Chapter 2
Design and Operation of a Pressurised Water Reactor

2.1. General information about reactor operation

The nuclei of some isotopes contained in nuclear fuel, such as ^{235}U and ^{239}Pu, can split up (fission) into two[1] smaller fragments called "fission products". These fragments have large amounts of kinetic energy that is mainly released as kinetic thermal energy in the surrounding fuel material. This release of energy is used to generate electricity in power reactors. Fission into two fragments can either be induced by neutrons (induced fission) or occur spontaneously in the case of heavy isotopes (spontaneous fission). Fission is accompanied by the release of two to three neutrons. Some of these neutrons may in turn initiate other fissions (the principle behind a nuclear chain reaction), be absorbed into the fuel without initiating any nuclear fission, or escape from the fuel.

Neutrons produced by fission from the neutrons of one generation form the neutrons of the next generation. The effective neutron multiplication factor, k, is the average number of neutrons from one fission that cause another fission. The value of k determines how a nuclear chain reaction proceeds:

– where $k < 1$, the system is said to be "subcritical". The system cannot sustain a chain reaction and ends up dying out;

1. In about 0.4%-0.6% of cases the fission can be into three fission products, this is termed "ternary fission".

– where $k = 1$, the system is "critical", i.e., as many neutrons are generated as are lost. The reaction is just maintained. This situation leads to a constant power level;

– where $k > 1$, the system is "supercritical". For every fission there will be an average of k fissions in the next generation. The result is that the number of fissions increases exponentially.

There are in fact two types of supercritical situation: prompt supercriticality and delayed supercriticality. Nearly all fission neutrons are immediately emitted (for example, 99.3% of neutrons are released as 10^{-7} s for ^{235}U); these neutrons are called "prompt neutrons". However, a small fraction of fission products are de-excited by beta decay (β decay) and subsequently emit what are termed "delayed neutrons". β decay occurs any time from a few tenths of a second to several tens of seconds after the fission event. The fraction of delayed neutrons is typically less than 1% of all the neutrons generated at any time in a chain reaction. During the interval between $k = 1$ and $k = 1/(1 - \beta) \approx 1 + \beta$, supercriticality is referred to as "delayed"; when $k > 1/1(1 - \beta) \approx 1 + \beta$, supercriticality is referred to as "prompt". The value of the fraction of delayed neutrons representing the interval between delayed and prompt supercriticality is defined as a "dollar" and depends on the isotope.

To produce energy, nuclear reactors operate in the region of delayed supercriticality for it is in this region that, thanks to the presence of delayed neutrons, changes in reaction rates occur much more slowly than with prompt neutrons alone. Without delayed neutrons, these changes would occur at speeds much too fast for neutron-absorbing systems to control.

The order of magnitude commonly used to express system departure from criticality is known as "reactivity" ρ, $\rho = 1 - 1/k$. Positive ρ values correspond to supercritical states and negative values correspond to subcritical states.

Chain reactions in nuclear reactors must be controlled, i.e., zero or negative reactivity must be maintained with the aid of neutron-absorbing elements. In pressurised water reactors, these elements are either placed inside mobile devices called control rods (containing chemical elements such as cadmium and boron) or dissolved in the cooling water (boron).

In some low-probability accidents, the reactivity of the reactor may reach high positive values that cause the chain reaction to become supercritical. If the measures taken are insufficient to bring the reactor back to a safe condition, such accidents could lead to an uncontrollable power increase that could result in severe reactor damage like that which occurred during the Chernobyl accident (Section 7.2).

The reactivity of a reactor is affected primarily by the temperature of both the fuel and the coolant and by the coolant void fraction. The influence of each of these parameters is characterised by a reactivity coefficient, which is the derivative of the reactivity with respect to the parameter considered. In the case of fuel, an increase in power results in an increase in fuel temperature and an increase in neutron capture by ^{238}U. The reactivity coefficient, called the temperature coefficient or the Doppler coefficient, is therefore negative. In the case of coolant, the reactivity coefficient is related to changes in the coolant density (temperature coefficient) or void fraction (void coefficient). These coefficients are negative in pressurised water reactors[2] to ensure reactor stability and limit the maximum power that could be reached during an accident.

Some fission products formed are radioactive. This radioactivity results in, even after the chain reaction stops, energy being released in the form of heat (called "decay heat"). This heat decreases over time and, one hour after reactor shutdown, amounts to approx. 1.5% of its level during operation[3].

The energy released by fissions and fission products must be continuously removed to avoid an excessive rise in reactor temperature. In pressurised water reactors, this energy is removed during normal conditions by three successive loops whose main purpose is to prevent the radioactive water exiting the core from leaving the plant (Figure 2.3):

- the first loop is the reactor coolant system (RCS). It cools the core by circulating water at an average temperature of around 300 °C and a pressure of 155 bar;
- the secondary loop extracts the heat from the RCS by means of steam generators, which supply steam to the turbine generator to produce electricity;
- the tertiary system consists of a condenser and rejects the remaining heat to a river or the sea or to the atmosphere by means of cooling towers.

This brief description of the operation of a nuclear reactor identifies the basic safety functions that must be ensured at all times:

- reactivity control;
- heat removal;
- containment of fission products and, more generally, radioactivity (some activation products in the RCS[4] are also radioactive).

2.2. The pressurised water reactors in France's nuclear power plant fleet

Various types of nuclear reactor are used to generate electricity in France. They use different fissile materials (natural uranium, uranium enriched in uranium-235, plutonium, etc.) and different neutron moderators (graphite, water, heavy water, etc.)[5]. They

2. Water is used as the moderator in pressurised-water reactors. It decelerates neutrons produced by fission (these neutrons lose their kinetic energy by colliding with the nuclei of the water's hydrogen atoms) and increases fission product yields. As the temperature inside the reactor core increases, the water expands. This reduces the water's ability to slow down neutrons and results in fewer fission reactions. The temperature coefficient of the water is thus negative.
3. One hour after reactor shutdown, a 900 MWe reactor generates 40 MW of heat and a 1300 MWe reactor generates 58 MW of heat. One day after shutdown, this heat output drops to 16 MW for a 900 MWe reactor and 24 MW for a 1300 MWe reactor.
4. Radioactive substances may be formed under irradiation by activation of the metal components in the RCS and be entrained into the reactor coolant by corrosion mechanisms.
5. The moderator reduces the velocity of the neutrons, thereby increasing their likelihood of producing a fission reaction.

are also characterised by the type of coolant (ordinary water in liquid or vapour form, heavy water, gas, sodium, etc.) used to remove heat from the core (where fission reactions occur) and transfer it either to the loops supplying the turbine generators or to the turbine generators directly.

The nuclear power plants currently in operation in France use enriched uranium in oxide form that may be mixed with plutonium oxide recovered from the reprocessing of spent fuel. They use ordinary water as the heat-transfer fluid. This water is maintained under high pressure (155 bar) so that it remains in liquid form at its operating temperature (300 °C). They are known as pressurise water reactors (PWRs) and belong to what is commonly known as the second generation of nuclear power reactors[6].

A distinctive feature of France's reactor fleet is its standardisation. The technical similarity of many of the country's reactors justifies the generic overview given in this chapter. The 19 nuclear power plants in operation in France have two to six PWRs, giving a total of 58 reactors. This reactor fleet consists of three series: the 900 MWe series, the 1300 MWe series, and the 1450 MWe (or N4) series (Figure 2.1).

The thirty-four 900 MWe reactors are split into two main types:

– CP0, which consists of the two reactors at Fessenheim and the four reactors at Bugey;
– CPY (consisting of types CP1 and CP2), which encompasses the 28 other reactors (four reactors at Blayais, four at Dampierre, six at Gravelines, four at Tricastin, four at Chinon, four at Cruas-Meysse and two at Saint-Laurent-des-Eaux).

The twenty 1300 MWe reactors are split into two main types:

– the P4, which consists of eight reactors: two at Flamanville, four at Paluel and two at Saint-Alban;
– the P'4, which consists of 12 reactors: two at Belleville-sur-Loire, four at Cattenom, two at Golfech, two at Nogent-sur-Seine and two at Penly.

Lastly, the N4 series consists of four 1450 MWe reactors: two at the Chooz nuclear power plant and two at the Civaux nuclear power plant.

Despite the deliberate standardisation of France's fleet of nuclear power reactors, technological innovations have been introduced during the design and construction of each plant. The creation of France's fleet occurred in four main stages:

– the CP0 900 MWe "preproduction" series was brought into operation between 1977 and 1979;
– the CPY 900 MWe series was brought into operation between 1980 and 1987;

6. Reactors built before the 1970s make up the first generation. The Generation-I reactors in France were graphite moderated, cooled by carbon dioxide, and fuelled with natural uranium metal. They were a type of gas-cooled reactor (GCR).

