Safety and Radiation Protection at Nuclear Power Plants in France in 2014

IRSN’S POSITION
IRSN is a public authority with industrial and commercial activities set up in 2001. Its activities were incorporated in Law 2015-992 of 17 August 2015 on green growth energy transition. It is supervised jointly by the Ministers of Ecology, Research, Industry, Health and Defence.

As a public expert on nuclear and radiological risks, IRSN, through its research, assessment and monitoring activities, evaluates the scientific and technical issues relating to such risks. The scope of its activities both in France and abroad is wide and varied, and includes the safety of nuclear facilities, transport and radioactive waste, monitoring the environment and the health of workers and patients, advice and response in the event of a radiological risk, and human radiation protection in normal and accident situations. Its expertise also comes into play in similar defence-related activities.

IRSN contributes directly to national policy in the field of nuclear safety, human and environmental protection against ionising radiation, and the protection of nuclear materials, facilities, and transport of radioactive materials against malicious acts. In this area, it interacts with all the stakeholders concerned by these risks: public authorities, particularly nuclear safety and security regulators, local authorities, businesses, research organisations, associations, and civil society stakeholders and representatives.

Another of its concerns is to keep the public informed by publishing the findings of its work. Through its activities, IRSN is also involved in major public policies in other areas such as research and innovation and occupational and environmental health.

IRSN has a workforce of some 1,700 employees including many engineers, doctors, agronomists, veterinarians, technicians, experts and researchers.

To carry out its work effectively, IRSN has a budget of some €300 million.
Safety and radiation protection require continuous vigilance on the part of all those involved and can never be taken for granted. They must remain an absolute priority to ensure continuous improvement.

For IRSN, achieving this goal implies constantly expanding knowledge gained from two complementary sources, namely research and careful analysis of national and international operating experience feedback. Constantly improving knowledge in this way is essential for performing state-of-the-art nuclear and radiological risk assessments that accurately reflect operational and ground realities.

As part of its activities, IRSN carries out a continuous technical watch on safety and radiation protection for civil basic nuclear installations and transport of radioactive materials for civil use in France.

This work involves analysing significant events concerning these installations and transport activities that are reported by licensees to ASN, the French Nuclear Safety Authority. The purpose of this analysis is to draw lessons to provide IRSN with additional feedback. IRSN carries out in-depth analysis of the most important events. It also performs a more general examination of these events to highlight overall lessons and trends and to identify areas for improvement that call for particularly close attention on the part of licensees. The results of these overall analyses are presented in three reports:

"IRSN’s Position on Safety and Radiation Protection at Nuclear Power Plants in France", a report published every year since 2008, concerns the 58 nuclear pressurised-water reactors currently operated by EDF.

"Safety at Basic Nuclear Installations other than Nuclear Power Plants: Lessons Learned from Significant Events Reported in (year)", a report published every two years since 2009, concerns nuclear fuel cycle facilities, research laboratories and reactors, radioactive waste treatment, storage or disposal facilities, as well as facilities that have been shut down and are currently undergoing clean-up or dismantling operations.

"Safety of the Transport of Radioactive Materials for Civilian Use in France - Lessons Learned by IRSN from Analysis of Significant Events Reported in (year)", a report published every two years since 2008, concerns the transport of radioactive materials for civil use in France.

As risks relating to nuclear activities are a major concern for the French public, as reflected in the annual IRSN Barometer on the perception of risks and safety, these reports are intended to inform stakeholders and the general public to improve their understanding of concrete issues in safety and radiation protection. With this in mind, the reports also address “general” or “cross-cutting” topics where IRSN’s expertise has helped to enhance safety and radiation protection.

http://www.irsn.fr/FR/IRSN/Publications/barometre/Pages/default.aspx#.VjeIGYRdf04
As in previous years, and in accordance with its policy of transparency, IRSN publishes an annual report giving its position on safety and radiation protection at nuclear power plants in France. The layout of this year’s report is the same as that adopted for the 2013 report, and is aimed at facilitating reading and helping stakeholders and the general public to grasp the concrete safety and radiation protection issues concerning these facilities.

None of the significant events reported in 2014 affected reactor safety and few of them had any impact on workers’ health or the environment. The total number of significant safety events reported by EDF dropped in 2014, although the trend observed over the past two years remains: errors during maintenance work still account for nearly half of all such events reported. The total number of significant radiation protection events reported also dropped in 2014. This was due to a considerable decrease in the number of gamma radiography inspections. However, the number of events due to workers in controlled areas failing to wear a dosimeter rose. EDF must therefore continue and amplify the efforts it has made in this area in recent years.

