest possible level.**
- **To withstand large airplane crash, the Reactor Building, Spent Fuel Building and two of the four Safeguard Buildings are protected by an outer shell made of reinforced concrete. The other two Safeguard Buildings are protected by a geographical separation. Similarly, the Diesel generators are located in two geographically separate buildings to avoid common failures.**

➡ **The EPR Nuclear Island design has undisputed advantages for operators, especially where radiation protection and ease of maintenance are concerned.**

- **The layout is optimized and based on the strict separation of redundant systems.**
- **The distinction between access-controlled areas containing radioactive equipment and non-controlled areas significantly contributes to reduce exposure of the operating personnel.**
- **Maintenance requirements were systematically taken into account at the earliest stage of the design. For example, large setdown areas have been designed to make maintenance operations easier for operating personnel.**

PRIMARY SYSTEM

PRIMARY SYSTEM CONFIGURATION

The EPR primary system is of a well proven 4-loop design. French 1,300 MWe and 1,500 MWe N4 reactors as well as German KONVOI reactors are also of 4-loop design.

In each of the four loops, the primary coolant leaving the reactor pressure vessel through an outlet nozzle goes to a steam generator – the steam generator transfers heat to the secondary circuit –, then the coolant goes to a reactor coolant pump before returning to the reactor pressure vessel through an inlet nozzle. Inside the reactor pressure vessel, the primary coolant is first guided downward outside the core periphery, then it is channeled upward through the core, where it receives heat generated by the nuclear fuel.

A pressurizer, part of the primary system, is connected to one of the four loops. In normal operation, its main role is to automatically maintain the primary pressure within a specified range.

Cattenom, France (4 X 1,300 MWe): inside a reactor building.

The EPR main reactor components: reactor pressure vessel, pressurizer and steam generators feature larger volumes than similar components from previous designs to provide additional benefit in terms of operation and safety margins.

The increased free volume in the reactor pressure vessel, between the nozzles of the reactor coolant lines and the top of the core, provides a higher water volume above the core and thus additional margin with regard to the core “dewatering” time in the event of a postulated loss of coolant accident. Therefore, more time would be available to counteract such a situation.

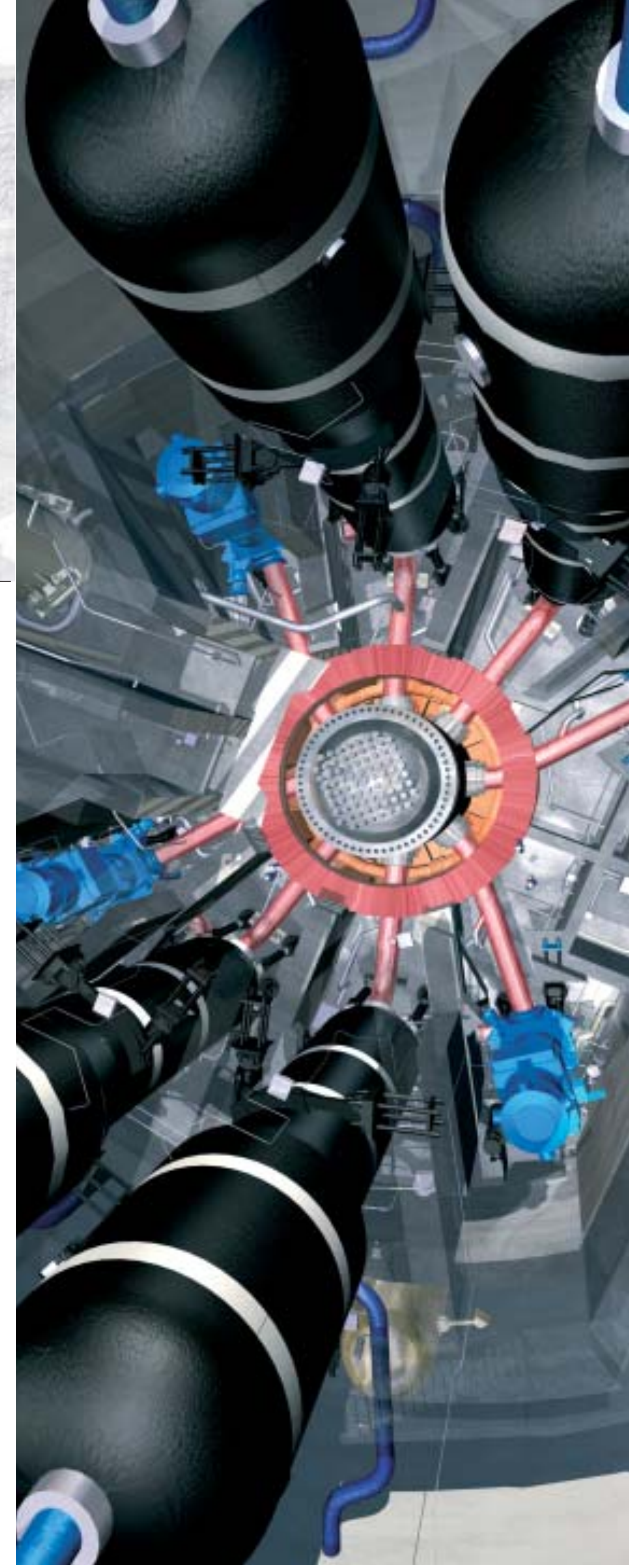
This increased volume would also be beneficial in shutdown conditions in case of loss of the Residual Heat Removal System function.

Larger water and steam phase volumes in the pressurizer smooth the response of the plant to normal and abnormal operating transients allowing extended time to counteract accident situations and extended equipment lifetime.

The larger volume of the steam generator secondary side results in increasing the secondary water inventory and the steam volume, which offers several advantages.

- During normal operation, smooth transients are obtained and thus the potential for unplanned reactor trips is reduced.
- Regarding the management of steam generator tube rupture scenarios, the large steam volume, in conjunction with a setpoint of the safety valves of the steam generators above the safety injection pressure, prevents liquid release outside the reactor containment.
- Due to the increased mass of secondary side water, in case of an assumed total loss of the steam generator feedwater supply, the dry-out time would be at least 30 minutes, sufficient time to recover a feedwater supply or to decide on other countermeasures.

In addition, the primary system design pressure has been increased in order to reduce the actuation frequency of the safety valves which is also an enhancement in terms of safety.



Computer-generated image of the EPR primary system

CHARACTERISTICS	DATA
Reactor coolant system	
Core thermal power	4,500 MWth
Number of loops	4
Coolant flow per loop	28,330 m^3/h
Reactor pressure vessel inlet temperature	295.9 °C
Reactor pressure vessel outlet temperature	327.2 °C
Primary side design pressure	176 bar
Primary side operating pressure	155 bar
Secondary side design pressure	100 bar
Saturation pressure at nominal conditions	78 bar
Main steam pressure at hot standby	90 bar

OVERALL FUNCTIONAL REQUIREMENTS AND FEATURES

Activation of safety systems

Activation of the safety systems, including safety valves, does not occur prior to reactor trip, which means that best possible use is made of the depressurizing effect of the reactor trip. This approach also ensures maximum safety by minimizing the number of valve activations and the potential for valves sticking open after response.

Preventing reactor trip

Reactor trip is prevented by a fast reactor power cutback to part load when one of the following events occurs:

- loss of steam generator feedwater pumps, provided at least one of them remains available,
- turbine trip,
- full load rejection,
- loss of one reactor coolant pump.

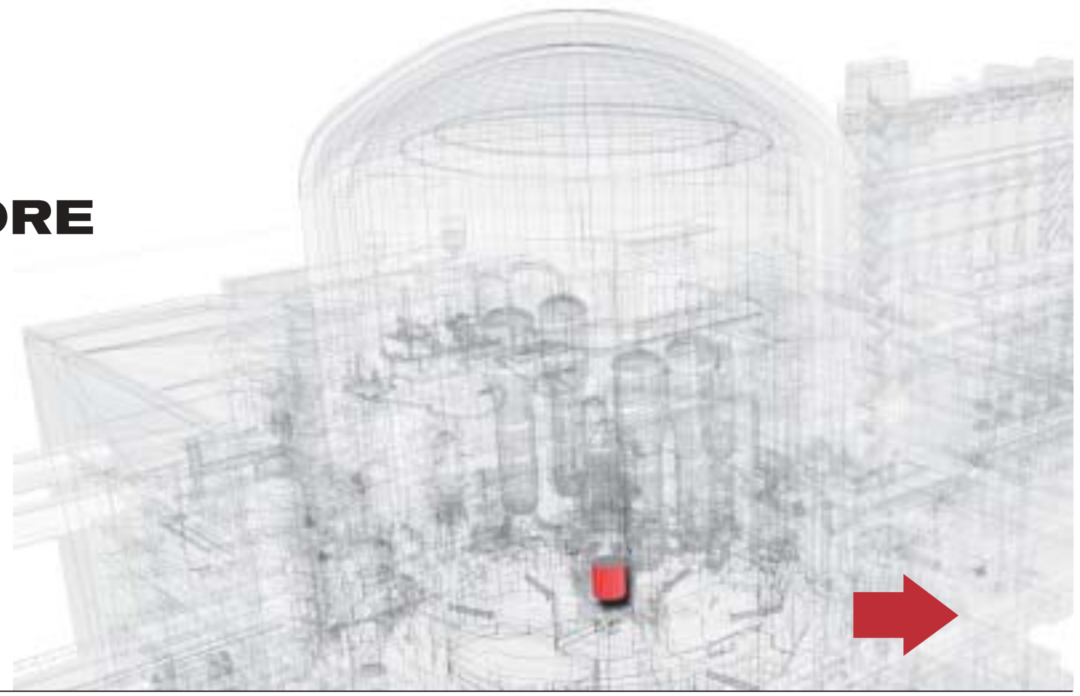
➡ **The increased volume of the primary system is beneficial for smoothing over many types of transients.**

➡ **The primary system design pressure has been increased to reduce the safety valve actuation frequency.**

➡ **The management of steam generator tube rupture scenarios prevents any liquid release outside the reactor containment.**

➡ **The large steam generator secondary side water inventory increases the time available to take action in case of assumed total loss of secondary feedwater.**

REACTOR CORE



The reactor core contains the fuel material in which the fission reaction takes place, releasing energy. The reactor internal structures serve to physically support this fissile material, control the fission reaction and channel the coolant.

The core is cooled and moderated by light water at a pressure of 155 bar and a temperature in the range of 300 °C. The coolant contains soluble Boron as a neutron absorber. The Boron concentration in the coolant is varied as required to control relatively slow reactivity changes, including the effects of fuel burnup. Additional neutron absorbers (Gadolinium), in the form of burnable absorber-bearing fuel rods, are used to adjust the initial reactivity and power distribution. Instrumentation is located inside and outside the core to monitor its nuclear and thermal-hydraulic performance and to provide input for control functions.

The EPR core consists of 241 fuel assemblies. For the first core, assemblies are split into four groups with different enrichments (two groups with the highest enrichment, one of them with Gadolinium). For reload cores, the number and characteristics of the fresh assemblies depend on the type of fuel management scheme selected, notably cycle length and type of loading patterns. Fuel cycle lengths up to 24 months, IN-OUT and OUT-IN fuel management are possible. The EPR is designed for flexible operation with UO_2 fuel and/or MOX fuel. The main features of the core and its operating conditions have been selected to obtain not only high thermal efficiency of the plant and low fuel cycle costs, but also extended flexibility for different fuel cycle lengths and a high level of maneuverability.

The core design analyses demonstrate the feasibility of different types of fuel management schemes to meet the requirements expressed by the utility companies in terms of cycle length and fuel cycle economy (reload fraction, burnup), and to provide the core characteristics needed for sizing of the reactor systems. The nuclear analyses establish physical locations for control rods, burnable poison rods, and physical parameters such as fuel enrichments and Boron concentration in the coolant. The thermal-hydraulic analyses establish coolant flow parameters to ensure that adequate heat is transferred from the fuel to the reactor coolant.

Core instrumentation

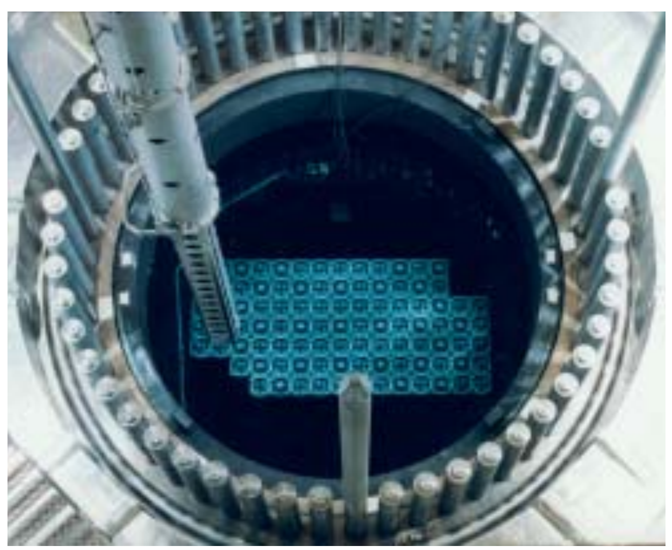
The core power is measured using the ex-core instrumentation, also utilized to monitor the process to criticality.

The reference instrumentation to monitor the power distribution in the core is an “aeroball” system. Vanadium balls are periodically inserted in the core. Their activation level is measured, giving values of the local neutron flux to construct the three-dimensional power map of the core.

The fixed in-core instrumentation consists of neutron detectors and thermocouples to measure the neutron flux distribution in the core and temperature distribution at the core outlet.

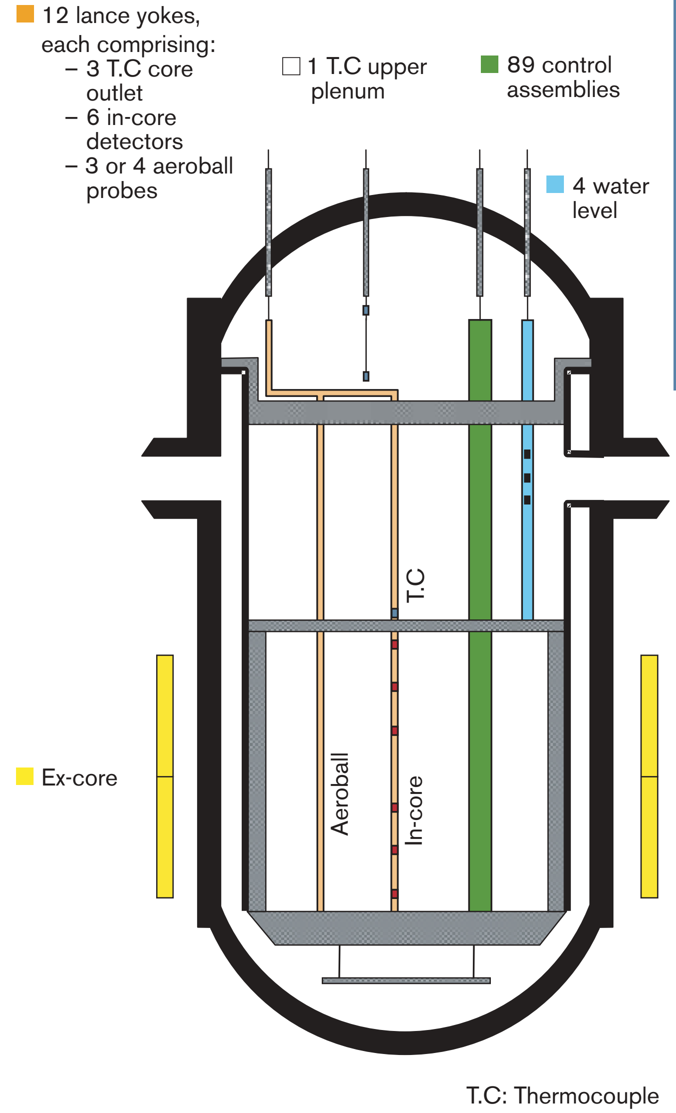
The whole in-core instrumentation package is introduced from the top of the reactor pressure vessel head. Therefore, the bottom of the reactor pressure vessel is free from any penetration.

For additional information see the “Instrumentation and Control systems” chapter, page 42.



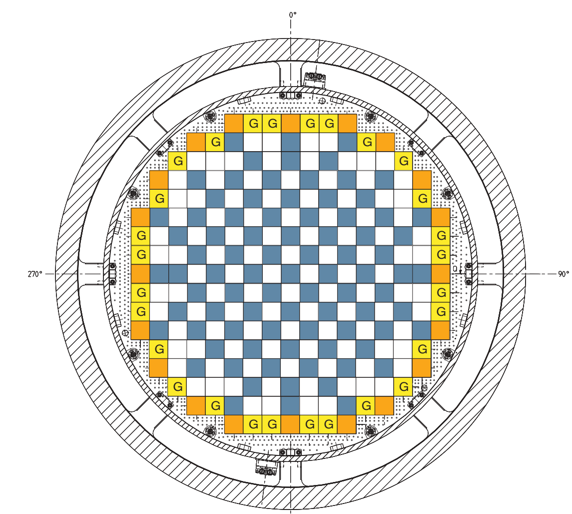
Isar 2 unit, Germany (KONVOI, 1,300 MWe): fuel loading operation.

In-core instrumentation



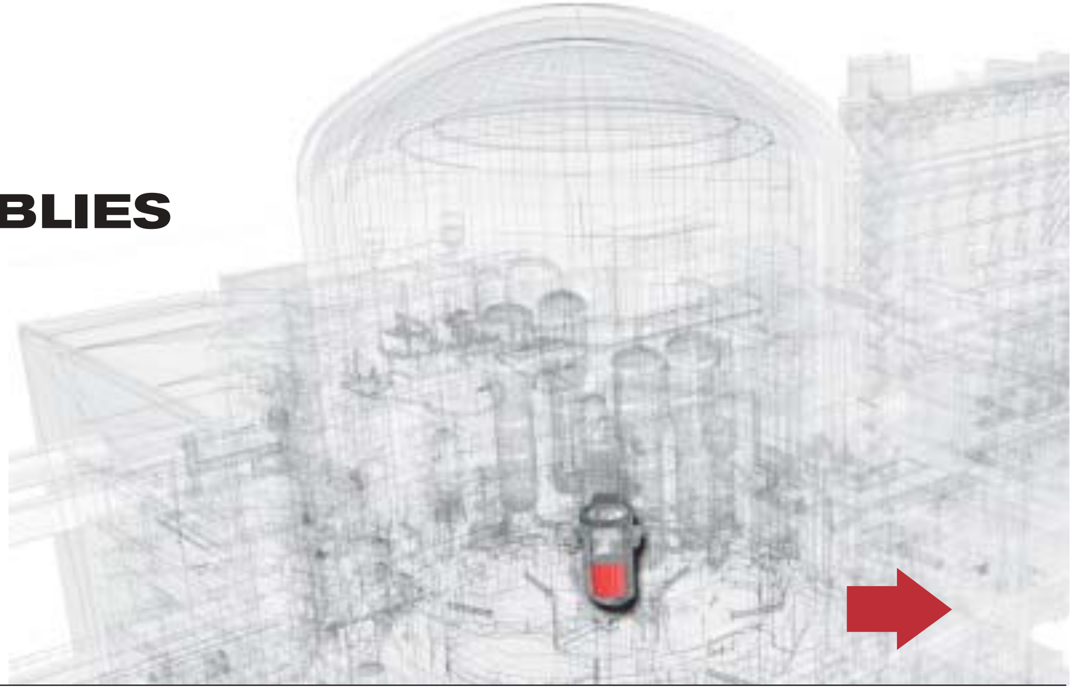
CHARACTERISTICS	DATA
Reactor core	
Thermal power	4,500 MWth
Operating pressure	155 bar
Nominal inlet temperature	295.6 °C
Nominal outlet temperature	328.2 °C
Equivalent diameter	3,767 mm
Active fuel length	4,200 mm
Number of fuel assemblies	241
Number of fuel rods	63,865
Average linear heat rate	156.1 W/cm

Typical initial core loading



- **The EPR core is characterized by considerable margins for fuel management optimization.**
- **Several types of fuel management (fuel cycle length, IN-OUT/OUT-IN) are available to meet utilities’ requirements.**
- **The main features of the core and its operating conditions give competitive fuel management cycle costs.**
- **The EPR core also offers significant advantages in favor of sustainable development:**
  - **17% saving on Uranium consumption per produced MWh,**
  - **15% reduction on long-lived actinides generation per MWh,**
  - **great flexibility for using MOX (mixed UO_2-PuO_2) fuel assemblies in the core, i.e. of recycling the plutonium extracted from spent fuel assemblies.**

FUEL ASSEMBLIES



Each fuel assembly is made up of a bundle of fuel rods that contain the nuclear fuel. The fuel rods and the surrounding coolant are the basic constituents of the active zone of the reactor core.