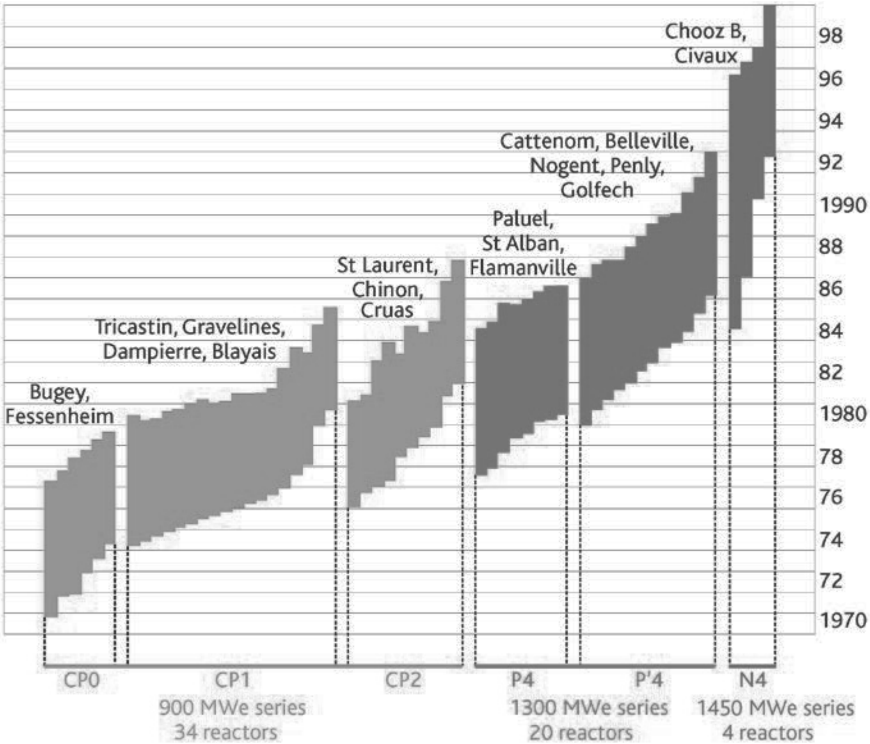

Figure 2.1. Construction periods and distribution of the three series of 900, 1300 and 1450 MWe power reactors in operation in France in 2015.

– the P4 and P'4 1300 MWe series were brought into operation between 1984 and 1993;

– the 1450 MWe (or N4) series was brought into operation between 2000 and 2002.

The CPY reactors benefited from the feedback obtained from the design studies, construction and operation of the CP0 reactors. Unlike the design studies for the CP0 series, which were conducted separately for each site, the design studies for the CPY series were conducted for all the sites. As a result, the CPY series differs from the CP0 series in terms of building design (in particular, the containment building was modified to facilitate operations), siting of the engineered safety systems (which were modified to increase the independence of the systems' trains and increase their reliability) and more flexible reactor control (particularly via the use of control rods and the addition of control rods with less neutron-absorbing capacities[7]). In the case of the CP2 reactors, the orientation of the

7. The control-rod clusters are made up of 24 rods. There are two types of control-rod cluster, "black" and "grey". Black clusters have 24 neutron-absorbing rods (consisting of a silver, indium and cadmium alloy (Ag-In-Cd) or boron carbide [B_4C]). Grey clusters consist of rods made of materials with varying degrees of absorbency (e.g., only eight Ag-In-Cd or B_4C absorbing rods and 18 rods made of steel, which is more transparent to neutrons). Moving these clusters at different rates in the core makes it possible to optimise the spatial power distribution, control changes in reactor power and adjust the mean temperature of the reactor coolant.

control room was shifted by 90 degrees to prevent projectiles generated by rupture of the turbine generator from damaging the reactor containment vessel (Figure 2.2).

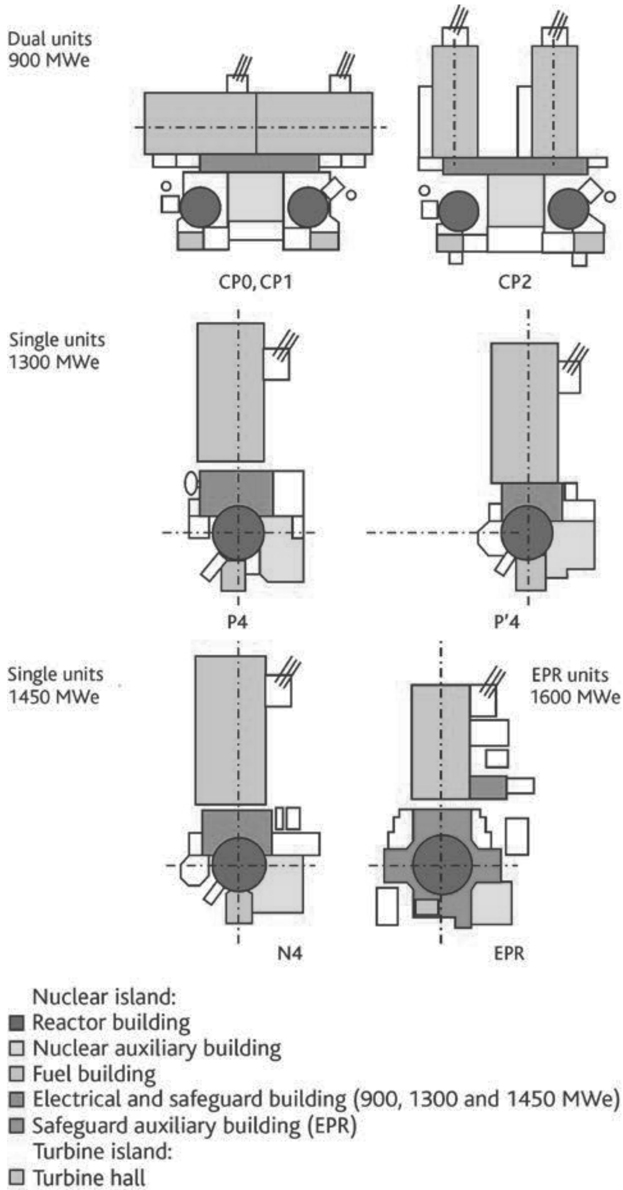

Figure 2.2. Schematic plant layout showing the buildings of the different reactor series in operation in France.

The 1300 MWe reactors differ from the 900 MWe reactors in terms of the design of their core, loops and reactor protection system as well as their buildings. The increase in power was achieved by increasing the size of the reactor. In order to remove the increased heat (from 900 to 1300 MWe), the cooling capacity of the RCS was increased by the installation of an additional cooling loop (thus changing the number of loops from three for the 900 MWe reactors to four for the 1300 MWe reactors) (Figure 2.3). The components of each RCS are also larger than those of the preceding series. In terms of the locations of the buildings, the new series are single-unit plants, whereas the preceding series were dual-unit plants (Figure 2.2). The engineered safety systems and auxiliary systems are located in buildings specific to each unit so as to improve the safety of their operation. In addition, each containment vessel has a double concrete wall (an inner wall of prestressed concrete and an outer wall of reinforced concrete) instead of the single wall of steel-lined prestressed concrete on the 900 MWe reactors. New microprocessor-based instrumentation and control technologies using programmable memory are used. The P'4 series differs from the P4 series in that the installation of the buildings and structures was optimised with the primary goal of reducing costs. The result is a denser plant layout and smaller buildings and structures.

Lastly, the main differences between the 1450 MWe reactors and those of the preceding series are the larger reactor core, smaller steam generators (SG) that delivery steam at higher pressure, the design of the reactor coolant pump (higher flow rate) and the computerised control system.

The next generation of reactor that EDF is planning to put into service in France will consist of a design known as the European Pressurised Water Reactor, or EPR). A reactor with a power output of around 1600 MWe is currently under construction at EDF's Flamanville site, on France's Cotentin Peninsula on the English Channel. These new PWRs incorporate evolutionary improvements over earlier designs. They therefore benefit from extensive operating experience feedback from the current fleet and meet more stringent safety objectives. They also benefit research findings, particularly regarding core melt accidents, which were factored in right from the design phase. Their main differences with the Generation-II PWRs are the design of the loops, the reactor protection system and the site buildings (particularly the containment), which offer a higher degree of protection in the event of an accident.

The design of the RCS and the main components and the configuration of the loops are quite similar to those of the N4 series. The main evolutionary improvements are as follows:

- increase in the volumes of primary and secondary water (particularly in the steam generators) to increase the thermal inertia of the reactor;
- organisation of the engineered safety systems and the support systems (safety injection system [SIS], steam generator emergency feedwater system [EFWS], component cooling-water system [CCWS], essential service-water system [ESWS], emergency power supplies [EPS]) into four independent trains located in physically separate rooms. This physical separation ensures that the engineered safety systems remain available in the event of an internal or external hazard (e.g., fire, earthquake or flood).

Regarding the containment, in addition to the reinforcement of its structure (more specifically the outer concrete wall, see Section 2.3.2.3), the following changes have been made in relation to those of the N4 series:

– placement of the borated-water storage tank inside the containment, hence the name "in-containment refuelling water storage tank" (IRWST). The IRWST feeds the safety injection system and the containment heat-removal system (CHRS);
– installation of a system for containing and cooling molten corium inside a special compartment in the event of a vessel melt-through during a core melt accident. The purpose of this system is to provide long-term protection of the basemat from erosion should such an accident occur;
– installation of a steel liner on the inner wall of the double-wall containment.

Another notable difference with the N4 series is that more rooms are protected by the reinforced-concrete outer wall (airplane crash [APC] shell). In addition to the reactor building, the fuel building and two of the rooms housing the engineered safety systems are covered by the outer concrete wall.

The layout of the buildings (Figure 2.2) was changed so that the four independent trains of the engineered safety systems and support systems could be housed in separate rooms and thus prevent leaks being released directly into the environment from the containment. All the containment penetrations lead into buildings located around the reactor building and equipped with ventilation and filtration systems.

To provide the reader with the information needed to understand the concepts presented in this document, the rest of this chapter provides a relatively generic, summary overview of the main components of the reactors in operation in France and of how these reactors function under normal and accident conditions. The specific features of the EPR are described whenever they relate to core melt accidents.

2.3. Description of a pressurised water reactor and its main loops

2.3.1. Facility overview

Each reactor comprises a nuclear island, a turbine island, water intake and discharge structures and, in some cases, a cooling tower (Figures 2.2 and 2.3).

