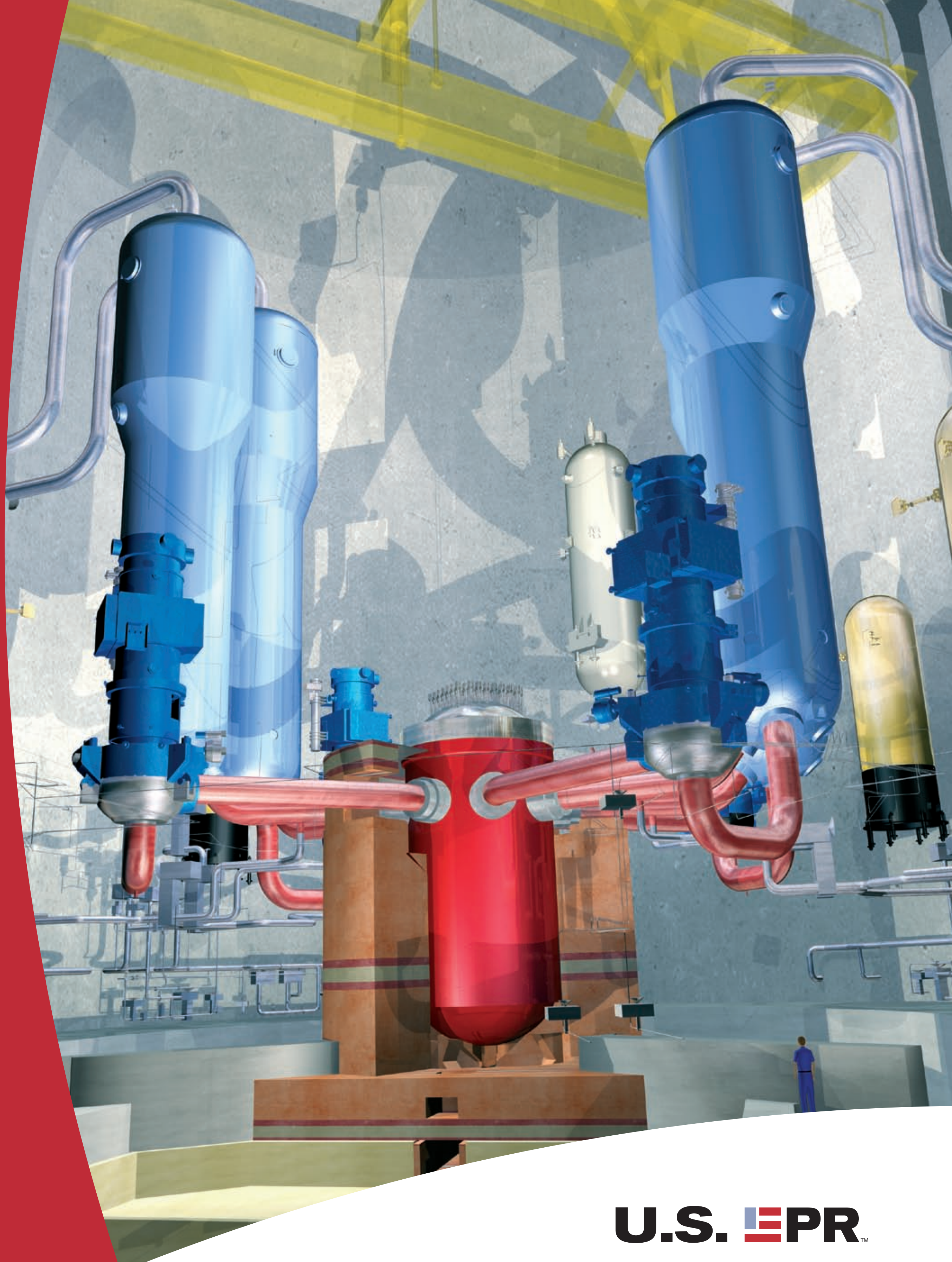


U.S. EPR™

AREVA

> FOREWORD

Mature design from a proven supplier – for the next 60 years

Security of energy supply. Long-term stability of energy costs. The need to combat the greenhouse effect and potential global warming. All of these factors argue in favor of greater diversity in energy supplies. Nuclear power has a vital role to play as a safe, economically competitive, reliable and environmentally friendly energy source.

As a world expert in energy, AREVA creates and offers solutions to generate, transmit and distribute electricity. AREVA is the largest company in the world comprising the entire nuclear cycle: the front end (Uranium ore mining, conversion and enrichment, and fuel fabrication), reactor design and construction, reactor services, the back end of the fuel cycle (spent fuel management), and transmission and distribution of electricity.

Utilities can ensure reliable electric service and meet the growing demands for power in their service areas by adding the U.S. EPR to their generation portfolio. The U.S. EPR (Evolutionary Power Reactor) is a large, advanced reactor of the pressurized water reactor (PWR) type offered by AREVA NP to provide the most predictable path for new baseload generation that is cost-effective, safe and environmentally conscious.

The EPR's key assets to support a strategic choice

An evolutionary, safe and innovative design that is part of a standardized global fleet

The U.S. EPR is a 1,600 MWe class PWR plant design based on the European Pressurized Water Reactor (EPR) design being built in Olkiluoto, Finland. While the EPR is a standardized global design, it is being marketed and licensed in the U.S. with a different name to reflect its conversion to meet U.S. codes, standards, regulatory requirements, and U.S. cycle frequency and grid voltages. Thus, when referring to the European plant design being built in Finland and scheduled for construction in France, the term "EPR" is used. "U.S. EPR" refers to the plant design being marketed, certified and licensed in the U.S.

The EPR's 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 operating 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 affiliation, the EPR benefits from an uninterrupted continuum of evolutionary development and innovation that has supported the development of the PWR since its introduction to the Western marketplace in the mid-1950s.

Offering a significantly enhanced level of safety, the EPR features major innovations, especially in 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 earthquakes. The EPR operating and safety systems provide progressive responses commensurate with any abnormal occurrences.

A number of technological advances place the EPR at the forefront of nuclear power plant design. Significant progress has been incorporated into its main features:

- The reactor core and its flexibility 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
- 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 certainty in terms of efficiency, operability, reliability and, therefore, economic competitiveness. The U.S. EPR is an important part of the solution to ensure the integrity of the world's future energy supply.

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 a number of participants 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 AREVA NP)
- 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

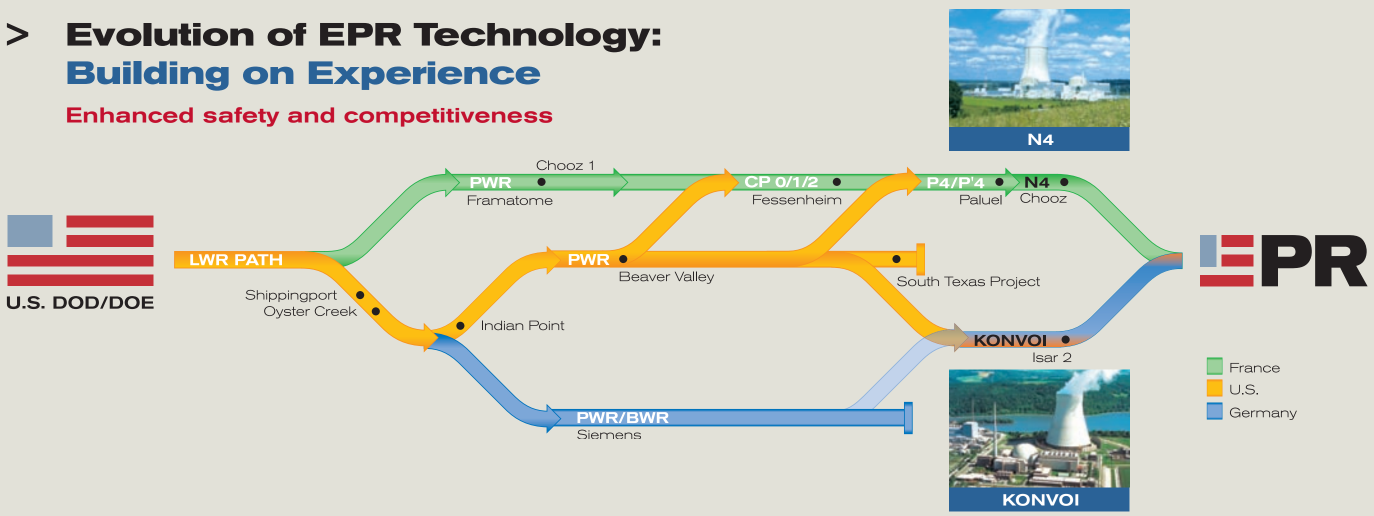
The EPR design takes into account the expectations of utilities as stated by the "European Utility Requirements" (EUR) and the "Utility Requirements Document" (URD) issued by the U.S. Electric Power Research Institute (EPRI). It complies with the 1993 joint recommendations of the French and German safety authorities. 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," the advisory committee for reactor safety to the French safety authority) supported by German experts.

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

On May 4, 2006, the EDF Board of Directors decided to launch the construction of its first EPR unit on the Flamanville site.

> Evolution of EPR Technology: Building on Experience

Enhanced safety and competitiveness



Continuity in technology

The N4 and KONVOI reactors are descendants of the earlier Framatome and Siemens KWU generation reactors, which are themselves derivative of those first implemented in the U.S., then refined and expanded upon by Framatome and Siemens KWU. The EPR is the direct descendant of the well proven N4 and KONVOI reactors, resulting in a fully mastered technology that minimizes risks linked to design, licensing, construction and operation – a unique certainty for EPR customers that carries distinct advantages. First, 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 more easily deployed and utilized to carry out new nuclear plant projects based on EPR technology.

- **U.S. EPR relies on sound, proven technology**
- **EPR complies with safety authorities' requirements for new nuclear plants**
- **Continuous in-house design and manufacturing achieves project optimization**
- **Design and licensing, construction and commissioning, operability and maintainability of all EPR units benefit from AREVA NP's decades of worldwide experience and expertise — uniquely minimizing customers' 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 the design process, the EPR offers significantly reduced power generation costs, estimated to be 10% lower than those of the most modern nuclear units currently in operation. According to the most recent international study, OECD NEA/IEA *Projected Costs of Generating Electricity– 2005 Update*, in which several countries in Europe chose the EPR as the reference model for their future nuclear programs, the average cost of electricity generated by an EPR would be significantly less than that generated using combined cycle gas turbine (CCGT) technology. The cost savings amount to around 20% for a gas price of $4-$6/Mbtu and a weighted average capital cost (WACC) of 8% to 9% in real terms.

This high level of competitiveness is achieved through:

- **Unit power in the 1,600 MWe range (the highest unit power to date), providing an attractive cost of the installed kWe**
- **35% overall efficiency — presently the highest value ever for water reactors**
- **Shortened construction time incorporating experience feedback and continuous improvement of construction methodology and task sequencing**
- **A design objective for a 60-year service life**
- **Enhanced and more flexible fuel utilization**
- **An average availability factor greater than 94% during the entire service life of the plant, obtained through longer fuel cycles, shorter refueling outages and in-operation maintenance**

Significant advances for sustainable development

Due to its optimized core design and higher overall efficiency compared to reactors in operation today, the U.S. EPR also offers many significant advantages in terms of sustainable development:

- **Uses 7% less Uranium/MWh**
- **15% reduction of long-lived actinides generation/MWh**
- **14% gain on the electricity generation vs. thermal release ratio (compared to 1,000 MWe-class reactors)**
- **Great flexibility to use MOX (mixed UO_2-PuO_2) fuel**

> 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 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-fueled plants.

In a pressurized water reactor (PWR) like the EPR, ordinary water is used to remove the heat formed inside the reactor core by the nuclear fission process. This water also slows down (or moderates) neutrons (constituents of atom nuclei 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.

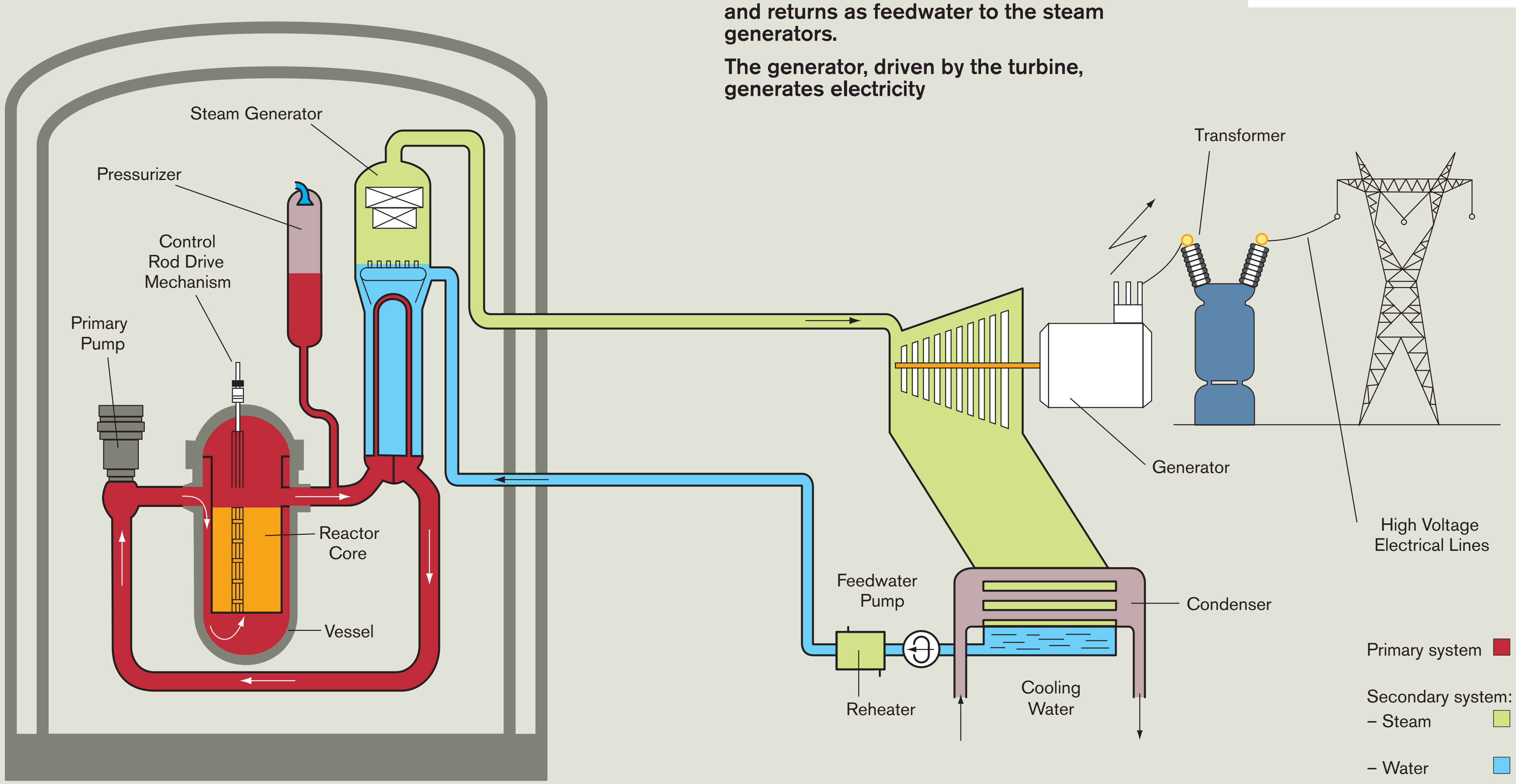
The primary water is pumped through the reactor core and the primary side of the steam generators, in four parallel closed loops, by electric motor-powered coolant pumps. Each loop is equipped with a steam generator and a coolant pump.

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 boils to produce saturated steam. The steam is dried in the steam generators then routed to the turbine to drive it. The steam is then condensed and returns as feedwater to the steam generators.

The generator, driven by the turbine, generates electricity

➡ The following chapters will provide detailed explanation about the U.S. EPR and operation of nuclear power stations based on its design.



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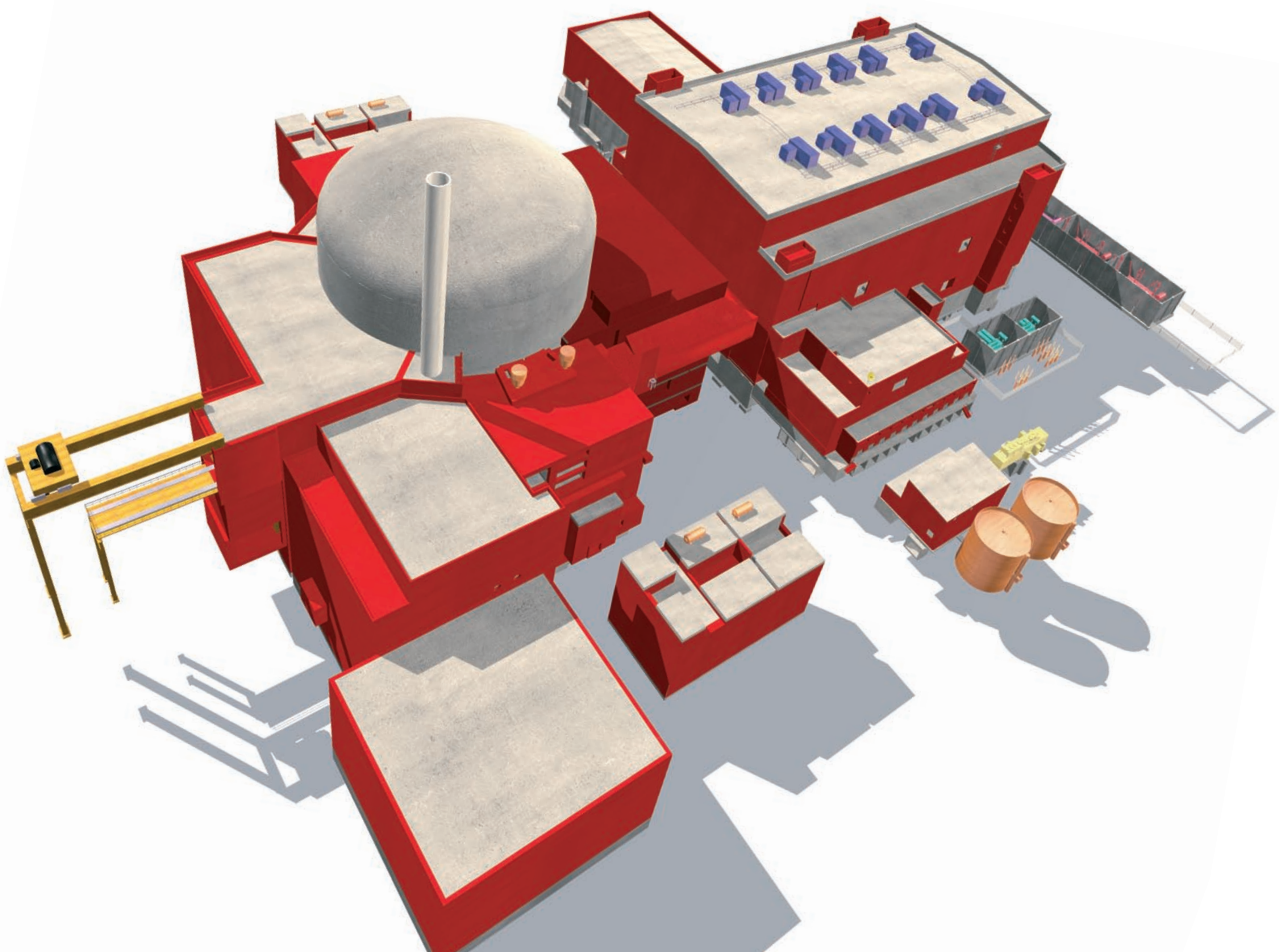


EPR NUCLEAR ISLAND

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

U.S. 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 the consequences of internal and external hazards. It consists of a cylindrical, post-tensioned concrete inner containment with a metallic liner surrounded by an outer reinforced concrete shell.