Fuel assembly structure

The fuel assembly structure supports the fuel rod bundle. It consists of a bottom and a top nozzles plus 24 guide thimbles and 10 spacer grids. The spacer grids are vertically distributed along the assembly structure. Inside the assembly, the fuel rods are vertically arranged according to a square lattice with a 17 x 17 array. 24 positions in the array are occupied by the guide thimbles, which are joined to the spacer grids and to the top and bottom nozzles. The bottom nozzle is equipped with an anti-debris device that almost eliminates debris-related fuel failures.

The guide thimbles are used as locations for the absorber rods of the Rod Cluster Control Assemblies (RCCA) and, when required, for fixed or moveable in-core instrumentation and neutron source assemblies. The bottom nozzle is shaped to direct and contributes to balance the coolant flow. It is also designed to trap small debris, which might circulate inside the primary circuit, in order to prevent damage to the fuel rods. The top nozzle supports the holddown springs of the fuel assembly. The spacer grids, except the top and bottom grids, have integrated mixing vanes to cause mixing of the coolant and improve the thermal exchange between the fuel rods and the coolant. The EPR spacer and mixing grids benefit from a proven design combining a mechanical robustness with a high level of thermal-hydraulic performance.

The guide thimbles and the structure of the mixing spacer grids are made of **M5™ alloy,** a Zirconium based alloy extremely resistant to corrosion and hydriding (the springs of the grids are made of Inconel 718).

Fuel rods

The fuel rods are composed of a stack of enriched Uranium dioxide (or Uranium and Plutonium Mixed Oxide, MOX) sintered pellets, with or without burnable absorber (Gadolinium), contained in a hermetically sealed cladding tube made of **M5™ alloy.** The fuel rod claddings, as the first of the three barriers against radioactive releases, isolate the fuel and fission products from the coolant. A plenum is provided inside the fuel rod to limit the build-up of pressure due to the release of fission gases by the pellets during irradiation. The fuel pellets are held in place by a spring which acts on the top end of the pellet stack. The fuel pellets consist of Uranium dioxide (UO_2) enriched in the fissile isotope U^{235} up to 5% or of Uranium-Plutonium mixed oxyde energetically equivalent.

Burnable poison

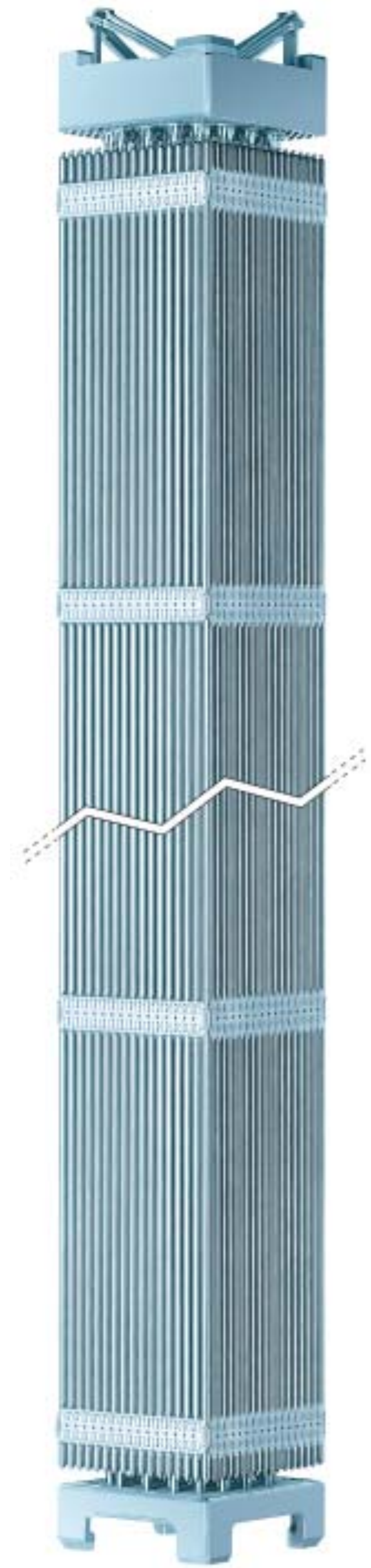
Gadolinium in the form of Gd_2O_3, mixed with the UO_2, is used as integrated burnable poison. The Gadolinium concentrations are in the range of 2% to 8% in weight. The number of Gadolinium-bearing rods per fuel assembly varies from 8 to 28, depending on the fuel management scheme. Enriched UO_2 is used as a carrier material for the Gd_2O_3 to reduce the radial power peaking factors once the Gadolinium has been consumed and makes it easier to meet the prescribed cycle length requirements.

The M5™ Zirconium based alloy

The M5™ alloy is a proven Zirconium based alloy which was developed, qualified and is industrially utilized by Framatome ANP, mainly due to its outstanding resistance to corrosion and hydriding under PWR primary coolant system conditions. Under high duty and high burnup conditions, resistance to corrosion and hydriding is a crucial characteristic for PWR fuel rod claddings and fuel assembly structures as well. Consequently, EPR fuel rod claddings, guide thimbles and spacer grids are made of M5™ alloy. M5™ is presently the most advanced high performance PWR fuel material.

Fuel rod cutaway, showing fuel pellets, cladding, end-plugs and spring.

17 x 17 fuel assembly

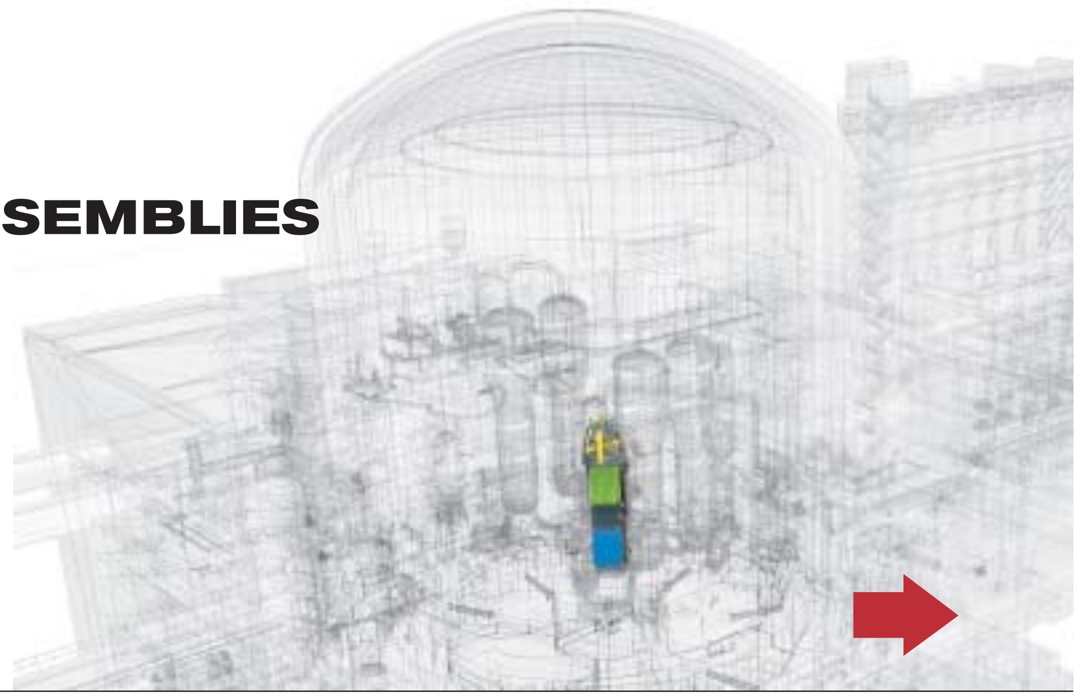


CHARACTERISTICS	DATA
Fuel assemblies	
Fuel rod array	17 x 17
Lattice pitch	12.6 mm
Number of fuel rods per assembly	265
Number of guide thimbles per assembly	24
Fuel assembly discharge burnup (maximum)	> 70,000 MWd/t
Materials	
– Mixing spacer grids	
• structure	M5™
• springs	Inconel 718
– Top & bottom spacer grids	Inconel 718
– Guides thimbles	M5™
– Nozzles	Stainless steel
– Holddown springs	Inconel 718
Fuel rods	
Outside diameter	9.50 mm
Active length	4,200 mm
Cladding thickness	0.57 mm
Cladding material	M5™

- **The U^{235} enrichment level up to 5% allows high fuel assembly burnups.**
- **The choice of M5™ for cladding and structural material results in outstanding resistance to corrosion and hydriding and excellent dimensional behavior at high burnup.**
- **The spacer grids design offers a low flow resistance and a high thermal performance.**
- **The use of an efficient anti-debris device almost eliminates debris-related fuel failures.**

Fuel manufacturing workshop, Lynchburg (Virginia, USA).

CONTROL ASSEMBLIES



The control assemblies, inserted in the core through the guide-thimbles of fuel assemblies, provide reactor power control and reactor trip.

RCCA manufacturing at the FBFC Pierrelatte (France) fuel fabrication plant.

Rod Cluster Control Assemblies

The core has a fast shutdown control system comprising 89 Rod Cluster Control Assemblies (RCCAs). All RCCAs are of the same type and consist of 24 identical absorber rods, fastened to a common head assembly. These rods contain neutron absorbing materials. When they are totally inserted in the core, they cover almost the whole active length of the fuel assemblies.

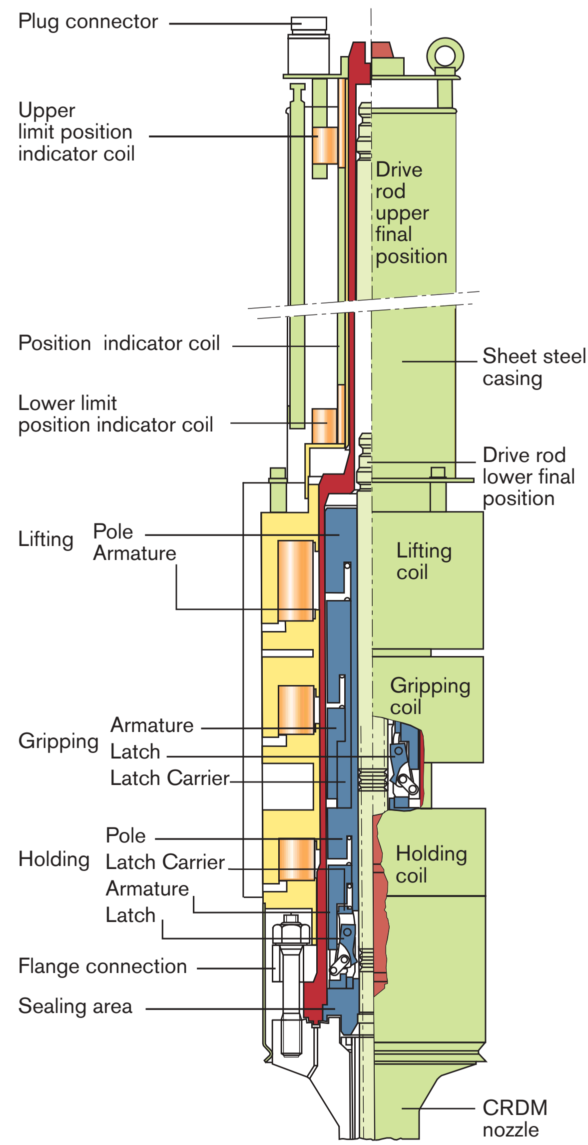
The EPR is equipped with RCCAs of the HARMONI™ type, a proven Framatome ANP design. The neutron absorbing components are bars made of an Ag, In, Cd alloy and sintered pellets of Boron carbide (B_4C). Each rod is composed of a stack of Ag, In, Cd bars and B_4C pellets contained in a stainless steel cladding under a Helium atmosphere (for efficient cooling of the absorbing materials).

Because mechanical wear of the rod claddings happens to be a limiting factor for the operating life of RCCAs, the HARMONI™ claddings benefit from a specific treatment (ion-nitriding) that makes their external surface extremely wear-resistant and eliminates the cladding wear issue.

The RCCAs are assigned to different control bank groups. 37 RCCAs are assigned to control average moderator temperature and axial offset, and 52 RCCAs constitute the shutdown-bank. The first set is divided into five groups split into quadruplets. These quadruplets are combined to form four different insertion sequences depending on cycle depletion. This sequence can be changed at any time during operation, even at full power. A changeover is performed at regular intervals, approximately every 30 equivalent full power days, to rule out any significant localized burnup delay. At rated power the control banks are nearly withdrawn. At intermediate power level, the first quadruplet of a sequence can be deeply inserted and the second may be also inserted. Shutdown margins are preserved by the RCCA insertion limits.

➡ **The EPR is equipped with RCCAs of the proven HARMONI™ design that guarantees a long operating life whatever the operating mode of the reactor.**

CRDM cutaway



CHARACTERISTICS	DATA
Rod cluster control assemblies (RCCAs)	
Mass	82.5 kg
Number of rods per assembly	24
Absorber	
AIC part (lower part)	
– Weight composition (%): Ag, In, Cd	80, 15, 5
– Specific mass	10.17 g/cm^3
– Absorber outer diameter	7.65 mm
– Length	1,500 mm
B_4C part (upper part)	
– Natural Boron	19.9% atoms of B^{10}
– Specific mass	1.79 g/cm^3
– Absorber diameter	7.47 mm
– Length	2,610 mm
Cladding	
Material	AISI 316 stainless steel
Surface treatment (externally)	Ion-nitriding
Outer diameter	9.68 mm
Inner diameter	7.72 mm
Filling gas	Helium
Control rod drive mechanisms (CRDMs)	
Quantity	89
Mass	403 kg
Lift force	$>$ 3,000 N
Travel range	4,100 mm
Stepping speed	375 mm/min or 750 mm/min
Max. scram time allowed	3.5 s
Materials	– Forged Z5 CN 18-10 stainless steel – Magnetic Z12 C13 – Amagnetic stainless steel

Control Rod Drive Mechanisms

A function of the Control Rod Drive Mechanisms (CRDMs), for reactor control purposes, is to insert and withdraw the 89 RCCAs over the entire height of the core and to hold them in any selected position. The other function of the CRDMs is to drop the RCCAs into the core, to shut down the reactor in a few seconds by stopping the chain reaction, in particular in case of an abnormal situation.

The CRDMs are installed on the reactor pressure vessel head and fixed to adapters welded to the vessel head. Each CRDM is a self-contained unit that can be fitted or removed independently of the others. These CRDMs do not need forced ventilation of the coils, which saves space on the reactor head. The control rod drive system responds to the actuation signals generated by the reactor control and protection system or by operator action. The pressure housings of the CRDMs are part of the second of the three barriers against radioactive releases, like the rest of the reactor primary circuit. Therefore, they are designed and fabricated in compliance with the same level of quality requirements.

The complete CRDM consists of:
- the pressure housing with flange connection,
- the latch unit,
- the drive rod,
- the coil housing.

When the reactor trip signal is given, all operating coils are de-energized, the latches are retracted from the rod grooves and the RCCA drops freely into the reactor core under the force of gravity.

➡ **CRDMs are of the same type as those used in the KONVOI reactors, thus they are well proven and based on excellent track record.**

➡ **CRDMs are latch mechanisms cooled by natural convection which saves space on the reactor head.**

REACTOR PRESSURE VESSEL AND INTERNAL STRUCTURES

Chalon manufacturing plant (France): Civaux 1 (N4, 1,500 MWe) reactor pressure vessel and its closure head.

Reactor Pressure Vessel

The Reactor Pressure Vessel (RPV) is the component of the Nuclear Steam Supply System that contains the core. A closure head is fastened to the top of the RPV by means of a stud-nut-washer set.

To minimize the number of large welds, and consequently reduce their manufacturing cost and time for in-service inspection, the upper part of the RPV is machined from one single forging and the flange is integral to the nozzle shell course. Nozzles of the set-on type facilitate the welding of the primary piping to the RPV and the welds in-service inspection as well.

The lower part of the RPV consists of a cylindrical part at the core level, a transition ring and a spherical bottom piece. As the in-core instrumentation is introduced through the closure head at the top of the RPV, there is no penetration through the bottom piece.

The RPV has been designed to facilitate the non-destructive testing during in-service inspections. In particular, its internal surface is accessible to allow 100% visual and/or ultrasonic inspection of the welded joints from the inside.

The RPV closure head is a partly spherical piece with penetrations for the control rod drive mechanisms and the in-core instrumentation.

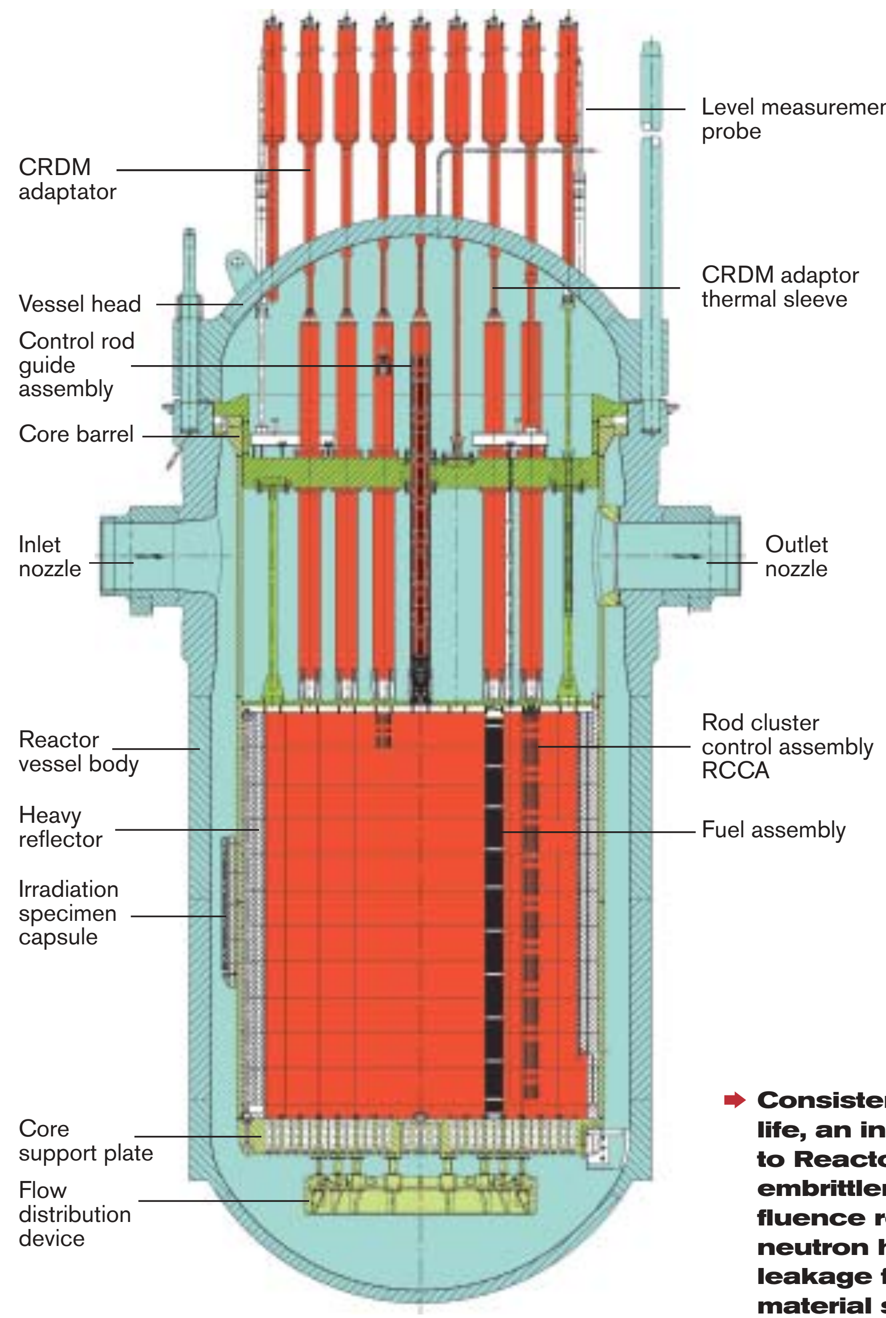
The RPV and its closure head are made of forged ferritic steel – 16 MND 5 – a material that combines adequate tensile strength, toughness and weldability. The entire internal surface of the RPV and its closure head are covered with a stainless steel cladding for corrosion resistance. To contribute to the reduction of the corrosion products radiation source term, the cladding material is specified with a low Cobalt residual content.