The main parts of the nuclear island are:

– the reactor building (RB), which contains the reactor and all the pressurised coolant loops as well as part of the loops and systems required for reactor operation and safety (Figures 2.3, 2.6 and 2.7);
– the fuel building (FB), which houses the facilities for storing and handling new fuel (pending its loading into the reactor) and spent fuel (pending its transfer to reprocessing plants). The fuel building also contains the equipment in the

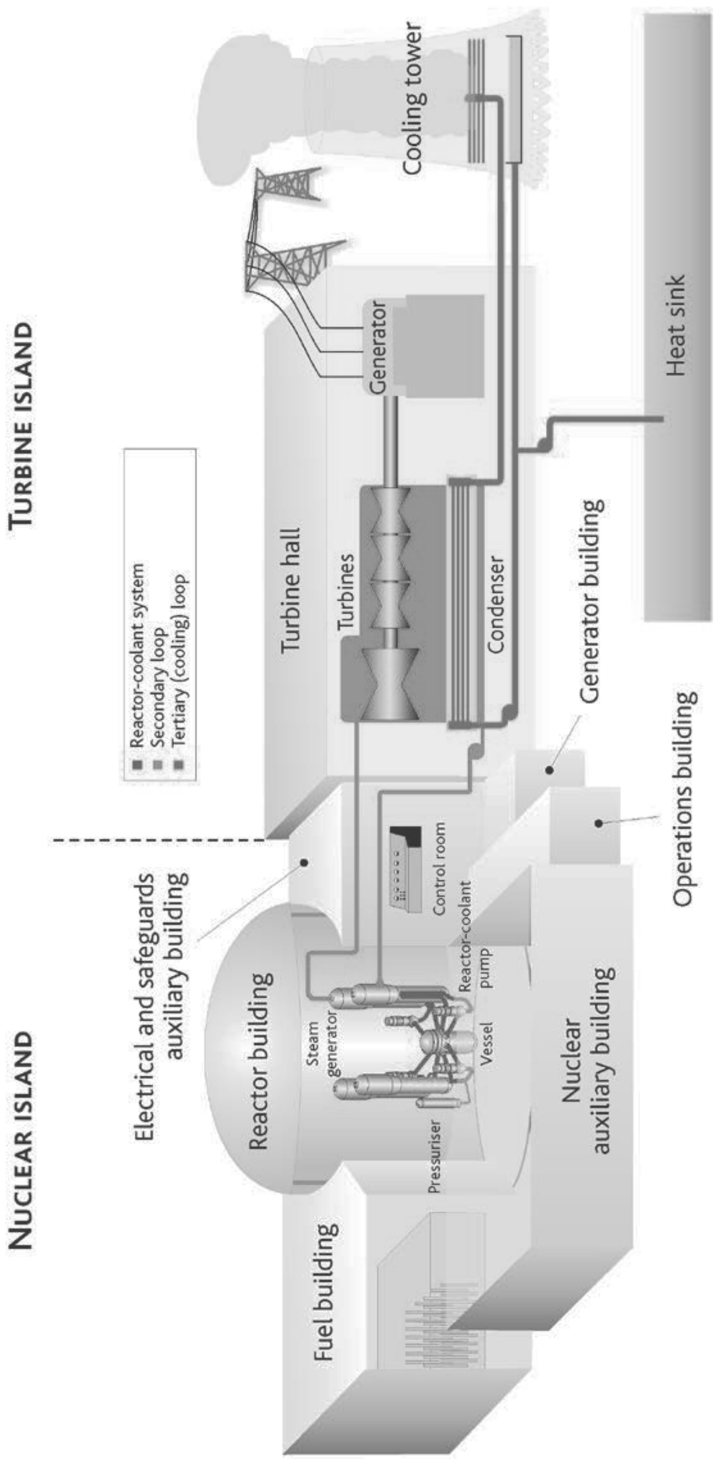

Figure 2.3. Schematic diagram of a PWR (1300 MWe or N4) and its main loops.

fuel pool cooling and purification system (FPCPS) and, for units in operation, the equipment in the steam generator emergency feedwater system (EFWS). The EPR itself has four independent steam generator emergency feedwater trains. Each train is located in one of the four divisions of the safeguard auxiliary building;

– a safeguard auxiliary building (SAB) with electrical equipment rooms. The main engineered safety systems are located in the SAB's bottom half and the electrical equipment rooms are located in its top half. These two halves do not communicate with each other. The rooms in the SAB contain equipment, particularly that of the safety injection system (SIS), the containment spray system (CSS), the component cooling water system (CCWS) and ventilation equipment. The electrical equipment rooms contain all the means for controlling the unit (the control room and operations facilities, electric power supplies, and the instrumentation and control [I&C] system). Note that, in the case of the 900 MWe series, there is only one SAB with electrical equipment rooms for two adjoining units. In the case of the 1300 MWe and N4 series, there is only one building per unit. The EPR has four independent engineered safety systems. Each is located, with its support systems, in a room that is physically separate from the others. These rooms are known as the "divisions" of the SAB. Divisions 2 and 3 of the SAB are protected by the reinforced-concrete outer wall. The control room is located in division 3 of the SAB;

– a nuclear auxiliary building (NAB) housing the auxiliary systems required for normal reactor operation. This building houses the equipment of the chemical and volume control system (CVCS), the gaseous waste processing system, the reactor coolant effluent processing system and the boron recycle system;

– two geographically separate buildings, each housing a diesel generator (emergency power supply). In the case of the EPR, the offsite emergency power supplies consist of two sets of four diesel generators (each set being housed in its own building) and two station blackout (SBO) generators;

– an operations building.

The turbine-island equipment converts the steam generated by the nuclear island into electricity and supplies this electricity to the transmission system. The main parts of the turbine island are:

– the turbine hall, which houses the turbine generator (it receives the steam generated by the nuclear island and converts it into electricity) and its auxiliary systems;

– a pump house to cool the facility under normal operating conditions and provide emergency cooling with the related hydraulic structures;

– a cooling tower in the case of closed-loop cooling.

Some of these items of equipment contribute to reactor safety. The secondary loops are the interface between the nuclear island and the turbine island.

2.3.2. *Description of the main components of a PWR*

2.3.2.1. Reactor core

The reactor core is made up of fuel assemblies (Figure 2.4). Each assembly consists of 264 fuel rods (Figure 2.4, left), 24 tubes to contain the rods of a control rod cluster and a guide tube. All are arranged in a 17 × 17 square lattice (Figure 2.4, right). The fuel rods are made up of zirconium alloy tubes also known as "cladding" (zirconium has low neutron-absorbing properties and good corrosion resistance). Zircaloy, which contains 98% zirconium, is the alloy most frequently used in France's PWRs. The cladding, which is 0.6 mm thick and 9.5 mm in diameter, is held in place by Zircaloy grids. Pellets made of uranium dioxide (UO_2) or a mixture of uranium and plutonium oxides ($(U,Pu)O_2$, commonly referred to as MOX fuel) and measuring 8.2 mm in diameter are stacked inside the rods. These pellets make up the nuclear fuel. The level of ^{235}U enrichment varies

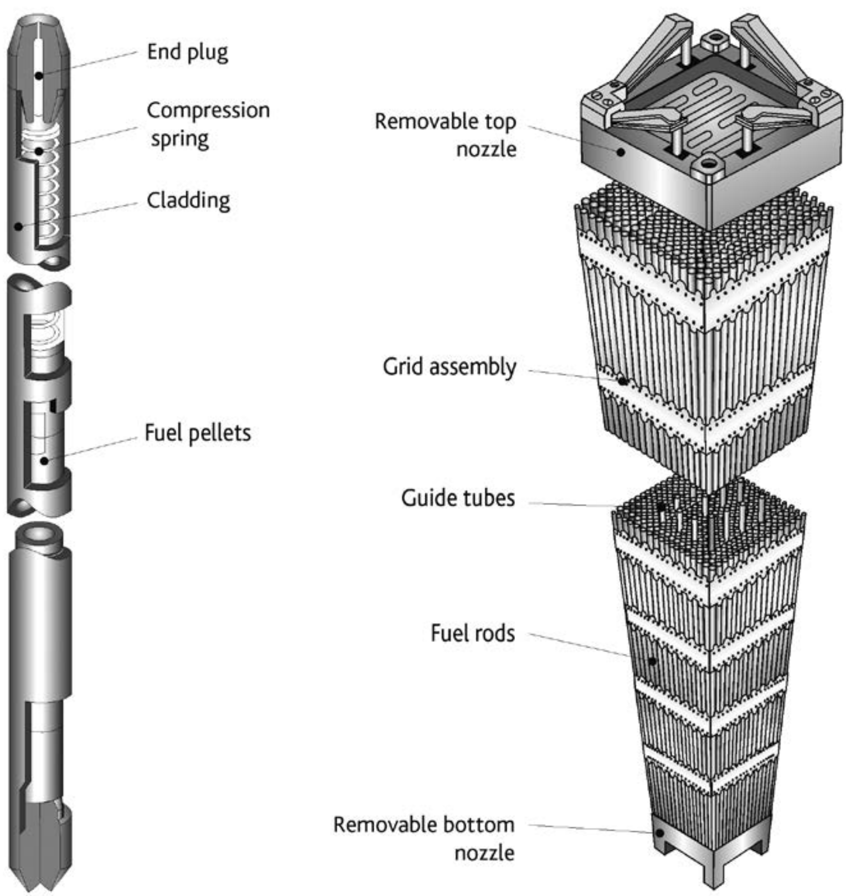

Figure 2.4. Diagram of a fuel rod (left) and of the main components of a fuel assembly (right).

between 3% and 4.5% depending on the method of fuel management[8]. The fuel assemblies are similar for all the series. Only their lengths change. One-third to one-fourth of the fuel is replenished once every 12 to 18 months during reactor outages.