The NSSS arrangement is characterized by:

- steam generators, reactor coolant pumps and reactor vessel, each in separate compartments;
- concrete walls between the loops and between the hot and cold legs of each loop;
- a pressurizer located in a separate area; and
- a 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 shielding. 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 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 radiological controlled areas, whereas the corresponding Component Cooling and Emergency Feedwater Systems are installed at the outer areas in the non-controlled areas. The main Control Room is in one of the Safeguard Buildings. The safe shutdown facility is in a different safeguard building.

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

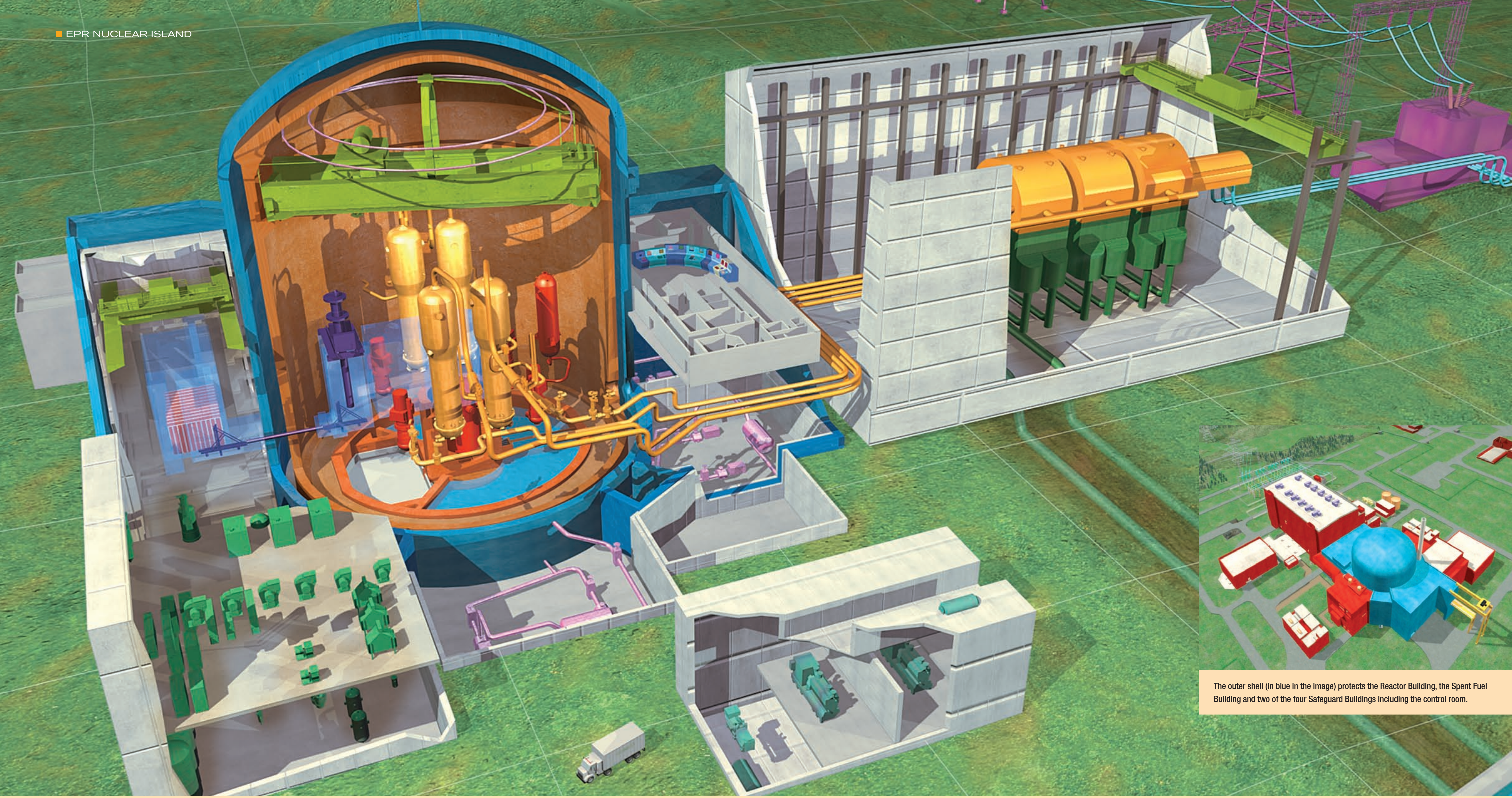
Part of the Nuclear Auxiliary Building (NAB) is designed as a radiological non-controlled area in which parts of the Operational Chilled Water System are located. Special laboratories for sampling systems are at the lowest level. The maintenance area and some setdown areas used during the refueling phase 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 the turbine, the generator set, the condenser and their auxiliary systems.



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.

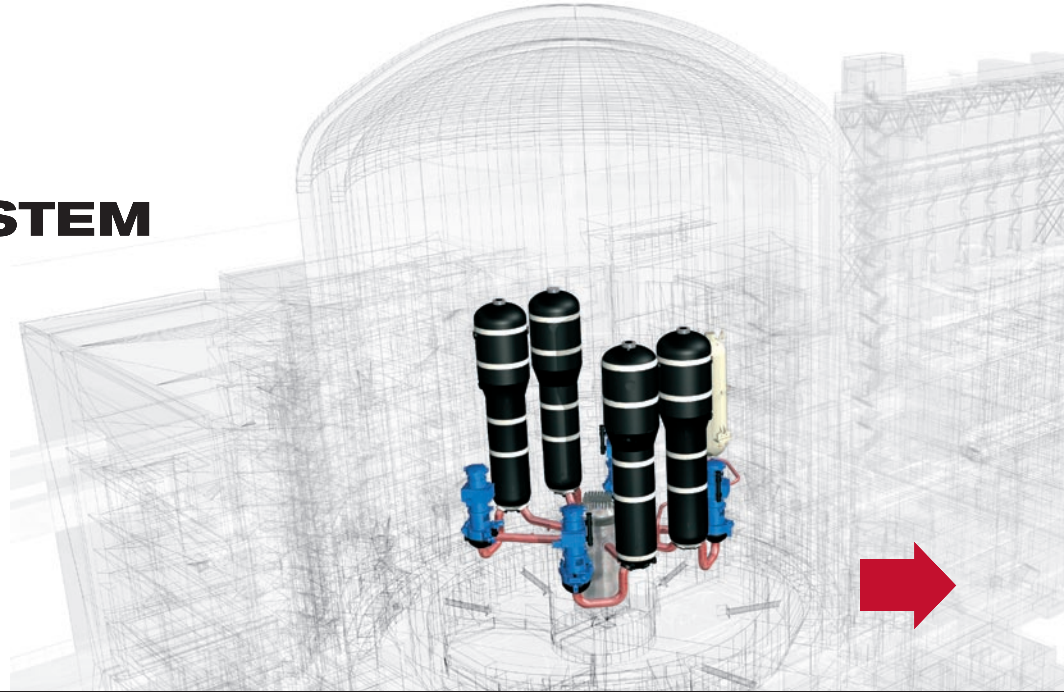
The U.S. EPR layout offers exceptional and unique resistance to external hazards, especially earthquakes and airplane crashes.

- **To withstand a 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 a 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 geographical separation. Similarly, the diesel generators are in two geographically separate buildings to avoid common failures.**

The U.S. EPR Nuclear Island design has clear 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 reduced 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 the well-proven 4-loop design. The French 1,300 MWe and 1,500 MWe N4 reactors and the German KONVOI reactors are of the 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, in which heat is transferred to the secondary circuit. The coolant then goes to a reactor coolant pump before returning to the reactor pressure vessel through an inlet nozzle. Inside the reactor pressure vessel, the coolant flows downward in the annular space between the core barrel and the vessel, then makes a U-turn upward and flows through the core to extract the heat generated by the nuclear fuel.

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

The U.S. EPR's main reactor components – reactor pressure vessel, pressurizer and steam generators – feature larger volumes than similar components from previous designs to provide additional operational and safety margins.

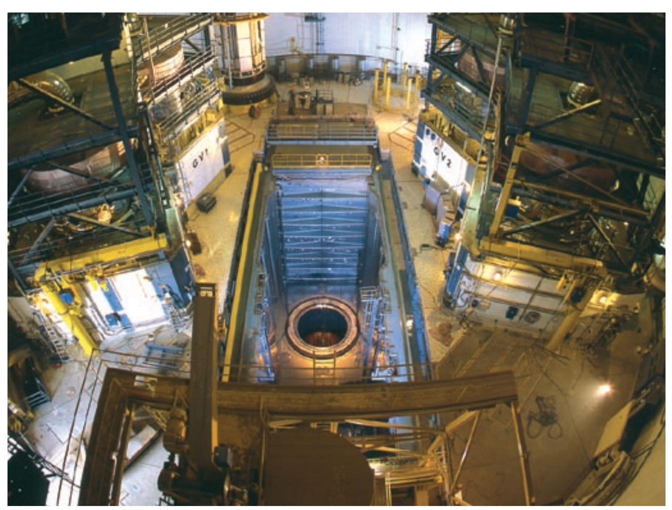
The increased free volume in the reactor pressure vessel (between the nozzles of the reactor coolant lines and the top of the core) provides a higher water volume above the core and thus additional margin with regard to the core "dewatering" time in the event of a postulated loss-of-coolant accident. 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.

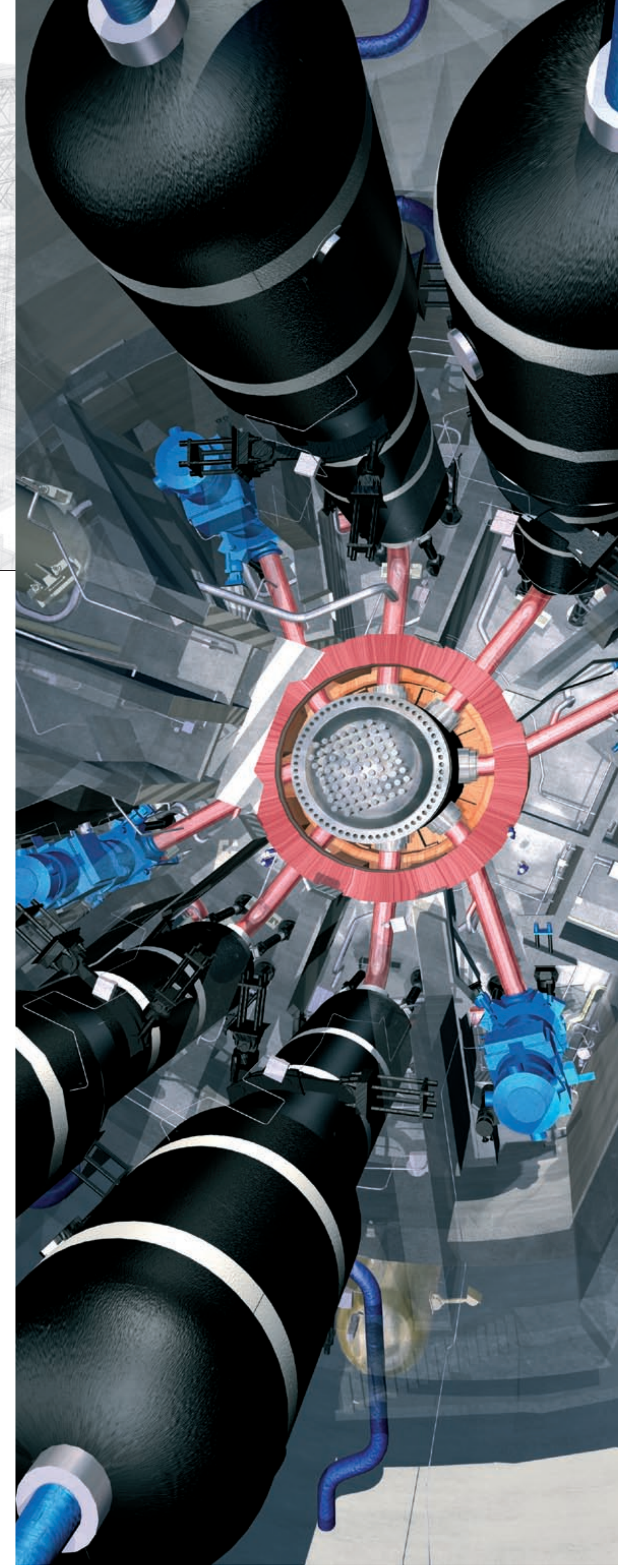
The larger volume of the steam generator secondary side results in an increased secondary water inventory and steam volume, which offers several advantages:
- During normal operation, smooth transients are obtained, reducing the potential for unplanned reactor trips.
- Regarding the management of steam 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.

Integration of design and manufacturing

As the world's largest fully integrated nuclear supplier, AREVA brings heavy component design and manufacturing activities together within one group. The close connection between the two functions is an important asset to project performance that is unique in the nuclear marketplace. This integrated structure, maintained by AREVA NP for many years, is of great advantage to utilities through interaction that optimizes design, manufacturing, scheduling and costs to achieve the best solutions.



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



Computer-generated image of the EPR primary system

CHARACTERISTICS	DATA
Reactor coolant system	
Core thermal power	4,590 MWth
Number of loops	4
Coolant flow per loop	124,730 gpm
Reactor pressure vessel inlet temperature	563.5° F
Reactor pressure vessel outlet temperature	624.5° F
Primary side design pressure	2,550 psia
Primary side operating pressure	2,250 psia
Secondary side	
Secondary side design pressure	1,450 psia
Steam pressure at nominal conditions	1,109 psia
Main steam pressure at hot standby	1,305 psia

FEATURES TO MAXIMIZE PLANT AVAILABILITY

Activation of safety systems

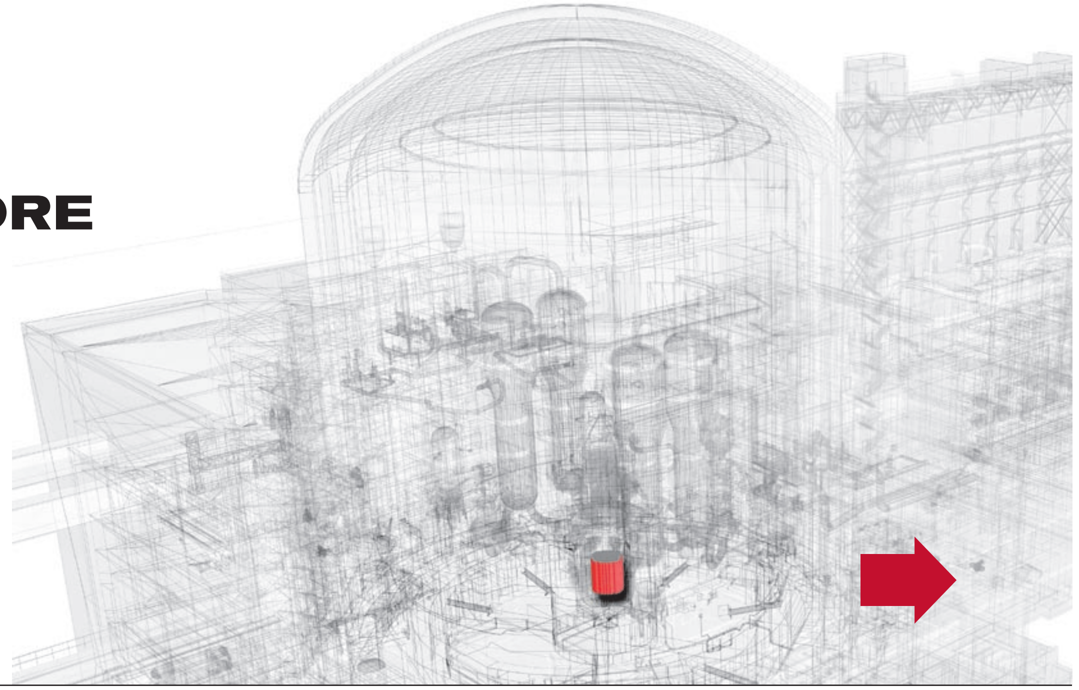
Activation of the safety systems – including safety valves – does not occur prior to reactor trip, which means that best possible use is made of the depressurizing effect of the reactor trip. This approach also ensures maximum safety by minimizing the number of valve activations and the potential for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

Fast reactor power cutback

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 managing many types of transients.**
- **The primary system design pressure has been increased to reduce the likelihood of safety valve actuation.**
- **The management of steam generator tube rupture scenarios prevents any release of contaminated liquid 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 support the fuel assemblies, channel the coolant and guide the control rods that control the fission reaction.

The core is cooled and moderated by light water at a pressure of 2,250 psia and a temperature of approximately 594° F. 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 U.S. 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 U.S. EPR is designed for flexible operation with UO2 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.

Core design analysis

The core design analyses demonstrate the feasibility of different types of fuel management 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

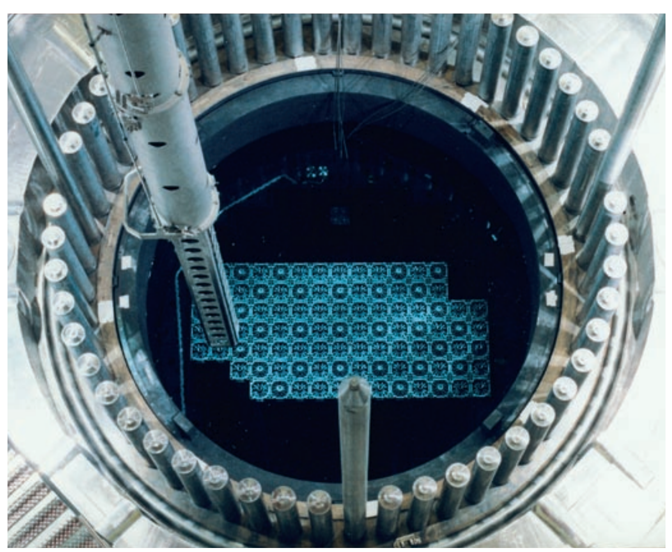
The core power is measured using the ex-core instrumentation, also used to monitor the approach to criticality during start-up.

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 the temperature distribution at the core outlet.