In this report, IRSN presents its analysis of some of the significant events and anomalies reported in 2014, including loose parts in a reactor coolant system which caused damage to fuel assembly cladding, an event in which electrical equipment came into contact with water that had leaked through openings, and the risk of safety-related equipment becoming unavailable due to overheating in the emergency turbine generator room.

Lastly, the report gives two examples that demonstrate how analyses performed by IRSN during the assessments it carries out at ASN’s request complement those conducted by EDF. Rather than simply responding to a request to verify compliance with regulations, these analyses genuinely enhance safety at nuclear facilities. The two examples in question are instrumentation and control renovation on 1300 MW reactors, and modifications to the Gravelines plant in response to the risks induced by the operation of the future LNG terminal in Dunkirk.

I hope that you will find this report informative and interesting to read and look forward to any comments you might have in line with our continuous improvement policy.

Jacques Repussard
IRSN Director-General
As part of its assessment work, IRSN analyses operating experi-
ence feedback from significant events concerning safety or radia-
tion protection. It draws in particular on reports submitted to the French Nuclear Safety Authority (ASN) by EDF two
months after the occurrence of the event in compliance with the
reporting system.

The drop in the number of significant safety events observed in 2013 was confirmed in 2014.

The drop in the number of significant safety events observed in 2013 compared with 2012 was confirmed in 2014 (-8%). Early detection by EDF teams of any anomalies may have contributed to the fact that none of the events reported had a very significant impact on safety. ASN did not class any of the events reported in 2014 on INES level 2.

IRSN’s analysis highlighted the following points:

- The number of anomalies affecting safety-related equipment in reactors was generally lower in 2014 than in 2013. This drop can largely be explained by correction of the many compliance gaps observed in 2013 on seismic-qualified valves used in many systems;
- The downward trend in the number of incorrect actions on equipment during maintenance or equipment changes observed in 2013 was confirmed in 2014. In particular, the number of significant safety events due to technical slips during maintenance - or maintenance non-quality - also fell sharply compared with 2013 (-40%). Nevertheless, half the significant safety events reported by EDF were caused by maintenance errors, mostly during reactor outages. Improved control of these activities, which require meticulous preparation for optimum planning, remains a key issue for EDF, and will be all the more important as the “Grand Carénage” major refit programme increases the volume of maintenance work.
- Another concern is the number of deviations from the authorised operating domain, more than half of which were due to operating errors during delicate manual reactor control phases. Analysis of this problem has shown that shift crews are aware of the risks involved in these operations, since half these events are detected and corrected in less than four minutes.

The number of events reported concerning system alignment errors was also higher in 2014 than in 2013. These errors could have led to reactor safety-related systems becoming unavailable. Among the most frequently observed errors are those involving the choice of valve to be operated, failure to set the valve correctly and operations that do not comply with the operating documents. “Lines of defence”, that is to say the preparations for an alignment operation and constant communication between the technicians and the operating team, play a vital role in ensuring the successful completion of alignment tasks.

Overall, analysis of these problems highlights the importance for EDF to keep a close watch over human and organisational factors, as these must ensure that operating and maintenance teams can perform their tasks at facilities safely.

Licensees of basic nuclear installations must report to ASN, the French Nuclear Safety Authority any significant safety or radiation protection events within forty-eight hours of detection. Significant safety events can considerably affect facility safety. Significant radiation protection events are liable to impair human health through exposure to ionising radiation.
Key Events in 2014

The number of significant radiation protection events fell in 2014 compared with 2013

After rising every year since 2010, the number of significant radiation protection events reported in 2014 (Figure B) fell in comparison with 2013, and was just below the number of similar events reported in 2012. The decrease in significant radiation protection events was mainly due to four types of event:

- those related to gamma radiography inspections for checking welds, the number of which was halved between 2013 and 2014;
- those related to access to or time spent in orange radiologically controlled areas;
- deviations regarding training or certification, which only concerned non-compliance with the validity dates of radiation protection training; their number has fallen steadily for the past three years;
- contamination outside radiologically controlled areas. IRSN has, however, observed a significant rise in the number of events related to worker dosimetry, in particular failure to wear a dosimeter (representing 17 out of 18 significant radiation protection events reported per reactor in 2014). EDF’s efforts in this area have apparently failed to achieve the expected results and must be strengthened. The number of events relating to contaminated clothing and surfaces, radiological monitoring faults (i.e. concerning radiation portal monitors at controlled area exits) and unauthorised access to red controlled areas were also slightly higher in 2014.

Some key events in 2014

The report describes three events that highlight the importance of ensuring optimum operating conditions for reactors. These events are briefly outlined below.