The main characteristics of the fuel and the core are given for each series in Table 2.1.

Table 2.1. Characteristics of the cores of each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Number of fuel assemblies	157	193	205	241
Total height of the fuel pellets in each assembly rod (m)	3.66	4.27	4.27	4.20
Number of control rod clusters Absorbing material	57 Ag-In-Cd	65 Ag-In-Cd + B_4C	73 Ag-In-Cd + B_4C	89 Ag-In-Cd + B_4C
Mass of enriched uranium (t)	72.5	104	110.5	144.2

The core is located inside a vessel made of 16MND5 low-carbon steel fitted with an upper head that is removed for refuelling purposes (Figure 2.5). Inside the vessel are

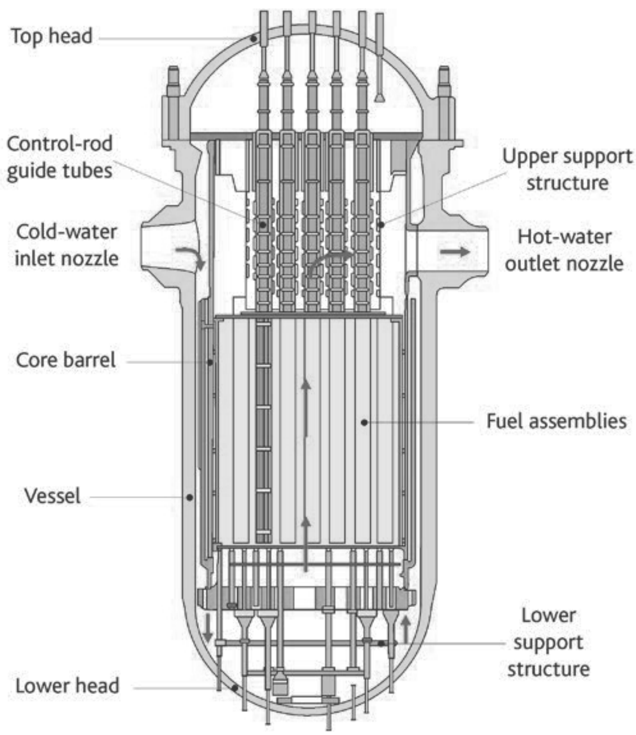

Figure 2.5. Cutaway of the PWR vessel at Fessenheim.

8. During reactor operation, the amount of fissile material in the fuel diminishes, requiring the spent fuel rods to be replaced by new assemblies. The method of managing this replacement depends on the initial enrichment of fissile material within the fuel.

metal structures (known as internals) that can be completely removed to facilitate periodic inspections:

- the lower structures support the core;
- the side structures (core barrel) separate the cold fluid entering the vessel from the hot fluid exiting the core;
- the upper structures are made up of the control rod guide tubes.

The dimensions of the vessels of each series are given in Table 2.2.

Table 2.2. Dimensions of the vessels of each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Inside diameter (m)	4.00	4.39	4.486	4.885
Height (m)	12.3	13.6	13.645	13.105
Cladding thickness at core level (m)	0.20	0.22	0.225	0.25

2.3.2.2. Reactor coolant system and secondary loops

The reactor coolant system (RCS) carries heat away from the reactor core by circulating pressurised water (known as reactor coolant) through the heat transport loops (there are three for a 900 MWe reactor, four for a 1300 MWe reactor, a 1450 MWe reactor or an EPR). Each loop is connected to the reactor vessel, which contains the core, and is equipped with a reactor coolant pump (RCP). This pump circulates the coolant heated through contact with the fuel elements to heat exchangers, called steam generators, where the coolant transfers its heat to the secondary loops and flows back to the reactor (Figures 2.3 and 2.6). The RCPs are fitted with seals that are continuously cooled by pressurised water to prevent reactor coolant from leaking outside the RCS.

The steam generators are evaporators composed of a bundle of U-tubes and a secondary side with integral moisture-separation equipment. The reactor coolant enters the inverted U-tubes and heats the secondary-side water, which flows in through a nozzle located above the tube bundle. The steam generated rises through the moisture separators and exits through the top of the steam generator.

A tank, called a pressuriser, allows the coolant to expand and maintains the RCS pressure at 155 bar so that the coolant (heated to over 300 °C) remains in liquid form. The reactors in operation have three letdown lines, each of which has an isolation valve and a safety valve. In particular, these valves enable emergency blowdown of the RCS to prevent high-pressure core melt.

The upper section of the EPR pressuriser has three letdown lines, each of which has a pilot valve fitted with a position sensor. The EPR also has an emergency RCS blowdown system consisting of a set of motor-operated valves that are actuated to avert high-pressure core melt. This system consists of two parallel letdown lines connected to the same nozzle at the top of the pressuriser. Each line is fitted with two motor-operated valves and is connected to a shared letdown line that leads to the pressuriser relief tank. This system is described in Section 4.3.4.

The normal operating conditions of the RCS for each series are given in Table 2.3.

For each unit, the RCS is completely located inside the containment.

Table 2.3. Normal operating conditions of the RCS for each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Number of loops	3	4	4	4
Nominal absolute RC pressure (bar)	155	155	155	155
Nominal flow rate (m^3/h)	21,250	23,325	24,500	27,195
RCS volume, pressuriser included (m^3)	271	399	406	460
Nominal temperature of the water at the vessel inlet (°C)	286	293	292	296
Nominal temperature of the water at the vessel outlet (°C)	323	329	330	330

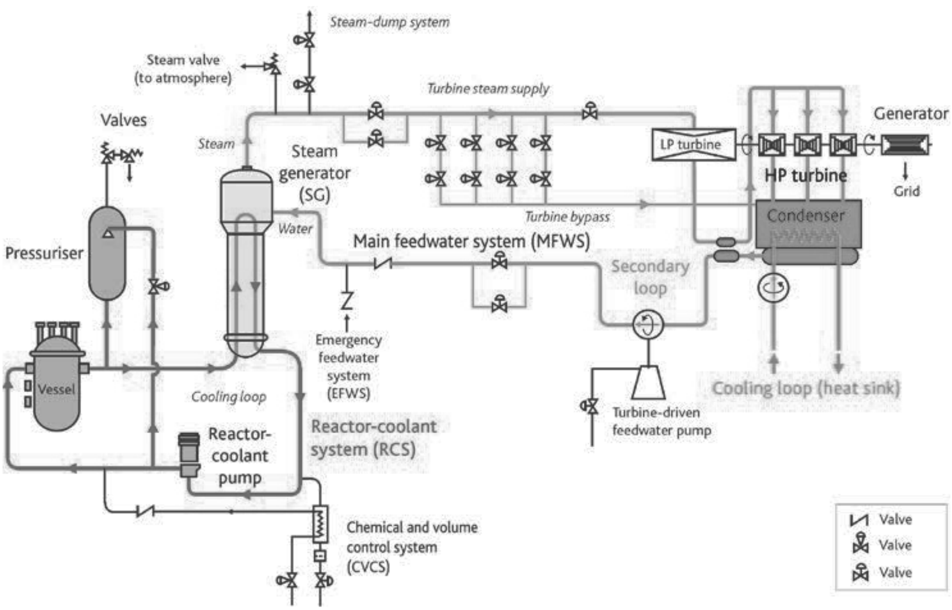

Figure 2.6. Schematic diagram of the main components of the RCS and the secondary loops.

During normal operation, the secondary loops convert the thermal energy produced by the core into electrical energy. To prevent radioactive coolant from leaving the containment, the secondary loops are separated from the RCS by the pipes of the steam generators. The reactor coolant flows through these pipes, where its heat is transferred to the water in the secondary loops. This water is vaporised then expands in the steam turbine connected to the generator (Figure 2.6). The steam is generated in these loops at a pressure of 58 bar absolute (900 MWe reactors), 65 bar absolute (1300 MWe reactors), 73 bar absolute (1450 MWe reactors) or 77 bar absolute (EPR). It exits the turbine and flows into a condenser that is cooled by water from a river or sea. In some instances, the water is cooled by contact with air inside a cooling tower.

The upper sections of the steam generators are connected to the turbine's steam chest *via* three or four lines[9] (one per steam generator) (Figure 2.6). Each line has:

– a flow restrictor inside the outlet pipe of the steam generator;
– a steam dump system equipped with an isolation valve and a control valve;
– seven (two for the N4 series and the EPR) safety valves with steam dump pipes;
– an isolation valve that closes in a matter of seconds.

The flow restrictor slows down the rate of cooling and depressurisation of the secondary loop and reduces the forces exerted on the tube bundle in the event of a steam line break. The valves protect the loop against overpressure if the steam can no longer be dumped. The bypass is used to temporarily send steam directly to the condenser without passing through the turbine or activating the valves. It is used especially to remove heat from the core during startup, hot shutdown or cold shutdown of the reactor and until the residual heat removal system (RHRS) is turned on (Figure 2.7). The steam dump system discharges the residual heat, thus cooling the reactor core if it can no longer be cooled by the normal systems, and avoids having to open the safety valves in the event of rupture of one or more steam generator lines. This system consists of one line per steam generator for the 900 and 1300 MWe series, two lines per steam generator for the N4 series, and only one line for the EPR. Each line has a dump valve and an isolation valve.

The characteristics of the secondary loops are given for each series in Table 2.4.