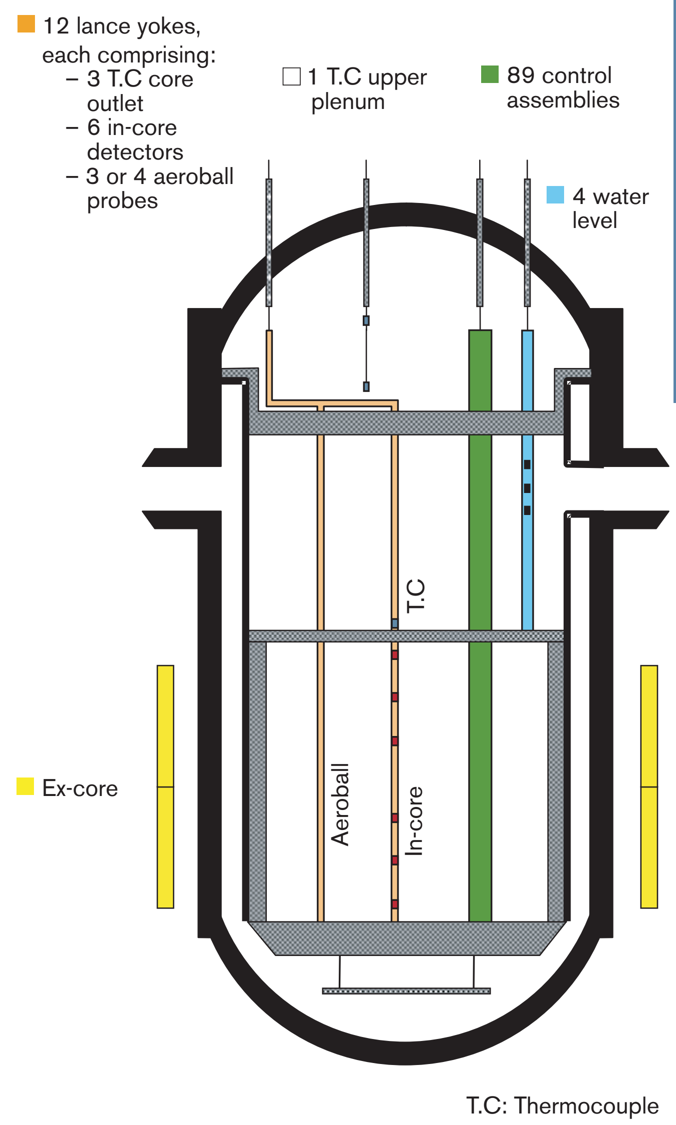
The in-core instrumentation is introduced through the vessel head, leaving the bottom of the reactor pressure vessel free from penetrations.

For additional information, see the "Instrumentation and Control Systems" chapter, page 42.



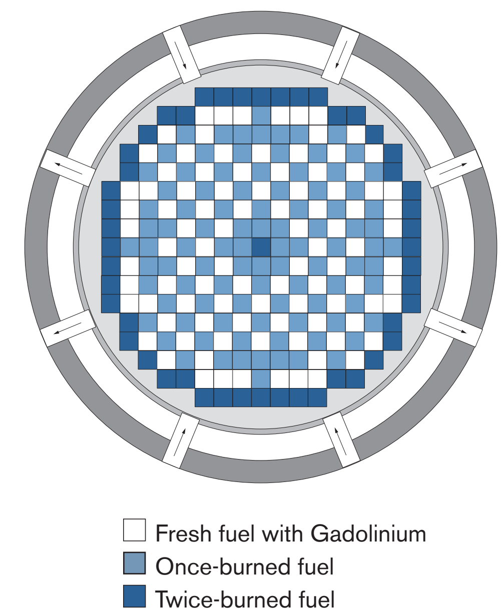
Isar 2 unit, Germany (KONVOI, 1,300 MWe): fuel loading operation.

In-core instrumentation



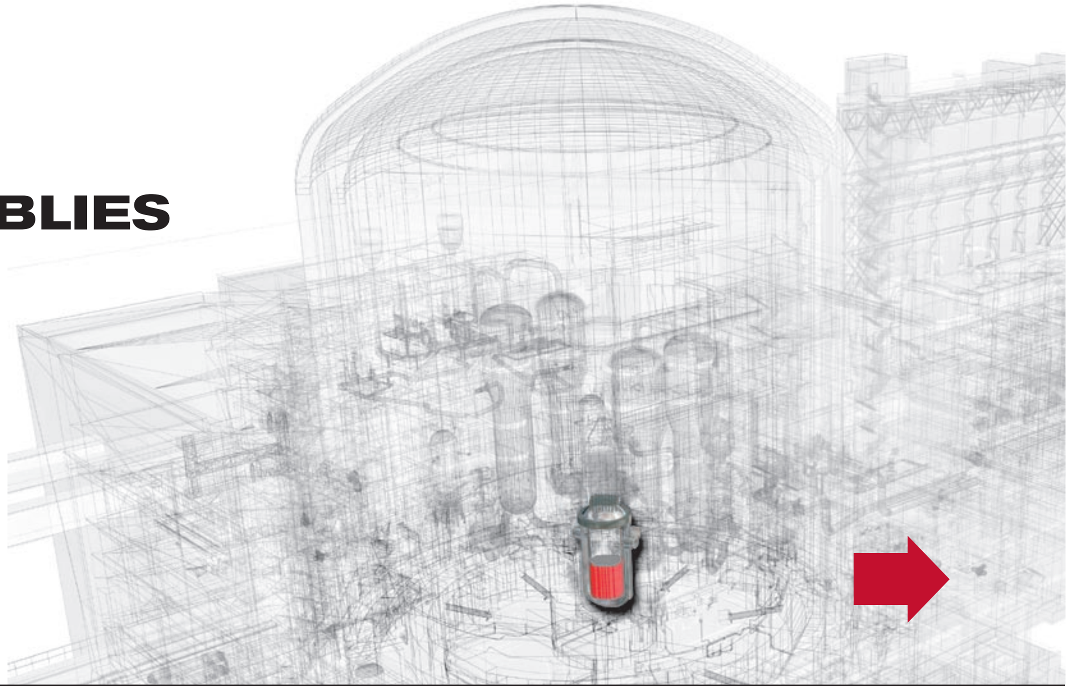
CHARACTERISTICS	DATA
Reactor core	
Thermal power	4,590 MWth
Operating pressure	2,250 psia
Nominal inlet temperature	563.5° F
Nominal outlet temperature	626.5° F
Equivalent diameter	12.4 ft
Active fuel length	13.8 ft
Number of fuel assemblies	241
Number of fuel rods	63,865
Average linear heat rate	5.11kW/ft

Typical core loading



- **The U.S. 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 U.S. EPR core also offers significant advantages in favor of sustainable development:**
  - **7% savings in Uranium consumption per produced MWh**
  - **15% reduction in long-lived actinide generation per MWh**
  - **Great flexibility for using MOX (mixed UO_2-PuO_2) fuel assemblies in the core, thus recycling 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, which consists of a bottom and a top nozzle, 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 in a square lattice with a 17x17 array. Twenty-four 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 (RCCAs) and, when required, for fixed or moveable in-core instrumentation and neutron source assemblies. The bottom nozzle is shaped to direct and help balance the coolant flow. It is also designed to trap small debris that 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 features to mix the coolant and improve the thermal exchange between the fuel rods and the coolant. The U.S. 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.

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 cladding, as the first of the three barriers against radioactive releases, isolates 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 on the top 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 oxide energetically equivalent.

Burnable poison

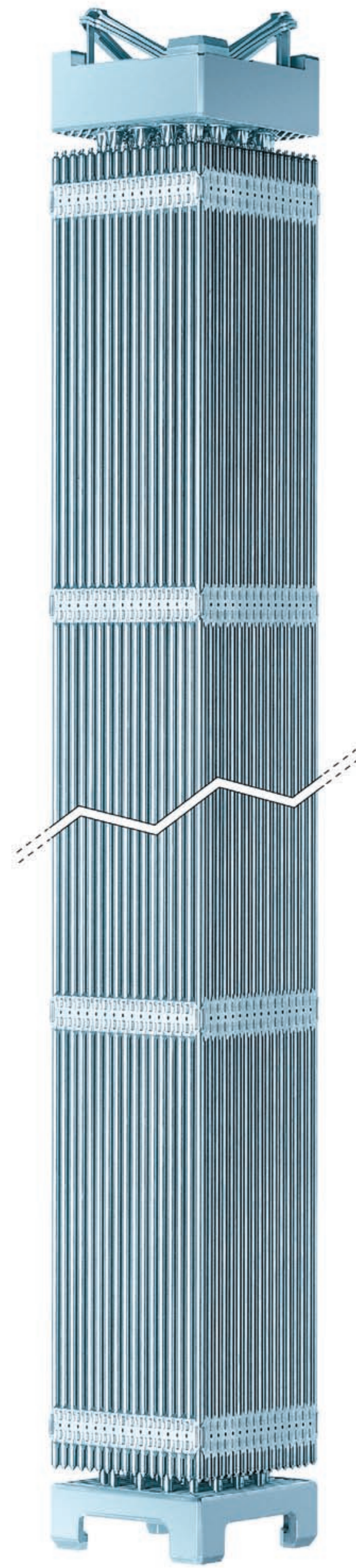
Gadolinium in the form of Gd_2O_3, mixed with the UO_2, is used as integrated burnable poison. The Gadolinium concentrations are in the range of 2% to 8% in weight. The number of Gadolinium-bearing rods per fuel assembly varies, depending on the fuel management. Enriched UO_2 is a carrier material for 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 developed, qualified and industrially utilized by AREVA NP, 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, fuel rod claddings, guide thimbles and spacer grids for all plants based on the EPR design are made of M5™ alloy. M5™ is the most advanced, high-performance PWR fuel material available.

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

17-x-17 fuel assembly

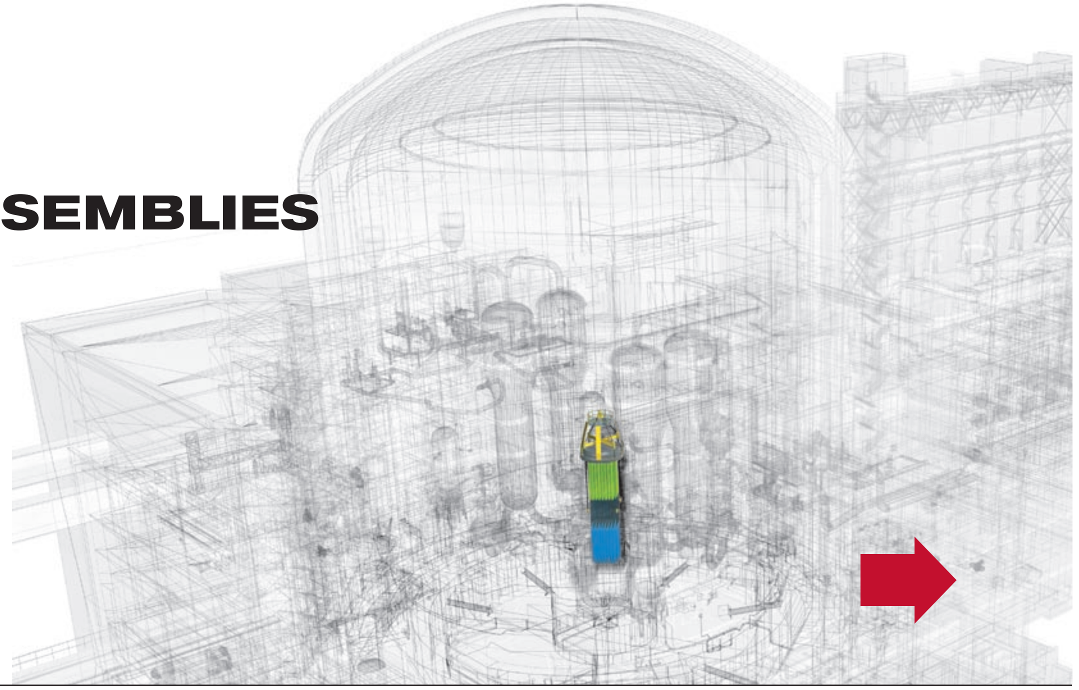


CHARACTERISTICS	DATA
Fuel assemblies	
Fuel rod array	17x17
Lattice pitch	0.50 in
Number of fuel rods per assembly	265
Number of guide thimbles per assembly	24
Fuel assembly discharge burnup (maximum)	70,000 MWd/t
Materials	
– Mixing spacer grids	M5™
– Top & bottom spacer grids	Inconel 718
– Guide thimbles	M5™
– Nozzles	Stainless steel
– Holddown springs	Inconel 718
Fuel rods	
Outside diameter	0.374 in
Active length	13.8 ft
Cladding thickness	0.022 in
Cladding material	M5™

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

Fuel manufacturing workshop, Lynchburg, Virginia.

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, and when fully inserted in the core, they cover almost the whole active length of the fuel assemblies.

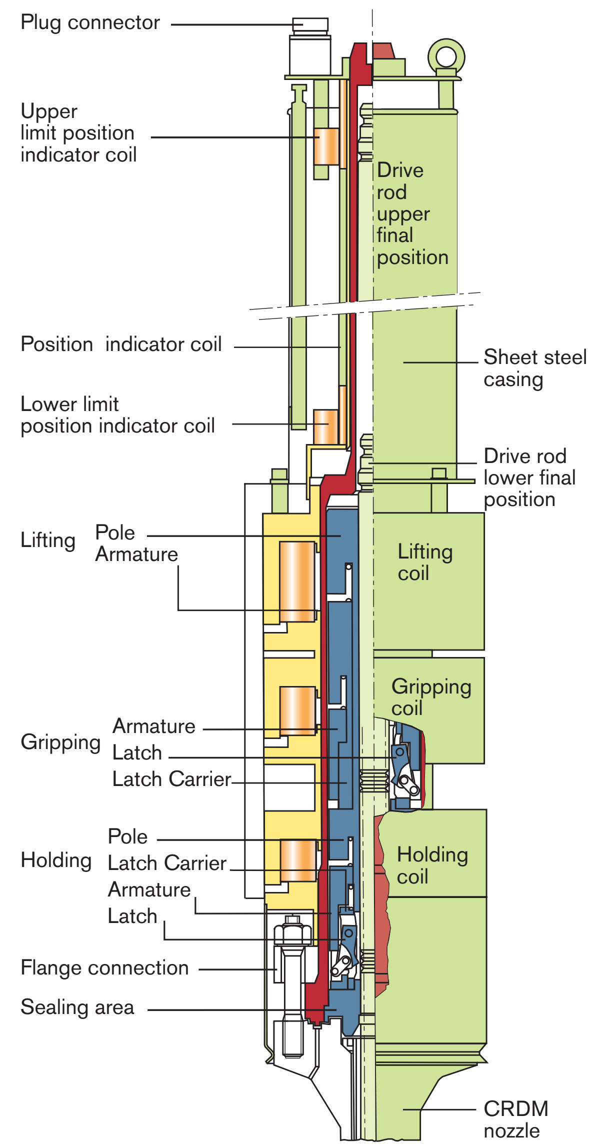
The U.S. EPR is equipped with HARMONI™ RCCAs, a proven AREVA NP design. The neutron-absorbing components are bars made of an Ag-In-Cd alloy. Each rod is composed of a stack of Ag-In-Cd bars contained in a stainless steel cladding under a Helium atmosphere (for efficient cooling of the absorbing materials).

Because mechanical wear of the rod cladding is a limiting factor for the operation life of RCCAs, the HARMONI™ cladding benefits from an ion-nitriding treatment that makes their external surface extremely wear-resistant and virtually 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, which are combined to form four different insertion sequences depending on cycle depletion. This sequence can be changed 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 also be inserted. Shutdown margins are preserved by the RCCA insertion limits.

➡ **The U.S. EPR is equipped with RCCAs of the proven HARMONI™ design for long operation life whatever the operating mode of the reactor.**

CRDM cutaway



CHARACTERISTICS		DATA
Rod cluster control assemblies (RCCAs)		
Number of rods per assembly		24
Absorber		
– Weight composition (%): Ag, In, Cd		80, 15, 5
– Absorber outer diameter		0.341 in
– Length		162 in
Cladding		
Material		Austenitic stainless steel
Surface treatment (externally)		Ion-nitriding
Outer diameter		0.381 in
Inner diameter		0.344 in
Filling gas		Helium
Control rod drive mechanisms (CRDMs)		
Quantity		89
Lift force		> 675 lb
Travel range		161 in
Stepping speed		14.8 in/min or 29.5 in/min
Max. scram time allowed		3.5 s
Materials	– pressure housing	Forged austenitic stainless steel
	– drive rod	Martensitic stainless steel
	– latch unit	Amagnetic austenitic 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, particularly in the case of an abnormal situation.

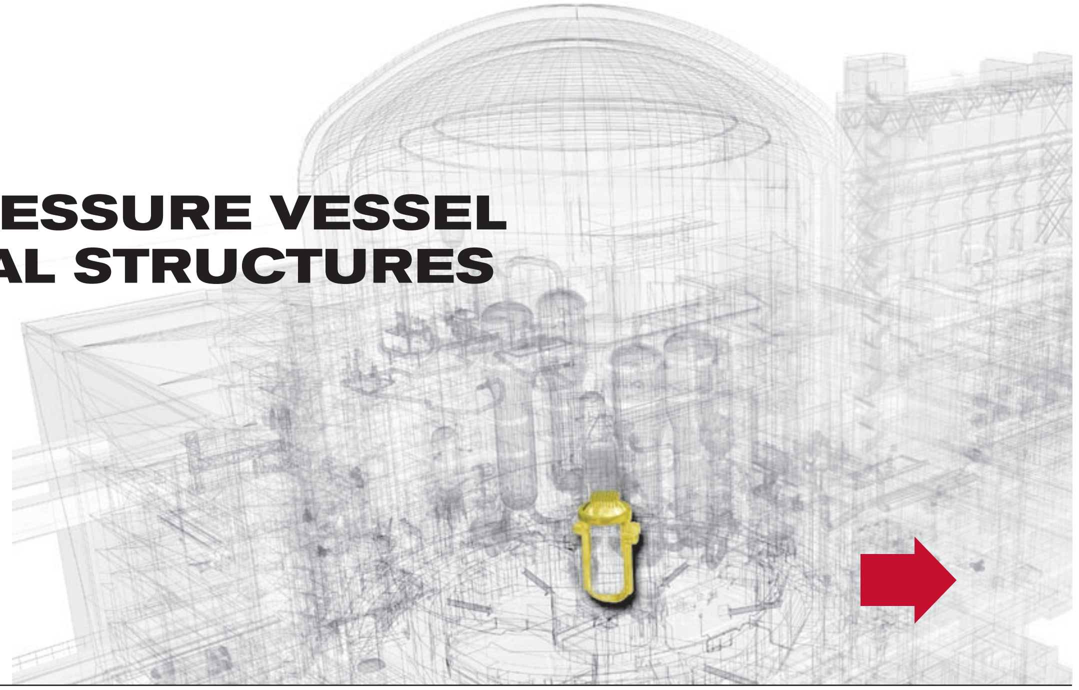
The CRDMs are installed on the reactor pressure vessel head and bolted to adapters welded to the vessel head. Each CRDM is self-contained and can be fitted or removed independently of the others. The CRDMs do not need forced ventilation of the coils, which saves space on the reactor head. The control 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 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 following parts:
- Pressure housing with flange connection
- Latch unit
- Drive rod
- Coil housing

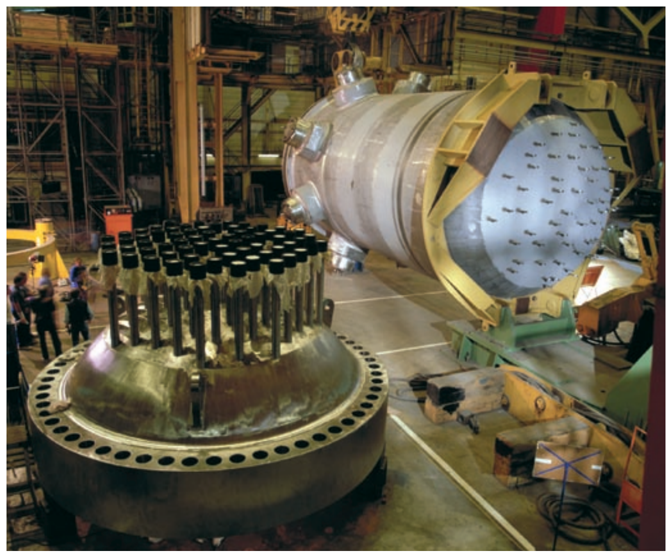
When the reactor trip signal is given, all operating coils are de-energized, the latches are retracted from the rod grooves, and the RCCA drops freely into the reactor core under the force of gravity.