Inside the reactor building, the entire RPV structure (including the reactor core) is supported by a set of integrated pads underneath the eight primary nozzles. These pads rest on a support ring which is the top part of the reactor pit.

Significant safety margin against the risk of brittle fracture (due to material aging under irradiation) during the RPV's 60 year design life is ensured.

Reactor pressure vessel monobloc upper shell for the Olkiluoto 3 (Finland) EPR.

Reactor pressure vessel and internals cutaway



The ductile-brittle transition temperature (RT_{NDT}) of the RPV material remains lower than 30 °C at the end of the design life. This result is obtained from the choice of the RPV material and its specified low content in residual impurities, and also thanks to a reduced neutron fluence to the RPV due to the implementation of a neutron reflector surrounding the core and protecting the RPV against the neutron flux.

The suppression of any weld between the flange and the nozzle shell course plus the set-on design of the nozzles allow an increase of the vertical distance between the nozzles and the top of the core. Therefore, in the assumption of a loss of coolant situation, more time is available for the operator to counteract the risk of having the core uncovered by the coolant.

- **Consistently with the EPR 60-year design life, an increased margin with regard to Reactor Pressure Vessel (RPV) embrittlement is obtained from neutron fluence reduction (RPV diameter enlarged, neutron heavy reflector, low neutron leakage fuel management) and from RPV material specifications (reduced RT_{NDT}).**
- **The nozzle axis raising improves the fuel cooling in the event of a loss of coolant accident.**
- **The elimination of any penetration through the RPV bottom head strengthens its resistance in case of postulated core meltdown and prevents the need for in-service inspection and potential repairs.**
- **The reduced number of welds and the weld geometry decrease the need for in-service inspection, facilitate non-destructive examinations and reduce inspection duration as well.**
- **A low Cobalt residual content of the stainless steel cladding is specified to less than 0.06% to contribute to the radiation source term reduction.**

Reactor Internals

The Reactor Pressure Vessel Internals (RPVI) support the fuel assemblies and maintain their orientation and position within the core, to ensure core reactivity control by the control assemblies and core cooling by the primary coolant in any circumstances, including postulated accident circumstances.

The RPVI allow insertion and positioning of the in-core instrumentation as well as protection against flow-induced vibrations during reactor operation.

The internals also contribute to the integrity of the second of the three barriers against radioactive releases by protecting the Reactor Pressure Vessel (RPV) against fast neutron fluence-induced embrittlement.

The internals accommodate the capsules containing samples of the RPV material which are irradiated then examined in the framework of the RPV material surveillance program.

The RPVI are removed partially from the RPV to allow fuel assembly loading/unloading, or are totally removed for complete access to the RPV inner wall for in-service inspection.

Chooz B1, France (N4, 1,500 MWe) upper internals.

The main parts of the RPVI

Upper internals

The upper internals house the Rod Cluster Control Assembly (RCCA) guides. The RCCA guide tube housings and columns are connected to an RCCA guide support plate and an upper core plate. In operation, the upper internals maintain axially the fuel assemblies in their correct position.

Core barrel assembly and lower internals

The core barrel flange sits on a ledge machined from the RPV flange and is preloaded axially by a large Belleville type spring. The fuel assemblies sit directly on a perforated plate, the core support plate. This plate is machined from a forging of stainless steel and welded to the core barrel. Each fuel assembly is positioned by two pins 180° apart.

Heavy reflector

To reduce neutron leakages and flatten the power distribution, the space between the polygonal core and the cylindrical core barrel is filled with a heavy neutron reflector. The **heavy reflector** is a stainless steel structure, surrounding the core, made of rings piled up one on top of the other. The rings are keyed together and axially restrained by tie rods bolted to the core support plate. The heat generated inside the steel structure by absorption of gamma radiation is removed by the primary coolant, through holes and gaps provided in the reflector structure.

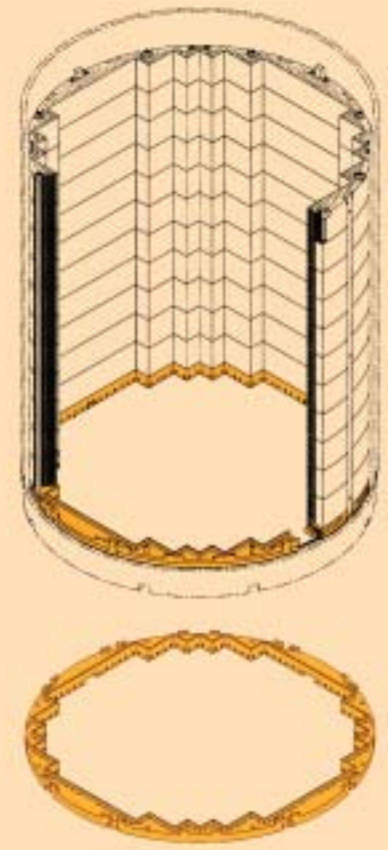
Materials

Most of the internals are made of low Carbon Chromium-Nickel stainless steel. The various connectors, such as bolts, pins, tie rods, etc., are made of cold-worked Chromium-Nickel-Molybdenum stainless steel. At some locations, hard-facing materials are used to prevent fretting wear. To contribute to the radiation source term reduction, stainless steels are specified with a very low Cobalt residual content and the use of Stellite hard-facing is reduced as much as possible.

Heavy reflector

The heavy reflector is an innovative feature with significant benefits:

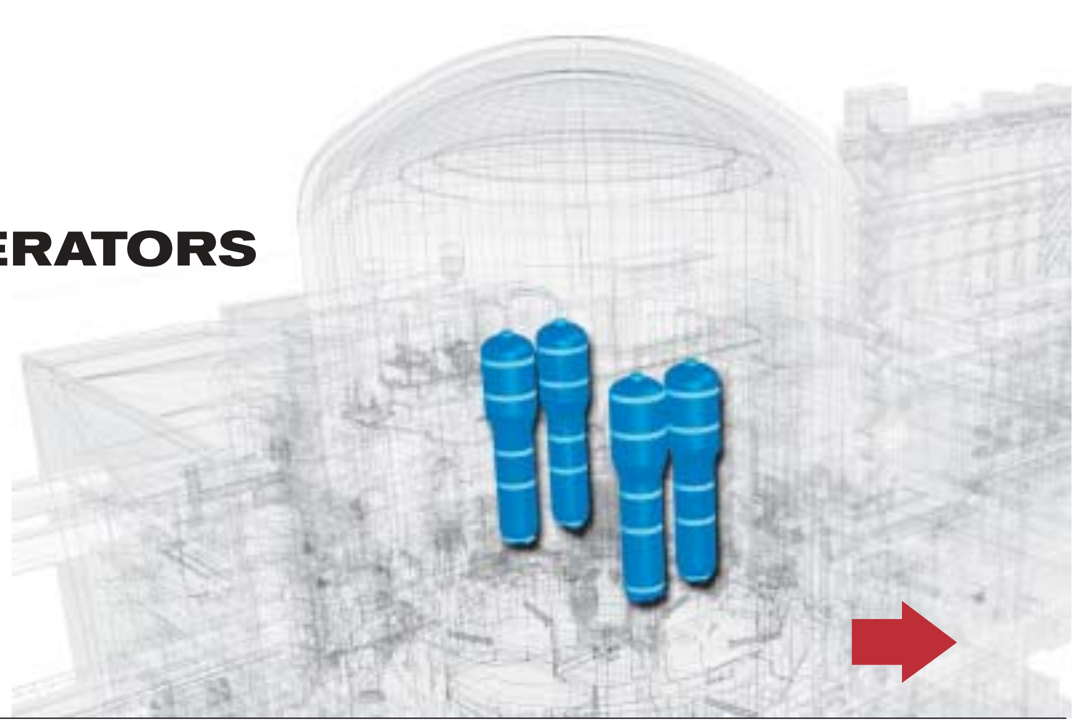
- **By reducing the flux of neutrons escaping from the core, the nuclear fuel is better utilized (more neutrons are available to take part in the chain reaction process), thereby making it possible to decrease the fuel cycle cost by reducing the fuel enrichment necessary to reach a given burnup, or to increase burnup with a given enrichment.**
- **By reducing the neutron leakages from the core, the Reactor Pressure Vessel is protected against fast neutron fluence-induced aging and embrittlement, helping to ensure the 60-year design life of the EPR.**
- **The reactor also provides advances in terms of mechanical behavior of the internal structure surrounding the core:**
  - **a smooth stress distribution inside the structure, due to an efficient inside cooling of the reflector, limiting loads and avoiding deformation,**
  - **no discontinuities, like welds or bolts, in the most irradiated areas,**
  - **a large decrease of depressurization loads to take into account in case of assumed loss of coolant accident, because there is no significant quantity of water trapped in the structure around the core.**



CHARACTERISTICS	DATA
Reactor pressure vessel	
Design pressure	176 bar
Design temperature	351 °C
Life time (load factor 0.9)	60 yrs
Inside diameter (under cladding)	4,885 mm
Wall thickness (under cladding)	250 mm
Bottom wall thickness	145 mm
Height with closure head	12,708 mm
Base material	16 MND 5
Cladding material	Stainless steel (Cobalt $\leq$ 0.06%)
Mass with closure head	526 t
End of life fluence level (E > 1 MeV) IN-OUT fuel management scheme with UO_2	$\approx 1 \times 10^{19}$ n/cm^2
Base material final RT_{NDT} (final ductile-brittle transition temperature)	$\approx$ 30 °C
Closure head	
Wall thickness	230 mm
Number of penetrations for:	
• Control rod mechanisms	89
• Dome temperature measurement	1
• Instrumentation	16
• Coolant level measurement	4
Base material	16 MND 5
Cladding material	Stainless steel (Cobalt $\leq$ 0.06%)
Upper internals	
Upper support plate thickness	350 mm
Upper core plate thickness	60 mm
Main material	Z3 CN 18–10/Z2 CN 19–10
Lower internals	
Lower support plate thickness	415 mm
Lower internals parts material	Z3 CN 18–10/Z2 CN 19–10
Neutron heavy reflector	
Material	Z2 CN 19–10
Mass	90 t

- **The design of the EPR reactor pressure vessel internals is based on the N4 and KONVOI proven designs.**
- **The heavy neutron reflector brings an enhanced fuel utilization and protects the reactor pressure vessel against aging and embrittlement.**
- **A low Cobalt residual content of the stainless steels is specified and the use of Stellite hard-facing is optimized so as to reduce radiation source term.**

STEAM GENERATORS



The steam generators (SG) are the interface between the primary water heated by the nuclear fuel and the secondary water which provides steam to the turbine generator. The primary water flows inside the steam generator tube bundle and transfers heat to the secondary water to produce steam.

The EPR steam generator is a vertical, U-tube, natural circulation heat exchanger equipped with an **axial economizer**. It is an enhanced version of the N4 steam generator.

It is composed of two subassemblies:
- one ensuring vaporization of the secondary feedwater,
- the other mechanically drying the steam-water mixture produced.

In conjunction with an increased heat exchange area, the EPR axial economizer makes it possible to reach a saturation pressure of 78 bar and a plant efficiency of 36 to 37% (depending on site conditions). The tube bundle is made of a proven stress-corrosion resistant alloy: Inconel 690 with a specified mean value Co content less than 0.015%. The steam generator bundle wrapper is made of 18 MND 5 steel.

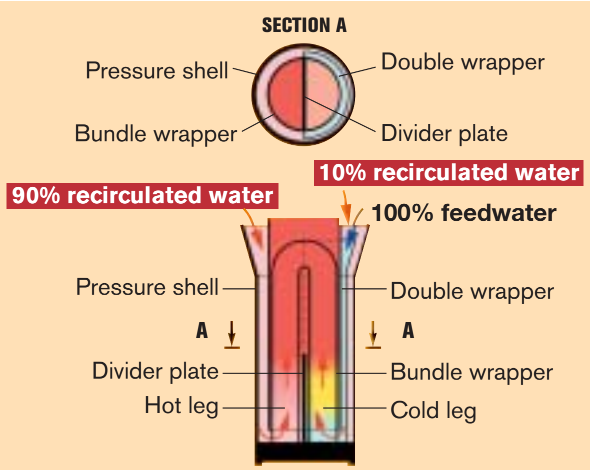
To increase the heat transfer efficiency, the axial economizer directs 100% of the cold feedwater to the cold leg of the tube bundle, and about 90% of the hot recirculated water to the hot leg. This is done by adding a wrapper to guide the feedwater to the cold leg of the tube bundle and a partition plate to separate the cold leg from the hot leg. This design improvement increases the steam pressure by about 3 bar compared to a conventional steam generator. There is an easy access to the tube bundle for inspection and maintenance is provided.

Particular attention was given during the design of the EPR steam generator to cancel out secondary cross-flows to protect the tube bundle against vibration risks.

The steam drum volume has been augmented. This feature, plus a safety injection pressure lower than the set pressure of the secondary safety valves, would prevent the steam generators from filling up with water in case of steam generator tube rupture to avoid liquid releases.

Compared to previous designs, the mass of water on the secondary side has been increased to get a dry-out time, in the event of a total loss of feedwater, of at least 30 minutes.

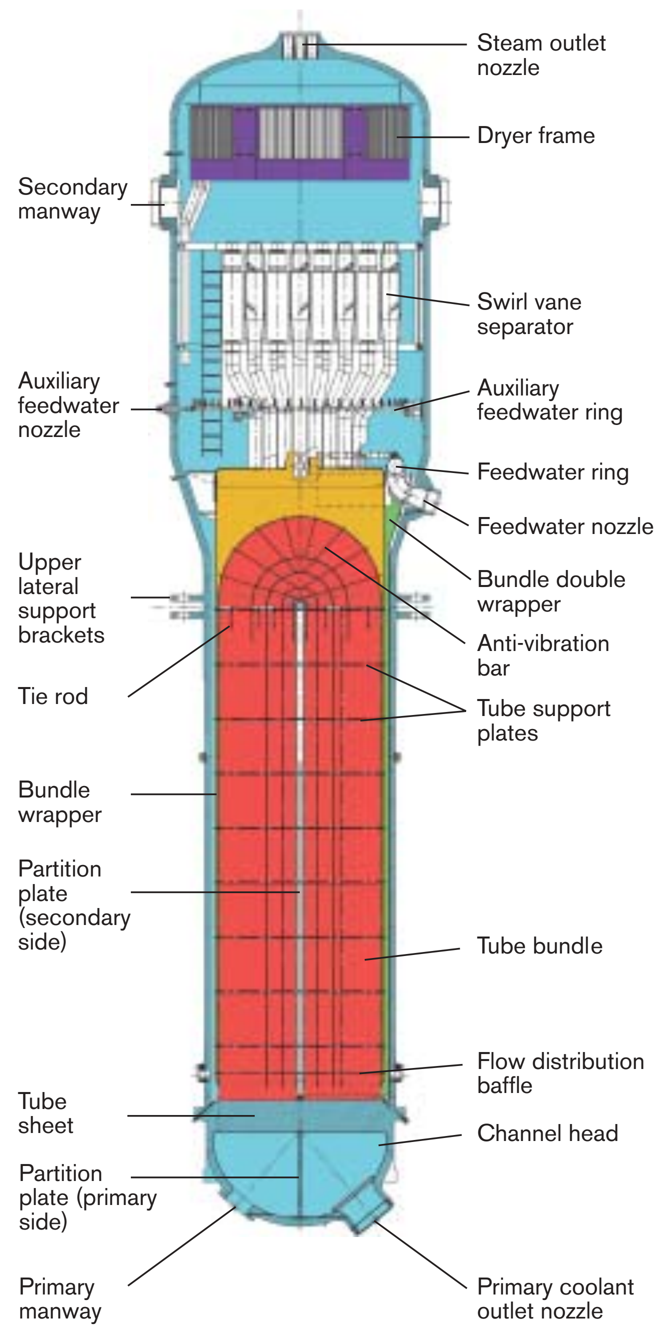
The steam generator is fully shop-built, transported to the plant site and installed in its reactor building cubicle in one piece.



The axial economizer

Its principle primarily consists in directing the feedwater to the cold leg of the tube bundle and about 90% of the recirculated water to the hot leg. In practice, this is done by adding to the standard natural circulation U-tube design a double wrapper in the cold leg of the downcomer to guide the feedwater to the cold leg of the tube bundle and a secondary side partition plate to separate the cold leg and the hot leg of the tube bundle. In conjunction with those two design features, the internal feedwater distribution system of the steam generator covers only the 180° of the wrapper on the cold side.

Steam generator cutaway



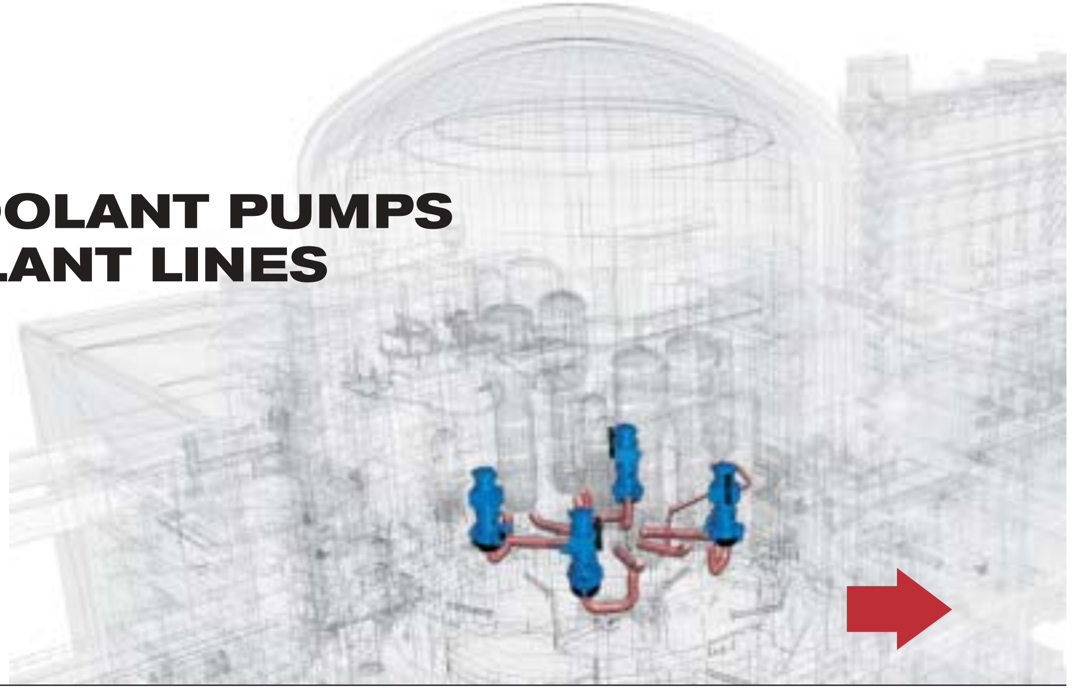
CHARACTERISTICS	DATA
Steam generators	
Number	4
Heat transfer surface per steam generator	7,960 m²
Primary design pressure	176 bar
Primary design temperature	351 °C
Secondary design pressure	100 bar
Secondary design temperature	311 °C
Tube outer diameter/wall thickness	19.05 mm / 1.09 mm
Number of tubes	5,980
Triangular pitch	27.43 mm
Overall height	23 m
Materials	
• Tubes	Alloy 690 TT*
• Shell	18 MND 5
• Cladding tube sheet	Ni Cr Fe alloy
• Tube support plates	13% Cr improved stainless steel
Miscellaneous	
Total mass	500 t
Feedwater temperature	230 °C
Moisture carry – over	0.1%
Main steam flow at nominal conditions	2,554 kg/s
Main steam temperature	293 °C
Saturation pressure at nominal conditions	78 bar
Pressure at hot stand by	90 bar

* TT: Thermally treated

Transportation of a steam generator manufactured in China for Ling-Ao 2.