- Significant $^{133}$Xe and $^{131}$I release was observed in the reactor coolant system of unit 2 of the Saint-Laurent-des-Eaux B plant. The cause of this fission product release was leakage in two fuel assemblies due to cladding defects. Video inspections revealed several loose parts to be the cause of the cladding defects. These parts came from a seal that had inadvertently been placed in the reactor coolant system.
- After a break in a potable water system pipe running through a service duct, water entered the top of the electrical building of reactor 1 at the Le Blayais NPP. The water seeped through openings that were not watertight, then ran along some cables, damaging electrical cabinets located lower down. As a result, several measurement indicators were unavailable in the control room. The leakage was caused by errors during the replacement of leaktight coverings on the openings.
- The plant operator of Fessenheim reactor 2 observed high temperatures in the emergency turbine generator room. Overheating in the room was due to operation of the actual equipment and its auxiliaries, and to the lack or poor performance of ventilation in the room. This generic compliance problem can rapidly lead to theshort-term unavailability of equipment required in the event of an accident involving total loss of reactor electrical power supplies (EDF grid and emergency diesel generators).
Under French regulations (Article L.593-18 of the Environment Code), licensees must carry out a periodic safety review of their facilities every ten years. Periodic safety reviews add to the continuous safety improvement process, which is built on analysis of experience feedback on the daily operation of reactors. They take into account the ageing of facilities and equipment obsolescence to ensure that the facilities concerned can be operated perfectly safely until the next review. Renovation of the I&C system on 1300 MW reactors, which is the subject of an article later in this report, or replacement of the steam generators for Blayais 3 (see Focus and Figure C) are examples of changes planned by EDF and considered adequate by IRSN.

What does a periodic safety review consist of?

➔ an examination to check that the condition of the facility complies with the safety baseline and the regulations in force; this examination is used to deal with any compliance gaps detected;

➔ a safety reassessment intended to bring the safety level of the oldest reactors up to that of the most recent ones where possible; the safety reassessment may lead EDF to revise its safety documentation;

➔ the implementation of improvements resulting from the safety reassessment.

The aim of the periodic safety reviews that have been carried out on power reactors in France for several decades is to guarantee the highest standards of safety at facilities.

**FOCUS**

Prolonged outage at Le Blayais pending compliance certification of the replacement steam generators

EDF had difficulties in 2014 demonstrating that the replacement steam generators for reactor 3 of the Le Blayais NPP met the regulatory requirements applicable to nuclear pressure equipment. The replacement operation was suspended until June 2015, delaying reactor restart (the reactor was connected to the grid on 5 September 2015), as the old steam generators had already been removed. Replacing steam generators is an answer to the problems that can be induced by ageing in steam generators equipped with tube bundles made from alloy 600, a nickel alloy that has proved more sensitive to stress corrosion than initially expected. The tubes in the new steam generators are made from alloy 690 and offer better performance in terms of stress corrosion resistance. Stress corrosion causes early damage to tubes, which must then be plugged.

For safety reasons, the number of tubes that can be plugged in a steam generator is limited, and the gradual increase in the number of plugged tubes has the effect of reducing the lifetime of the steam generator. It was not possible to authorise the installation of the replacement steam generators for reactor 3 at the Le Blayais NPP in 2014, because the substantiating evidence provided by the manufacturer was not considered to meet the requirements of the Order of 12 December 2005 on nuclear pressure vessels. The manufacturer and licensee prepared the substantiating evidence required by ASN which, after examining and approving the evidence submitted, declared the replacement steam generators to be compliant and authorised their installation in June 2015.

In a pressurised water reactor, the heat produced in the reactor core is transmitted by water circulating in a closed system, called the reactor coolant system, to a secondary system. The water in the secondary system is converted into steam that drives turbines to generate electricity. Heat is exchanged between the reactor coolant system and the secondary system through thousands of inverted U-tubes which form the tube bundle and are grouped together in vessels called steam generators. The tubes are held in place by anti-vibration bars and tube support plates. The water in the reactor coolant system flows inside the tubes, and the secondary system water flows along them and is gradually converted into steam. A steam generator tube is about one millimetre thick.

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IRSN’s Position on Safety and Radiation Protection at Nuclear Power Plants in France in 2014

Key Events in 2014

The 58 reactors operated by EDF in France are highly standardised: they have the same reactor system (they are all pressurised water reactors or PWRs); their nuclear steam supply system is built by the same constructor; and they have the same industrial architect, which is also a licensee. PWRs, divided into three power levels (Figure D), thus share common design and operating bases.