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

➡ **CRDMs are 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) contains the core. The closure head is fastened to the top of the RPV by a set of studs.

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 a single forging with the flange integral to the nozzle shell course. Set-on type nozzles facilitate 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. Because the in-core instrumentation is introduced through the closure head at the top of the RPV, there is no penetration through the bottom piece.

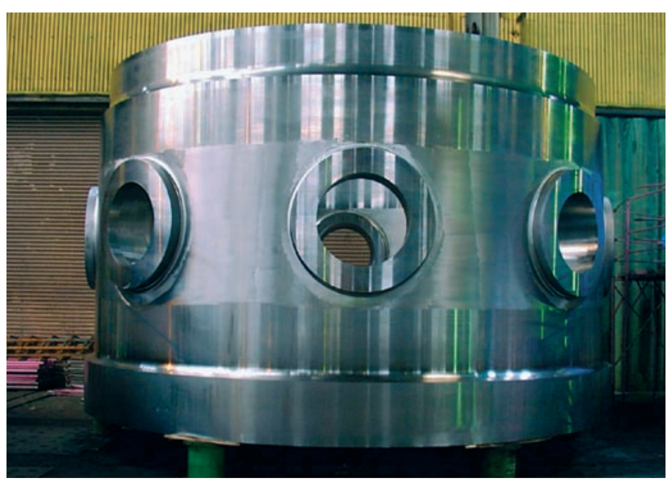
The RPV is designed to facilitate non-destructive testing during in-service inspections. In particular, its internal surface is accessible to allow 100% visual and/or ultrasonic inspection of the welded joints from the inside.

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 – SA 508, Gr. 3, Cl. 1 – a material that combines adequate tensile strength, toughness and weldability. The entire internal surface of the RPV and closure head is clad with stainless steel for corrosion resistance. To help reduction of the corrosion products radiation source term, the cladding material is specified with a low Cobalt residual content.

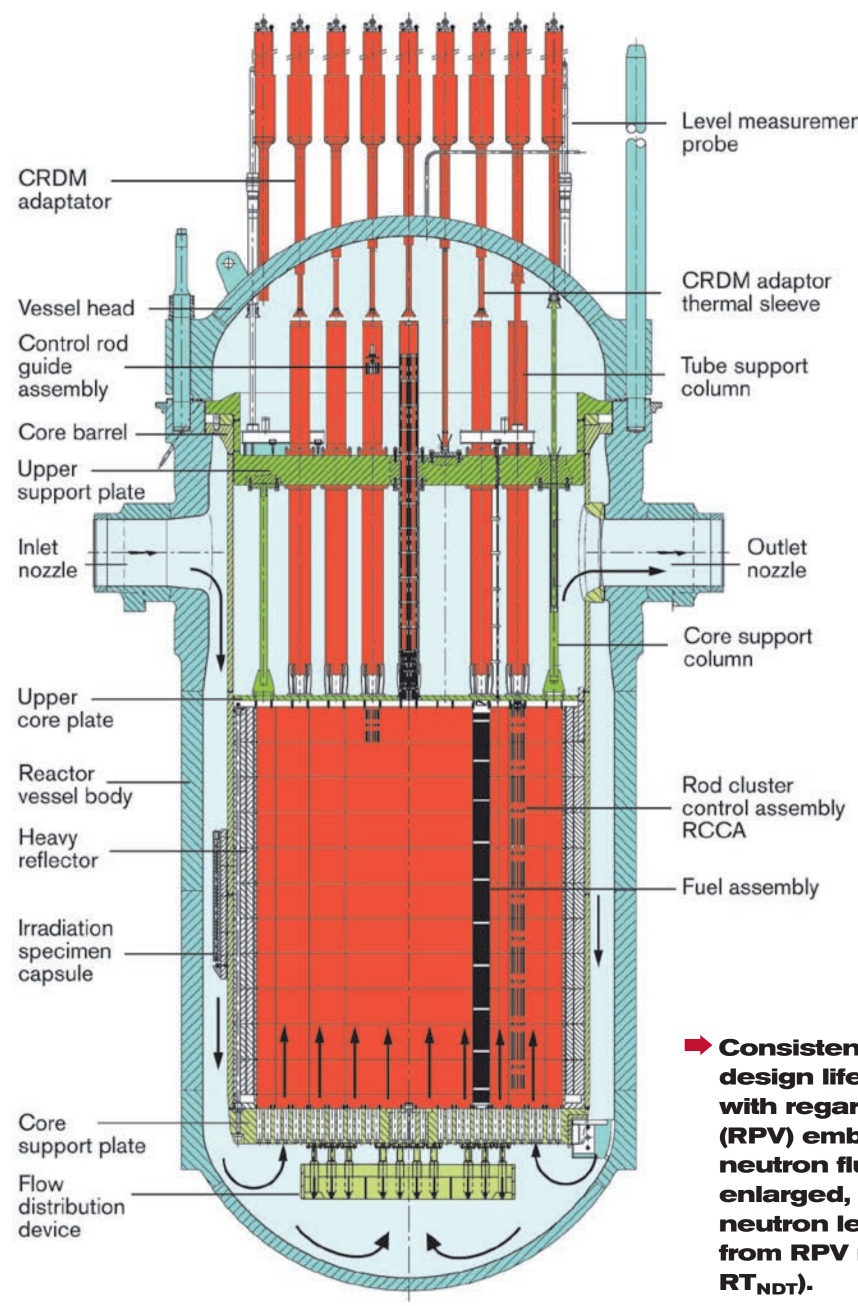
The RPV is supported by a set of integrated pads underneath the eight primary nozzles. These pads rest on a support ring, which is the top part of the reactor pit.

Significant safety margin against the risk of brittle fracture (due to material aging under irradiation) during the RPV's 60-year design life objective is ensured. The ductile-brittle transition temperature (RT^{NDT}) of the RPV material remains lower than 86° F at the end of the design life, resulting from the RPV material and its specified low content in residual impurities. It also results from 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.



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

Reactor pressure vessel and internals cutaway



The elimination of any welds between the flange and the nozzle shell course plus the set-on design of the nozzles allows 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, the operator has more time to counteract the risk of the core becoming uncovered by the coolant.

- **Consistent with the U.S. EPR's 60-year design life objective, 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}).**
- **A higher nozzle axis improves fuel cooling in the event of a loss-of-coolant accident.**
- **No penetration through the RPV bottom head 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

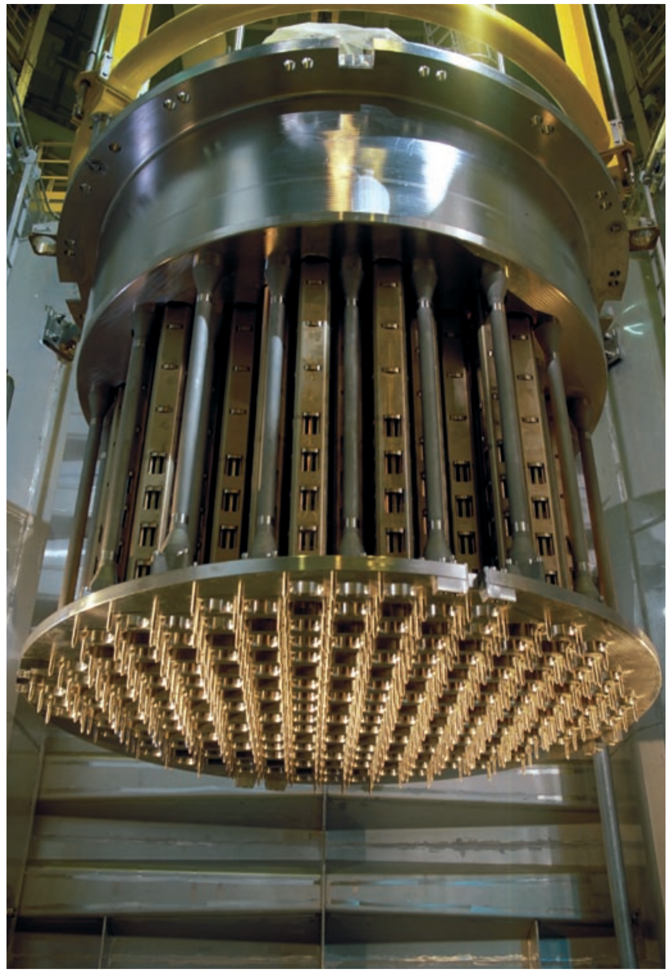
The Reactor Pressure Vessel Internals (RPVI) support the fuel assemblies and keep them properly aligned and spaced to ensure free motion of the control rods and core cooling by the primary coolant in any circumstances, including postulated accidents.

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 (see page 45) by protecting the Reactor Pressure Vessel (RPV) against fast neutron fluence-induced embrittlement.

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

The RPVI are partially removed from the RPV to allow fuel assembly loading and 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 the upper core plate. The upper internals axially maintain the fuel assemblies in their correct position during operation.

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 – which 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 stainless steel heavy reflector surrounds the core, and is made of a stack of rings 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.

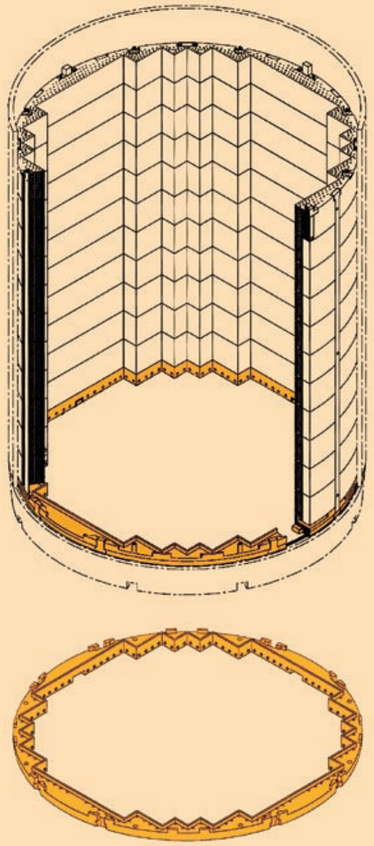
Materials

The base material of the internals is a low-Carbon Chromium-Nickel stainless steel. The various connectors (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:

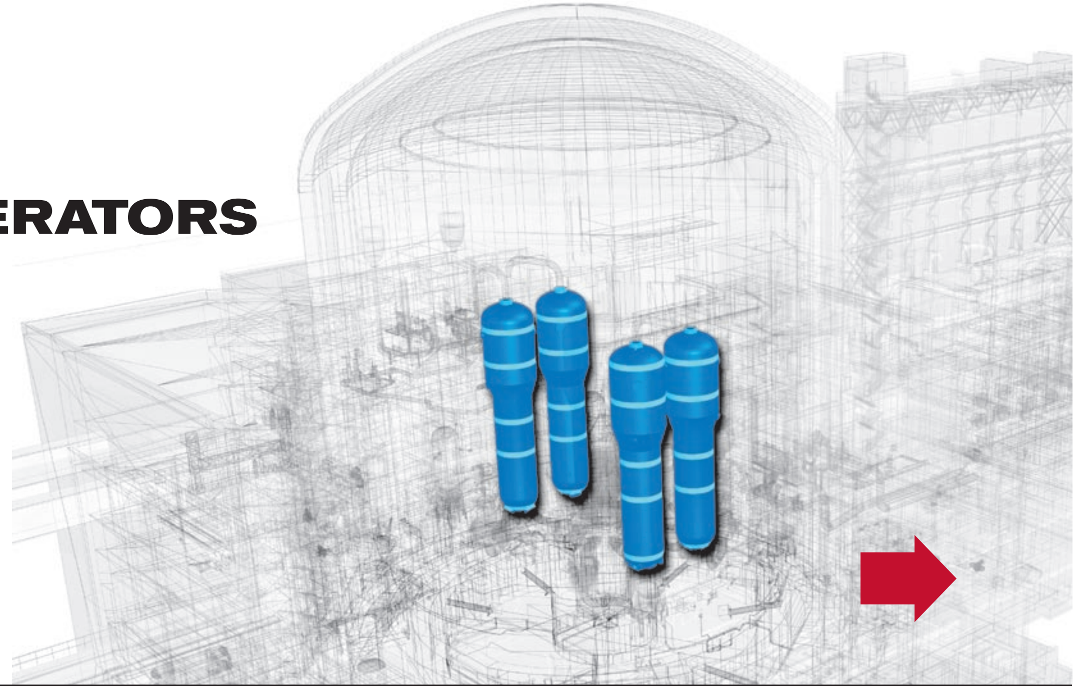
- **By reducing the flux of neutrons escaping from the core, the nuclear fuel is better utilized (more neutrons can take part in the chain reaction process), thereby decreasing the fuel cycle cost by reducing the fuel enrichment necessary to reach a given burnup, or to increase burnup with a given enrichment.**
- **By reducing the neutron leakage from the core, the RPV is protected against fast neutron fluence-induced aging and embrittlement, helping to ensure the 60-year design life of the U.S. EPR.**
- **The reactor also provides advances in terms of mechanical behavior of the internal structure surrounding the core:**
  - **Smooth stress distribution inside the structure, due to 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 in case of assumed loss-of-coolant accident, because no significant quantity of water is trapped in the structure around the core**



CHARACTERISTICS	DATA
Reactor pressure vessel	
Design pressure	2,550 psia
Design temperature	664° F
Lifetime (load factor 0.9)	60 yrs
Inside diameter (under cladding)	192.3 in
Wall thickness (under cladding)	9.8 in
Bottom wall thickness	5.7 in
Height with closure head	500 in
Base material	SA-508 Gr. 3 Cl. 1
Cladding material	Stainless steel (Cobalt $\leq$ 0.06%)
Mass with closure head	580 tons
End of life fluence level (E > 1 MeV) IN-OUT fuel management with UO_2	≈ 1×10^{19} n/cm^2
Base material final RT_{NDT} (final ductile-brittle transition temperature)	≈ 86° F
Closure head	
Wall thickness	9.1 in
Number of penetrations for:	
• Control rod mechanisms	89
• Dome temperature measurement	1
• Instrumentation	16
• Coolant level measurement	4
Base material	SA-508 Gr. 3 Cl.1
Cladding material	Stainless steel (Cobalt $\leq$ 0.06%)
Upper internals	
Upper support plate thickness	13.8 in
Upper core plate thickness	2.4 in
Main material	304LN SS
Lower internals	
Lower support plate thickness	16.3 in
Lower internals parts material	304LN SS
Neutron heavy reflector	
Material	304LN SS
Mass	99 tons

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

STEAM GENERATORS



The steam generators (SGs) are the interface between the primary water heated by the nuclear fuel and the secondary water, which provides steam to the turbine generator.

The U.S. 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.

The U.S. EPR SG is comprised of two subassemblies:

- Lower section – where the heat exchange process between the primary water and the secondary water takes place
- Upper section – where the steam-water mixture is mechanically dried before it is routed to the turbine

In conjunction with an increased heat exchange area, the U.S. EPR axial economizer makes it possible to reach a steam generator exit pressure of 1109 psia and a plant efficiency of 35%. 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 side of the tube bundle, and a partition plate to separate the cold side tubing from the hot side tubing. This design improvement increases the steam pressure by about 43 psi compared to a conventional SG. The tube bundle is easily accessed for inspection and maintenance.

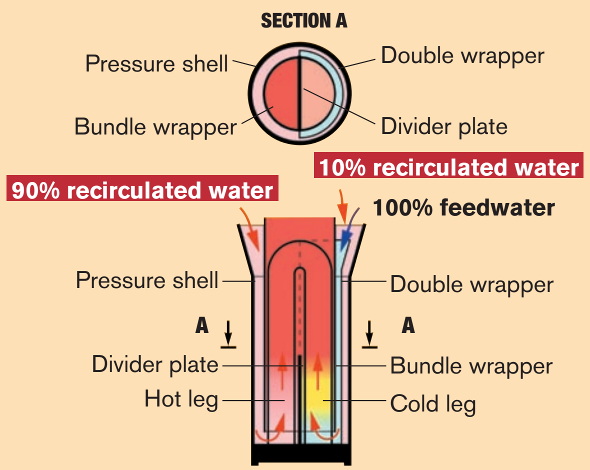
The tube bundle is made of thermally treated Inconel 690, a proven stress-corrosion resistant alloy with a specified mean value Co content less than 0.015%. The SG bundle wrapper is made of SA-508 steel.

The EPR steam generator was designed to prevent 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, prevents the SGs from filling up with water in case of SG tube rupture in order 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 of at least 30 minutes in the event of a total loss of feedwater.

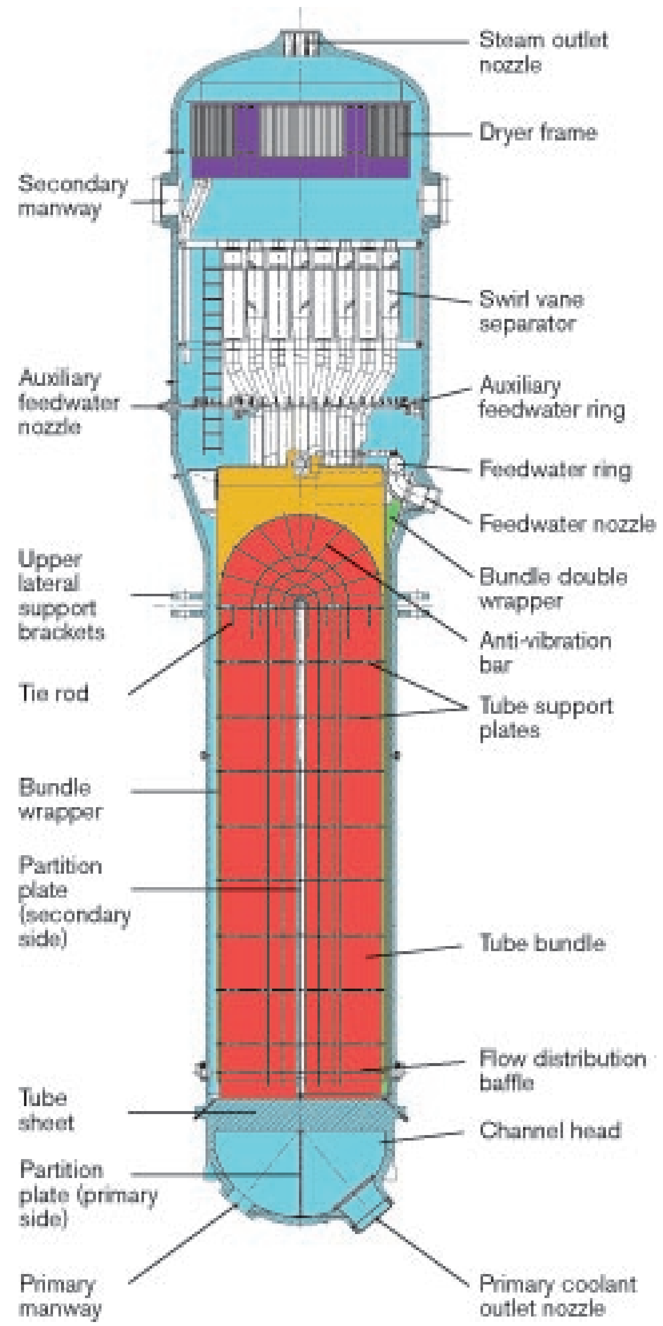
The steam generator is fully shop-built, transported to the plant site and installed in its reactor building cubicle in one piece.