Table 2.4. Characteristics of the secondary loops for each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Number of steam generators (SG)	3	4	4	4
Secondary-side steam pressure at the SG outlet (bar absolute)	58	65	73	77
Heat-transfer area in an SG (m^2)	4746	6940	7308	7960
Steam flow rate (t/h) per SG	1820	1909	2164	2197
Steam temperature at the SG outlet (°C)	273	281	288	293

9. In the EPRs, each of the four lines is located in a separate room.

2.3.2.3. Containment

The containment is made up of the reactor building, which houses the RCS, a portion of the secondary loops (including the steam generators), and a number of auxiliary operating and safety systems. The reactor building is a concrete cylinder topped by a concrete dome. It forms a strong barrier that offers the specified level of integrity (see Chapter 6 for more details), prevents radioactive substances from escaping into the outside environment, and protects the reactor from external hazards. The reactor buildings of PWRs currently in operation are designed to withstand the pressure (4 to 5 bar absolute) expected during a loss-of-coolant accident (LOCA with a double-ended guillotine break of a main coolant pipe) or rupture of a steam line inside the containment. They ensure a satisfactory level of integrity should either situation occur. The containment of the EPR is designed to withstand a higher pressure of approx. 6.5 bar absolute.

Whatever the reactor type, the concrete walls of the containment rest on a foundation, or basemat, which is also made of concrete. The walls are topped by a concrete dome that forms the roof of the building. The reactor building is designed to withstand the effects of a seismic margin earthquake (SME) (the magnitude of the SME is determined based on the magnitudes of the maximum historically probable earthquakes [MHPE] and by taking into account a safety margin that covers uncertainties, amongst other aspects) and environmental hazards (extreme weather conditions, aircraft crashes, explosions, etc.).

The reactor building penetrations are distinctive points of the containment. Pipes, electrical wiring and ventilation ducts are routed through orifices in the containment walls. There are also access hatches, or locks, for personnel and large items of equipment. Lastly, there is a canal, or pipe, for transferring fuel assemblies between the reactor building and the fuel building. Some water and steam pipes, particularly the portions of the secondary loops inside the reactor building and the outer portions leading to the isolation valves, are an extension of the containment. The secondary shell of the steam generators and the tube bundles on the primary side are also an extension of the containment.

All these penetrations have a specified level of integrity (see Chapter 6 for more details). With the exception of the water and steam penetrations on the secondary loops, these penetrations are fitted with isolation devices located inside the containment. These isolation devices, which are closed before or during an accident, are located on the fluid inlets and outlets. The isolation valves for the water and steam penetrations on the secondary loops are located inside the containment and after the safety valves (see the description of the secondary loops in the preceding section).

Before the reactor is first brought online, the containment is inspected and tested to determine its overall integrity and its resistance to forces under normal and accident conditions. All these aspects are explained in Chapter 6 of this document.

Internal components (known as internals) support equipment, provide biological shielding of personnel, and physically separate the loops (particularly the coolant loops) and some items of equipment.

Table 2.5. Characteristics of the containments of each series.

Series	900 MWe	1300 MWe P4	1300 MWe P'4	N4	EPR
Total volume (m^3)	60,000	98,000	83,700	86,000	102,700
Free volume (m^3)	50,400	82,000	70,500	73,000	75,500
Overall height above ground level (m)	51.3 (FES)* 52.9 (BUG)* 56.6 (CPY)	71.9	61.8	63.2	62.2
Inside diameter of the cylindrical portion (m)	37	45.00 (inner wall) 50.80 (outer wall)	43.80 (inner wall) 49.80 (outer wall)	43.80 (inner wall) 49.80 (outer wall)	48.00 (inner wall) 53.00 (outer wall)
Standard thickness of the cylindrical portion (m)	0.85 (CP0) 0.90 (CPY)	0.90 (inner wall) 0.55 (outer wall)	1.20 (inner wall) 0.55 (outer wall)	1.20 (inner wall) 0.55 (outer wall)	1.30 (inner wall) 1.30 (outer wall)
Height above ground level of the cylindrical portion (m)	41	54.15 (inner wall) 55.04 (outer wall)	46.60 (inner wall) 51.15 (outer wall)	51.00 (inner wall) 55.55 (outer wall)	43.90 (inner wall) 46.60 (outer wall)
Standard thickness of the dome (m)	0.75 (CP0) 0.80 (CPY)	0.87 (inner wall) 0.40 (outer wall)	0.82 (inner wall) 0.40 (outer wall)	0.82 (inner wall) 0.40 (outer wall)	1.00 (inner wall) 1.80 (outer wall)
Steel liner thickness (mm)	6	no liner	no liner	no liner	6

* FES = Fessenheim; BUG = Bugey.

With the exception of EPRs[10], the containments of France's reactors are fitted with a decompression and filtration system (also known as a filtered venting system) to prevent sudden containment failure in the event of a slow rise in the internal pressure during a core melt accident. To reduce the release of radioactive substances, the steam inside the containment is sent through this system to a system fitted a metal prefilter with a sand bed to trap most of the radioactive aerosols. This system is opened according to a specific procedure, known as U5 (see Section 2.5.2.1).

► Description of the containment walls

The 900 MWe reactor containments consist of a single wall of prestressed reinforced concrete that is lined on the inside with steel plate (known as the liner). The purpose of the steel liner is to act as a leaktight barrier, including during an accident. The purpose of the concrete containment is to withstand pressures and temperatures during an accident, seismic loads and external hazards.

10. In its opinion 2012-AV-0139 of 3 January 2012 concerning the complementary safety assessments of the priority nuclear facilities in the light of the accident at the Fukushima-Daiichi nuclear power plant, ASN recommends that "EDF will have to identify the existing or additional systems to be included in the EPR's "hardened safety core", in particular to control the pressure inside the containment in the event of a severe accident." This recommendation may result in a reconsideration of the installation of a filtered venting system on reactors of this type.

The containments of the 1300 MWe and 1450 MWe reactors have two walls:

- an inner wall made of unlined prestressed concrete and designed to withstand pressures and temperatures resulting from an accident and participate in ensuring a degree of leak tightness;
- an outer wall of reinforced concrete. Leakage from the inner wall is collected in the space between the inner and outer walls, or annulus. The annulus is maintained at negative pressure by the annulus ventilation system (AVS) so that any leaks from the inner wall and the penetrations can be collected and filtered before their release. The outer wall also protects the reactor from external hazards (extreme weather conditions, aircraft crashes, explosions, etc.).

Like the containments of the 1300 MWe and 1450 MWe reactors, the containment of the EPR (Flamanville 3) has two walls with a dynamic containment system. Furthermore, the inner wall of the containment is lined with steel plate, which ensures most of the integrity. The reinforced-concrete outer wall on the EPR has been made stronger than that of the preceding generation so that, in the event of a severe accident, it will withstand hydrogen explosions, meet requirements for no direct radioactive leaks to the environment, and consolidate its protection against external hazards.

► Description of the containment basemat

The reactor building sits on a prestressed-concrete slab, or basemat. This basemat forms the foundation of the containment's concrete walls and internal structures and confines the bottom half of the building. The configuration of the basemat varies with each site and is designed according to the seismic and geotechnical characteristics of each site. The thickness of the basemat also varies with each site. It is 1.5 m thick at Fessenheim, 2.25 m at Bugey, approx. 4 m for the CPY units, 3 m for the P4 units, 2.8 m for the P'4 units, approx. 3 m for the N4 units, and approx. 4 m for EPRs.

The basemat has access galleries for stretching the prestressing tendons, a drainage system and, where necessary, measuring systems.

The basemat under the EPR is also fitted with a core catcher, a system for containing and cooling molten corium in the event of vessel melt-through during a core melt accident. This system is described in detail in Section 4.3.4 of this document (see also Figure 4.7).

► Description of the reactor pit

The reactor pit is bordered by a cylindrical concrete hoop around the reactor vessel. The bottom has an opening to allow access inside the reactor pit (this opening is closed when the reactor is operating). The void between the vessel and the concrete of the reactor pit is occupied mainly by the vessel insulation. The reactor pit supports the reactor building's internal structures and rests on the containment basemat (Table 2.6).

Table 2.6. Characteristics of the reactor pits of each series.

Series	900 MWe	1300 MWe P4	1300 MWe P'4	1450 MWe N4	EPR
Inside diameter (m)	5.20	5.85	5.26	5.56	6.15
Thickness (m)	1.80 to 2.10	2.00	2.00	2.00	2.70

Chases at the top accommodate the neutron flux measurement systems and shafts accommodate the reactor coolant piping (hot legs and cold legs).

The role of the reactor pit is to support the reactor vessel as well as shield workers from ionising radiation during work on the RCS and adjacent equipment during operation of the reactor.

The reactor pit of the 900, 1300 and 1450 MWe reactors may contain water, particularly after an accident with RCS break. The water in the pit may also come from the CSS (Section 2.3.2.4).

The reactor pit of the EPR is designed to prevent molten corium from spreading into the containment after a vessel melt-through during a core melt accident. The aim of this system is to eliminate the risk of direct heating of the containment (Section 5.2.1). It is also designed to remain dry to prevent potential steam explosions from a corium-water interaction in the event of vessel melt-through (Section 5.2.3) and allow molten corium to flow to the core catcher (Section 5.4.3). The consideration of core melt accidents in the design of the EPRs is explained in detail in Section 4.3.4.

2.3.2.4. The main auxiliary systems and engineered safety systems

During normal operation, shutdown or restart of the reactor, the auxiliary systems contribute to fulfilling the basic safety functions (reactivity control, removal of heat from the RCS and of residual heat, containment of radioactive materials; see Section 2.1). The two main auxiliary systems are the chemical and volume control system (CVCS) and the residual heat removal system (RHRS). They are schematically illustrated in Figure 2.7, which relates to reactors in operation (EPRs excluded).