- **The steam generator is an enhanced version of the axial economizer steam generator implemented on N4 plants.**
- **The axial economizer allows increasing by 3 bar the steam pressure output compared to a conventional design, without impairing access to the tube bundle for inspection and maintenance.**
- **The very high steam saturation pressure at tube bundle outlet (78 bar) is a major contributor to the high efficiency of the EPR (37%).**
- **The secondary water mass is consistent with the 30 min. time period before steam generator dry-out in case of loss of all feedwater systems.**
- **The increase of the steam volume and the set pressure of the secondary safety valves prevent any liquid release to the environment in case of steam generator tube rupture.**

REACTOR COOLANT PUMPS & MAIN COOLANT LINES



Reactor Coolant Pumps

The Reactor Coolant Pumps (RCP) provide forced circulation of water through the reactor coolant system. This circulation removes heat from the reactor core to the steam generators, where it is transferred to the secondary system.

A reactor coolant pump is located between the steam generator outlet and the reactor vessel inlet of each of the four primary loops.

The reactor coolant pump design is an enhanced version of the model used in the N4 reactors. This pump model is characterized by the very low vibration level of its shaft line, due to the hydrostatic bearing installed at the end of the impeller. The pump capacity has been increased to comply with the EPR operating point. In addition, a new safety device, a **standstill seal**, has been added as shaft seal back-up.

➡ An enhanced version of the reactor coolant pump in operation on N4 plants which is characterized by the very low vibration level of its shaft line.

The EPR coolant pump consists of three major components: the pump itself, the shaft seals and the motor.

• **The pump hydraulic cell** consists of the impeller, diffuser, and suction adapter installed in a casing. The diffuser, in one piece, is bolted to the closure flange. The whole assembly can be removed in one piece. The torque is transmitted from the shaft to the impeller by a "Hirth" assembly which consists in radial grooves machined on the flat end of the shaft and symmetrically on the impeller. The shaft is made of two parts rigidly connected by a "spool" piece bolted to each half and removable for maintenance of the shaft seals. It is supported by three radial bearings, two oil bearings on the upper part and one hydrostatic water bearing located on the impeller. The static part of the hydrostatic bearing is part of the diffuser. The axial thrust is reacted by a double acting thrust bearing located at the upper end of the motor shaft below the flywheel.

• **The shaft seal system** consists of three dynamic seals staggered into a cartridge and a standstill seal. The first dynamic seal is a hydrostatic-controlled leakage, film-riding face seal that takes the full primary pressure; the second one is a hydrodynamic seal that takes the remaining pressure in normal operation but can take the full primary pressure in the assumed event of a first stage failure; the third one is also a hydrodynamic seal with no significant differential pressure. Its purpose is to complete final leak tightness and prevent spillage of water. The three seals are rubbing-face seals.

The shaft seals are located in a housing bolted to the closure flange. The closure flange is clamped to the casing by a set of studs together with the motor stand.

In normal operation, the shaft seals are cooled by the seal injection water which is injected just under the shaft seals at a pressure slightly higher than that of the reactor coolant. A thermal barrier, a low-pressure water coil, would cool the primary water before it comes in contact with the shaft seals in the event of a disruption of the seal injection water.

The standstill seal

The shaft seals are backed up with a standstill seal that closes, once the pump is at rest and all seals of the leak-off lines are closed. It creates a sealing surface with a metal-to-metal contact ensuring the shaft tightness in case of:
- **simultaneous loss of water supply by the Chemical and Volume Control System and by the Component Cooling Water System used to cool the shaft sealing system,**
- **cascaded failure of all the stages of the shaft sealing system.**

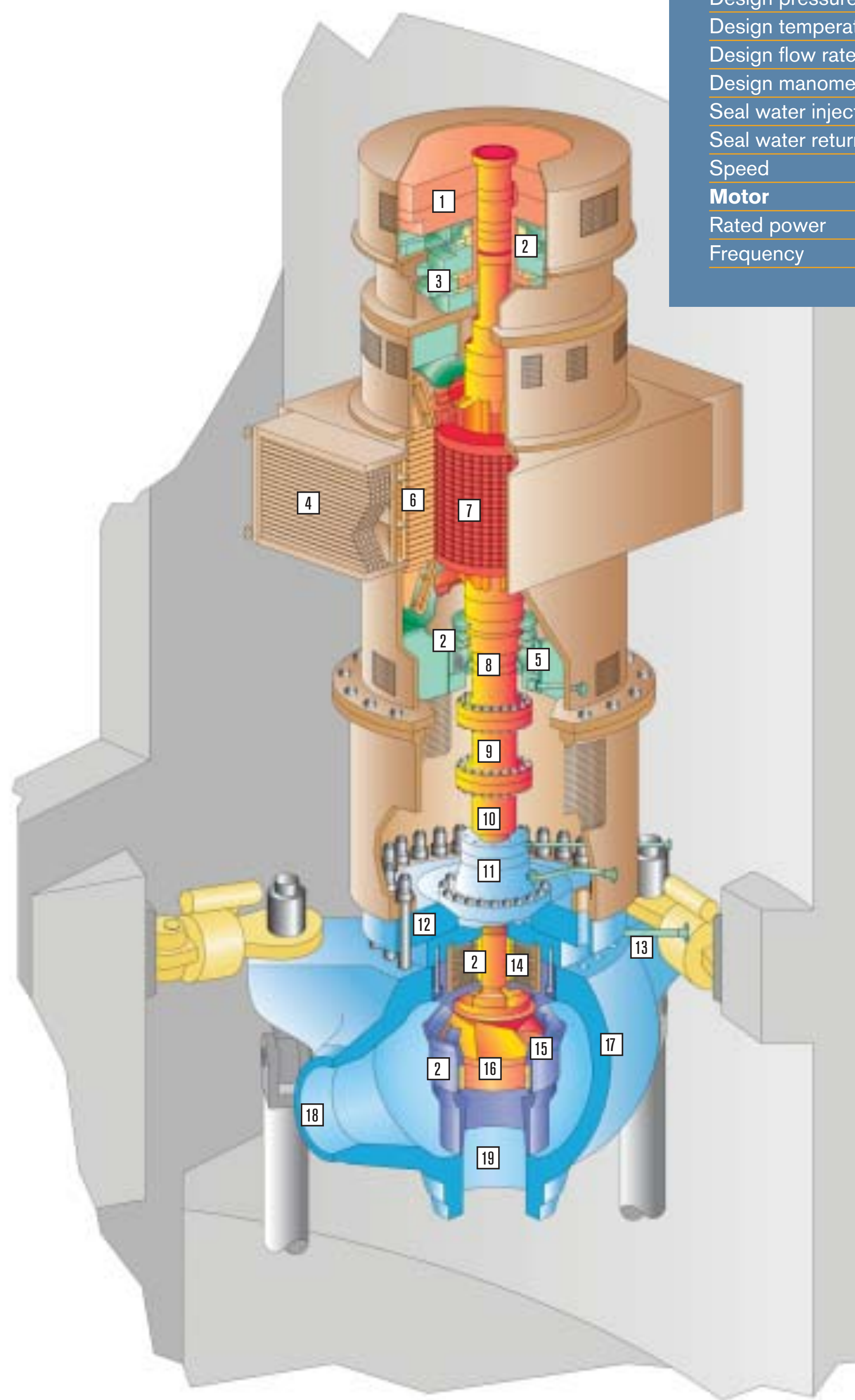
This feature ensures that even in case of total station blackout or failure of the main seals no loss of coolant would occur.

• **The motor** is a drip-proof squirrel-cage induction motor.

All parts of the reactor coolant pump are replaceable. Pump internals can be easily removed from the casing. The spool piece between the pump shaft and the motor shaft enables rapid maintenance of the controlled leakage seal with the motor in place.

CHARACTERISTICS	DATA
Reactor coolant pumps	
Number	4
Overall height	9.3 m
Overall mass w/o water and oil	112 t
Pump	
Design pressure	176 bar
Design temperature	351 °C
Design flow rate	28,330 m^3/h
Design manometric head	100.2 m ± 5%
Seal water injection	1.8 m^3/h
Seal water return	0.680 m^3/h
Speed	1,485 rpm
Motor	
Rated power	9,000 kW
Frequency	50 Hz

Reactor coolant pump cutaway



1 Flywheel
2 Radial bearings
3 Thrust bearing
4 Air cooler
5 Oil cooler
6 Motor (stator)
7 Motor (rotor)
8 Motor shaft
9 Spool piece
10 Pump shaft
11 Shaft seal housings
12 Main flange
13 Seal water injection
14 Thermal barrier heat exchanger
15 Diffuser
16 Impeller
17 Pump casing
18 Discharge
19 Suction

- **The shaft seal system consists of three dynamic seals staggered into a cartridge and a standstill seal.**
- **The standstill seal ensures that, in case of station blackout or failure of the shaft seals after the reactor coolant pump is at rest, no loss of coolant would occur.**
- **The shaft spool piece and the shaft seal cartridge design enable quick maintenance of the shaft seal with the motor in place.**

Jeumont manufacturing plant (France): reactor coolant pump (N4,1,500 MWe).

Chalon manufacturing plant (France): machining of primary piping elbow.

CHARACTERISTICS	DATA
Main coolant lines	
Primary loops	
Inside diameter of straight portions	780 mm
Thickness of straight portions	76 mm
Material	Z2 CN 19–10
Surge line	
Inside diameter	325.5 mm
Thickness	40.5 mm
Materials	Z2 CN 19–10 (low carbon austenitic stainless steel)

Main Coolant Lines

The piping of the four primary loops and the pressurizer surge line are part of the Reactor Coolant System installed in the reactor building. The reactor main coolant lines convey the reactor coolant from the reactor pressure vessel to the steam generators and then to the reactor coolant pumps, which discharge it back to the reactor pressure vessel.
The surge line connects one of the four primary loops with the pressurizer.

Each of the four reactor coolant loops comprises:
– a hot leg, from the reactor pressure vessel to a steam generator,
– a cross-over leg, from the steam generator to a reactor coolant pump,
– a cold leg, from the reactor coolant pump to the reactor pressure vessel.

A large inner diameter of 780 mm was chosen for all the legs to minimize the pressure drop and to reduce the coolant flow velocity in the coolant lines.

The surge line routing has been designed to avoid thermal stratification during steady state operation.

The main coolant line materials and manufacturing processes have been selected to yield a high quality product with high toughness properties, and to improve inspectability and significantly reduce the number of welds.

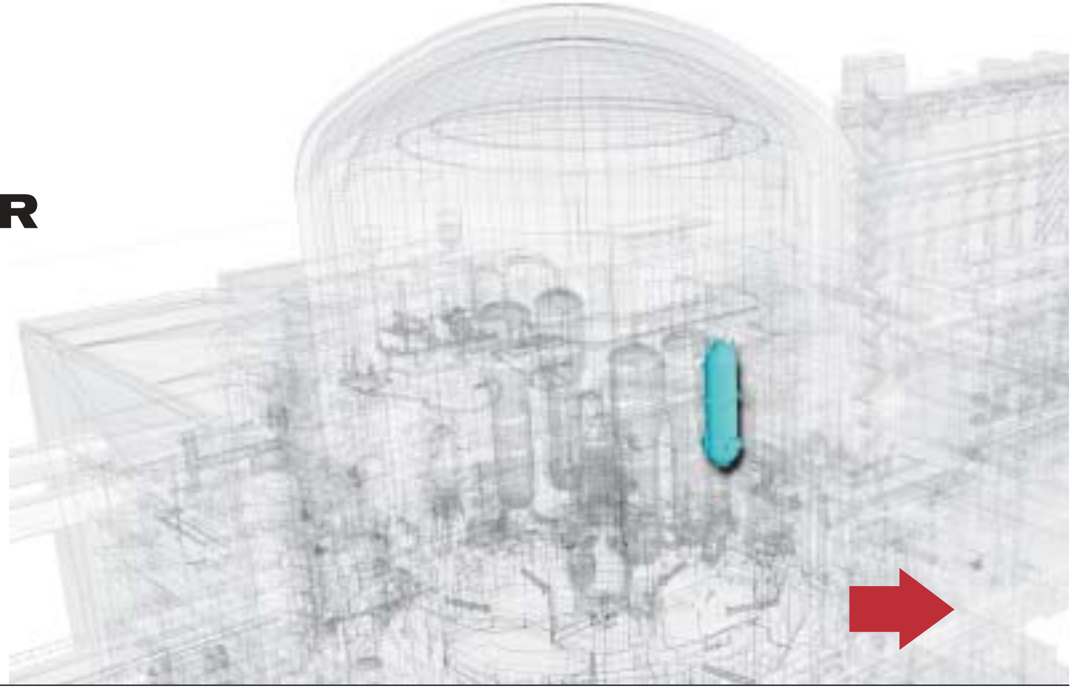
As already experienced on N4 reactors at the Civaux site, the material is a forged austenitic steel, which exhibits excellent resistance to thermal aging and permeability for ultrasonic testing. The hot leg is forged, with separate forged elbows. The cold leg is made using “one-piece technology” with an elbow machined out of the forging. The cross-over leg is made of three parts, mainly for erection convenience. The surge line also consists of several segments. Major advances concerning welding processes are implemented. The homogeneous circumferential welds are made using the orbital narrow gap TIG welding technology. The weld is made with an automatic TIG machine, which enables a large reduction of the volume of weld metal and an enhanced quality level. The bimetallic weld joining austenitic to ferritic parts (like reactor pressure vessel or steam generator nozzles) is made by direct automatic narrow gap welding of Inconel 52.

Several nozzles, branches and piping connections are mounted on each leg for auxiliary and instrumentation lines. Large nozzles are integral with the main coolant lines. They are machined out of the forging of the piping. Small nozzles are set on welded, except for the nozzles of the Chemical and Volume Control System, which are integral with the main coolant line to improve their resistance to thermal fatigue.

These design improvements strongly contribute to the capability for the main coolant lines to fulfill the Leak Before Break requirements.

- **The main coolant lines design and material are based on the technology already implemented on N4 reactor at the Civaux site.**
- **They are made of forged austenitic stainless steel parts (piping and elbows) with high mechanical strength, no sensitivity to thermal aging and are well suited to in-service ultrasonic inspection.**
- **Large nozzles for connection to auxiliary lines are integral and machined out of the forged piping (same for the Chemical and Volume Control System nozzles to avoid thermal fatigue effects).**
- **The main coolant lines design and material provide justification of the application of the Leak Before Break concept.**

PRESSURIZER



Pressurizer erection in a reactor building.

The pressurizer (PZR) role is to maintain the pressure of the primary circuit inside prescribed limits. It is a part of the primary circuit, and is connected through a surge line to the hot leg of one of the four loops of that circuit.

The pressurizer is a vessel containing primary water in its lower part, and steam water in its upper part. To accommodate some primary coolant volume variation, the pressurizer is equipped with electric heaters at its bottom to vaporize more liquid water, and with a spray system at its top to condense more steam. Compared to previous designs, the volume of the EPR pressurizer has been significantly increased in order to smooth the response to operational transients. This improvement provides a gain in terms of equipment life duration and a gain in terms of time available to counteract potential abnormal situations in operation.

Relief and safety valves at the top of the pressurizer protect the primary circuit against overpressure. Compared to previous designs, the EPR features an additional set of motorized valves; in case of postulated accident with a risk of core melting, these valves would provide the operator with an additional efficient mean to rapidly depressurize the primary circuit and avoid a high pressure core melt situation.

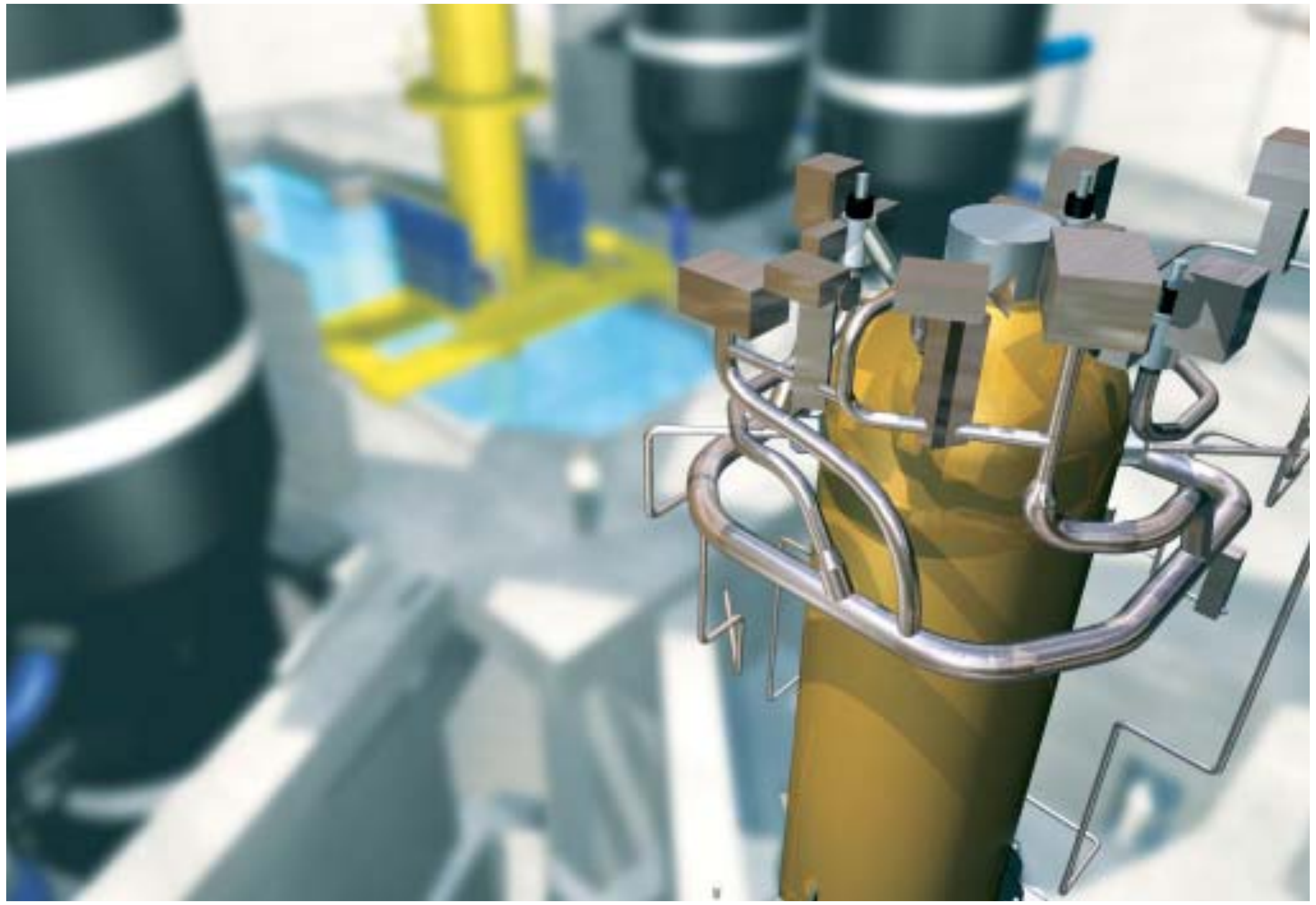
A number of construction provisions have improved maintainability. In particular, a floor between the pressurizer head and the valves eases heater replacement and reduces radiological dose during valve service.

All the pressurizer boundary parts, with the exception of the heater penetrations, are made of forged ferritic steel with two layers of cladding. The steel grade is the same as that for the reactor pressure vessel. The heater penetrations are made of stainless steel and welded with Inconel.

The pressurizer is supported by a set of brackets welded to the main body. Lateral restraints would preclude rocking in the event of a postulated earthquake or accident.

CHARACTERISTICS	DATA
Pressurizer	
Design pressure	176 bar
Design temperature	362 °C
Total volume	75 m³
Total length	14.4 m
Base material	18 MND 5 (low alloy ferritic steel)
Cylindrical shell thickness	140 mm
Number of heaters	108
Total weight, empty	150 t
Total weight, filled with water	225 t
Number and capacity of safety valve trains	3 x 300 t/h
Depressurization valves capacity	900 t/h

Computer-generated image of the EPR pressurizer head with its safety and relief valves.



➡ The pressurizer has a larger volume to smooth the operating transients in order to:

- **ensure the equipment 60-year design life,**
- **increase the time available to counteract an abnormal operating situation.**

➡ Maintenance and repair (concerning safety valves, heaters) are facilitated and radiological doses are reduced.

➡ A dedicated set of valves for depressurizing the primary circuit is installed on the pressurizer, in addition to the usual relief and safety valves, to prevent the risk of high pressure core melt accident.

SYSTEMS

CHEMICAL AND VOLUME CONTROL

The Chemical and Volume Control System (CVCS) performs several operational functions.