The axial economizer

The principle function of the axial economizer is to direct the feedwater to the cold leg of the tube bundle, and about 90% of the recirculated water to the hot leg. This is done by adding to the standard natural circulation U-tube design a double wrapper in the downcomer to guide the feedwater to the cold leg side of the tube bundle, and a secondary side partition plate to separate the cold leg and hot leg sides of the tube bundle. With those two design features, the internal feedwater distribution system of the steam generator covers only 180° of the wrapper on the cold side.

Steam generator cutaway



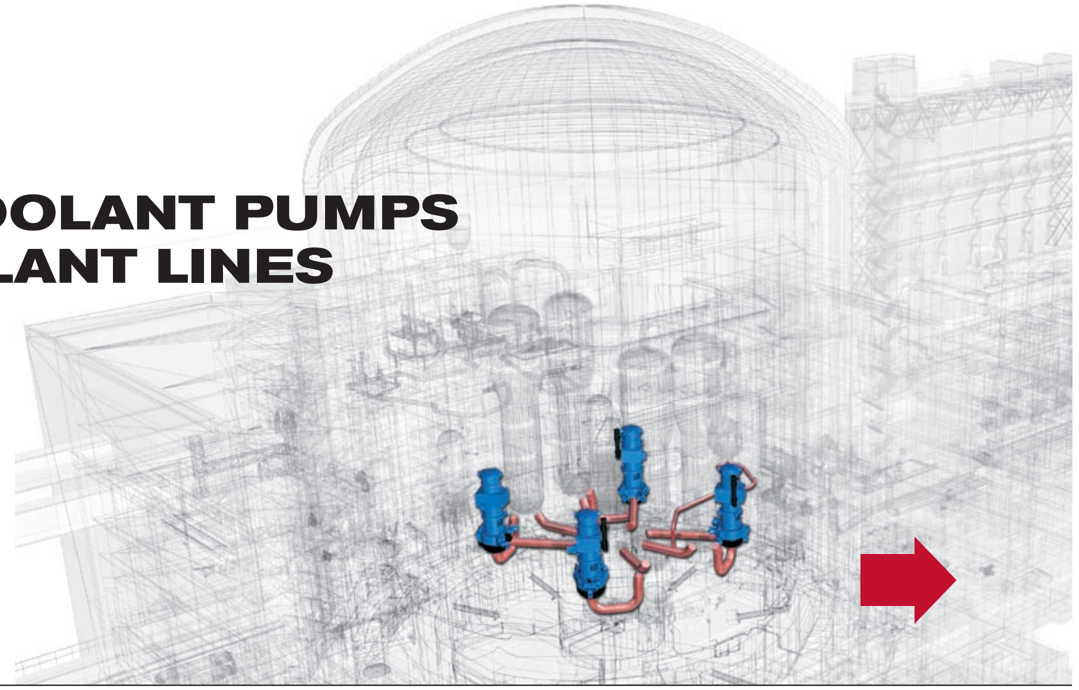
CHARACTERISTICS	DATA
Steam generators	
Number	4
Heat transfer surface per steam generator	85,680 ft²
Primary design pressure	2,550 psia
Primary design temperature	664° F
Secondary design pressure	1,450 psia
Secondary design temperature	592° F
Tube outer diameter/wall thickness	0.75 in / 0.043 in
Number of tubes	5,980
Triangular pitch	1.08 in
Overall height	79.3 ft
Total mass	551 tons
Materials	
• Tubes	Alloy 690 TT*
• Shell	SA-508 Gr. 3 Cl. 2
• Cladding tube sheet	Ni Cr Fe alloy
• Tube support plates	13% Cr improved stainless steel
Miscellaneous	
Feedwater temperature	448° F
Moisture carryover	0.1%
Main steam flow at nominal conditions	5.1 Mlb/hr per SG
Main steam temperature	559° F
Steam pressure at nominal conditions	1,109 psia
Pressure at hot standby	1,305 psia

* TT: Thermally treated

- **The steam generator is an enhanced version of the axial economizer steam generator implemented on N4 plants.**
- **The axial economizer increases the steam pressure by 43 psi compared to a conventional design, without impairing access to the tube bundle for inspection and maintenance.**
- **The high steam exit pressure (1109 psia) is a major contributor to the high efficiency of the U.S. EPR.**
- **The secondary water mass is consistent with the 30-minute 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.**

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

REACTOR COOLANT PUMPS & MAIN COOLANT LINES



Reactor Coolant Pumps

The Reactor Coolant Pumps (RCPs) 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.

An RCP is located between the steam generator outlet and the reactor vessel inlet of each of the four primary loops.

The RCP 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 called a standstill seal has been added as shaft seal back-up.

The U.S. EPR coolant pump consists of three major components:

• **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 of 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 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 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 seal is a hydrodynamic seal that takes the remaining pressure in normal operation, but can take the full primary pressure in the event of a first-stage failure. The third seal is also a hydrodynamic seal with no significant differential pressure. Its purpose is to complete final leak tightness and prevent water spillage.

The shaft seals are in a housing bolted to the closure flange, which 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 – will 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**

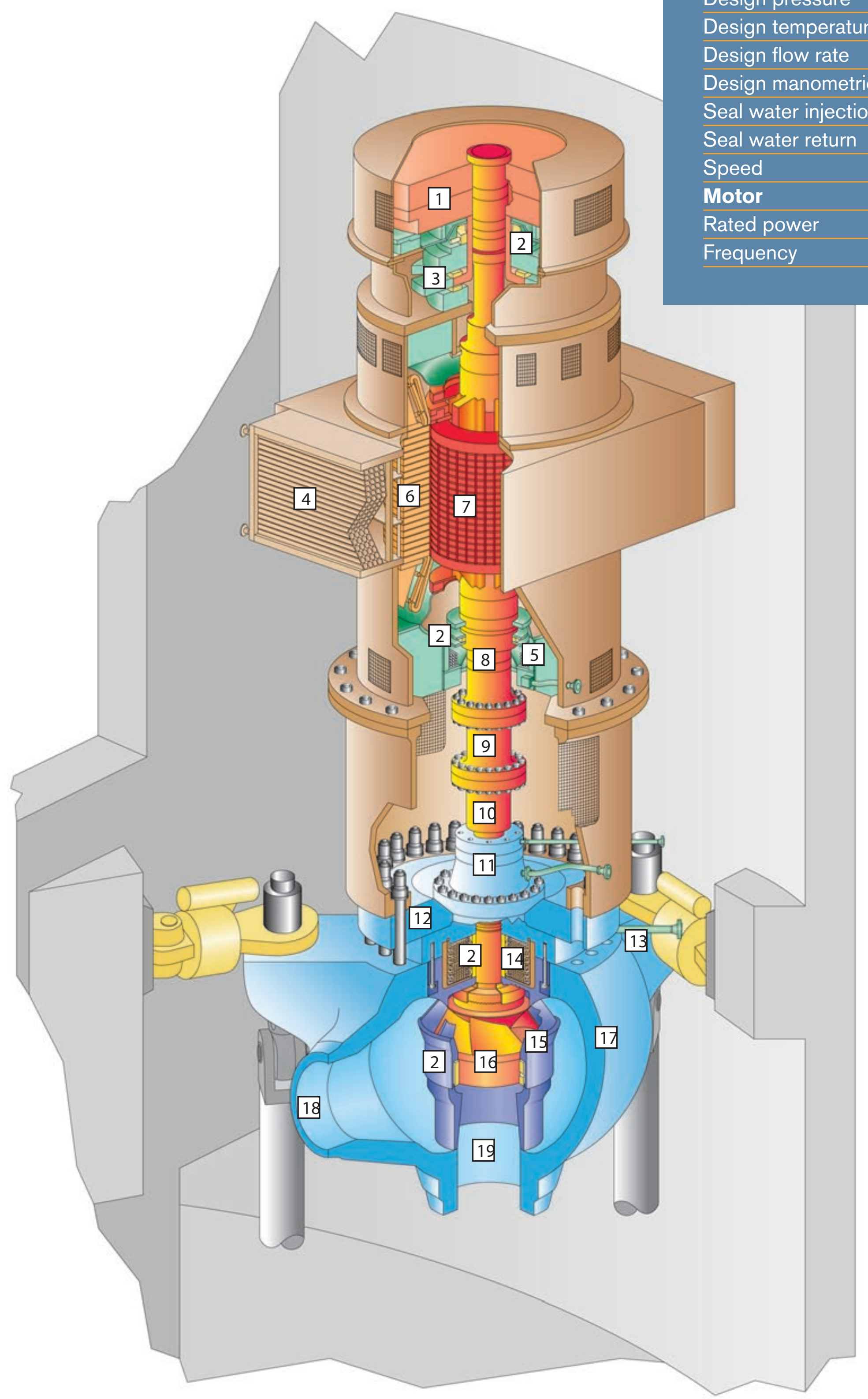
This feature ensures that no loss of coolant would occur even in case of total station blackout or failure of the main seals.

• **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	30.5 ft
Overall mass w/o water and oil	124 tons
Pump	
Design pressure	2,550 psia
Design temperature	664° F
Design flow rate	124,730 gpm
Design manometric head	329 ft
Seal water injection	7.9 gpm
Seal water return	3.0 gpm
Speed	1,190 rpm
Motor	
Rated power	9,000 kW
Frequency	60 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

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

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	30.7 in
Thickness of straight portions	3 in
Material	Low carbon austenitic stainless steel
Surge line	
Inside diameter	12.815 in
Thickness	1.6 in
Material	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

All the legs have a large inner diameter of 30.7 inches to minimize the pressure drop and to reduce the coolant flow velocity in the coolant lines.

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

The main coolant line materials and manufacturing processes were selected to yield a high-quality, tough product, 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 in welding processes are being implemented. For example, the homogeneous circumferential welds are made using 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 and are machined from 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 main coolant lines’ ability to fulfill the Leak Before Break requirements.

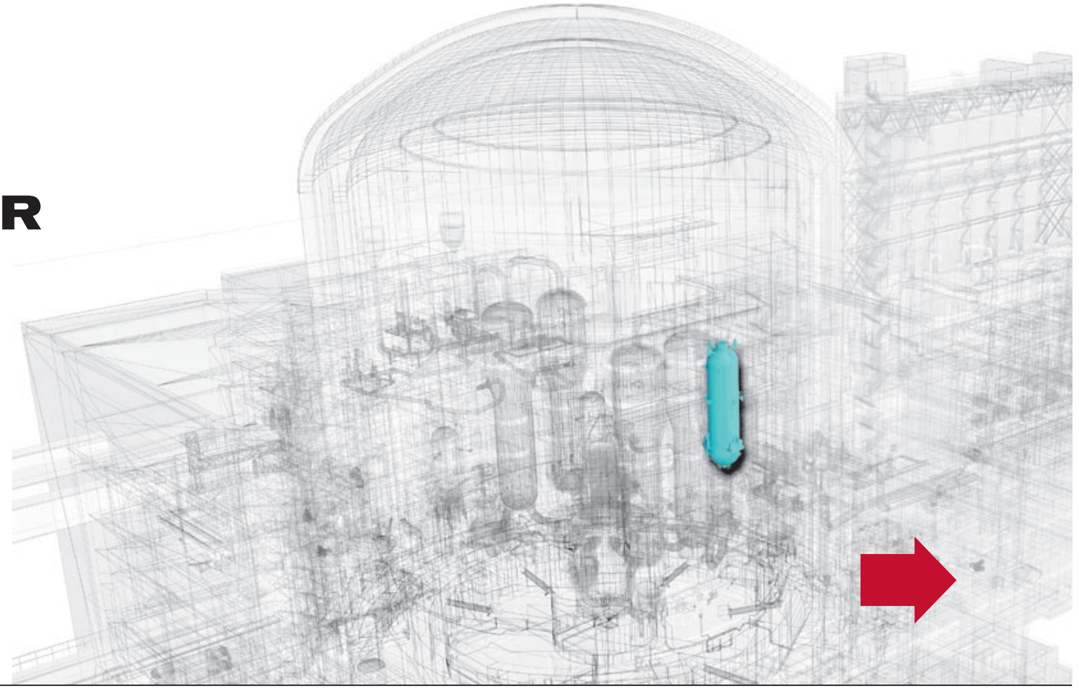
The main coolant lines' design and material are based on the technology already implemented on N4 reactors at the Civaux site.

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 from 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) maintains the pressure of the primary circuit inside prescribed limits. It is 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 vessel contains primary water in the lower part, and steam in the upper part. To accommodate some primary coolant volume variation, the pressurizer is equipped with electric heaters at the bottom to vaporize more liquid, and with a spray system at the top to condense more steam. Compared to previous designs, the volume of the EPR pressurizer is significantly increased to smooth the response to operational transients. This improvement increases equipment life duration and 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 a postulated accident with a risk of core melting, these valves would provide the operator an additional efficient means of rapidly depressurizing the primary circuit and avoiding a high-pressure core melt situation.

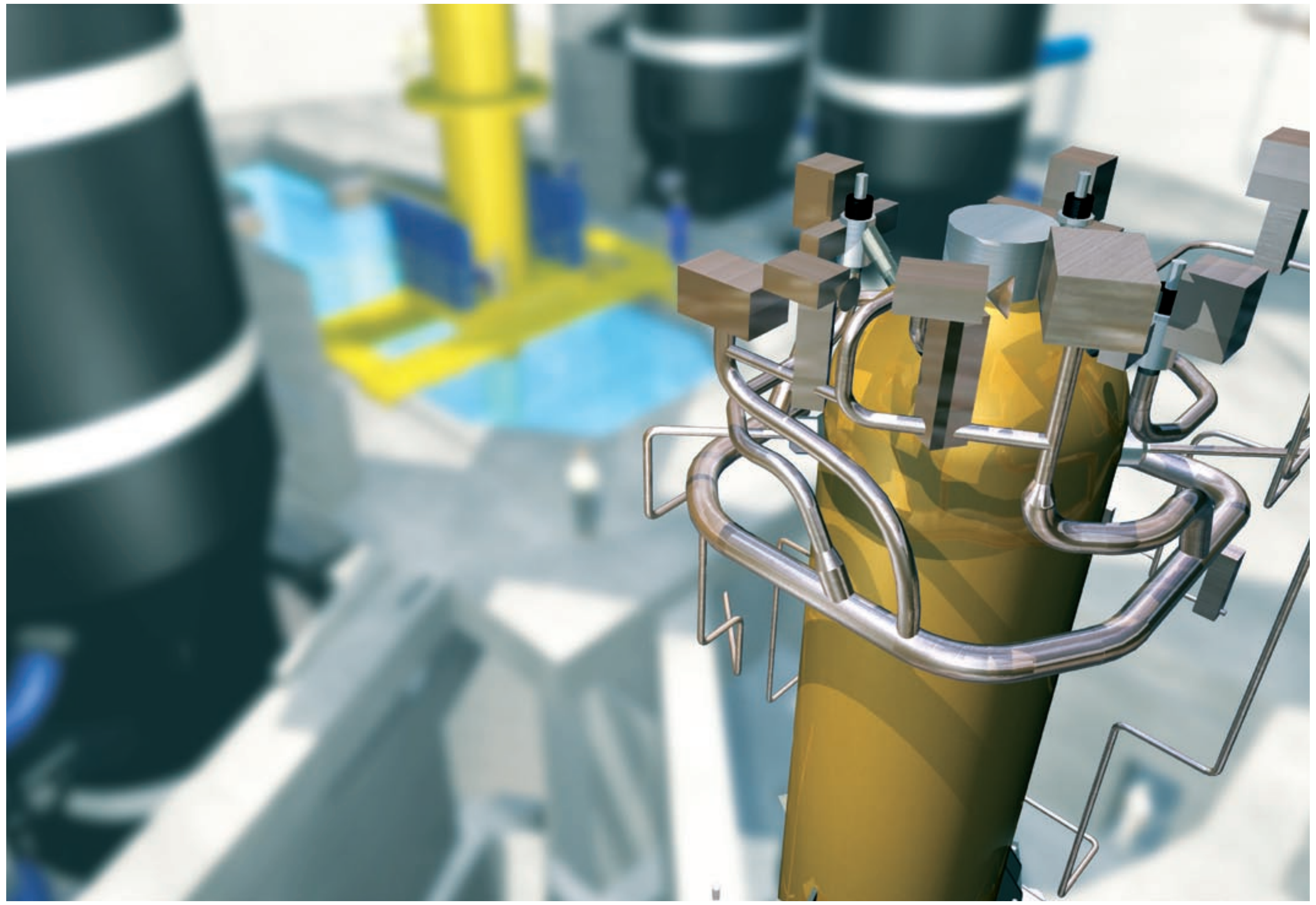
A number of design features have been incorporated to improve maintainability. In particular, a floor between the pressurizer head and the valves eases heater replacement and reduces radiological dose during valve service.

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 will preclude rocking in the event of a postulated earthquake or accident.

CHARACTERISTICS	DATA
Pressurizer	
Design pressure	2,550 psia
Design temperature	684° F
Total volume	2,649 ft^3
Total length	47.2 ft
Base material	SA-508 Gr. 3 Cl. 2
Cylindrical shell thickness	5.5 in
Number of heaters	108
Total weight, empty	165 tons
Total weight, filled with water	248 tons
Number and capacity of safety valve trains	3 x 660,000 lb/hr
Depressurization valves capacity	2 x 1,980,000 lb/hr

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:**
  - **Help achieve the 60-year equipment design life objective**
  - **Increase 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.

- Continuously 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, and 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 spray nozzle in order to reach Residual Heat Removal System (SIS/RHRS) operating conditions.
- 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" mode of diverse core cooling.

SAFETY INJECTION / RESIDUAL HEAT REMOVAL

The Safety Injection System (SIS/RHRS) comprises the Medium Head Safety Injection System (MHSI), the Accumulators, the Low Head Safety Injection System (LHSI) and the In-Containment Refueling Water Storage Tank (IRWST). 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, an MHSI pump and an LHSI pump, with a heat exchanger at the pump outlet.