During reactor operation, the CVCS is used to adjust the boron concentration in the reactor coolant by drawing in demineralised or borated water during reactor power changes. It is also used to adjust the water inventory in the RCS during temperature variations. The CVCS is also used to maintain the chemistry of the reactor coolant by adding chemicals (e.g., corrosion inhibitors) to reduce its concentration of corrosion products. This system has one or more water letdown lines leading from the RCS, a boric-acid tank, a purification unit, and one or more charging lines for reinjecting water into the RCS. Lastly, it continuously supplies water to the seals of the RCPs to ensure their integrity.

During normal reactor shutdown, the functions of the RHRS are to remove the residual heat generated by the fuel in the vessel and maintain the reactor coolant at a moderate temperature while fuel remains in the core. When the chain reaction stops, the reactor core continues to produce heat. This heat must be removed, otherwise the fuel may become severely damaged. The RHRS, which has two motor-driven circulation pumps, collects water from a primary loop at the reactor outlet, transfers it to heat exchangers, and sends it back into another primary loop at the reactor inlet. The heat exchangers are cooled by the component cooling water system (CCWS), which is cooled by the essential service-water system (ESWS).

In the case of the EPR, residual-heat removal (RHR) is carried out by the low-head safety injection system (LHSI). The EPR therefore has four separate, independent RHR trains.

The function of the engineered safety systems is to control accident situations and mitigate their consequences. They consist primarily of the safety injection system (SIS), the containment spray system (CSS) for the reactors in operation (EPRs excluded), and the steam generator emergency feedwater system (EFWS). These systems are schematically illustrated in Figure 2.7, which relates to non-EPR reactors.

In the event of a loss-of-coolant accident (LOCA), the SIS is used to inject borated water into the reactor core in order to halt the nuclear reaction and maintain the water inventory in the RCS. In the case of the reactors in operation (EPRs excluded) it is also used, in some cases of system operation[11], to remove residual heat.

In the case of the reactors in operation (EPRs excluded), the SIS has pressurised accumulator tanks of borated water, a boric-acid tank (refuelling water storage tank, RWST), and pumps with discharge rates and pressures that can handle the various LOCA cases (breaks of different sizes). The reactors in operation have a high-head safety-injection system and a low-head safety-injection system. The 1300 MWe reactors also have a medium-head safety-injection system.

The EPR has four separate, independent low-head and medium-head safety-injection trains. The four trains are supplied with borated water from the in-containment refuelling water storage tank (IRWST), so named because it is located inside the containment[12] (whereas the RWSTs of the reactors in operation are located outside the containments).

The operation of these systems is described in Section 2.4.2.

In the event of an accident leading to a significant increase in pressure in the reactor building, a water-spray system (CSS) is turned on to lower the pressure and thus preserve the integrity of the containment. This system is also used to wash radioactive

11. As with the CSS described below, the SIS can inject water either directly from the fuel pool cooling and purification system (FPCPS) or indirectly using the water collected at the bottom of the containment (recirculation). Residual heat removal by the SIS is achieved with direct injection only. In recirculation mode, the residual heat is removed by the CSS (Section 2.4.2.2).
12. The heat exchangers in the low-head safety injection system remove residual heat from the containment of the EPR without having to use the CSS, like on the reactors in operation. The EPR has a residual heat removal system, but it is used only for severe accident situations. The IRWST also provides the water needed to cool the molten corium in the core catcher in the event of a core melt accident with vessel melt-through.

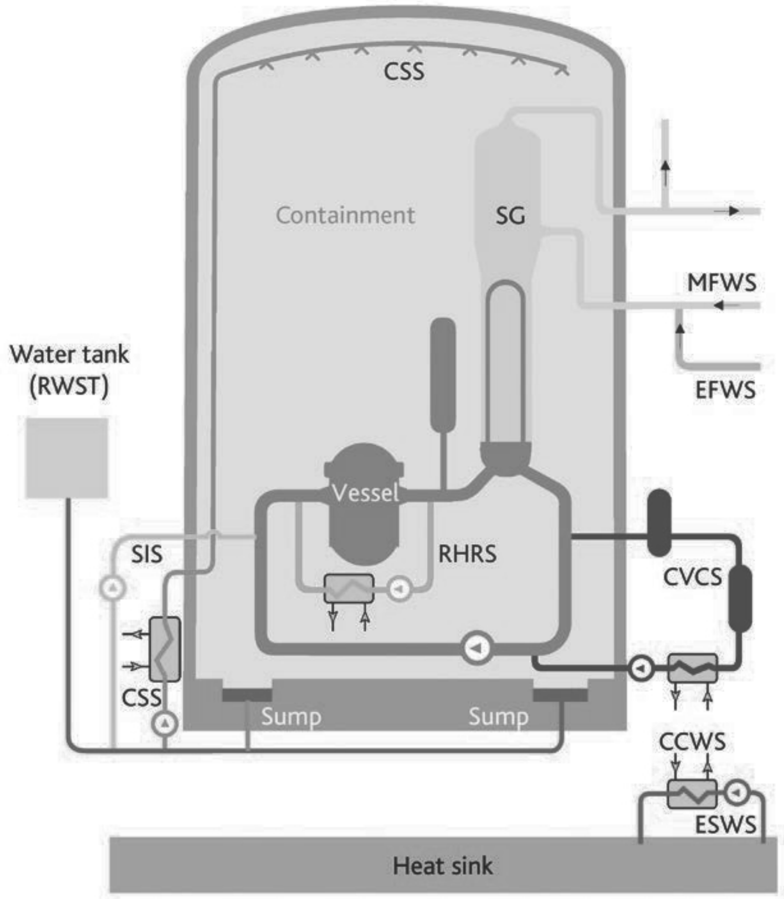

Figure 2.7. Schematic diagram of the main auxiliary systems and engineered safety systems (reactors in operation, EPRs excluded).

aerosols out of the air inside the containment. In the case of the reactors currently in operation, the CSS, which is partially outside the containment, is used to spray water inside the reactor building. This water is pumped in from an external water tank (RWST) fed with sodium hydroxide, or from the bottom half of the containment (sump).

In the case of the EPR, heat is removed from the containment during a severe accident by spraying borated water from the IRWST and draining the water inside the containment. This engineered safety system that is used only in the case of severe accidents is also known as the containment heat-removal system (CHRS). Its use is also described in Section 2.4.2.

The EFWS is used to maintain the level of water on the secondary side of the steam generators and thus cool the RCS in the event the main feedwater system (MFWS) is not available. In the case of the 900, 1300 and 1450 MWe reactors, it is also used during normal operation while the reactor is at hot shutdown to keep water in the steam generators. The EFWS has two motor-driven pumps and either one pump (900 MWe) or two pumps (other series) that are driven by a steam turbine supplied by the steam generators. These pumps draw in feedwater from a tank with a capacity of 700 m^3 (900 MWe), 1440 m^3 (P4), 1723 m^3 (P'4) or 1488 m^3 (N4) and inject this water on the secondary side of the steam generators.

In the case of the EPR, the EFWS has four separate, independent trains, each of which has its own water tank that is supplied by a shared 2600 m^3 tank. These trains are used only if the MFWS fails. The EPR is also equipped with a system that feeds water to the steam generators during reactor startup and shutdown.

2.3.2.5. Other systems

Other systems or circuits important to reactor safety include:

– the CCWS, which cools a number of items of equipment important to reactor safety (the RCPs and the CVCS pumps; the ventilation systems; the SIS and CSS for units at operation; the residual-heat-removal system (RHRS or SIS/RHRS for EPRs). The CCWS operates in a closed loop between the auxiliary systems and the engineered safety systems on the one hand, and the ESWS on the other (see next bullet, below) It should be noted that the CCWS at the Fessenheim reactors does not contribute to cooling of the CVCS, the RCPs, the CSS or the ventilation systems. These systems and items of equipment are cooled directly by the ESWS. The engineered safety systems in all the other reactors are cooled by two CCWS trains. In the EPRs, they are cooled by four CCWS trains;

– the ESWS, which cools the CCWS through the heat sink (river or sea). The series in operation have two ESWS trains; the EPRs have four trains;

– the fuel pool cooling and purification system (FPCPS), which, amongst other things, is used to remove decay heat generated by the fuel elements stored in the spent-fuel pool;

– the ventilation systems, which play an essential role in the containment of radioactive materials by placing rooms at negative pressure and filtering releases;

– the fire-protection circuits and systems;

– the instrumentation and control (I&C) system and the electrical systems. The systems important to reactor safety are powered by redundant power supplies consisting of two independent electrical trains for the reactors in operation and four independent electrical trains for the EPRs. Each electrical train is supplied by a switchboard that itself is supplied by either the transmission grid (two independent high-voltage lines) or a diesel generator. In addition, a third diesel generator (900 MWe series) or a backup turbine (1300 MWe and 1450 MWe series) or two SBO diesel generators (EPRs) may be used if necessary.

2.4. Reactor operation under normal and accident conditions

2.4.1. Systems used under normal reactor operating conditions

Normal reactor operating conditions refer to the following states:

– at-power and hot-shutdown states, during which cooling is provided by the steam generators, which are supplied on their secondary sides by the MFWS);
– shutdown states, with the RCS closed, during which cooling is provided by either the RHRS or the steam generators supplied on their secondary sides by the EFWS);
– shutdown states, with the RCS open, during which cooling is provided by the RHRS.

The basic safety functions – reactivity control, heat removal and containment of radioactive substances – must be ensured at all times for each reactor state. The systems used to ensure each of these functions are described on the following pages.

2.4.1.1. Reactivity-control systems

Two methods are used to control reactivity. The first method consists of adding boron (an effective neutron absorber) to the RCS to offset slow reactivity changes in the long term. The second method entails using the control-rod clusters (Section 2.3.2.1), which consist of 24 stainless-steel rods containing a silver-indium-cadmium alloy (all reactors) or boron carbide (1300 MWe and 1450 MWe reactors, EPRs) and which slide up and down inside Zircaloy guide tubes. The clusters are inserted into or removed from the core either by the facility's control systems or manual controls operated by facility operators.