- Continuous controls the water inventory of the Reactor Coolant System (RCS) during all normal plant operating conditions, using the charging and letdown flow.
- Adjusts the RCS Boron concentration as required for control of power variations and for plant start-up or shutdown, or to compensate for core burnup, using demineralized water and borated water.
- Ensures permanent monitoring of the Boron concentration of all fluids injected into the RCS, control of the concentration and the nature of dissolved gases in the RCS by providing the means of injecting the required Hydrogen content into the charging flow and allowing degassing of the letdown flow.
- Enables the adjustment of the RCS water chemical characteristics by allowing injection of chemical conditioning agents into the charging flow.
- Ensures a high flow rate capability for primary coolant chemical control with coolant purification, treatment, degassing and storage.
- Injects cooled, purified water into the Reactor Coolant Pump (RCP) seals system to ensure cooling and leaktightness and collection of the seal leakage flow.
- Supplies borated water to the RCS up to the concentration required for a cold shutdown condition and for any initial condition.
- Allows a reduction in pressure by condensing steam in the pressurizer by diverting the charging flow to the auxiliary pressurizer spray nozzle in order to reach Residual Heat Removal System (SIS/RHRS) operating conditions.
- Allows filling and draining of the RCS during shutdown.
- Provides a pressurizer auxiliary spray, if the normal system cannot perform its function, and make-up of the RCS in the event of loss of inventory due to a small leak.
- Ensures the feed and bleed function.

SAFETY INJECTION / RESIDUAL HEAT REMOVAL

The Safety Injection System (SIS/RHRS) comprises the Medium Head Safety Injection System, the Accumulators, the Low Head Safety Injection System and the In-Containment Refueling Water Storage Tank. The system performs a dual function both during the normal operating conditions in RHR mode and in the event of an accident.

The system consists of four separate and independent trains, each providing the capability for injection into the RCS by an Accumulator, a Medium Head Safety Injection (MHSI) pump and a Low Head Safety Injection (LHSI) pump, with a heat exchanger at the pump outlet.

During normal operating conditions, the system in RHR mode:

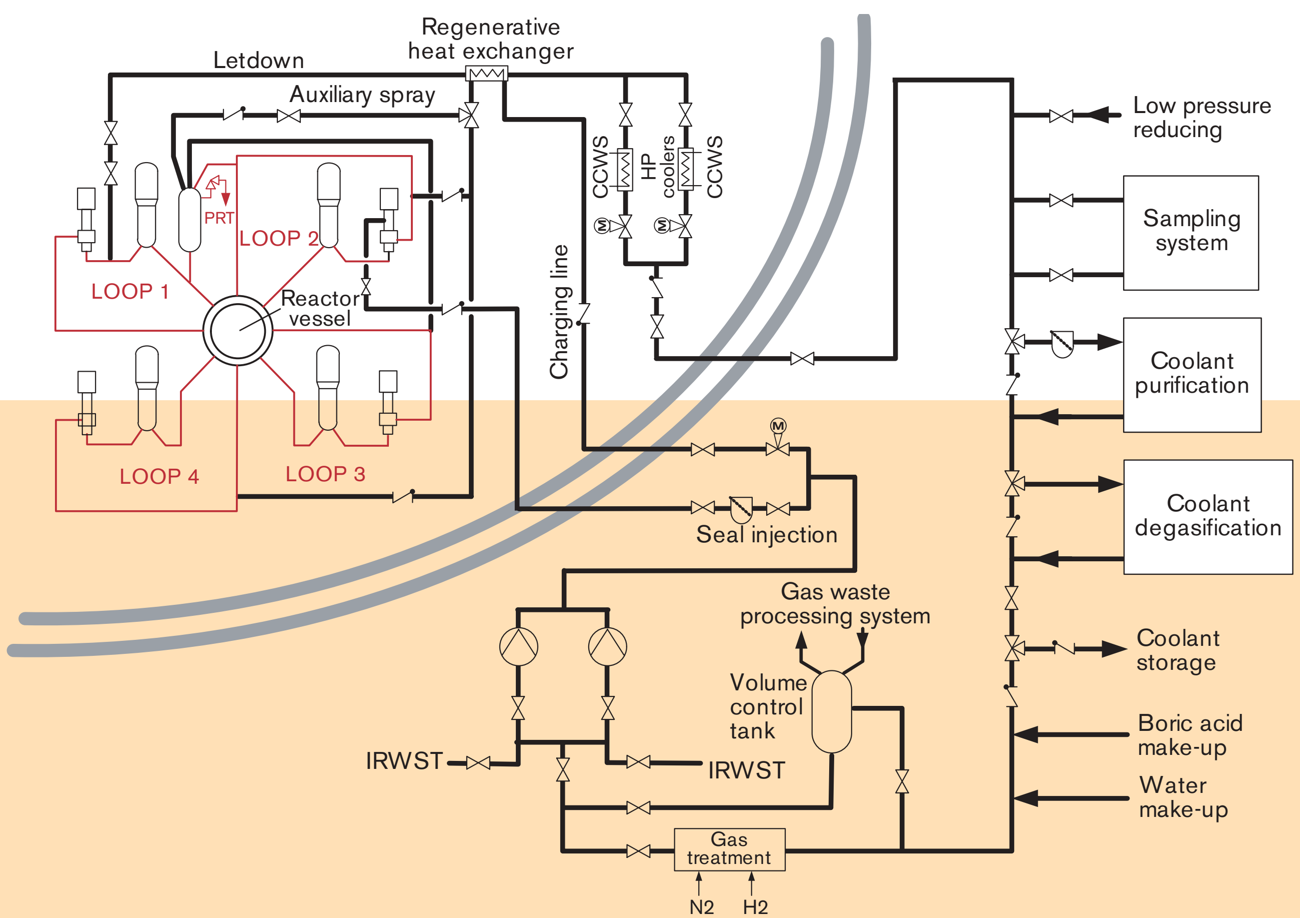
- provides the capability for heat transfer from the RCS to the Component Cooling Water System (CCWS) when heat transfer via the Steam Generators (SG) is no longer sufficiently effective (at an RCS temperature of less than 120 °C in normal operation),
- transfers heat continuously from the RCS or the reactor refueling pool to the CCWS during cold shutdown and refueling shutdown, as long as any fuel assemblies remain inside the containment.

In the event of an assumed accident and in conjunction with the CCWS and the Essential Service Water System (ESWS), the SIS in RHR mode maintains the RCS core outlet and hot leg temperatures below 180 °C following a reactor shutdown.

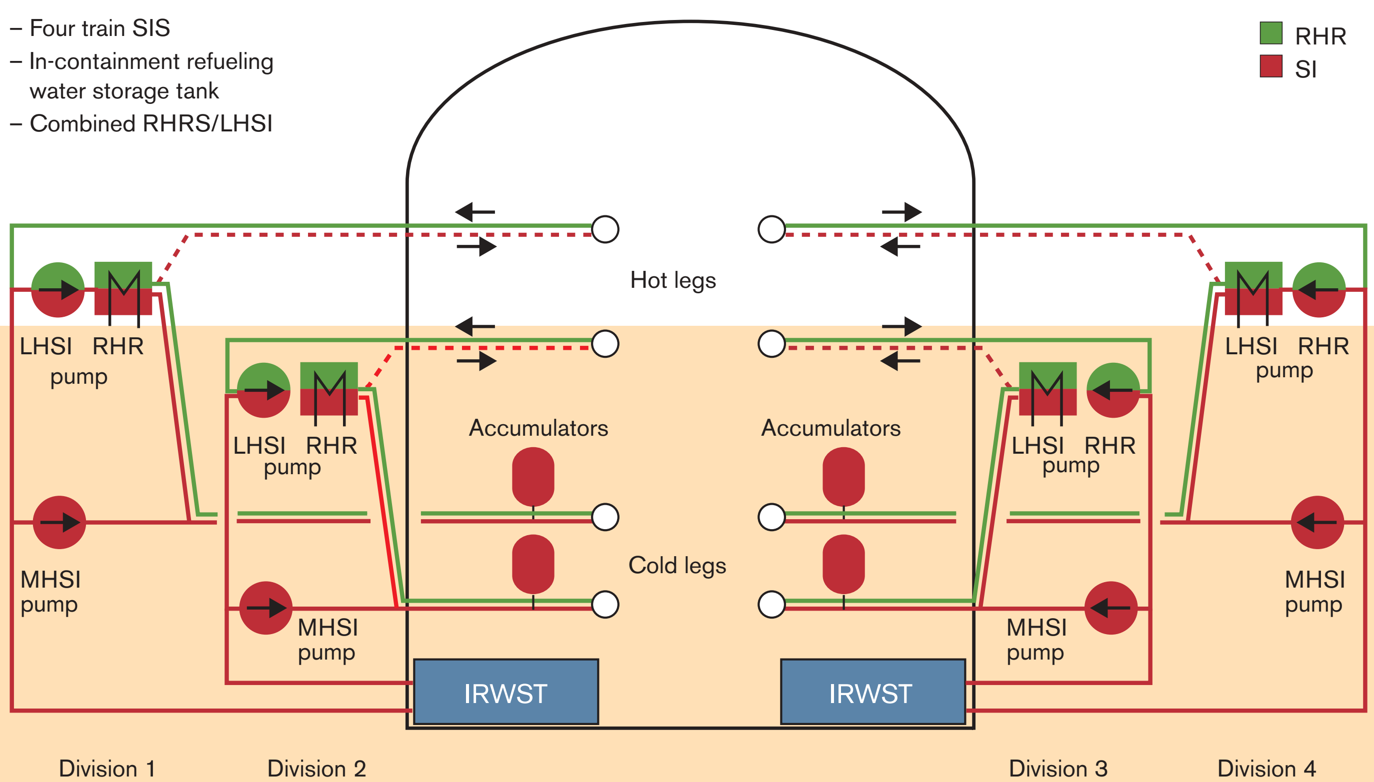
The four redundant and independent SIS/RHRS trains are arranged in separate divisions in the Safeguard Buildings. Each train is connected to one dedicated RCS loop and is designed to provide the necessary injection capability required to mitigate accident conditions. This configuration greatly simplifies the system design.

The design also makes it possible to have extended periods available for carrying out preventive maintenance or repairs. For example, preventive maintenance can be carried out on one complete safety train during power operation.

Chemical and Volume Control System



SI/RHR System



In safety injection mode, the main function of the SIS is to inject water into the reactor core following a postulated loss of coolant accident in order to compensate for the consequence of such events. It would be also activated during a steam generator tube rupture or during loss of a secondary-side heat removal function.

The MHSI system injects water into the RCS at a pressure (92 bar at mini-flow) set to prevent overwhelming the secondary side safety valves (100 bar) in the event of steam generator tube leaks. The accumulators and the LHSI system also inject water into the RCS cold legs when the primary pressure is sufficiently low (accumulator: 45 bar, LHSI: 21 bar at mini-flow).

Back-up functions are provided in the event of total loss of the redundant safety systems. For example:

- the loss of secondary side heat removal is backed up by primary side feed and bleed through an appropriately designed and qualified primary side overpressure protection system,
- the combined function comprising secondary side heat removal, accumulator injection and the LHSI systems can replace the MHSI system in the event of a small break loss of coolant accident,
- similarly, complete loss of the LHSI system is backed up by the MHSI system and by the Containment Heat Removal System (CHRS) for IRWST cooling.

IN-CONTAINMENT REFUELING WATER STORAGE TANK (IRWST)

The IRWST is a tank that contains a large amount of borated water, and collects water discharged inside the containment.

Its main function is to supply water to the SIS, Containment Heat Removal System (CHRS) and Chemical and Volume Control System (CVCS) pumps, and to flood the spreading area in the event of a severe accident.

The tank is located at the bottom of the containment below the operating floor, between the reactor cavity and the missile shield.

During the management of a postulated accident, the IRWST content should be cooled by the LHSI system.

Screens are provided to protect the SIS, CHRS and CVCS pumps from debris that might be entrained with IRWST fluid under accident conditions.

EMERGENCY FEEDWATER

The Emergency Feedwater System (EFWS) is designed to ensure that water is supplied to the steam generators when all the other systems that normally supply them are unavailable.

Its main safety functions are to:

- transfer heat from the RCS via the steam generators to the atmosphere, down to the connection of the RHRS following any plant incidents other than those involving a reactor coolant pressure boundary rupture; this is done in conjunction with the discharge of steam via the Main Steam Relief Valves (MSRV),
- ensure that sufficient water is supplied to the steam generators following a loss of coolant accident or a steam generator tube rupture accident,
- rapidly cool the plant down to LHSI conditions following a small loss of coolant associated with total MHSI failure, in conjunction with steam release from the Main Steam Relief Valves (MSRV).

This system consists of four separate and independent trains, each providing injection capability through an emergency pump that takes suction from an EFWS tank.

For start-up and operation of the plant, a dedicated system, separate from EFWS, is provided.

OTHER SAFETY SYSTEMS

The Extra Borating System (EBS) ensures sufficient boration of the RCS for transfer to the safe shutdown state with the Boron concentration required for cold shutdown. This system consists of two separate and independent trains, each capable of injecting the total amount of concentrated boric acid required to reach the cold shutdown condition from any steady state power operation.

Outside the containment, part of **the Main Steam System (MSS)** is safety classified. This part consists of four geographically separated but identical trains. Each includes one main steam isolation valve, one main steam relief valve, one main steam relief isolation valve and two spring-loaded main steam safety valves.

Outside the containment, part of **the Main Feedwater System (MFS)** is safety classified. It consists of four geographically separated but identical trains. Each includes main feedwater isolation and control valves.

In addition to the safety systems described above, other safety functions are performed **to mitigate postulated severe accidents,** as described in the section dealing with safety and severe accidents.

COMPONENT COOLING WATER

The Component Cooling Water System (CCWS) transfers heat from the safety related systems, operational auxiliary systems and other reactor equipment to the heat sink via the Essential Service Water System (ESWS) under all normal operating conditions.

The CCWS also performs the following safety functions:

- removes heat from the SIS/RHRS to the ESWS,
- removes heat from the Fuel Pool Cooling System (FPCS) to the ESWS for as long as any fuel assemblies are located in the spent fuel storage pool outside the containment,
- cools the thermal barriers of the Reactor Coolant Pump (RCP) seals,
- removes heat from the chillers in divisions 2 and 3 and cools the Containment Heat Removal System (CHRS) by means of two separate trains.

The CCWS consists of four separate safety trains corresponding to the four divisions of the safeguard buildings.

ESSENTIAL SERVICE WATER

The Essential Service Water System (ESWS) consists of four separate safety trains which cool the CCWS heat exchangers with water from the heat sink during all normal plant operating conditions and during incidents and accidents. This system also includes two trains of the dedicated cooling chain for conditions associated with the mitigation of postulated severe accidents.

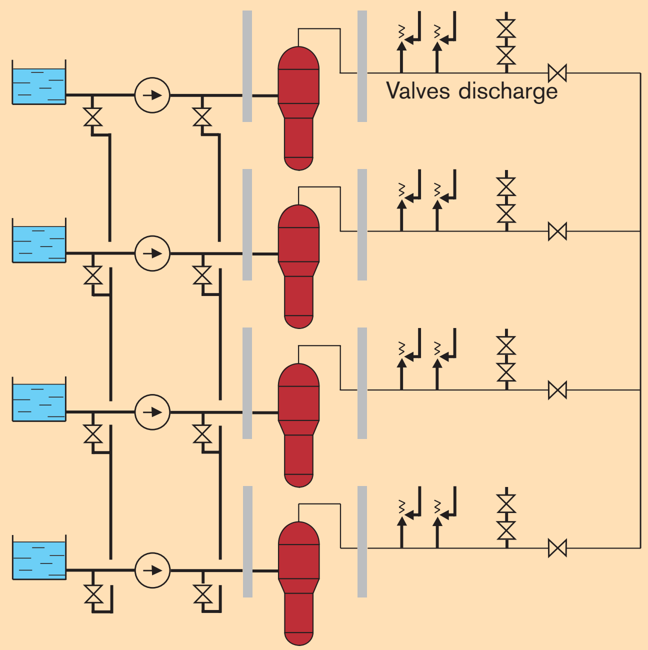
OTHER SYSTEMS

Other systems include the Nuclear Sampling, Nuclear Island Vent and Drain, Steam Generator Blowdown, and Waste Treatment Systems.

- The **Nuclear Sampling System** is used for taking samples of gases and liquid from systems and equipment located inside the reactor containment.
- The **Vent and Drain System** collects gaseous and liquid waste from systems and equipment so that it can be treated.
- The **Steam Generator Blowdown System** prevents the build-up of solid matter in the secondary side water.
- The **Waste Treatment System** ensures the treatment of solid, gaseous and liquid wastes.

Emergency Feedwater System (EFWS)

– Interconnecting headers at EFWS pump suction and discharge normally closed.
– Additional diverse electric power supply for 2/4 trains, using two smalls Diesel generator sets.



SAFETY SYSTEMS AND FUNCTIONS

- **Simplification by separation of operating and safety functions.**
- **Fourfold redundancy applied to the safeguard systems and to their support systems. This architecture allows their maintenance during plant operation, thus ensuring a high plant availability factor.**
- **The different trains of the safety systems are located in four different buildings in which strict physical separation is applied.**
- **With systematic functional diversity, there is always a diversified system which can perform the desired function and bring the plant back to a safe condition in the highly unlikely event of a redundant system becoming totally unavailable.**

POWER SUPPLY

The outline design of the power supply system is shown below.

The Emergency Power Supply is designed to ensure that the safety systems are powered in the event of loss of the preferred electrical sources.

It is designed as four separate and redundant trains arranged in accordance with the four division concept. Each train is provided with an Emergency Diesel Generator (EDG) set.

The emergency power supply system is designed to meet the requirements of the N+2 concept (i.e. assuming a single failure on one train and a maintenance operation on another).

The safety loads connected to the emergency power supply correspond to those required to safely shut down the reactor, remove the residual and stored heat and prevent release of radioactivity.

In the event of total loss of the four EDGs (Station BlackOut or SBO), two additional generators, the SBO Emergency Diesel Generators, provide the necessary power to the emergency loads. They are connected to the safety busbars of two divisions.

Isar 2, Germany (Konvoi, 1,300 MWe) emergency Diesel generator.

FUEL HANDLING AND STORAGE

The reactor core is periodically reloaded with fresh fuel assemblies. The spent fuel assemblies are moved to and stored in the Spent Fuel Pool (SFP). These operations are carried out using several handling devices and systems (fuel transfer tube, spent fuel crane, fuel elevator, refueling machine and spent fuel cask transfer machine).