While operating in residual heat removal (RHR) mode, the system:

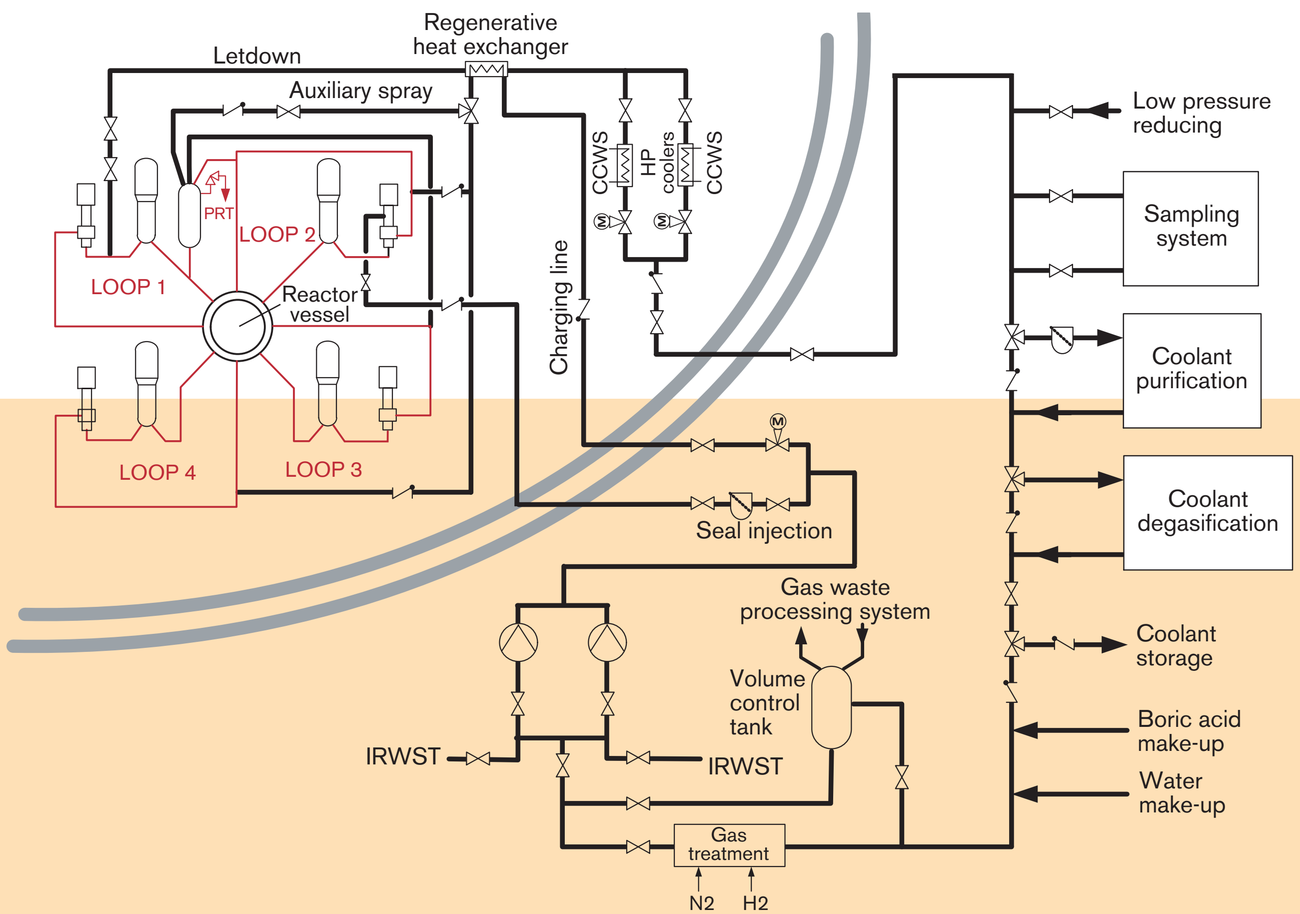
- Provides the capability for heat transfer from the RCS to the Component Cooling Water System (CCWS) when heat transfer via the Steam Generators (SG) is no longer sufficiently effective (at an RCS temperature of less than 250° F in normal operation)
- Transfers heat continuously from the RCS or the 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 356° F following a reactor shutdown.

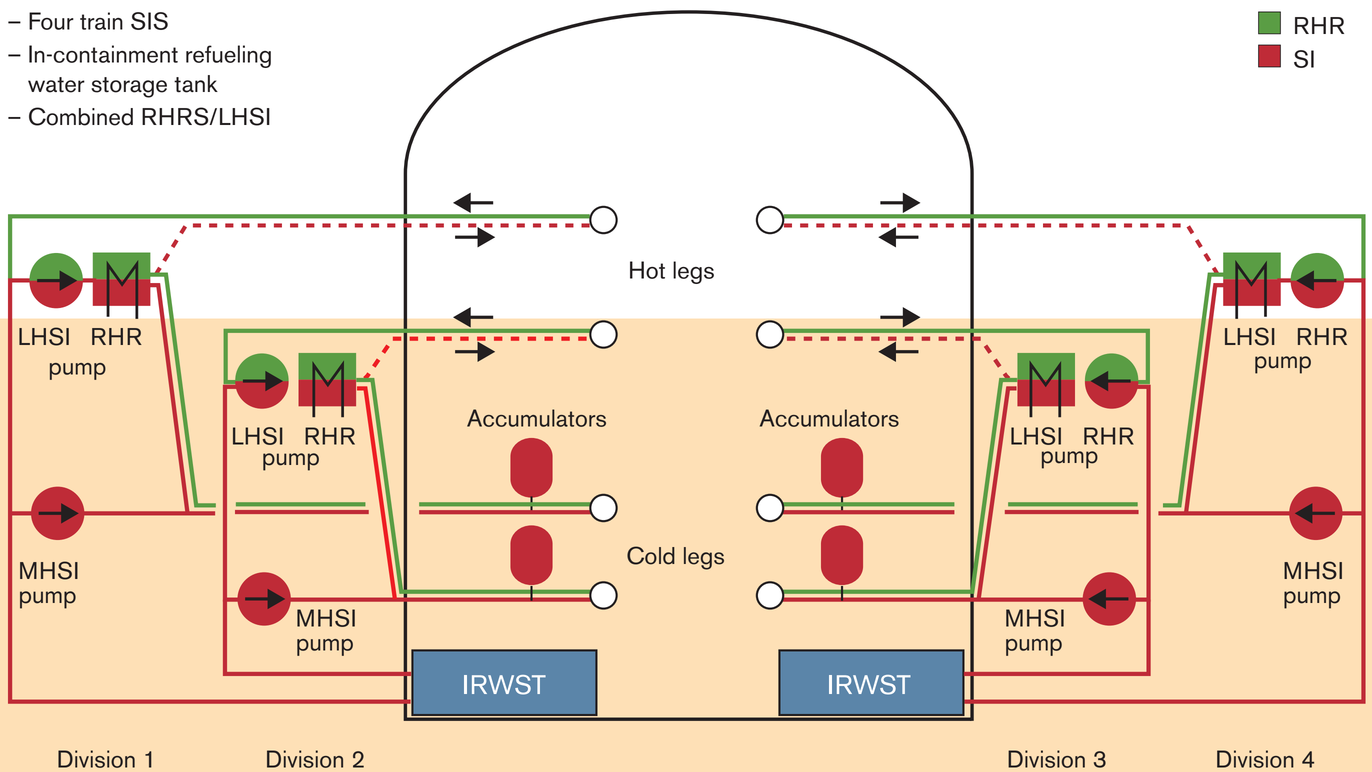
The four redundant and independent SIS/RHRS trains are arranged in separate divisions in the Safeguard Buildings. Each train is connected to one dedicated RCS loop and is designed to provide the injection capability required to mitigate accident conditions. This configuration greatly simplifies the system 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. It would be also activated during a steam generator (SG) tube rupture or during loss of a secondary side heat removal function.

The MHSI system injects water into the RCS at a pressure (1,335 psia at mini-flow) set to prevent overwhelming the secondary side safety valves in the event of SG tube leaks. The accumulators and the LHSI system also inject water into the RCS cold legs when the primary pressure is sufficiently low (accumulator: 650 psia, LHSI: 305 psia at mini-flow).

IN-CONTAINMENT REFUELING WATER STORAGE TANK (IRWST)

The IRWST contains a large amount of borated water, and collects water that might be discharged from the RCS to 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 is cooled by the LHSI system.

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

EMERGENCY FEEDWATER

The Emergency Feedwater System (EFWS) ensures that water is supplied to the SGs when all the other systems that normally supply them are unavailable.

Its main safety functions are to:
- Transfer heat from the RCS via the SGs to the atmosphere to cool the plant to RHRS start-up conditions 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 SGs following a loss-of-coolant accident or an SG tube rupture accident.
- In conjunction with steam release from the MSRV, provide a diverse method to rapidly cool the plant down to LHSI conditions in case of a small loss of coolant associated with total MHSI failure.

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 physically 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 physically 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 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

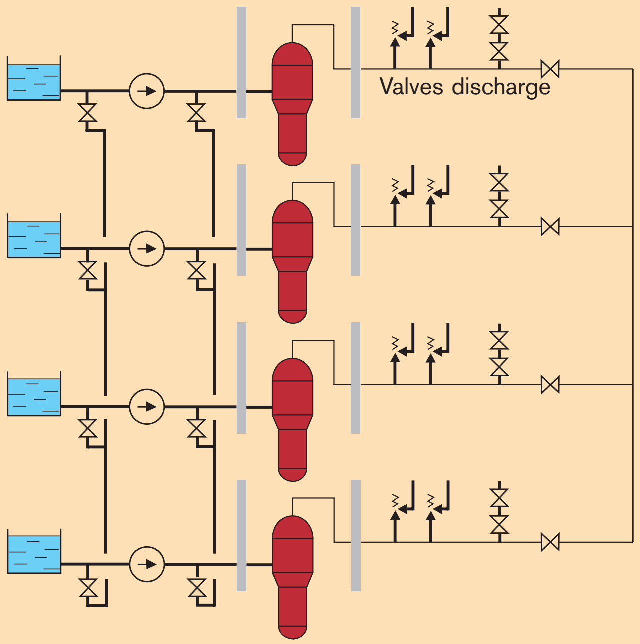
The ESWS consists of four separate safety trains that cool the CCWS heat exchangers with water from the heat sink during all normal plant operating conditions and during incidents and accidents. This system also includes two trains of the dedicated cooling chain for conditions associated with the mitigation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

OTHER SYSTEMS

- **Nuclear Sampling System –** takes samples of gases and liquids from systems and equipment inside reactor containment
- **Vent and Drain System –** collects gaseous and liquid waste from systems and equipment so it can be treated
- **Steam Generator Blowdown System –** prevents build-up of contaminants in the secondary side water
- **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.



BACK-UP FUNCTIONS IN THE EVENT OF TOTAL LOSS OF THE REDUNDANT SAFETY SYSTEMS

- **The loss of secondary side heat removal is backed up by primary side feed and bleed cooling 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 CHRS for IRWST cooling.**

SAFETY SYSTEMS AND FUNCTIONS

- **Simplification by separation of operating and safety functions**
- **Fourfold redundancy applied to the safeguard systems and to their support systems – allows maintenance during plant operation, ensuring a high plant availability factor**
- **Safety systems trains are located in four different, physically separated buildings**
- **With systematic functional diversity, there is always a diversified system that 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

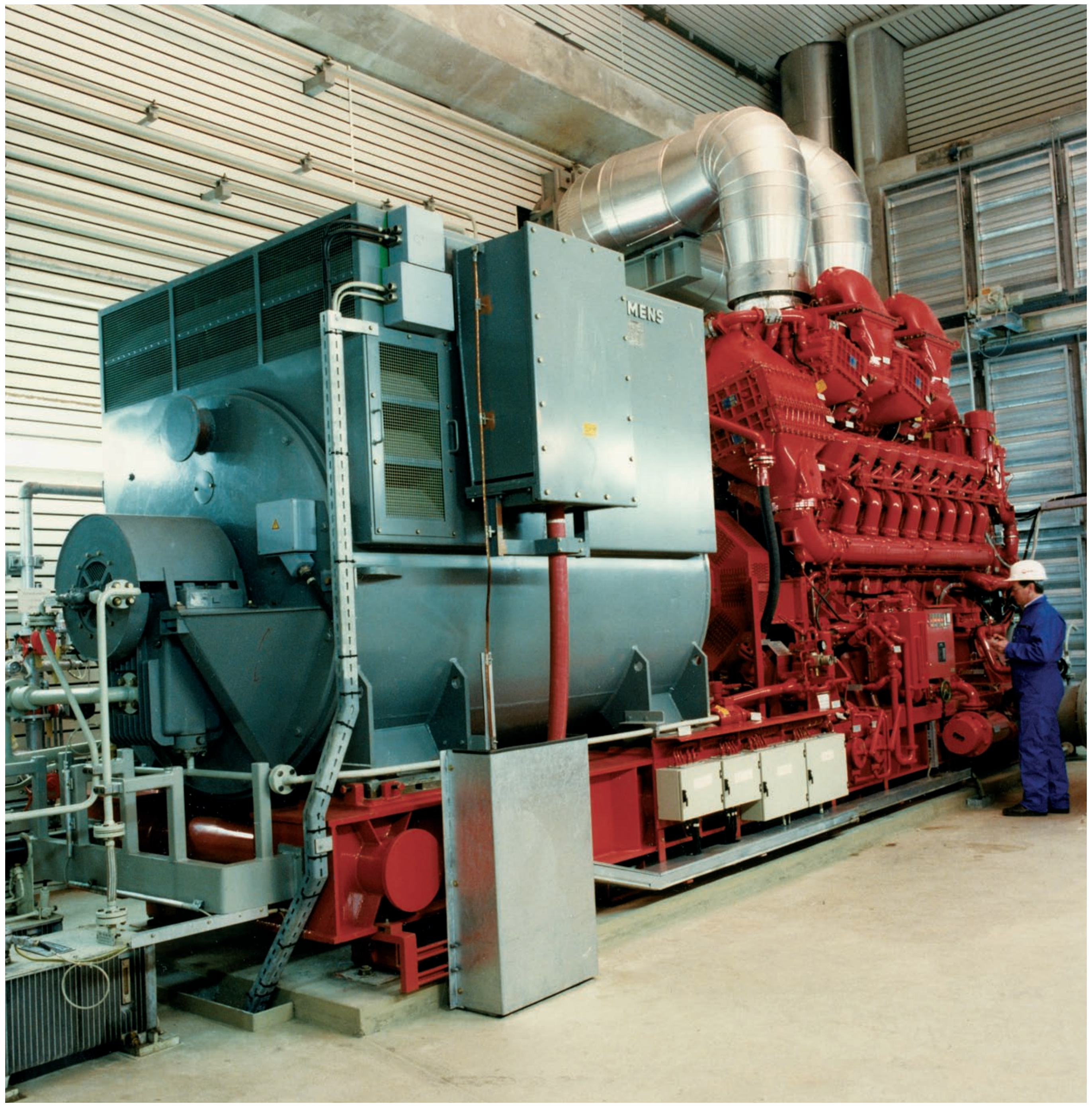
The Emergency Power Supply ensures the safety systems are powered in the event of loss of the preferred electrical sources.

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.

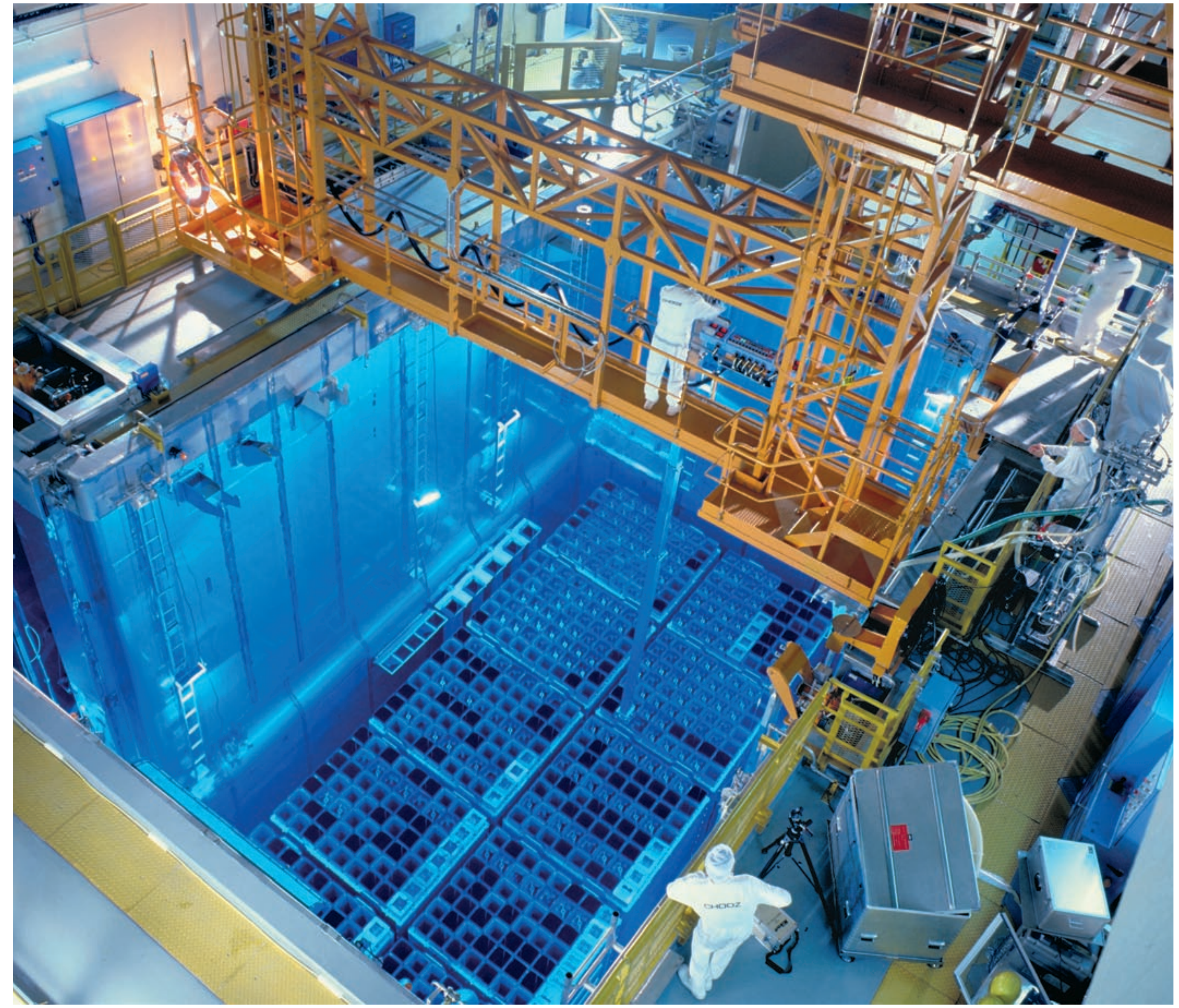
The emergency power supply system is designed to meet the requirements of the N+2 concept (assuming a single failure on one train and a maintenance operation on another).

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 power to critical loads. They are connected to the safety busbars of two divisions.



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



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

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, including fuel transfer tube, spent fuel crane, fuel elevator, refueling machine and spent fuel cask transfer machine.

The underwater fuel storage racks are used for storing:

- Fresh fuel assemblies from on-site delivery to loading into the reactor core
- Spent fuel assemblies following fuel unloading from the core and prior to shipment off site

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

- **FuelPool Cooling System (FPCS)** provides heat removal from the SFP and keeps the SFP temperature at the required level during normal plant operation (power operation and refueling outage). This system is arranged in two separate and independent trains, with two FPCS pumps operating in parallel in each train.
- **FuelPool Purification System (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.

INSTRUMENTATION & CONTROL SYSTEM

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

The I&C system is composed of sensors to transform physical data into electrical signals, programmable controllers to process these signals, and the control actuators, monitors and other means of control by 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.

In designing the EPR and the associated I&C system, specific attention was paid to ensure a high level of operational flexibility in order to meet the needs of electricity companies. As a result, the U.S. 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 by its functions (measurement, actuation, automation, man-machine interface) and its role in safety or operation of the plant.

A multi-level structure

Consideration of the different roles played by the different I&C systems leads to a multi-level structure for I&C architecture:

- **Level 0: Process Interface –** comprises the sensors and the switchgears.
- **Level 1: System Automation –** encompasses I&C systems to perform
  - – Reactor protection,
  - – Reactor control, surveillance and limitation functions,
  - – Safety automation, and
  - – Process automation.
- **Level 2: Process Supervision and Control –** consists of
  - – The workstations and panels located in the Main Control Room, the Remote Shutdown Station and the Technical Support Center, which are also called the man-machine interface (MMI); and
  - – The I&C systems that act as a 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 with the appropriate quality level.