In the interval between a hot and cold reactor shutdown, the temperature coefficient is negative. This drop in temperature causes the reactivity in the core to increase (Section 2.1). In this situation, the boron concentration in the RCS is increased to make up for the inability of the control rods to control the reactivity.

Boric acid is injected into the RCS during plant operation or shutdown by the CVCS (see Section 2.3.2.4 for a description of the CVCS).

2.4.1.2. Heat-removal systems

At power, the heat generated in the reactor core is removed by the RCS and transferred to the steam generators by the RCPs. The steam produced on the secondary side of the steam generators is expanded in the steam turbine and exhausted to the condenser. The condenser is cooled by a tertiary loop, which is a heat sink consisting of the sea or a river (open-loop system) or the atmosphere (via cooling towers [closed-loop system]). The condensed water is pumped back to the steam generators by the MFWS (Figures 2.3 and 2.6).

When the reactor is shut down, the decay heat from the fission products is much lower (less than 1% of nominal power) and decreases over time. This heat is removed by various systems depending on whether the RCS is open or closed. When the RCS is closed, the decay heat may be transferred to the steam generators by means of natural convection and without using the RCPs. Water may be supplied to the steam generators by either the MFWS or the EFWS. When it is supplied by the EFWS, the steam generated is dumped to the atmosphere by control valves (Figure 2.6).

An alternative method of heat removal for reactors in operation (EPRs excluded) is the RHRS. The RHRS may be used when the vessel is either closed or open (see Section 2.3.2.4, Figure 2.7 for a description of these systems).

In the case of the EPR, residual heat is removed by the low-head safety-injection system (LHSI, Section 2.3.2.4).

2.4.1.3. Containment systems for radioactive substances

During normal operation, the reactor coolant contains small amounts of radioactive substances. These substances consist of corrosion products in the RCS, which are irradiated during their time in the reactor core, and fission products in the form of gases or particles from leaks in the fuel-rod cladding.

The reactor coolant is continuously purified by the CVCS. The particles contained in it are captured by filters and the gaseous products are stored in tanks for subsequent treatment.

Containment systems are used to prevent these radioactive substances from leaking out into the environment:

- rooms or buildings containing radioactive substances in the form of gases or particles are placed at a pressure below the outdoor air pressure. So-called "iodine-risk" rooms are placed at a pressure below that of the rooms surrounding them;
- gas leaks are collected by either special systems (in particular the gaseous waste processing system) for storage and inspection before being released, or by the nuclear auxiliary building's ventilation systems, which are fitted with iodine traps;
- liquid leaks are collected by sumps, retention pits, containment basins and collection lines for treatment and inspection before being stored in special tanks.

In the case of the EPR, design provisions have been made to prevent radioactive substances from leaking directly into the environment. All the containment penetrations lead into buildings equipped with ventilation and filtration systems.

Radiation is measured in the rooms housing the auxiliary systems (activity of the ambient air, activity of the sump water) to monitor the integrity of these systems, and in the CCWS and the steam system to monitor the integrity of these heat exchangers.

2.4.2. Systems used under reactor incident or accident conditions

During normal facility operation (including normal operation transients), the essential physical parameters remain within their set value ranges. In the event of an accident, some of these parameters may go beyond their ranges, tripping the systems (protection and engineered safety systems, see Sections 2.3.2.4 and 2.3.2.5) designed to bring the reactor back to a state ensuring the three basic safety functions: reactivity control, heat removal and containment of radioactive substances.

2.4.2.1. Reactivity-control systems

Reactivity is controlled by inserting the control rods[13] into the core. These rods fall by gravity into the core within 2-3 seconds of a power failure (reactor trip). The values of some parameters are continuously compared against thresholds (e.g., reactor power, RCS pressure, RCP velocity, temperatures). When any of these thresholds is exceeded, the protection system initiates a reactor trip and may also trip other systems. For example, under some accident conditions, the SIS are also tripped to pump borated water into the RCS to control the reactivity. The CVCS may also be activated to make up the borated water lost from the RCS through small leaks.

2.4.2.2. Residual-heat-removal systems

In accident situations without a break in the RCS, residual heat may be removed first by the EFWS, which is automatically initiated if the MFWS is not available (Figure 2.7).

In some accident situations, a break in the RCS may be caused by a loop failure or the opening of the safety valves. If this occurs, residual heat can be removed only if the water inventory in the RCS remains sufficient and the heat transferred into the containment by the coolant flowing out of the break is removed. For example, if a small break occurs on the RCS, the heat from the reactor core is not completely carried away by the coolant flowing out of the break and into the containment. A portion of this heat must be removed by the EFWS.

► Maintaining the RCS water inventory

A sufficient water inventory is maintained in the RCS by the SIS, which pumps sufficient amounts of water into the RCS to compensate for breaks up to and including double-ended breach (complete rupture) of the RCS.

In the case of the 900 MWe reactors, this function is carried out for breaks of all sizes by two pumps that inject borated water at high pressure (trip threshold of 170 bar) and two pumps that inject borated water at low pressure (trip threshold: 10 bar). In addition, accumulator tanks containing borated water and pressurised with nitrogen empty their

13. Failure of a drive mechanism may lead to ejection of a control rod and uncontrolled increase in the reactivity of the affected assembly. This type of reactivity accident is the subject of many studies and much research, particularly within the scope of IRSN's international Cabri programme. This programme is beyond the scope of this document.

contents into the RCS if its pressure drops below 40 bar. The 1300 MWe and 1450 MWe reactors have a medium-head safety-injection system (consisting of two pumps that inject borated water into the cold legs at a trip threshold of 120 bar), a low-head safety-injection system (two pumps that inject borated water into the cold and hot legs at a trip threshold of 20 bar) and four accumulator tanks that empty their contents into the RCS if its pressure drops below 40 bar.

In the case of the reactors in operation (EPRs excluded), the SIS is automatically initiated by the protection system if the pressure measured in the pressuriser becomes low (trip thresholds given earlier for each series). When the SIS is initiated, borated water is pumped into the RCS from a storage tank located in the reactor building and known as the in-containment refuelling water storage tank (IRWST). When there is no more water in the IRWST, the SIS actuates the CSS. The CSS operates in a closed loop using water condensed from the steam inside the containment and which flows into sumps located at the bottom of the containment.

The SIS can also maintain a sufficient water inventory in the RCS during accident situations without RCS breaks and where the steam generators are unavailable. In these situations, the RCS safety valves and the SIS are actuated to maintain a sufficient water inventory in the RCS, protect the RCS against overpressure, and remove the residual heat ("feed and bleed"). The low-temperature water (approx. a few dozen °C) injected by the SIS flows through the core and exits the valves in the form of steam.

The EPRs have a medium-head safety-injection system (consisting of four pumps that inject borated water into the cold legs at a trip threshold of approx. 90 bar), a low-head safety-injection system (four pumps that inject borated water into the cold and hot legs at a trip threshold of 20 bar) and four accumulator tanks that empty their contents into the RCS if its pressure drops below 40 bar. The water injected into the RCS comes from the in-containment refuelling water storage tank (IRWST).

► Removal of heat released into the containment

If an RCS break occurs in the reactors in operation (EPRs excluded), the CSS is actuated to lower the heat and pressure inside the containment. It does so by drawing in water from the RWST by means of two motor-driven pumps. When the RWST is empty, the CSS draws water from the sumps at the bottom of the containment (all series). The water used by the CSS is cooled by the CCWS, which itself is cooled by the ESWS (at Fessenheim, this cooling is provided directly by the ESWS) (Figure 2.7).

In some accidents with core melt that jeopardise the integrity of the containment, the heat inside the containment may be removed by the containment's filtered venting system. This system limits the peak pressure inside the containment (U5 procedure; Section 2.5.2.1).

If an RCS break occurs on the EPR, the temperature and pressure inside the containment are decreased by the low-head safety-injection system via the heat exchangers on the SIS/RHRS, which has four motor-driven pumps that draw in water from the IRWST.

A specific system, the CHRS, has been installed to lower the temperature and pressure inside the containment during a core melt accident. The CHRS consists of a two-train spray system, heat exchangers and a specific heat sink.

In the event of a core melt accident with vessel melt-through, the water in the IRWST is used to flood and cool the molten corium in the core catcher. The CHRS, which is used for severe accidents, supplies water to the core catcher, limits vaporisation of the water covering the molten corium, and limits the rise in pressure inside the confinement.

2.4.2.3. Containment systems for radioactive substances

Under accident conditions, the integrity of the fuel-rod cladding may become compromised when the heat inside the fuel rods is not adequately removed. The rise in temperature causes creepdown and collapse of the cladding as well as oxidation of the zirconium by water vapour.

In the case of incident conditions estimated to occur relatively frequently over the life of the reactor, containment of fission products is ensured primarily by the fuel-rod cladding, which is designed to remain leaktight during such conditions. The reactor coolant pressure boundary (RCPB) and the containment are two additional barriers ensuring containment of fission products in the event of fuel-rod-cladding failure and in the presence of the activation products in the reactor coolant in the primary circuit.

In the case of situations estimated to occur less often and for which the cladding and the RCPB are no longer leaktight, containment is ensured by the reactor building, which is designed to remain adequately leaktight. Leaks from the systems penetrating the containment are prevented by isolation valves placed on all the containment penetrations and which automatically close when the pressure inside the containment exceeds a set threshold (containment isolation). The purpose of these provisions is to ensure a very low rate of leakage from the containment to the atmosphere.