The underwater fuel storage racks are used for underwater storage of:

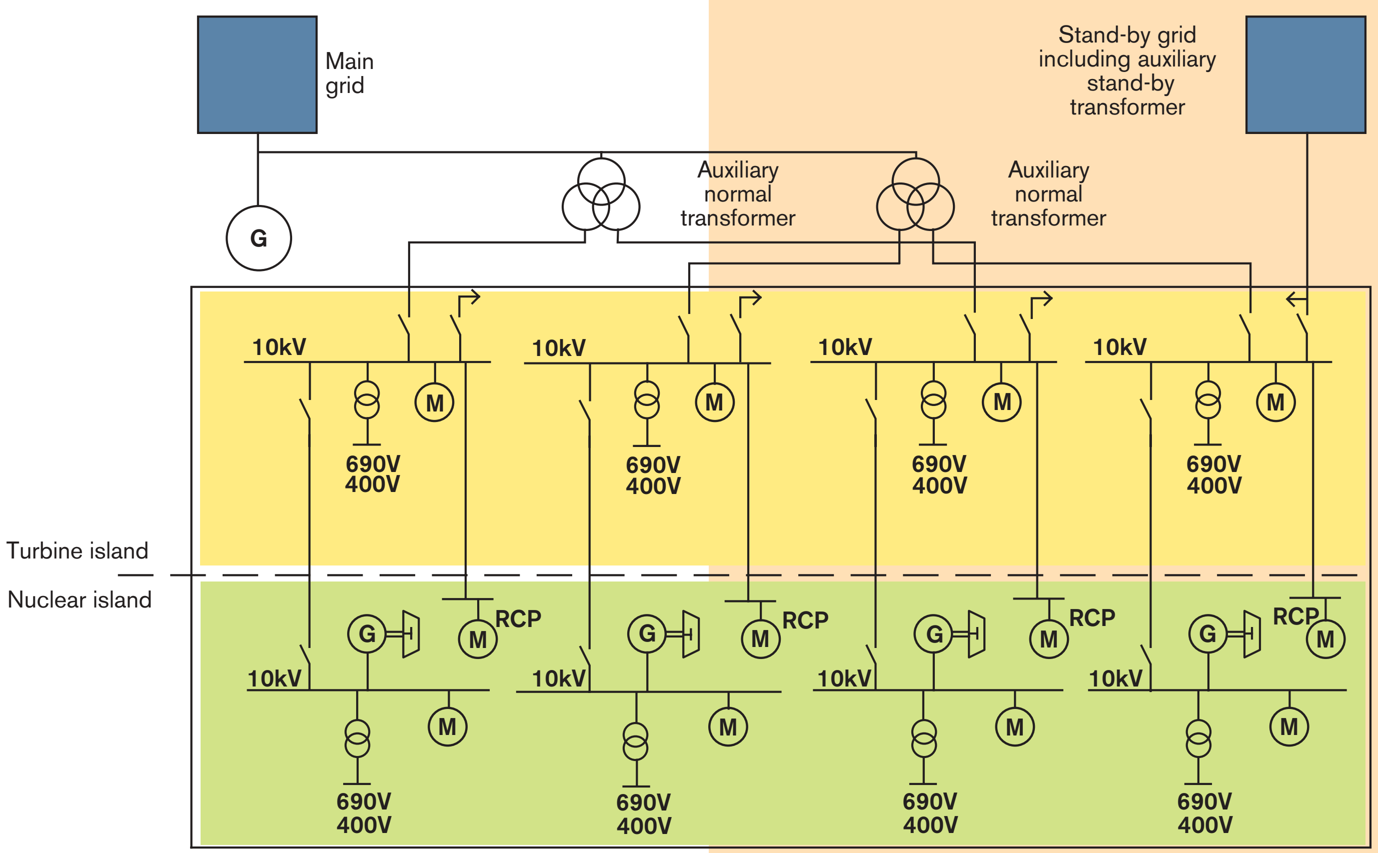
- fresh fuel assemblies, from the time they are delivered on site to the time they are loaded into the reactor core,
- spent fuel assemblies following fuel unloading from the core and prior to shipment out of the site.

The Fuel Pool Cooling and Purification System (FPCPS) is divided into two subsystems: the Fuel Pool Cooling System (FPCS) and the Fuel Pool Purification System (FPPS).

The FPCS provides the capability for heat removal from the SFP and is designed to keep the SFP temperature at the required level during normal plant operation (power operation and refueling outage). This system is arranged in a two separate and independent train configuration with two FPCS pumps operating in parallel in each train.

The FPPS comprises a purification loop for the SFP, a purification loop for the reactor pool and the IRWST, and skimming loops for the SFP and the reactor pool. The system includes two cartridge filters, a demineralizer and a resin trap filter used for purification of pool water.

Electrical systems of an EPR nuclear power station



Chooz B1, France (N4, 1,500 MWe) fuel building.

INSTRUMENTATION & CONTROL SYSTEM

A nuclear power plant, like any other industrial facility, needs technical means to monitor and control its processes and equipment. These means, as a whole, constitute the plant Instrumentation & Control (I & C) processes, which actually comprises several systems and their electrical and electronic equipment.

Basically, the I & C system is composed of sensors to transform physical data into electrical signals, programmable controllers to process these signals and control actuators, monitoring and control means at the disposal of the operators.

The overall design of the I & C system and associated equipment has to comply with requirements imposed by the process, nuclear safety and operating conditions.

To design the EPR and its I & C system, specific attention has been given to ensure a high level of operational flexibility in order to fit with electricity companies' needs. As a result, the EPR is particularly well adapted to load follow and remote control operation modes.

➨ **A plant I & C system, completely computerized, supported by the most modern digital technologies, for high-level operational flexibility**

EPR I & C OVERALL ARCHITECTURE

Inside the overall I & C architecture, each system is characterized depending on its functions (measurement, actuation, automation, man-machine interface) and its role in safety or operation of the plant.

A several level structure

Consideration of the different roles played by the different I & C systems leads to a several level structure for I & C architecture:
- level 0: process interface,
- level 1: system automation,
- level 2: process supervision and control.

(A level 3 deals with site management functions).
Different general requirements are assigned to each level.

The "process interface" (level 0) comprises the sensors, and the switchgears.

The "system automation" level (level 1) encompasses I & C systems to perform:
- reactor protection,
- reactor control, surveillance and limitation functions,
- safety automation,
- process automation.

The "process supervision and control" (level 2) consists of:
- the workstations and panels located in the Main Control Room, the Remote Shutdown Station and the Technical Support Centre, which are also called the Man-Machine Interface (MMI),
- the I & C systems which act as link between the MMI and the "system automation" level.

Safety classification

I & C functions and equipment are categorized into classes in accordance with their importance to safety. Depending on their safety class, I & C functions must be implemented using equipment having the appropriate quality level.

Redundancy, division, diversity and reliability

I & C systems and equipment of the EPR comply with the principles of redundancy, division and diversity enforced for designing EPR safety-related systems. As an illustration, the Safety Injection System and the Emergency Feedwater System, which consist of four redundant and independent trains, have four redundant and independent I & C channels.

Each safety-related I & C system is designed to satisfactorily fulfil its functions even if one of its channels is not available due to a failure and if, at the same time, another of its channels is not available for preventive maintenance reasons or due to an internal hazard (e.g. fire).

I & C systems and equipment participating in safety functions are specified with a level of availability in compliance with the safety probabilistic targets adopted to design the EPR.

➨ **A quadruple redundant safety-related I & C for a further increased level of safety.**

Description of the I & C architecture

Functional safety class		Equipment quality level
F1A	Functions required in case of accident to bring the reactor to controlled state.	E1A
F1B	Functions required after an accident to bring the reactor to safe state. Functions intended to avoid the risk of radioactive releases.	E1B
F2	Other functions contributing to plant safety (adherence to limit operating conditions, surveillance of safety system availability, protection against the effects of internally-generated hazards, detection/monitoring of radioactive releases, functions used in post-accident operation…).	E2
NC	Non-classified functions.	NC

I & C technology

Concerning I & C technology, Framatome ANP uses a consistent I & C system based on its TELEPERM-XS technology for safety applications and on a diversified technology for standard applications.

Computer-generated image of the EPR control room.

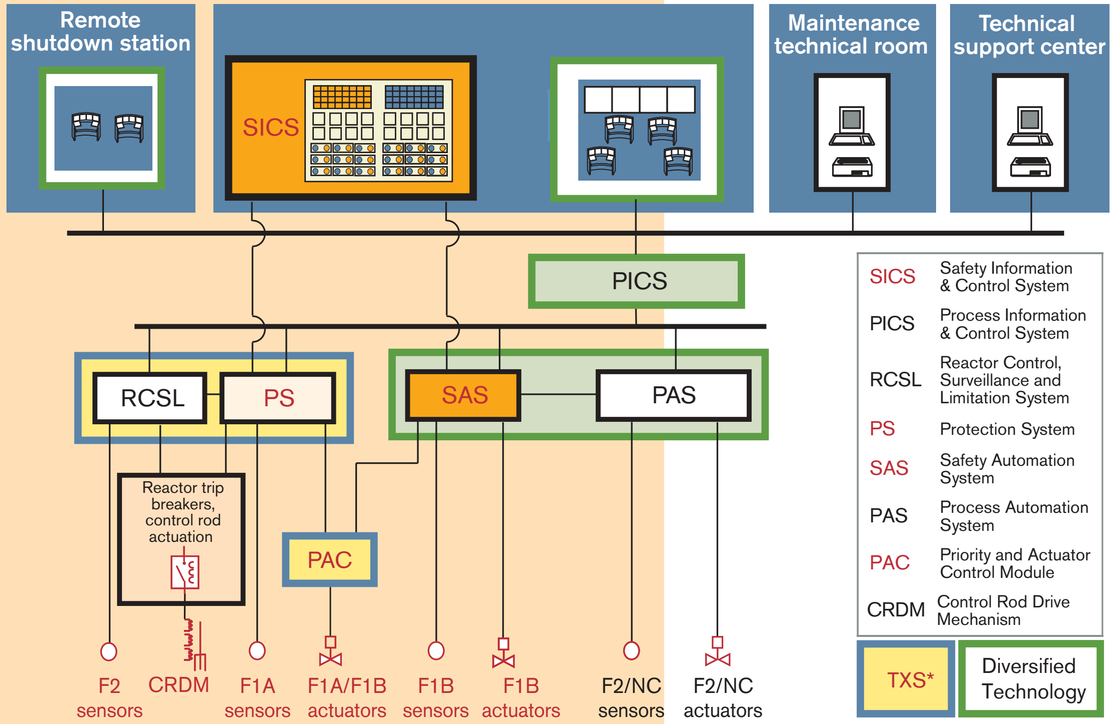
ROLE OF THE I & C SYSTEMS

The I & C systems act in accordance with the "defense in depth" concept.

Three lines of defense are implemented:
- the control system maintains the plant parameters within their normal operating ranges,
- in case a parameter leaves its normal range, the limitation system generates appropriate actions to prevent protective actions from having to be initiated,
- if a parameter exceeds a protection threshold, the reactor protection system generates the appropriate safety actions (reactor trip and safeguard system actuation).

Normally, to operate and monitor the plant, the operators use workstations and a plant overview panel in the Main Control Room. In case of unavailability of the Main Control Room, the plant is monitored and controlled from the Remote Shutdown Station.

I & C architecture



*TELEPERM-XS Framatome ANP technology.

Instrumentation (level 0)

A number of instrumentation channels supply measured data for control, surveillance and protection systems and for information of the control room staff. Multiple-channel acquisition is used for important controls such as control of pressure and temperature of the primary coolant, liquid level in the reactor pressure vessel. Multiple-channel and diversified data acquisition means are implemented.

Concerning the protection of the reactor, a major aspect is the capacity to predict and measure the nuclear power (or neutron flux) level and the three dimensional distribution of power in the core.

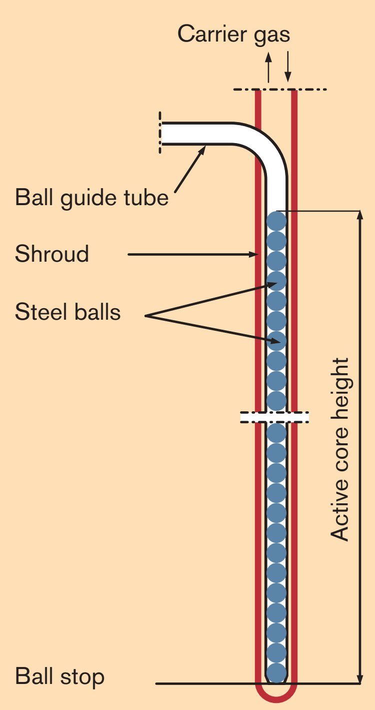
The measurement of the power level is performed using ex-core instrumentation which also provides signals to monitor the core criticality. Relying on temperature measurements in the cold and hot legs of the four primary loops, a quadruple-redundant primary heat balance is achieved and complemented by neutron flux measurements with very short response time.

Prediction and measurement of the three-dimensional power distribution relies on two types of in-core instrumentation:
- "movable" reference instrumentation to validate the core design and to calibrate the other sensors utilized for core surveillance and protection purposes,
- "fixed" instrumentation to deliver online information to the surveillance and protection systems which actuate appropriate actions and countermeasures in case of anomalies or exceeding of predefined limits.

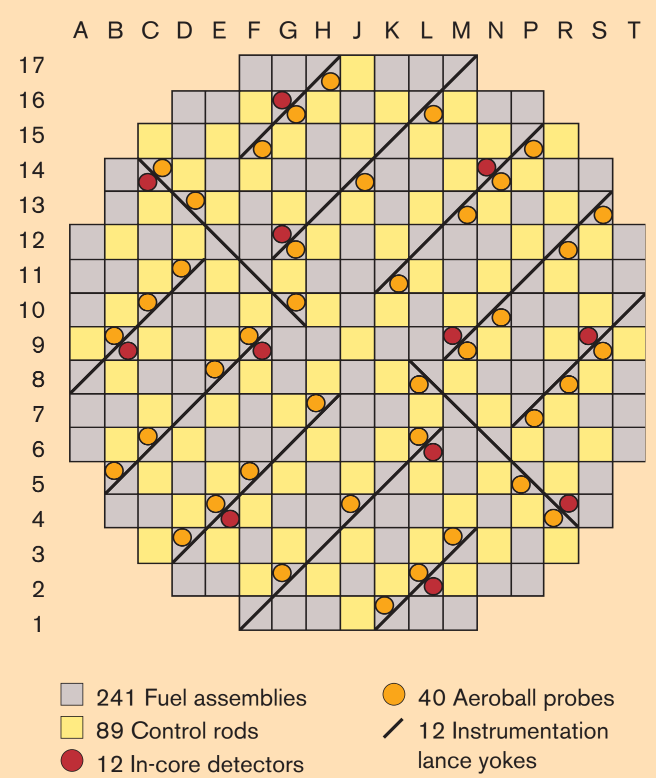
The movable reference instrumentation for power distribution assessment is an "aeroball" system. Stacks of vanadium-alloy balls, inserted from the top of the pressure vessel, are pneumatically transported into the reactor core (inside guide thimbles of fuel assemblies), then, after three minutes in the core, to a bench where the activation of each probe is measured at 30 positions in five minutes. This gives values of the local neutron flux in the core, which are processed to construct the three-dimensional power distribution map.

The fixed in-core instrumentation consists of neutron detectors and thermocouples to measure the neutron flux radial and axial distribution in the core and temperature radial distribution at the core outlet. The neutron flux signals are utilized to control the axial power distribution, and for core surveillance and protection. The core outlet thermocouples continuously measure the fuel assembly outlet temperature and provide signals for core monitoring in case of loss of coolant event. They also provide information on radial power distribution and thermal-hydraulic local conditions.

Aeroball system



EPR in-core instrumentation



Limitation functions and protection of the reactor (level 1)

Four-channel limitation functions are implemented to rule out impermissible operational conditions that would otherwise cause reactor trip actions to be initiated. They also ensure that process variables are kept within the range on which the safety analysis is based, and they initiate actions to counteract disturbances that are not so serious as to require the protection system to trip the reactor.

The protection system counteracts accident conditions, first by tripping the reactor, then by initiating event-specific measures. As far as reasonably possible, two diverse initiation criteria are available for every postulated accident condition.

Reactor trip is actuated by cutting off the power to the electro-magnetic gripping coils of the control rod drive mechanisms. All the control assemblies drop into the core under their own weight and instantaneously stop the chain reaction.

➨ **An enhanced and optimized degree of automated plant control, associated to an advanced Man-Machine interface for operator information and action.**

Man-Machine interface (level 2)

At the design stage of the EPR, due consideration has been given to the human factor for enhancing the reliability of operators' actions, during operation, testing and maintenance phases. This is achieved by applying appropriate ergonomic design principles and providing sufficiently long periods of time for the operators' response to encountered situations or events.

Sufficient and appropriate information is made available to the operators for their clear understanding of the actual plant status, including in the case of a severe accident, and for a relevant assessment of the effects of their actions.

The plant process is supervised and controlled from the Main Control Room which is equipped, regarding information and control, with:
- two screen-based workstations for the operators,
- a plant overview panel which gives information on the status and main parameters of the plant,
- a screen-based workstation for presenting information to the shift supervisor and the safety engineer,
- an additional workstation for a third operator to monitor auxiliary systems.

The Remote Shutdown Station is provided with the same information and data on the process as the Main Control Room.

The plant also comprises a Technical Support Centre. It is a room with access to all the data concerning the process and its control, to be used, in case of accident, by the technical team in charge of analysing the plant conditions and supporting the post accident management.

A computerized screen-based control room designed to maximize operator efficiency. Chooz B1, France (N4, 1,500 MWe).

SAFETY

Golfech 2, France (1,300 MWe): reactor pressure vessel and internals.

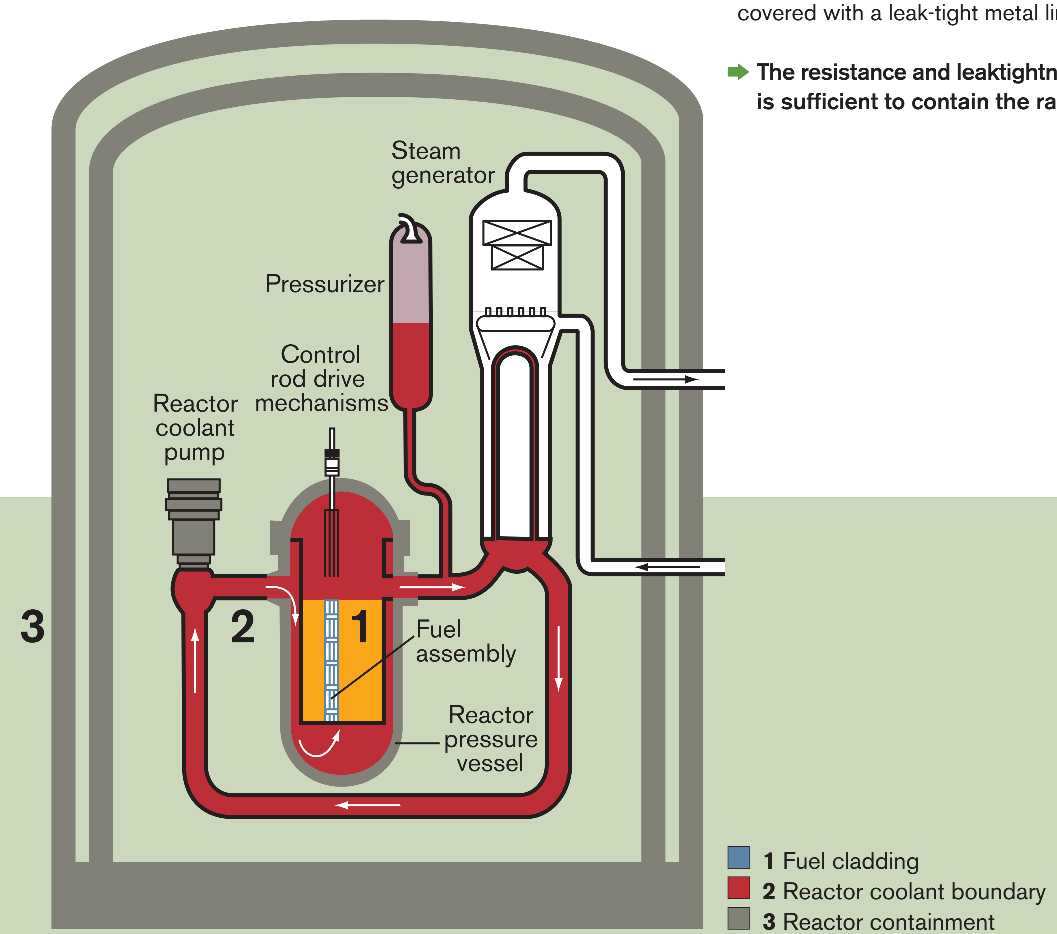
NUCLEAR SAFETY

The fission of atomic nuclei, performed in reactors to generate heat, brings into play large quantities of radiation-emitting radioactive substances from which people and the environment must be protected.
This explains the need for nuclear safety, which consists of the set of technical and organizational provisions taken at each stage in the design, construction and operation of a nuclear plant to ensure normal service, prevent the risks of an accident and limit its consequences in the unlikely event of its occurrence.

Nuclear reactor safety requires that three functions should be fulfilled at all times:
- control of the chain reaction, and therefore of the power generated,
- cooling of the fuel, including after the chain reaction has stopped, to remove residual heat,
- containment of radioactive products.

It relies upon two main principles:
- the three protective barriers,
- defense in depth.

The three protective barriers



THREE PROTECTIVE BARRIERS

The concept of the "three protective barriers" involves placing, between the radioactive products and the environment, a series of strong, leak-tight physical barriers to contain radioactivity in all circumstances:
- **first barrier:** the fuel, inside which most of the radioactive products are already trapped, is enclosed within a metal cladding,
- **second barrier:** the reactor coolant system is housed within a metal enclosure which includes the reactor vessel containing the core constituted by the fuel within its cladding,
- **third barrier:** the reactor coolant system is also enclosed within a high-thickness concrete construction (for the EPR, this construction is a double shell resting upon a thick basemat, whose inner wall is covered with a leak-tight metal liner).