Redundancy, division, diversity and reliability

U.S. EPR I&C systems and equipment comply with the principles of redundancy, division and diversity enforced for designing EPR safety-related systems. For example, 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 fulfill its functions even if one of the 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.**

I&C technology

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

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

ROLE OF THE I&C SYSTEMS

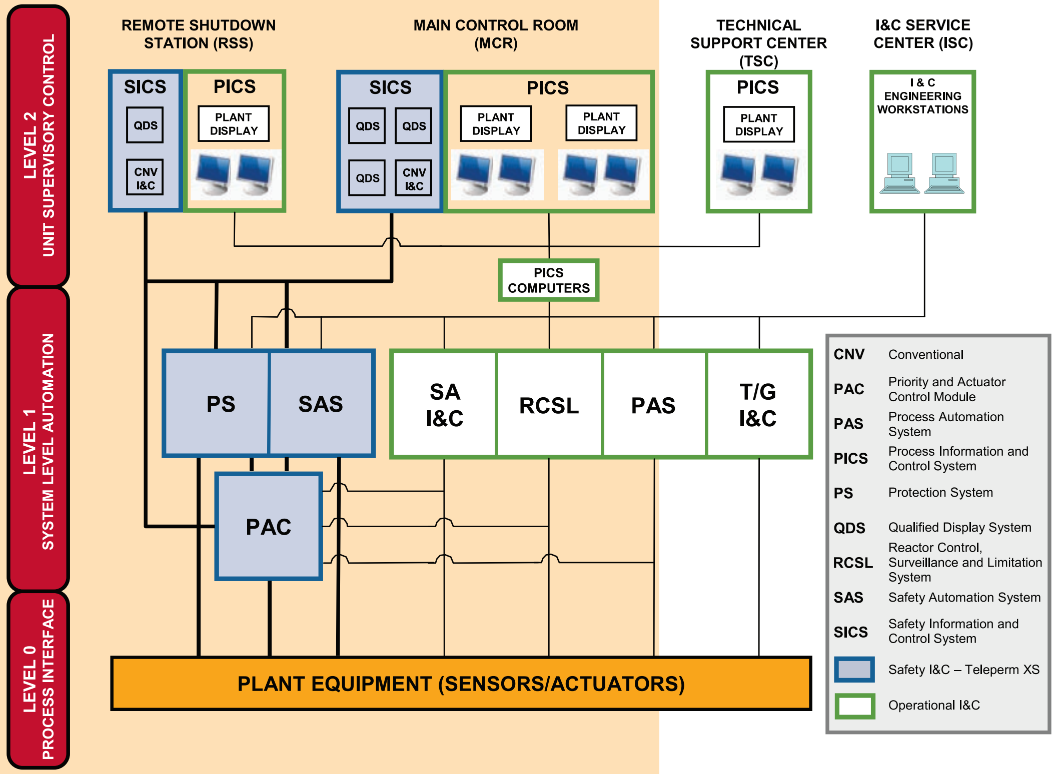
The I&C systems act in accordance with the "defense in depth" concept.

Three lines of defense are implemented:

- The control system maintains the plant parameters within their normal operating ranges.
- In case a parameter leaves 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



Instrumentation (Level 0)

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

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

Measurement of the power level is performed using ex-core instrumentation, which also provides signals to monitor 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.

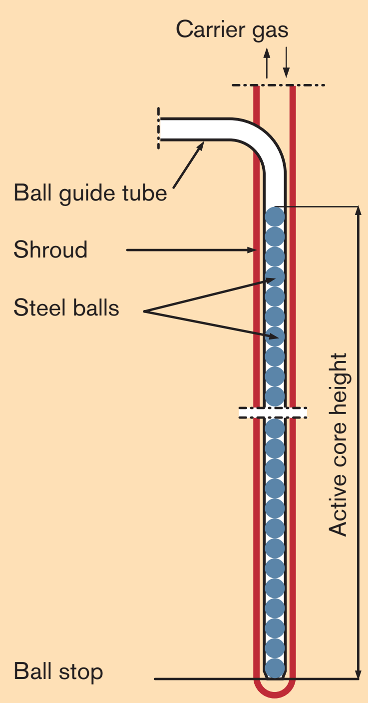
Prediction and measurement of the three-dimensional power distribution relies on two types of in-core instrumentation:

- "Movable" reference instrumentation validates the core design and calibrates other core surveillance and protection sensors.
- "Fixed" instrumentation delivers online information to the surveillance and protection systems, which actuate appropriate actions and countermeasures in case of anomalies or exceeding of predefined limits.

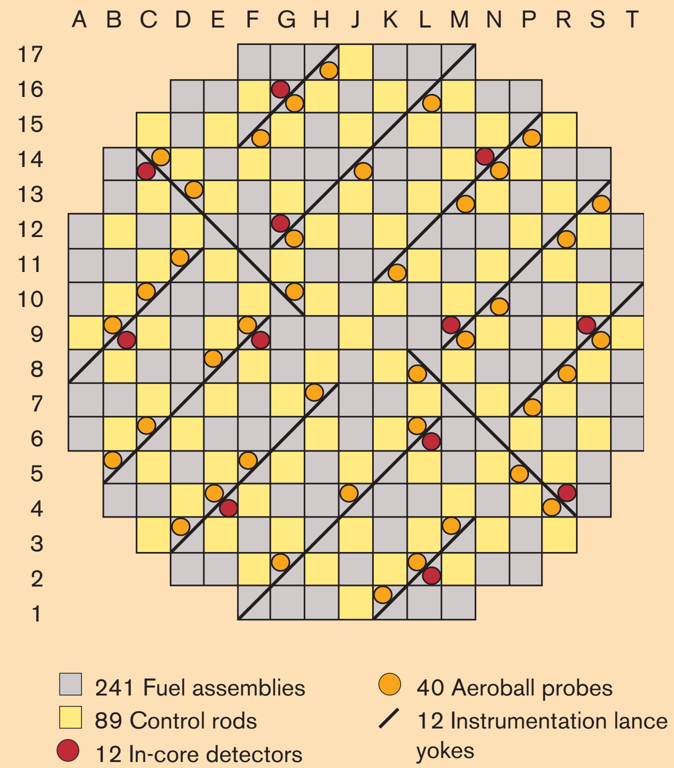
The movable reference instrumentation for power distribution assessment is an "aeroball" system. Stacks of vanadium-alloy balls, inserted from the top of the pressure vessel, are pneumatically transported into the reactor core (inside guide thimbles of fuel assemblies), then – after three minutes – to 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 that 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 used 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 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 electromagnetic 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 with an advanced Man-Machine interface for operator information and action.**

Man-Machine interface (Level 2)

Due consideration was given at the design stage to the human factor for enhancing the reliability of operators' actions during operation, testing and maintenance phases. This enhancement is achieved by applying appropriate ergonomic design principles and providing sufficient time for operators' responses 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 for information and control with:

- Two screen-based workstations for the operators
- 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
- An auxiliary panel to bring the plant to cold shutdown using safety-grade displays and control
- Large plant overview panels that give information on the status and main parameters of the plant

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

The plant also has a Technical Support Center with access to all the data concerning the process and its control. In case of accident, this room is to be used by the technical team in charge of analyzing plant conditions and supporting post-accident management.

The EPR's computerized control room features control screens providing relevant summary information on the process (computer-generated picture).

SAFETY

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

NUCLEAR SAFETY

The fission of atomic nuclei in reactors to generate heat creates large quantities of radiation-emitting radioactive substances from which people and the environment must be protected. Thus 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, to prevent the risks of an accident, and to limit its consequences in the unlikely event of its occurrence.

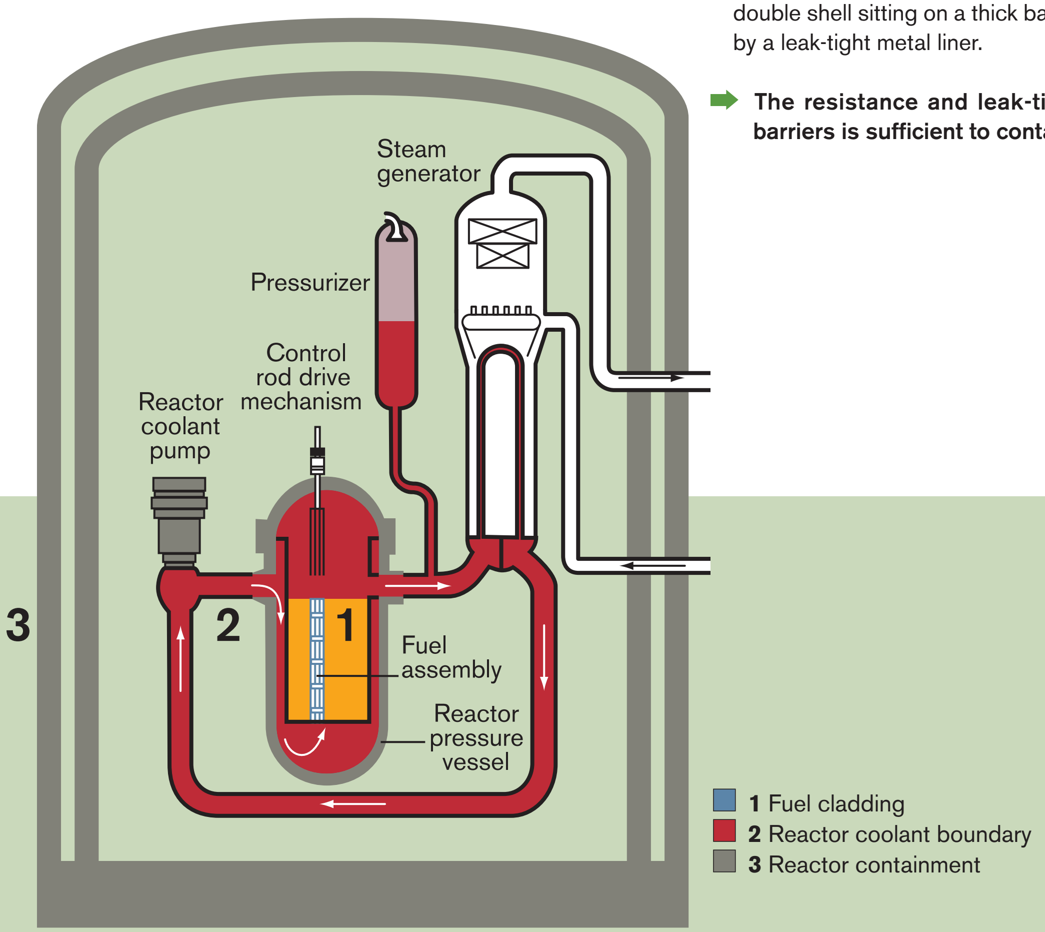
Nuclear reactor safety requires that three functions be fulfilled at all times:

- 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

Nuclear reactor safety relies on two main principles:

- Three protective barriers
- Defense in depth

The three protective barriers



THREE PROTECTIVE BARRIERS

The concept of the "three protective barriers" involves placing a series of strong, leak-tight physical barriers between the radioactive products and the environment 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, comprised by the fuel within its cladding.
- **Third barrier:** The reactor coolant system is also enclosed within a thick concrete construction. For the EPR, this construction is a double shell sitting on a thick basemat with an inner wall covered by a leak-tight metal liner.

➡ **The resistance and leak-tightness 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 that will guarantee a high degree of effectiveness.

- **First level:** safe design, quality workmanship, diligent operation, with incorporation of the lessons learned and experience feedback in order to prevent occurrence of failures
- **Secondlevel:** 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
- **Thirdlevel:** means of action for mitigating the consequences of failures and preventing core melt down. This level includes use of redundant systems to bring the reactor to safe shutdown automatically. 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 – ensure 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.**

➡ **These systems are diversified to prevent a single failure from concurrently affecting several of the systems providing the same function.**

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

The training for steam generator inspection illustrates:

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

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

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

EPR SAFETY

The first important consideration was to design the EPR using an evolutionary approach based on the experience feedback from the 96 reactors built by AREVA NP around the world. This approach enables AREVA NP to offer an evolutionary reactor based on the latest projects (N4 reactors in France and KONVOI reactors in Germany) and to avoid the risk of unproven technologies.

Innovative solutions, backed by the results of large-scale research and development programs have also been incorporated. Indeed, these solutions contribute to the accomplishment of the EPR design objectives, especially in terms of safety and, in particular, regarding the prevention and mitigation of hypothetical severe accidents.

These 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 to further reduce the probability of core melt, and
- simultaneously incorporating 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

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 from the design stage
- Choices made regarding equipment and systems to reduce the risk of an abnormal situation deteriorating into an accident
- Advances in reliability of operator action

The EPR complies with international safety objectives for future PWR power plants:

➡ **Further reduction of core melt probability**

➡ **Practical elimination of accident situations that 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.

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

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. The large-scale probabilistic assessments conducted for the EPR from the design phase 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 failures and hazards, which fully meets the objective for new nuclear power plants set 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 of 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

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, or explosion).

The safeguard systems are designed on the basis of quadruple redundancy, both for the mechanical and electrical portions and for the I&C. This means that each system is comprised of four sub-systems or "trains," each capable 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:

- **Low-headand medium-head injection systems for borated water safety injection into the reactor vessel in case of loss-of-coolant accident**
- **A steam generator emergency feedwater system**
- **The electrical systems and I&C linked to these systems**

The reactor building, fuel-storage building, and the four safeguard buildings are given special protection against external 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 4.3-foot thick reinforced concrete and an inner shell made of post-tensioned concrete (also 4.3 feet thick), which is internally covered with a 0.25-inch thick metallic liner. The thickness and the reinforcement of the outer shell alone 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 safeguard buildings, 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 that 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 to reduce the risk of an abnormal situation deteriorating into an accident.

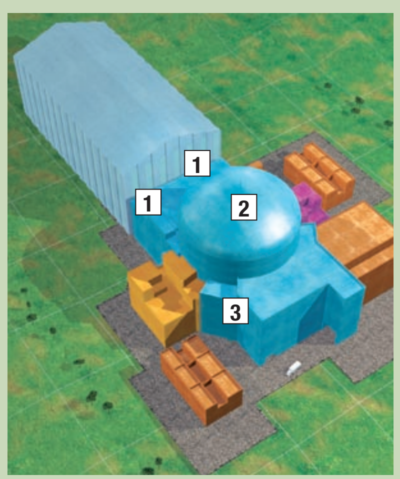
Elimination of the risk of a large reactor coolant pipe break

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

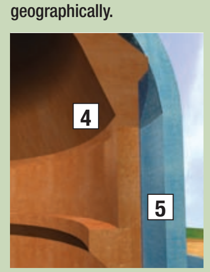
The major safety systems comprise four sub-systems or trains, each capable of performing the entire safety function on its own. There is one train in each of the four safeguard buildings (1) surrounding the reactor building (2) to prevent a simultaneous failure of the trains.

➡ **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 4.3 ft 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. If a given pressure threshold is reached, the activation of the safety injection of water into the reactor vessel will occur automatically. In such a case for the EPR, 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. This has the dual advantage of avoiding potential liquid releases and considerably reducing 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 systems and their support systems each feature 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 (cold legs are 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 (hot legs are located between the steam generators and the reactor vessel). The water reserve intended to feed the safety injection system is at the bottom of the inside of reactor containment, and the injection pumps take suction only 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 dedicated system for cooling the inside of the reactor containment, which would be activated only in the event 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 event of a leak or break occurring in 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 event 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 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 of 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.

The ergonomics of the EPR control room benefits from the latest developments (computer-generated picture).

➡ **Design of components, a high degree of automation, advanced solutions for I&C and man-machine interface combine to increase reliability of operator actions.**

DESIGN CHOICES FOR LIMITING THE CONSEQUENCES OF A SEVERE ACCIDENT

➡ **Although highly unlikely, a core melt accident would necessitate only very limited off-site measures.**

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

The policy of mitigation of the consequences of a severe accident that guided the EPR design is aimed to:

➡ **Practically eliminate the situations that could lead to large early radiological releases such as –**
- **High-pressure core melt**
- **High-energy corium/water interaction**
- **Hydrogen detonation inside the reactor containment**
- **Containment bypass**

➡ **Ensure the integrity of the reactor containment even in the event of a low-pressure core melt followed by ex-vessel progression, through –**
- **Retention and stabilization of the corium inside containment**
- **Cooling of the corium**

➡ **Situations that could generate significant radioactivity release are virtually eliminated.**

Prevention of high-pressure core melt

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

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

The relieving capacity provides fast primary depressurization to low pressure, precluding risk of containment pressurization through dispersion of corium debris in the event of vessel rupture.

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

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

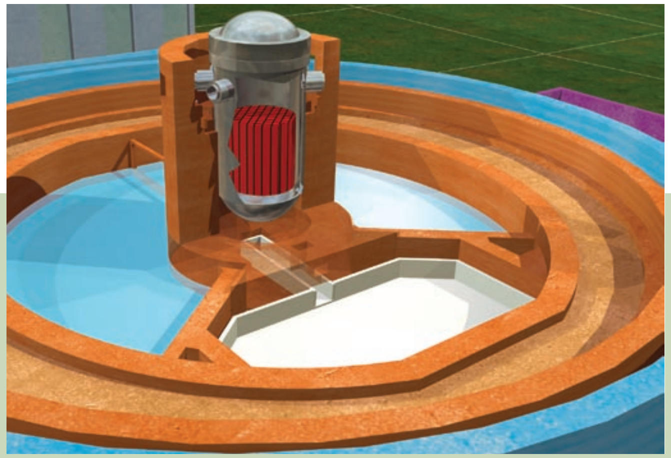
Containment design with respect to the Hydrogen risk

In the unlikely case of a severe accident, Hydrogen would be released in large quantities inside containment. This would happen first 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 structure is designed to withstand the pressure that could result from the combustion of the Hydrogen. Further, catalytic Hydrogen recombiners are installed inside the containment to keep the average concentration below 10% at all times to avoid any risk of detonation.

Corium retention and stabilization to protect the basemat

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 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 equipped with a solid metal structure and covered with "sacrificial" concrete. It protects 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 (1,830 ft²) 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 passively cooled by gravity injection of water from the tank located inside containment and by evaporation.

This effective cooling process would stabilize the corium within a few hours, and completely solidify it within a few days.