The radioactive products (primarily fission products) carried under accident conditions by the water flowing through the RCS, the SIS and the containment spray system (CSS for the reactors in operation; CHRS in the event of a severe accident in the case of the EPR) are another source of radioactive releases. In order to reduce atmospheric releases from leaks on these systems, parts of which are located outside the reactor building, the buildings in which they are housed are maintained at negative pressure by ventilation systems equipped with filters.

Containment of radioactive substances must therefore be ensured under any situation that, due to certain kinds of equipment damage, would allow the reactor coolant to leak directly outside the containment, i.e., either inside the peripheral buildings or directly into the outdoor environment. Known as containment bypass events, these situations are described in detail in Section 6.4. A case in point is rupture of the steam generator tubes, which allows reactor coolant to enter the secondary loops and can result in radioactivity being released to the atmosphere by the steam dump valves and safety valves on the steam generators. The risks of such an event occurring are prevented by periodically inspecting the condition of the tubes; placing plugs on weak or corroded tubes; replacing the steam generators when necessary; controlling the chemistry and activity of the reactor coolant; and implementing operating procedures that avoid actuating the dump valves of the steam generators.

2.5. Reactor control under normal and accident conditions

2.5.1. Control room

A reactor is run by operators located in the control room. The control room contains all the control, display and monitoring equipment and systems required to operate a reactor under normal, incident and accident conditions.

After the TMI-2 accident of 1979 (Section 7.1), ergonomic improvements were made to the control rooms of the units in operation (900 MWe and 1300 MWe reactors). The layouts of the controls and display systems were improved and the information on the display systems was made clearer and more reliable. The goal was to display clearer information about the state of the systems used to operate the reactor. An aid, called the safety parameter display system (SPDS) was installed in each control room to aid in implementation of incident and accident procedures. Connected to the data-acquisition system, the SPDS provides a summary of the parameters required for implementing the facility's operational procedures. It also allows operators to quickly ascertain the availability of systems important to safety (containment isolation systems, engineered safety systems, electric power supplies, etc.). In an accident situation, it provides operators with the most relevant information (state of the systems, water level and margin to in-core boiling, containment integrity, etc.).

Reactor operation is centralised in the control room provided it is accessible by staff. If the control room is not accessible (for example, after being evacuated during a fire or other emergency), a safe-shutdown panel located in another room is used, under some conditions, to shut down the unit and maintain the reactor in safe state. This supplementary control room must remain accessible in the event the main control room has to be evacuated. In this case, the controls on the safe-shutdown panel override the controls in the main control room. There are also distributed control panels for specific functions (waste processing, water demineralisation, local control of diesel generators, etc.).

Each unit also has an emergency-response centre that is made available to the emergency-response team formed on the site during an accident. The equipment in this room helps the local emergency-response team to ascertain the main parameters of the unit and share them with other emergency-response teams located around the country and which are familiar with the unit. This way, the local and national emergency-response teams have the same information about the parameters of the situation and can manage it accordingly.

The control rooms of the N4 series differ from those of the preceding series in that heavy use is made of computerised systems. In the control rooms of the N4 series, operating procedures are displayed on screens and logic processing and monitoring are automated. An additional means of emergency response, the auxiliary panel, is installed in the control room. Its role is to safely shut down the unit and control accident situations if the computer system is down. The control rooms of the EPRs are technologically identical to those of the N4 series.

2.5.2. Reactor control

2.5.2.1. Operating procedures

During normal operation and incident and accident transients, the facility is controlled according to a set of procedures whose purpose is to keep the reactor in a safe state or drive it into this state.

Each initiating event liable to lead to an incident or accident is associated with a standard method of operation that is denoted by an "I" (for incidents) or an "A" (for accidents). These operating procedures are established based on the foreseeable development of an incident or accident so as to keep the reactor in a safe state or drive it into this state. They apply if an event occurs alone (not in a combination) and has been correctly identified. This method is known as the "event-based approach".

Supplementary procedures have been established for operating conditions involving simultaneous failure of the redundant trains of systems important to safety and for failure of equipment used over the long term (several months) after a loss-of-coolant accident (LOCA). Known as "H procedures" ("H" for *hors dimensionnement*, or, beyond design basis), these supplementary procedures may require the installation of new, supplementary equipment (e.g., addition of a turbine generator that produces electricity from the steam in the secondary loop to supply a power source for some essential systems, or the installation of a special generator [for the 900 MWe reactors] or a combustion turbine [for the 1300 MWe and 1450 MWe reactors]). These procedures are as follows:

- procedure H1 for total loss of the heat sink or associated systems;
- procedure H2 for total loss of the steam generator feedwater supply (MFWS and EFWS);
- procedure H3 for total loss of the offsite and onsite power sources (loss of both offsite power sources, unsuccessful house-load operation, and loss of both generators);
- procedure H4 for total loss of the SIS or CSS over the long-term phase following a LOCA (future total loss of pumping or heat-exchange systems);
- procedure H5 for protection of some riverside sites against flooding above the thousand-year flood level.

In addition to the aforementioned accidents, there remains the possibility that a series of events could lead to radioactivity being released outside the facility. This is the case of core melt accidents. The following emergency procedures have been created to mitigate or delay core damage and radiological consequences:

- procedure U1 for averting core meltdown in situations where no A or H procedures would be suitable or effective. This procedure recommends, based on changes in the core outlet temperature and the availability of the systems and equipment, the best actions to be taken in terms of using the steam generators, SIS, and the relief valves on the pressuriser and the RCPs to prevent core meltdown;
- procedure U2 for locating and isolating containment leaks;
- procedure U3 for implementing mobile emergency equipment for the SIS and CSS and which supplements procedure H4;
- procedure U4 for implementing means of prevention of early radioactive releases in the event of vessel breach and corium erosion of the basemat;
- the U5 procedure for relieving the pressure inside the containment *via* the sand-bed filter.

In such a situation, the emergency-response teams use the Assistance Guide for Emergency-Response Teams (GAEC) and the Severe Accident Operating Guidelines (GIAG), which define the actions to be taken to ensure containment of radioactive substances for as long as possible.

2.5.2.2. Choice of procedure and the state-oriented approach

To determine the appropriate operating procedure, the state of the reactor must first be diagnosed. This diagnosis is made based on an analysis of the relevant alarms and physical variables.

Making this diagnostic and selecting the right procedure are not always easy in the case of situations with multiple failures. Indeed, the combinations of events caused by multiple failures can be endless. On the other hand, the possible physical states of the reactor are limited in number. They can be identified from a limited number of data characterising the physical state of the main reactor components: subcriticality of the reactor core; RCS water inventory; efficiency of (residual) heat removal; integrity of, and water level in, the steam generators; and integrity of the containment. In addition, the actions to be taken may generally be inferred from knowledge of the reactor's physical state without necessarily having to identify the sequence of events that led up to this state. The entire approach that consists of the identifying the physical state of the facility, defining the operation to be achieved based on this state, and setting priority actions to be taken to control the situation is known as the "state-oriented approach".

In the state-oriented approach, the operational objective and strategy may be redefined at any time based on developments in the situation (physical state of the reactor, equipment failures, human errors). Unlike with the event-based approach, operation is no longer defined solely by an initial diagnosis of the cause of the incident or accident. This type of operation makes it possible to cover all thermal-hydraulic incidents or accidents (RCS breaks, secondary-loop breaks, core heating, etc.), be they single or multiple, occurring alone or compounded by system failures, power failures or even human errors.

2.5.2.3. Control under incident and accident conditions

The operating procedures describe the operations to be carried out to return the reactor to a safe state. They primarily address:

- controlling reactivity through operation of the systems used to add boron to the RCS after rod drop;
- maintaining the RCS water inventory through operation of the CVCS and the SIS;
- removing residual heat through operation of the core cooling systems:
  - the steam generators (if available): the heat generated in the reactor is removed by spraying water from the secondary loops. This water is then either cooled by the condenser or discharged to the atmosphere. The necessary makeup water is provided by either the MFWS or the EFWS);
  - the engineered safety systems (CSS and SIS) in the event of a LOCA or total loss of the steam generators;
  - the RHRS after reactor shutdown (RHRS for the units in operation – EPRs excluded – or LHSI for the EPRs);
- containment of radioactive substances through closure, where necessary, of the devices used to ensure the integrity of the containment.

By monitoring the systems used, the operators can detect if any of them fail and, if possible, implement stopgap measures while the failed systems are being repaired.

In order ensure human redundancy, the shift manager then the facility safety engineer are called into the control room as soon as an accident procedure is implemented. Their role is to monitor the situation as it progresses and meet the following objectives:

- ensure that the necessary safety-related actions are carried out;
- ensure that operators correctly follow the procedure relating to the reactor's state;
- monitor the state and availability of the safety systems used.

In the event of an accident involving a risk of radioactive release, the local and national emergency-response teams are set up in a matter of hours and use the Assistance Guide for Emergency-Response Teams (GAEC) and the Severe Accident Operating Guidelines (GIAG) to manage the situation (Section 2.5.2.1).

2.6. Conclusion

The preceding sections present the main components, systems and loops found in PWRs as well as their main principles of operation under normal and accident conditions. PWRs are complex facilities with specific risks related to the presence of large amounts of radioactive products. Safety must be a constant concern at each stage of their life (design, construction, operation, dismantling) to reduce risks, particularly the dissemination of radioactive substances.

Provisions are made at each stage in the life of a reactor to protect people and the environment against the effects of radiation. These provisions aim to:

- ensure the normal operation of facilities;
- prevent incidents and accidents;
- mitigate the consequences of a potential incident or accident.

The approach used to implement these safety measures is described in the next chapter.

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