➡ **The resistance and leaktightness of just one of these barriers is sufficient to contain the radioactive products.**

DEFENSE IN DEPTH

The concept of "defense in depth" involves ensuring the resistance of the protective barriers by identifying the threats to their integrity and by providing successive lines of defense which will guarantee high effectiveness:

- **first level:** safe design, quality workmanship, diligent operation, with incorporation of the lessons of experience feedback in order to prevent occurrence of failures,
- **second level:** means of surveillance for detecting any anomaly leading to departure from normal service conditions in order to anticipate failures or to detect them as soon as they occur,
- **third level:** means of action for mitigating the consequences of failures and prevent core melt down; this level includes use of redundant systems to automatically bring the reactor to safe shutdown; the most important of these systems is the automatic shutdown by insertion of the control rods into the core, which stops the nuclear reaction in a few seconds; in addition, a set of safeguard systems, also redundant, are implemented to ensure the containment of the radioactive products,
- **beyond,** the defense in depth approach goes further, as far as postulating the failure of all these three levels, resulting in a "severe accident" situation, in order to provide all the means of minimizing the consequences of such a situation.

➡ **By virtue of this defense in depth concept, the functions of core power and cooling control are protected by double or triple systems – and even quadruple ones as in the EPR – which are diversified to prevent a single failure cause from concurrently affecting several of the systems providing the same function.**

➡ **In addition, the components and lines of these systems are designed to automatically go to safe position in case of failure or loss of electrical or fluid power supply.**

Lynchburg technical center (Va, USA): training for steam generator inspection.

The training for steam generator inspection illustrates:

➡ **the first level of defense in depth relating to the quality of workmanship,**

➡ **the second barrier, as the training relates to steam generator tubes which form part of the primary system.**

EPR SAFETY

The first important choice, in line with the recommendations of the French and German Safety Authorities, was to build the EPR design upon an evolutionary approach based on the experience feedback from the 96 reactors previously built by Framatome or Siemens. This choice enables the AREVA Group to offer an evolutionary reactor based on the latest constructions (N4 reactors in France and KONVOI in Germany) and to avoid the risk arising from the adoption of unproven technologies.

This does not mean that innovative solutions, backed by the results of large-scale research and development programs, have been left out; indeed, they contribute to the accomplishment of the EPR progress objectives, especially in terms of safety and in particular regarding the prevention and mitigation of hypothetical severe accidents.

These progress objectives, motivated by the continuous search for a higher safety level, involve reinforced application of the defense in depth concept:

- by improving the preventive measures in order to further reduce the probability of core melt,
- by simultaneously incorporating, right from the design stage, measures for limiting the consequences of a severe accident.

➡ **A two-fold safety approach against severe accidents:**
- **further reduce their probability by reinforced preventive measures,**
- **drastically limit their potential consequences.**

DESIGN CHOICES FOR REDUCING THE PROBABILITY OF ACCIDENTS LIABLE TO CAUSE CORE MELT

In order to further reduce the probability of core melt, which is already extremely low for the reactors in the current nuclear power plant fleet, the advances made possible with the EPR focus on three areas:

- extension of the range of operating conditions taken into account right from design,
- the choices regarding equipment and systems, in order to reduce the risk of seeing an abnormal situation deteriorate into an accident,
- the advance in reliability of operator action.

Extension of the range of operating conditions taken into account right from design

Provision for the shutdown states in the dimensioning of the protection and safeguard systems

The probabilistic safety assessments highlighted the importance that should be given to the reactor shutdown states. For the EPR, these shutdown states were systematically taken into account, both for the risk analyses and for the dimensioning of the protection and safeguard systems.

The use of the probabilistic safety assessments

Although the EPR safety approach is mainly based on the defense in depth concept (which is part of a deterministic approach), it is reinforced by probabilistic analyses. These make it possible to identify the accident sequences liable to cause core melt or to generate large radioactive releases, to evaluate their probability and to ascertain their potential causes so that they can be remedied. In their large scale right from the design phase, the probabilistic assessments conducted for the EPR constitute a world first. They have been a decisive factor in the technical choices intended to further strengthen the safety level of the EPR.

With the EPR, the probability of an accident leading to core melt, already extremely small with the previous-generation reactors, becomes infinitesimal:

- smaller than 1/100,000 (10^{-5}) per reactor/year, for all types of failure and hazard, which fully meets the objective set for the new nuclear power plants by the International Nuclear Safety Advisory Group (INSAG) with the International Atomic Energy Agency (IAEA) – INSAG 3 report,
- smaller than 1/1,000,000 (10^{-6}) per reactor/year for the events generated inside the plant, making a reduction by a factor 10 compared with the most modern reactors currently in operation,
- smaller than 1/10,000,000 (10^{-7}) per reactor/year for the sequences associated with early loss of the radioactive containment function.

The EPR complies with the safety objectives set up jointly by the French and German safety authorities for future PWR power plants:

➡ **further reduction of core melt probability,**

➡ **practical elimination of accident situations which could lead to large early release of radioactive materials,**

➡ **need for only very limited protective measures in area and time*, in case of a postulated low pressure core melt situation.**

* No permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long-term restriction in the consumption of food.

Greater provision for the risk arising from internal and external hazards

The choices taken for the installation of the safeguard systems and the civil works minimize the risks arising from the various hazards (earthquake, flooding, fire, aircraft crash).

The safeguard systems are designed on the basis of a quadruple redundancy, both for the mechanical and electrical portions and for the I & C. This means that each system is made up of four sub-systems, or “trains”, each one capable by itself of fulfilling the whole of the safeguard function. The four redundant trains are physically separated from each other and geographically shared among four independent divisions (buildings).

Each division includes:

- **for borated water safety injection into the reactor vessel in case of loss of coolant accident, a low-head injection system and its cooling loop, together with a medium-head injection system,**
- **a steam generator emergency feedwater system,**
- **the electrical systems and I & C linked to these systems.**

The building housing the reactor, the building in which the spent fuel is interim-stored, and the four buildings corresponding to the four divisions of the safeguard systems, are given special protection against externally-generated hazards such as earthquakes and explosions.

This protection is further strengthened against an airplane crash. The reactor building is covered with a double concrete shell: an outer shell made of 1.30 m thick reinforced concrete and an inner shell made of pre-stressed concrete and also 1.30 m thick which is internally covered with a 6 mm thick metallic liner. The thickness and the reinforcement of the outer shell on its own have sufficient strength to absorb the impact of a military or large commercial aircraft. The double concrete wall protection is extended to the fuel building, two of the four buildings dedicated to the safeguard systems, the main control room and the remote shutdown station which would be used in a state of emergency.

The other two buildings dedicated to the safeguard systems, those which are not protected by the double wall, are remote from each other and separated by the reactor building, which shelters them from simultaneous damage. In this way, should an aircraft crash occur, at least three of the four divisions of the safeguard systems would be preserved.

The choices regarding the equipment and systems, in order to reduce the risk of an abnormal situation deteriorating into an accident

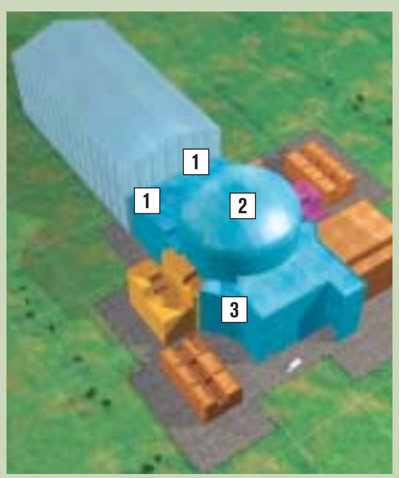
Elimination of the risk of a large reactor coolant pipe break

The reactor coolant system design, the use of forged pipes and components, construction with high mechanical performance materials, combined with the measures taken to detect leaks at the earliest time and to promote in-service inspections, practically rule out any risk of large pipe rupture.

The major safety systems comprise four sub-systems or trains, each capable of performing the entire safety function on its own. There is one train in each of the four safeguard buildings (1) surrounding the reactor building (2) to prevent common-mode failure of the trains.

➨ **A set of quadruple redundant safeguard systems, with independent and geographically separated trains, minimize consequences of potential internal and external hazards.**

➨ **This protection is even reinforced against the airplane crash risk by the strong double concrete shell implemented to shelter the EPR.**



The outer shell (5) covers the reactor building (2), the spent fuel building (3) and two of the four safeguard buildings (1). The other two safeguard buildings are separated geographically.

The reactor containment building has two walls: an inner prestressed concrete housing (4) internally covered with a metallic liner and an outer reinforced concrete shell (5), both 1.30 m thick.

Optimized management of accidental steam generator tube break

Steam generator tube break is an accident which, if it occurs, leads to a transfer of water and pressure from the primary system to the secondary system. The primary side pressure drop automatically induces a reactor shutdown then, if a given pressure threshold is reached, the activation of the safety injection of water into the reactor vessel. The choice, for the EPR, of a safety injection pressure (medium-head injection) lower than the set pressure of the secondary system safety valves prevents the steam generators from filling up with water in such a case. This has a dual advantage: it avoids the production of liquid releases and considerably reduces the risk of a secondary safety valve locking in open position.

Simplification of the safety systems and optimization of their redundancy and diversification

The safety-important systems and their support systems are – as already set out – quadrupled, each featuring four trains shared among four separate divisions.

The structure of these systems is straightforward and minimizes the changes that have to be made to their configuration depending on whether the reactor is at power or in shutdown; the design of the EPR safety injection system and residual heat removal system is an illustration of this.

The safety injection system, which would be activated in case of a loss of coolant accident, is designed to inject water into the reactor core to cool it down. In a first phase, water would be injected into the core via the cold legs of the reactor coolant system loops (legs located between the reactor coolant pumps and the reactor vessel). In the longer term, the water would be simultaneously injected via the cold and hot legs (legs located between the steam generators and the reactor vessel). The water reserve intended to feed the safety injection system is located on the inside and at the bottom of the reactor containment, and the injection pumps only take suction from this reserve. Therefore, there is no need (compared to previous designs) for switching over from a so-called “direct injection” phase to a “recirculation” phase. The EPR safety injection system is equipped with heat exchangers in its low-head portion, to be capable of ensuring core cooling on its own. The EPR is further equipped with a severe accident dedicated system for cooling the inside of the reactor containment, which would be only activated in the eventuality of an accident leading to core melt.

Residual heat removal is provided by the four trains of the low head portion of the safety injection system, which are then configured to remove the residual heat in closed loop (suction via the hot legs, discharge via the cold legs). Safety injection remains available for action in the eventuality of a leak or break occurring on the reactor coolant system.

➨ **The safety-related systems are simple, redundant and diversified to ensure reliability and efficiency.**

Increased reliability of operator action

Extension of action times available to the operator

The protection and safeguard actions needed in the short term in the eventuality of an incident or accident are automated. Operator action is not required before 30 minutes for an action taken in the control room, or one hour for an action performed locally on the plant.

The increase in the volumes of the major components (reactor pressure vessel, steam generators, pressurizer) gives the reactor extra inertia which helps to extend the time available to the operators to initiate the first actions.

Increased performance of the Man-Machine Interface

The progress accomplished in the digital I & C field and the analysis of the experience feedback from the design and operation of the N4 reactors, among the first plants to be equipped with a fully-computerized control room, have conferred on the EPR a high-performance, reliable and optimized solution in terms of Man-Machine Interface. The quality and relevance of the summary data on the reactor and plant status made available in real time to the operators further boost the reliability of their actions.

Computer-generated image of the EPR control room.

➨ **Design of components, high degree of automation, advanced solutions for I & C and Man-Machine Interface combine to further add to reliability of operator actions.**

DESIGN CHOICES FOR LIMITING THE CONSEQUENCES OF A SEVERE ACCIDENT

➡ **Although highly unlikely, a core melt accident would cause only very limited off-site measures in time and space.**

In response to the new safety model for the future nuclear power plants, introduced as early as 1993 by the French and German safety authorities, the plant design must be such that a core melt accident, although highly unlikely, causes only very limited off-site measures in time and space.

The policy of mitigation of the consequences of a severe accident, which guided the design of the EPR, therefore aimed to:

- **practically eliminate the situations which could lead to early important radiological releases, such as:**
  - **high-pressure core melt,**
  - **high-energy corium/water interaction,**
  - **Hydrogen detonation inside the reactor containment,**
  - **containment by-pass,**
- **ensure the integrity of the reactor containment, even in the eventuality of a low-pressure core melt followed by ex-vessel progression, through:**
  - **retention and stabilization of the corium inside the containment,**
  - **cooling of the corium.**

➡ **Practically, situations which could generate a significant radioactivity release are eliminated.**

Prevention of high-pressure core melt

In addition to the usual reactor coolant system depressurization systems on the other reactors, the EPR is equipped with valves dedicated to preventing high-pressure core melt in the eventuality of a severe accident. These valves would then ensure fast depressurization, even in the event of failure of the pressurizer relief lines.

Controlled by the operator, they are designed to safely remain in open position after their first actuation.

Their relieving capacity guarantees fast primary depressurization down to values of a few bars, precluding any risk of containment pressurization through dispersion of corium debris in the event of vessel rupture.

➡ **Even in case of extremely unlikely core melt accident with piercing of the reactor pressure vessel, the melted core and radioactive products would remain confined inside the reactor building whose integrity would be ensured in the long term.**

Prevention of high-energy corium/water interaction

The high mechanical strength of the reactor vessel is sufficient to rule out its damage by any reaction, even high-energy, which could occur on the inside between corium* and coolant.

The portions of the containment with which the corium would come in contact in the eventuality of a core melt exacerbated by ex-vessel progression – namely the reactor pit and the core spreading area – are kept “dry” (free of water) in normal operation. Only when it is spread inside the dedicated area, therefore already partially cooled, surface-solidified and less reactive, would the corium be brought into contact with the limited water flow intended to cool it down further.

*Corium: product which would result from the melting of the core components and their interaction with the structures they would meet.

Containment design with respect to the Hydrogen risk

In the unlikely case of a severe accident, Hydrogen would be released in large quantities inside the containment. This would happen first of all by reaction between the coolant and the Zirconium which is part of the composition of the fuel assembly claddings, then, in the event of core melt and ex-vessel progression, by reaction between the corium and the concrete of the corium spreading and cooling area.

For this reason, the pre-stressed concrete inner shell of the containment is designed to withstand the pressure which could result from the combustion of this Hydrogen. Further, devices called catalytic Hydrogen recombiners are installed inside the containment to keep the average concentration below 10% at all times, to avoid any risk of detonation. Besides, the pressure in the containment does not exced 5.5 bar, assuming an Hydrogen deflagration.

Corium retention and stabilization aiming to protect the base mat

The reactor pit is designed to collect the corium in case of ex-vessel progression and to transfer it to the corium spreading and cooling area. The reactor pit surface is protected by “sacrificial” concrete which is backed-up by a protective layer consisting of zirconia-type refractory material.

In the event of core meltdown, molten core escaping from the reactor vessel would be passively collected and retained, then cooled in a specific area inside the reactor building.

The dedicated corium spreading and cooling area is a core-catcher equipped with a solid metal structure and covered with “sacrificial” concrete. It aims to protect the nuclear island basemat from any damage, its lower section features cooling channels in which water circulates. The aim of its large spreading surface area (170 m²) is to promote the cooling of the corium.

The transfer of the corium from the reactor pit to the spreading area would be initiated by a passive device: a steel “plug” melting under the effect of the heat from the corium.

After spreading, the flooding of the corium would also be initiated by a passive fusible plug-based device. It would then be cooled, still passively, by gravity injection of water from the tank located inside the containment and by evaporation.

The effectiveness of the cooling would then provide stabilization of the corium in a few hours and its complete solidification in a few days.

Containment heat removal system and long-term residual heat removal device

In the eventuality of a severe accident, to prevent the containment from losing its long-term integrity, means would have to be provided to control the pressure inside the containment and to stop it from rising under the effect of residual heat. A dedicated dual-train spray system with heat-exchangers and dedicated heat sink is provided to fulfil this function. A long time period would be available for the deployment of this system by the operators: at least 12 hours owing to the large volume of the containment (80,000 m³).

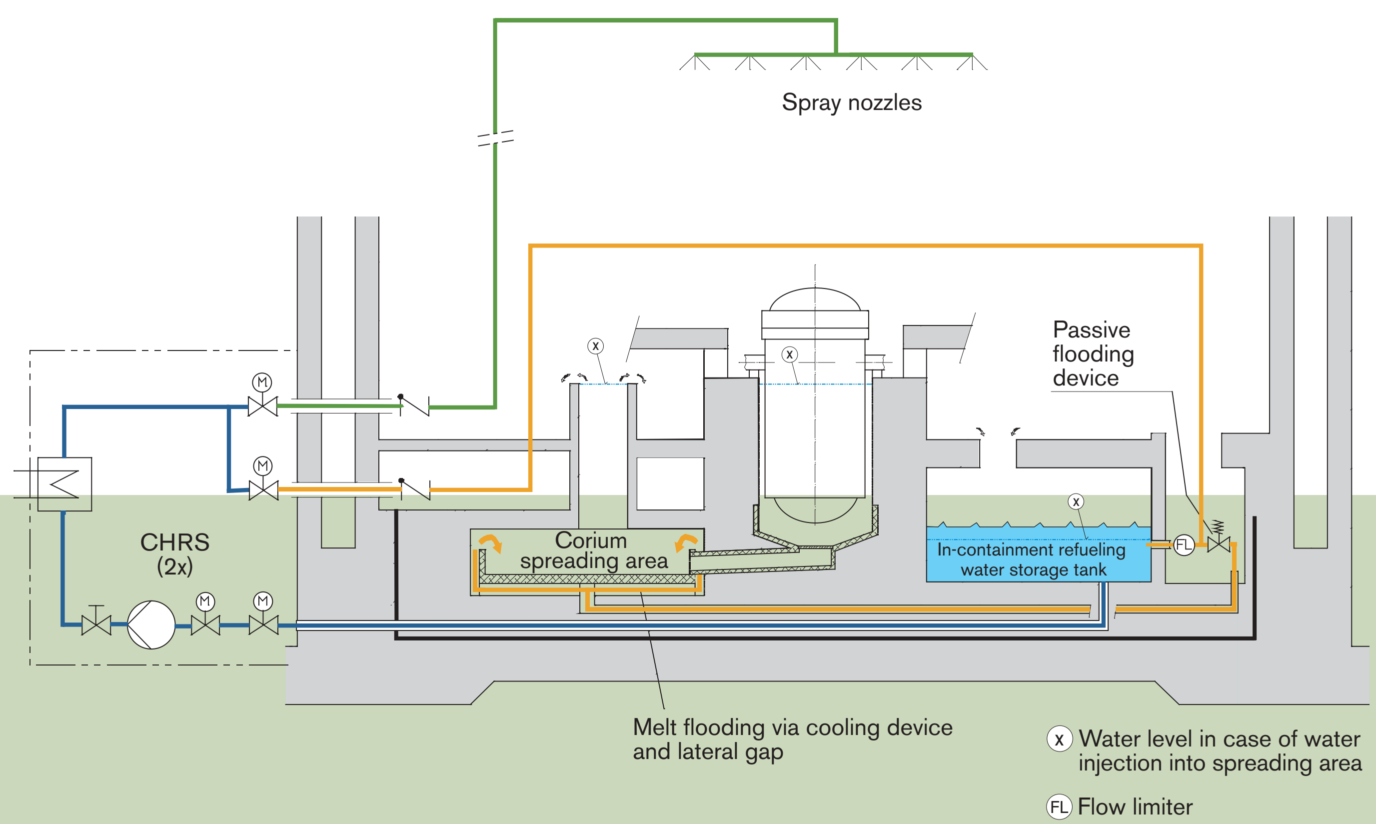
A second mode of operation of the containment heat removal system enables to feed water directly into the core-catcher, instead of into the spray system.

Collection of inter-containment leaks

In the eventuality of a core melt leading to vessel failure, the containment remains the last of the three containment barriers; this means that provisions must be taken to make sure that it remains undamaged and leak-tight. For the EPR, the following measures have been adopted:

- a 6 mm thick metal liner internally covers the pre-stressed concrete inner shell,
- the internal containment penetrations are equipped with redundant isolation valves and leak recovery devices to avoid any containment bypass,
- the architecture of the peripheral buildings and the sealing systems of the penetrations rule out any risk of direct leakage from the inner containment to the environment,
- the space between the inner and outer shells of the containment is passively kept at slight negative pressure to enable the leaks to collect there,
- these provisions are supplemented by a containment ventilation system and a filter system upstream of the stack.