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

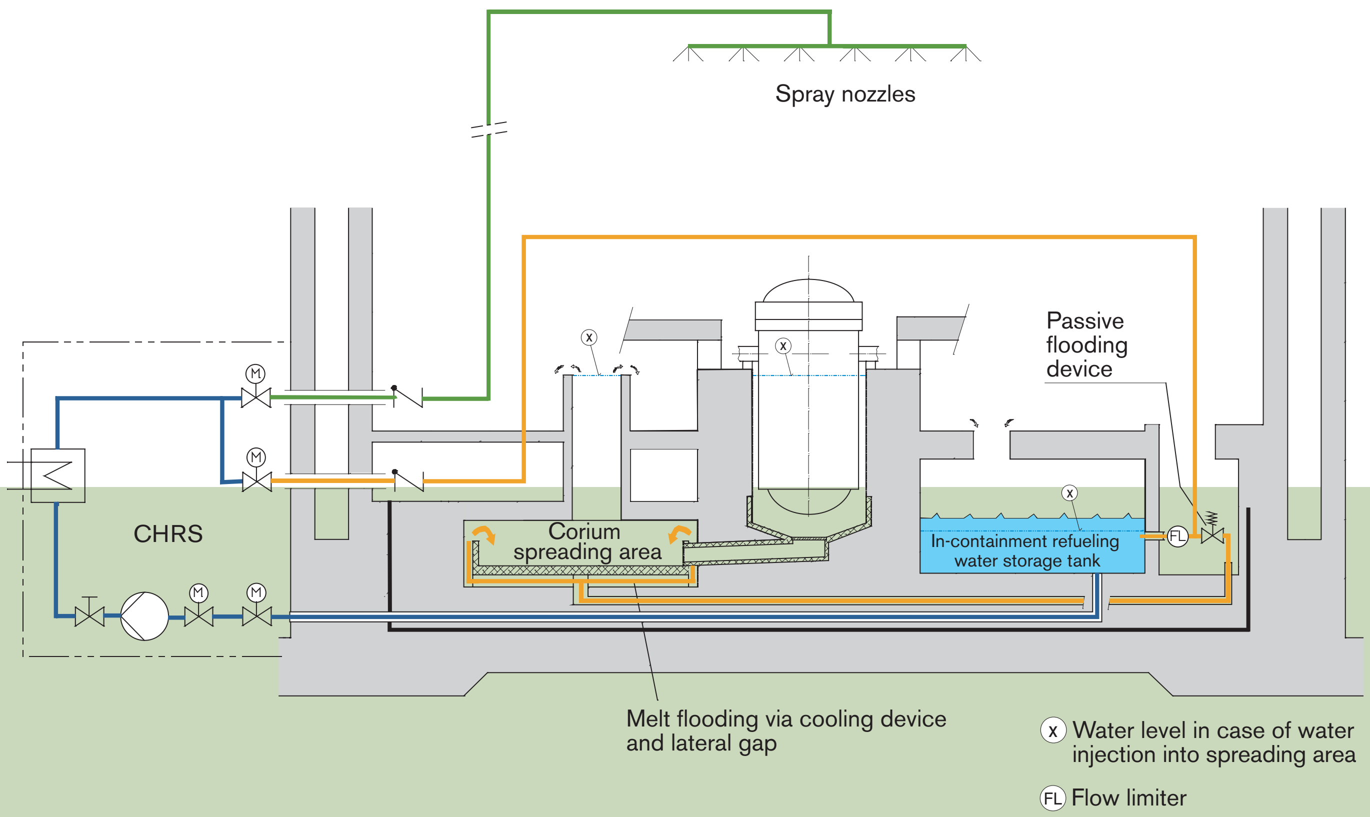
To prevent the containment from losing its long-term integrity in the event of a severe accident, a dedicated spray system with heat-exchangers and dedicated heat sink controls the pressure inside containment and keeps it from rising under the effect of residual heat. A long time period is available for the deployment of this system by the operators: at least 12 hours because of the large containment volume (~2.82 million ft³).

A second mode of operation for the containment heat removal system enables the water to flow directly into the spreading area instead of into the spray system.

Collection of inter-containment leaks

In the event of a core melt leading to vessel failure, reactor containment remains the last of the three protective barriers, thus it must remain undamaged and leak-tight. For the EPR, the following measures have been adopted to ensure its integrity:
- A 0.25-inch thick metal liner covers the inside of the pre-stressed concrete inner shell.
- 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 (the annulus) is 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 task sequencing implemented by AREVA NP and our partners worldwide. Improvements continue to be made in the design, construction and commissioning methods to accelerate the EPR construction schedule as much as possible.

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 construction activities for all work disciplines.

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

CONSTRUCTION AND INSTALLATION METHODS

Three main principles are applied to EPR construction and installation:

Coordination of the interfaces between civil works and installation of mechanical components. The ongoing pursuit for the optimization of interfaces between civil and mechanical/electrical installation 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 in parallel.

Open-top installation of civil and mechanical assemblies. "Open-top" construction practices will be employed to allow equipment and materials to be positioned by cranes into the buildings before the next floor is constructed. This approach allows the installation and final assembly of those components to start while civil work on the floors above is being performed. To enhance "open-top" construction, emphasis will be placed on prefabrication, pre-assembly, and modularization to the maximum extent practical.

Use of prefabrication, pre-assembly and 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 burden or cost disavantage due to advanced detail design, procurement or prefabrication.

Opportunities for pre-assembly and modularization will be considered where critical-path construction schedule benefits can be realized, or where significant work can be performed by an alternative off-site workforce. Pre-assembly and modularization allow for a greater percentage of the bulk item construction/installation to be shifted from the traditional "on-site, in power block," to off-site fabrication or "on-site but out of power block" fabrication, and transported to the site/power block for large-volume, high-quantity single installations. This approach shifts labor to fabrication areas where critical-path activities can be performed in parallel, capitalizes on the production efficiencies of fabrication outside the power block, and reduces power block congestion.

Prefabrication and pre-assembly will be implemented for fabrication of items such as re-bar mats, piping and equipment skids, heat exchangers, civil dependent tanks, and switchgear cabinets to minimize construction man-hours and schedule risk. Modules are implemented mainly for portions of the civil works of the reactor building such as the reactor pit, the reactor cavity and liner, the containment liner, and the containment dome, as they are all on the critical path for the construction of the reactor building.

MAIN COMPONENTS MANUFACTURING

AREVA NP's Chalon/Saint-Marcel and Jeumont plants along with our U.S. supply partners have over 30 years of experience in the manufacture of heavy nuclear components. These vital facilities have the know-how necessary to optimize heavy nuclear component production time. Construction of the U.S. EPR will benefit from their unique manufacturing capability and expertise.

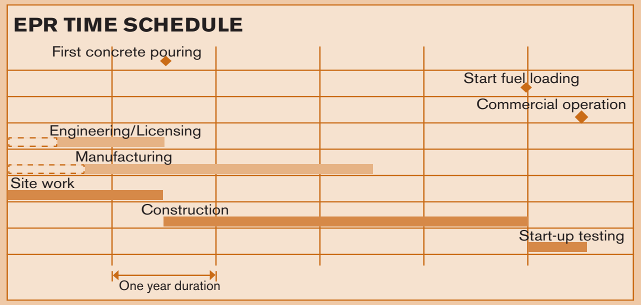
COMMISSIONING TESTS

As with the interfaces between civil and mechanical/electrical installation, the interfaces between construction and testing 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 carried out only once.

Experience gained from AREVA NP's past achievements, associated with the systematic analysis of possible improvements and optimization of construction and test activities results in an optimal technical and economical construction schedule for EPR projects.

Indicative planning and overall time-scale

The overall construction schedule is dependent on site conditions, organization and policies, and local working conditions. Shown below is the typical U.S. EPR construction timeline, which can be tailored to meet site-specific conditions and requirements.



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

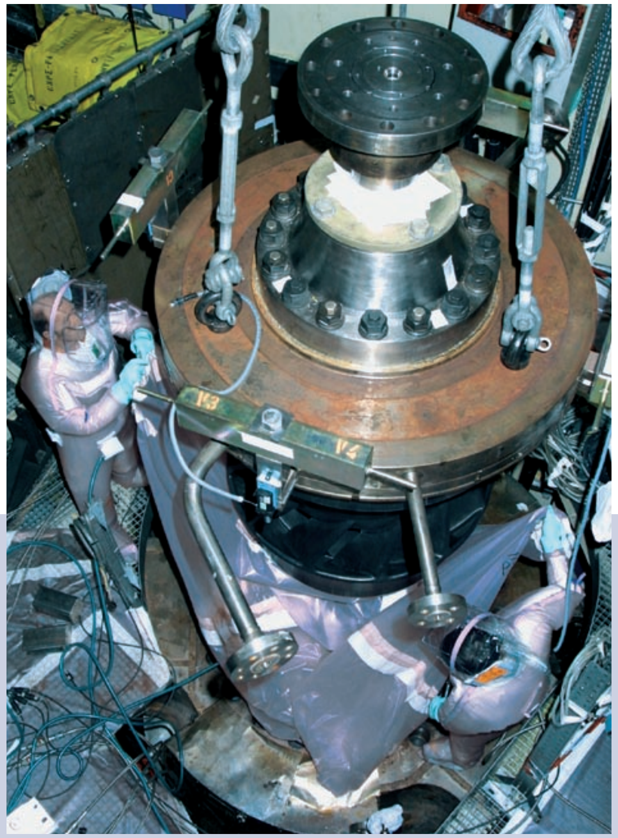
From the beginning, EPR equipment and systems have been designed to allow efficient refueling outages, and to simplify and optimize inspection and maintenance. The result is increased plant availability and reduced maintenance costs, two major objectives of plant operators worldwide to meet the demands of increasingly competitive power markets.

94% AVAILABILITY FACTOR OVER THE ENTIRE PLANT LIFE

The U.S. EPR is designed to achieve greater than 94% availability on average over its entire 60-year design life objective. This high availability is made possible by shorter outage durations for fuel reloading, in-service inspections and maintenance, and by reduced downtimes attributable to unscheduled outages.

The high degree of equipment reliability and the decrease in unplanned reactor trips (in particular due to special EPR design features) means 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 accessibility enables outage and maintenance operations to be prepared and demobilized with no loss of availability. Accessibility of the reactor while on line also facilitates field services, which could be needed outside of scheduled outage periods. Based on experience feedback, standardization and ease of access to reactor components allow simple, rapid performance of inspection and maintenance work.

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

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 also reduced from a reduction in the time needed to run the beginning-of-cycle core physics tests (gain supplied by the "aeroball" in-core instrumentation system).

➡ **The typical duration of a regular outage for preventive maintenance and refueling is reduced to 16 days. Duration of a refueling outage only should not exceed 11 days. Decennial outages for in-service inspection of main equipment, turbine overhaul and containment pressure testing are planned to last 30 days.**

The EPR is designed to:

- **Maximize plant availability and maneuverability**
- **Ease operation and maintenance and reduce their costs**
- **Enhance radiological protection of personnel**
- **Protect the environment and contribute to sustainable development**

A HIGH LEVEL OF OPERATIONAL MANEUVERABILITY

In terms of operation, the U.S. EPR is designed to provide 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 U.S. EPR's maneuverability is a particularly well adapted response to scheduled and unscheduled power grid demands for load variations, managing of grid disturbances or mitigation of grid failures.

ENHANCED RADIOLOGICAL PROTECTION

Design objectives for the EPR included allowance for operating constraints and human factors, improvement of worker radiation protection, and limitation of radioactive releases, radwaste quantity and activity. For this purpose, the designers drew heavily upon the experience feedback from the operation of existing nuclear power plant fleets

Accordingly, major progress has been made, particularly in the following areas:

- Choice of materials – for example, 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"
- Choices in the design and layout of the components and systems liable to convey radioactivity, taking into account the various plant operating states
- Optimization of the radiation shielding thicknesses in response to forecasted reactor maintenance during outages or in service

Thanks to these significant advances, collective doses of less than 40 person-rem 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 100 person-rem per reactor/year.

PLANT SERVICES

Optimization of plant processes and implementation of innovative maintenance technologies and concepts are also significant contributors to the achievement of operators' cost and availability objectives. In this area, AREVA NP supplies the most comprehensive range of nuclear services and technologies in the world.

AREVA NP's power plant services include:

- In-service inspection and non-destructive examination
- Outage services
- Component repair and replacement, including steam generators and reactor pressure vessel heads
- Supply of spare parts
- Off-site maintenance of components in "hot" workshops
- Fuel inspection, repair and management
- Services in instrumentation and diagnosis, I&C and electrical systems, and chemistry
- Plant engineering and plant upgrading
- Plant decommissioning and waste management
- Training of operating personnel
- Expert consultancy

The "FROG" network (see page 57) offers member electricity companies a cost-effective way to exchange information and experience. FROG's members have access to broad operational and maintenance feedback. They also benefit from the results of joint study programs 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 ambitious 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, AREVA NP 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.

AREVA NP'S SPIRIT OF SERVICE

➡ To satisfy customers and help them to succeed in a highly competitive energy market by:

- **Improving safety and performance**
- **Reducing operating and maintenance costs**
- **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 – AREVA NP 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 institutions involved in the extensive long-term nuclear program currently underway in China. An excellent example of this cooperation is the close relationship 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 AREVA NP, which fully benefits from AREVA NP's expertise and technologies in its activity field.

AREVA NP Technical Center (TC), with locations in France, Germany and the U.S., is the first link in 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.

AREVA NP'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**

THE "FROG" OWNERS GROUP

The FROG (formely Framatome Owners Group) is dedicated to building strong, 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 established 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, Ringhals AB from Sweden (June 1997), LANPC, owner of the Ling Ao plant in China (October 2000), and British Energy, owner of Sizewell B in the United Kingdom (October 2002) joined the FROG as members. In 2003, GNPJVC and LANPC merged operation of their plants into 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 challenges.

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 AREVA NP. Committee participants are the FROG members plus the companies NSP and AmerenUE from the U.S., NOK from Switzerland and NEK from Slovenia.

> CONCLUDING REMARKS

Electric utilities can meet the growing demand for power in their service areas by adding the U.S. EPR to their generation portfolio. The U.S. EPR provides the greatest certainty of success for reliable, new baseload generation in the near term by providing the following advantages:

Project certainty

- Most reliable knowledge base for cost, licensing and construction
- An established global supply chain and re-establishing the U.S. supply chain
- Predictable, timely licensing schedule supporting new nuclear generation in 2015 timeframe
- Proven, evolutionary design derived from familiar PWR technology minimizes risk
- Completed, detailed design for the EPR

Lowest total lifecycle costs

- Efficient fuel use – consumes 7% less Uranium/MWh
- 94% availability on average across a 60-year design life objective
- Allows online maintenance, greatly reducing outage time and costs
- Fewer components to maintain than an existing plant of comparable power output, reducing operating and maintenance costs
- Part of a global standardized fleet for tremendous savings in training, component manufacture, spare parts and many other areas

Robust safety and operating margins

- Enhanced safety for multiple 21st century scenarios
- Double-walled containment, physical separation of the four safety systems, a core melt retention system and many other unique features designed to respond to severe accident challenges
- Greatest physical protection against external hazards such as earthquake and airplane crash
- Heightened protection against accidents and their radiological consequences
- Supports future power uprates

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

On May 4, 2006, the Board of Directors of EDF decided to launch the building of its first EPR unit in France on the Flamanville site.

> U.S. EPR

Key to nuclear plant cutaway

1 Reactor building: inner and outer shell
- 2 Polar crane
- 3 Severe accident heat removal spray system
- 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 SG blowdown system heat exchanger
- 17 Safety injection accumulator tank
- 18 Pressurizer
- 19 Main steam isolation valves
- 20 Feedwater isolation valves
- 21 Main steam safety and relief valve exhaust silencer

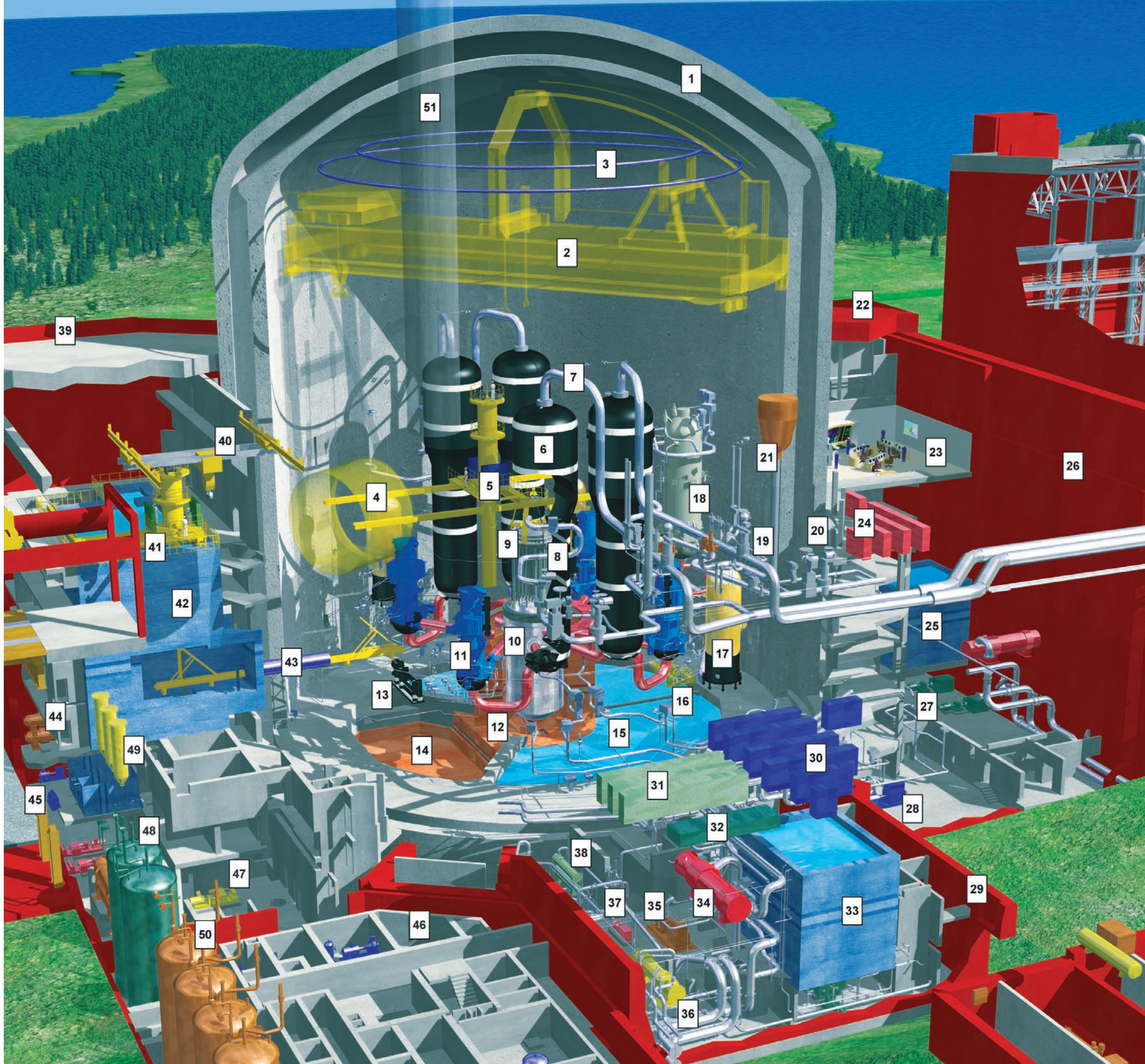
22 Safeguard building division 2
- 23 Main control room
- 24 Computer room
- 25 Emergency feedwater storage pool, division 2

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 Emergency feedwater storage pool, division 4
- 34 CCWS heat exchanger, division 4
- 35 Low head safety injection pump, division 4
- 36 Residual heat removal system heat exchanger, division 4
- 37 Severe accident heat removal system pump, division 4
- 38 Severe accident 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 40 countries and a sales network in more than 100, AREVA offers customers reliable technological solutions for CO_2-free power generation and electricity transmission and distribution. We are the world leader in nuclear power and the only company to cover all industrial activities in this field.

Our 58,000 employees are committed to continuous improvement on a daily basis, making sustainable development the focal point of the group's industrial strategy.

AREVA's businesses help meet the 21st century's greatest challenges: making energy available to all, protecting the planet, and acting responsibly towards future generations.

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