Containment heat removal system



EPR CONSTRUCTION

Emsland nuclear power plant, Germany (KONVOI, 1,300 MWe).

EPR CONSTRUCTION TIME SCHEDULE

The evolutionary approach adopted for the EPR allows its construction schedule to benefit from vast construction experience feedback and from the continuous improvement process of the methodologies and tasks sequencing implemented by Framatome ANP worldwide.
Provisions have been made in the design, construction, erection and commissioning methods to further shorten the EPR construction schedule as far as possible. Significant examples can be given as follows.

DESIGN FEATURES

The general layout of the main safety systems in four trains housed in four separate buildings simplifies, facilitates and shortens performance of the erection tasks for all work disciplines.

Location of electromechanical equipment at low levels means that it can be erected very early on in the program, thus shortening the critical path of the construction schedule.

CONSTRUCTION AND ERECTION METHODS

Three main principles are applied to the EPR construction and erection: minimization of the interfaces between civil works and erection of mechanical components, modularization and piping prefabrication.

Minimization of the interfaces between civil works and erection. The on-going search for the optimization of interfaces between civil and erection works results in the implementation of a construction methodology "per level" or "grouped levels" enabling equipment and system erection work at level "N", finishing construction works at level "N+1" and main construction work at levels "N + 2" and "N + 3" to be carried out simultaneously; this methodology is used for all the different buildings except for the reactor building, where it cannot apply.

Use of modularization for overall schedule optimization. Modularization techniques are systematically considered, but retained only in cases where they offer a real benefit to the optimization of the overall construction schedule without inducing a technical and financial burden due to advanced detailed design, procurement or prefabrication. This approach enables the site preparation schedule to be optimized, delays investment costs with regard to start of operation, and so offers financial savings.

For instance, modules are mainly implemented for the civil works of the reactor building, such as the reactor pit, the internal structures and the containment dome, as well as for the structures of the reactor building (and fuel building) pools, as they are all on the critical path for the construction of the reactor building.

Maximization of piping and support prefabrication. Piping and support prefabrication is maximized in order to minimize erection man-hours and especially welding and controls at erection places; this measure also results in an even better quality of the piping spools with lower cost.

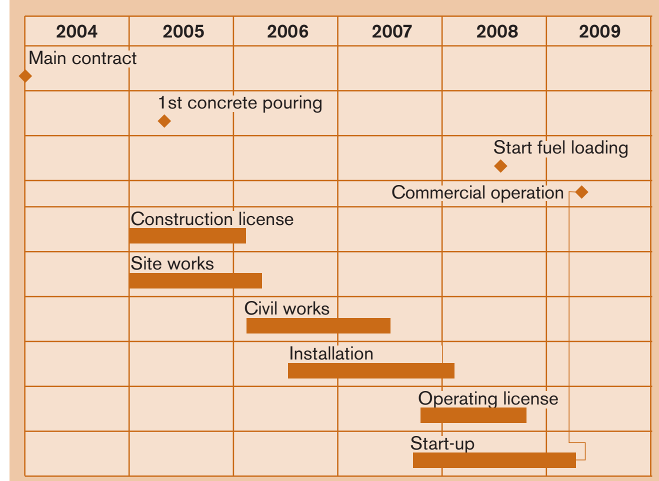
COMMISSIONING TESTS

As with the interfaces between civil and erection works, the interfaces between erection and tests have been carefully reviewed and optimized. For instance, teams in charge of commissioning tests are involved in the finishing works, flushing and conformity checks of the systems, so that these activities are only carried out once.

Instrumentation & Control factory acceptance tests are carried out on a single test platform with all cabinets interconnected, which ensures a shorter on-site test period together with improved overall quality.

The benefits drawn from the unique experience feedback gained from Framatome ANP's past achievements, associated with the systematic analysis of possible improvements and optimization of construction, erection and test activities together with their interfaces, results in an optimal technical and economical construction schedule for the implementation of the EPR projects. This experience, and current EPR projects provide confidence that the EPR schedule is actually feasible and a reality.

The short Olkiluoto 3 construction time-schedule, adapted to this particular project, is provided below as an illustration.



➨ **The overall construction schedule of a new unit depends largely on site conditions, industrial organization and policies, and local working conditions. So accurate figures are valid only for the specific project to which they are related.**

PLANT OPERATION, MAINTENANCE & SERVICES

Neckarwestheim nuclear power plant (Germany): unit 2 (right foreground) is of the KONVOI type (1,300 MWe).

PLANT OPERATION, MAINTENANCE & SERVICES

From the beginning, the EPR and its equipment and systems have been designed to allow for efficient refueling outages and to simplify and optimize inspection and maintenance in order to increase plant availability and reduce maintenance costs, two major objectives of plant operators worldwide to meet the demands of more and more competitive power markets.

A 92% AVAILABILITY FACTOR OVER THE ENTIRE PLANT LIFE

Regarding availability, the EPR is designed to reach 92% over the entire 60 years of its design lifetime. This is made possible by short-scheduled outages for fuel loading/unloading and in-service inspections and maintenance, and also through reduced downtimes attributable to unscheduled outages.

The high degree of equipment reliability on the one hand, and the decrease in reactor trip causes (in particular due to the deployment of the limitation system related to reactor operation) on the other hand lead to an unscheduled unavailability not exceeding 2%.

The quadruple redundancy of the safeguard systems allows a large part of the preventive maintenance operations to be performed while the reactor is at power.

Chooz B1, France (N4, 1,500 MWe): removal of the hydraulic section of a reactor coolant pump for maintenance.

Moreover, the reactor building is designed to be accessible, under standard safety and radiation protection conditions, while the reactor is at power. This enables the outage and maintenance operations to be prepared and demobilized with no loss of availability. This possibility of access with the reactor on line also facilitates field services which could be needed outside scheduled outage periods. Based on experience feedback, standardization and ease of access of the components of the reactor allow simple and rapid performance of inspection and maintenance work.

Access to the reactor building during power operation allows to start preventive maintenance and refueling tasks up to seven days before reactor shutdown and to continue their demobilization up to three days after reactor restart.

The duration of the plant shutdown phase is reduced by a time gain for reactor coolant system cooldown, depressurization and vessel head opening. Similarly the length of the restart phase is reduced as well and benefits from the reduction in the time needed to run the beginning-of-cycle core physics tests (gain supplied by the "aeroball" in-core instrumentation system). Durations of about 70 and 90 hours are respectively scheduled for the shutdown and restart phases. For the fuel loading/unloading operations, a time period of about 80 hours is scheduled.

➡ **Duration of a regular outage for preventive maintenance and refueling is reduced to 16 days. Duration of an outage for refueling only does not exceed 12 days. Decennial outages for main equipment in-service inspection, turbine overhaul and containment pressure test are planned with a 38-day duration.**

The EPR is designed to:

- ➡ **maximize plant availability and maneuverability,**
- ➡ **ease operation and maintenance and reduce their costs,**
- ➡ **enhance radiological protection of the personnel,**
- ➡ **protect the environment and contribute to a sustainable development.**

A HIGH LEVEL OF OPERATIONAL MANEUVERABILITY

In terms of operation, the EPR is designed to offer the utilities a high level of maneuverability. It has the capacity to be permanently operated at any power level between 20 and 100% of its nominal power in a fully automatic way, with the primary and secondary frequency controls in operation.

The EPR capability regarding maneuverability is a particularly well adapted response to scheduled and unscheduled power grid demands for load variations, managing of grid perturbations or mitigation of grid failures.

AN ENHANCED RADIOLOGICAL PROTECTION

Allowance for operating constraints and for the human factor, with the aim of improving worker radiation protection and limiting radioactive releases, together with radwaste quantity and activity, was a set objective as soon as EPR design got underway. For this purpose, the designers drew heavily upon the experience feedback from the operation of the French and German nuclear power plant fleets.

Accordingly, major progress has been made, particularly in the following areas:
- the choice of materials, for example the optimization of the quantity and location of the Cobalt-containing materials and liners, in order to obtain a gain on the Cobalt 60 “source term”,
- the choices regarding the design and layout of the components and systems liable to convey radioactivity, taking into account the various plant operating states,
- the optimization of the radiation shielding thicknesses in response to forecast reactor maintenance during outages or in service.

Thanks to these significant advances, collective doses less than 0.4 Man.Sievert per reactor/year can be expected for operation and maintenance staff (to date, for the major nuclear power plant fleets of OECD countries like France, Germany, the United States and Japan, the average collective dose observed is about 1 Man.Sievert per reactor/year).

PLANT SERVICES

Optimization of plant processes and implementation of innovative maintenance technologies and concepts are also significant contributors to the achieving of operators’ cost and availability objectives. In this area, Framatome ANP, an AREVA and Siemens company, supplies the most comprehensive range of nuclear services and technologies in the world.

Thanks to its experience from designing and constructing 96 nuclear power plants worldwide, its global network of maintenance and services centers with highly trained teams (more than 3,000 specialists mainly based in France, Germany and the USA) committed to excellence, Framatome ANP provides a full range of inspection, repair and maintenance services for all types of nuclear power plants, based on the most advanced techniques available today. Its field of expertise covers the whole scope of customers’ needs from unique one-of-a-kind assignments to the implementation of integrated service packages.

Framatome ANP’s offer of power plant services encompasses:
- in-service inspection and non destructive testing,
- outage services,
- component repair and replacement (including steam generators, reactor pressure vessel heads),
- supply of spare parts,
- off-site maintenance of components in “hot” workshops,
- fuel inspection, repair and management,
- services in the fields of instrumentation and diagnosis, I & C and electrical systems, chemistry,
- plant engineering and plant upgrading,
- plant decommissioning and waste management,
- training of operating personnel,
- expert consultancy.

The Framatome Owners Group network (FROG) offers member electricity companies a cost-effective means for exchange of information and experience. FROG’s members have access to broad operational and maintenance feedback. They also benefit from the results of study programs jointly decided to deal with issues of shared interest.

In-service inspection machine for ultrasonic testing of reactor pressure vessels.

Operators have developed ambitious outage optimization plans to decrease outage duration. Their objectives are even more ambitions and include plant upgrades and component replacement for life extension of plant operation. Aware of the strategic importance of the operators’ goal of reducing outage duration, Framatome ANP has created an International Outage Optimization Team that spans all regions and capabilities of the company for customer benefit in terms of quality, safety and costs.

FRAMATOME ANP’S SPIRIT OF SERVICE

➨ **To satisfy customers and help them to succeed in a highly competitive energy market, by:**
- **reducing operating and maintenance costs,**
- **improving safety and performance,**
- **extending plant life,**
- **reducing radiation exposure.**

CONTINUOUSLY IMPROVING SERVICE TO CUSTOMERS

To continuously improve service to customers, with particular attention to respect of local cultures and practices, especially in geographical areas outside its European and American bases, Framatome ANP has established special links and partnerships with entities well positioned to locally propose and perform power plant services. A significant illustration is the company’s long-lasting and successful cooperation with Chinese companies and institutes involved in the extensive long-term nuclear program currently underway in China. An excellent example of this cooperation is the tight links with the ShenZhen Nuclear Company Ltd (SNE), which is mainly engaged in maintenance and refueling outages of commercial power stations in China and has also diversified its activities to cover other industrial projects.

SNE was created in the Guangdong province at the end of 1998. Since July 2003, SNE is a joint venture between Company 23 of China Nuclear Engineering and Construction Corporation (CNEC) and Framatome ANP, which fully benefits from Framatome ANP’s expertise and technologies in its activity field.

Framatome ANP Technical Center (TC), with its locations in France, Germany and the USA, is the first link for the development of new technologies. A major objective of the TC is to provide support in solving technical issues in specific fields. More than 300 scientific engineers and technicians work in the TC laboratories which are equipped with the most up-to-date technology and test loops. Their fields of excellence cover material engineering, welding, chemistry and radiochemistry, corrosion, non-destructive examination, thermal-hydraulics and fluid dynamics, testing of components and systems, manufacture of special components.

FRAMATOME ANP’S COMMITMENT

➨ **Flexibility to accommodate customers’ needs, cultures and practices, through:**
- **optimized organization and processes,**
- **consolidation of expertise and experience,**
- **rapid mobilization of skilled and highly qualified multi-cultural teams,**
- **technical and contractual innovation,**
- **partnerships with customers and local service partners.**

FROG: THE FRAMATOME OWNERS GROUP

The FROG, the Framatome Owners Group, is dedicated to building strong and efficient teaming for mutual cooperation, assistance and sharing of its members’ experience and expertise, to support the safe, reliable, cost-effective operation of its members’ nuclear power units.

The FROG was set up in October 1991 by five utility companies that were either operating or building nuclear power plant units incorporating a Framatome nuclear steam supply system or nuclear island.

These utility companies are Electrabel from Belgium, Electricité de France, Eskom from the Republic of South Africa, GNPJVC from the People’s Republic of China and KHNP from the Republic of Korea.

Later on, Ringhals AB from Sweden (in June 1997), LANPC, owner of the Ling Ao plant in China (in October 2000), British Energy owner of Sizewell B in the United Kingdom (in October 2002) joined the FROG as members. In 2003, GNPJVC and LANPC merged operation of their plants in one company DNMC.

The Owners group provides a forum for its members to share their experiences in all domains of nuclear power plant operation, enabling a cost-effective exchange of information to identify and solve common issues or problems.

Several working groups and technical committees are actively dealing with specific technical and management issues. Among them, a specific Steam Generator Technical Committee, has been formed by utilities having steam generators served by Framatome ANP. Committee participants are the FROG members plus the companies NSP from the USA, NOK from Switzerland and NEK from Slovenia.

> CONCLUDING REMARKS

Let us summarize the advantages offered by the EPR from an electricity utility point of view:

- culminating from the legacy of Western PWR technology,
- evolutionary design, uniquely minimizing design, licensing, construction and operation technical risks and their financial impacts,
- assurance to be backed in the long run by the world's largest company comprising the entire nuclear cycle,
- continuity in the mastery of PWR technology,
- competitiveness in terms of installed kW cost and kWh production cost: a 1,600 MWe-class reactor, with high efficiency, reduced construction time, extended service life, enhanced and more flexible fuel utilization, increased availability,
- safety:
  - heightened protection against accidents, including core meltdown, and their radiological consequences,
  - robustness against external hazards, in particular airplane crash and earthquake,
- optimized operability,
- enhanced radiological protection of operating and maintenance personnel,
- efficiency in the use of nuclear fuel, fostering sustainable development.

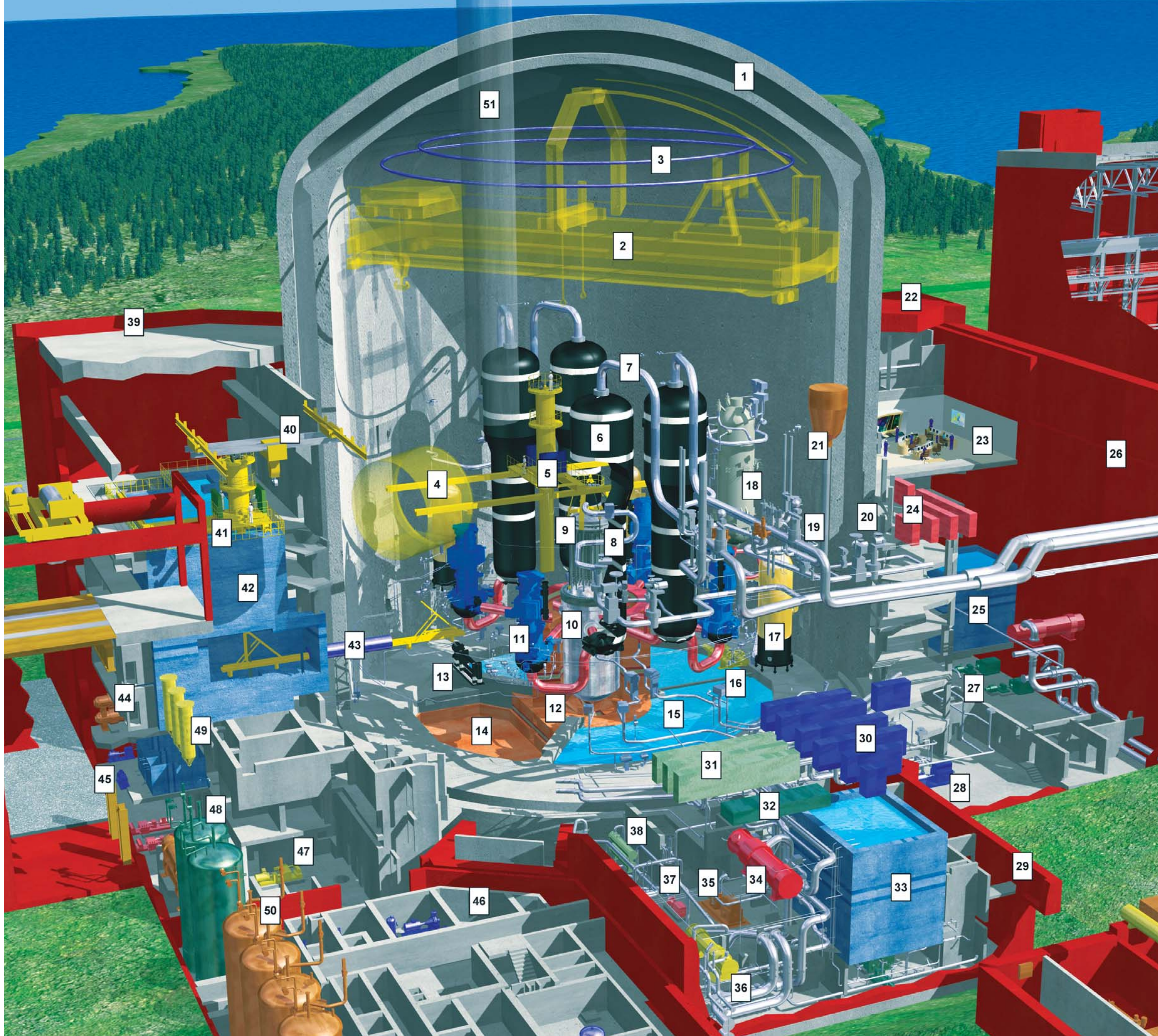
On December 18, 2003, the Finnish electricity utility, Teollisuuden Voima Oy (TVO) signed a contract with the consortium set up by AREVA and Siemens for the construction of the Olkiluoto 3 EPR in Finland.

This first EPR is scheduled to start commercial operation in 2009.

> EPR

Key to power station cutaway

1 **Reactor building: inner and outer shell**
2 Polar crane
3 Ultimate heat removal system: sprinklers
4 Equipment hatch
5 Refueling machine
6 Steam generator
7 Main steam lines
8 Main feedwater lines
9 Control rod drives
10 Reactor pressure vessel
11 Reactor coolant pump
12 Reactor coolant piping
13 CVCS heat exchanger
14 Corium spreading area
15 In-containment refueling water storage tank
16 Residual heat removal system, heat exchanger
17 Safety injection accumulator tank
18 Pressurizer
19 Main steam valves
20 Feedwater valves
21 Main steam safety and relief valve exhaust silencer
22 **Safeguard building division 2**
23 Main control room
24 Computer room
25 Demineralized water pool, division 2
26 **Safeguard building, division 3**
27 Emergency feedwater pump, division 3
28 Medium head safety injection pump, division 3
29 **Safeguard building, division 4**
30 Switchgear, division 4
31 I & C cabinets
32 Battery rooms, division 4
33 Demineralized water pool, division 4
34 CCWS heat exchanger, division 4
35 Low head safety injection pump, division 4
36 Component cooling water surge tank, division 4
37 Ultimate heat removal system pump, division 4
38 Ultimate heat removal system heat exchanger, division 4
39 **Fuel building**
40 Fuel building crane
41 Spent fuel pool bridge
42 Spent fuel pool and fuel transfer pool
43 Fuel transfer tube
44 Spent fuel pool cooler
45 Spent fuel pool cooling pump
46 **Nuclear auxiliary building**
47 CVCS pump
48 Boric acid tank
49 Delay bed
50 Coolant storage tank
51 Vent stack



With manufacturing facilities in over 40 countries and a sales network in over 100, AREVA offers customers technological solutions for nuclear power generation and electricity transmission and distribution.

The group also provides interconnect systems to the telecommunications, computer and automotive markets.

These businesses engage AREVA's 70,000 employees in the 21st century's greatest challenges: making energy and communication resources available to all, protecting the planet, and acting responsibly towards future generations.

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