Major events in 2014 for EDF included:
- preparing the action programme for the fourth ten-yearly safety reviews of the thirty-four 900 MW reactors (VD4-900), the first of which is scheduled in 2020;
- validating the actions programmed for the third ten-yearly safety reviews of the twenty 1300 MW reactors (VD3-1300), the first of which began in 2015 at Paluel (see Focus);
- laying down guidelines for the action programme that will be implemented for the second ten-yearly safety reviews of the four 1450 MW or N4 reactors (VD2-N4), scheduled to start in 2018.

FOCUS

Paluel NPP review inspection prior to the 3rd ten-yearly review

Every year, ASN conducts a large-scale inspection of a particular nuclear power plant, with technical support from IRSN.

In 2014, it carried out this inspection at the Paluel nuclear power plant, whose reactor 2 was to be the first 1300 MW reactor to undergo a third ten-yearly review in 2015. This type of review calls for a particularly long outage with extensive work on maintenance and facility changes. The focus of the inspection was "safety management during refuelling outages" in preparation for this major outage.

Before the inspection began, IRSN proposed to ASN a number of topics that could be examined as part of the inspection, which took place from 3 to 7 November 2014 with twenty ASN inspectors and IRSN experts going to the Paluel plant. The inspection mainly consisted of observations of works and meetings to discuss on-going activities.

On the whole, the findings of the inspection were positive regarding outage preparations. The licensee had completed a considerable amount of work in planning ahead in terms of human resources and training requirements. As far as reactor outage safety was concerned, however, there was room for improvement in several areas. Examples of this included contractor monitoring, and consideration by the plant managers of the positions expressed by the independent safety review team.
The nuclear power plants currently in operation in France comprise a total of 58 pressurised water reactors (PWRs), referred to as "second generation", by comparison with the European Pressurised Water Reactor (EPR), which is currently under construction and part of the "third generation".

One specific feature of the French NPP fleet is its standardisation, with many technically similar reactors spread over 19 nuclear sites (Figure 1.1). Each site includes two to six PWRs. The nuclear reactor fleet is divided into three series according to electrical power output:

- The 34 reactors in the **900 MW** series include six in the CP0 series (two at Fessenheim and four at Bugey) and 28 in the CPY series (four at Tricastin, six at Gravelines, four at Dampierre-en-Burly, four at Le Blayais, four at Chinon, four at Cruas and two at Saint-Laurent-des-Eaux).
- The 20 reactors in the **1300 MW** series are subdivided into two trains, those in the P4 train (four at Paluel, two at Saint-Alban and two at Flamanville) and those in the P'4 train (two at Belleville-sur-Loire, four at Cattenom, two at Golfech, two at Nogent-sur-Seine and two at Penly).
- The four reactors in the **1450 MW** series, also referred to as the N4 series (two at Chooz and two at Civaux).

The rest of this chapter provides a relatively general and simplified description of the main components of the PWRs operating in France to provide a basis for understanding this report.
General layout

Broadly speaking, a nuclear reactor consists of two parts (Figure 1.2): the "nuclear island", where nuclear fission produces heat, and the "conventional island", where that heat is transformed into electric current, and where the facility cooling system is also located.

**Nuclear island**

In 1300 MW reactors, the nuclear island primarily includes:

- the reactor building (RB) which houses the actual reactor and the entire reactor coolant system, as well as some of the systems that ensure reactor operation and safety;
- the fuel building (FB), which houses, in particular, the facilities for storing and handling fresh fuel (until it is loaded into the reactor) and spent fuel (until it is transferred to the reprocessing plant);
- the safeguard auxiliary building and electrical equipment rooms (SAB/BL), with the main engineered safeguard systems located on the lower level of the building and the electrical equipment rooms (control room and operations facilities, electrical power supplies, and the I&C system of the reactor) on the upper level;
- the nuclear auxiliary building (NAB), which houses the auxiliary systems required for normal reactor operation; two physically separate buildings, each housing a diesel generator (backup electrical power supplies);
- an operations building.

**Conventional island**

The conventional 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 conventional island are:

- the turbine hall, which houses the turbine generator (this converts the steam generated by the nuclear island into electricity) and its auxiliary systems;
- the pumping station, which cools the facility through the heat sink (river or sea) – this is known as once-through cooling;
- a cooling tower, if a closed-loop cooling system is used.

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Figure 1.2
General diagram of a pressurised water reactor (1300 MW or 1450 MW) and its main systems
Description of a nuclear reactor

Reactor core

The reactor core is made up of fuel assemblies. Each fuel assembly includes 264 fuel rods, 24 tubes that may contain rods from a control rod assembly and an instrumentation tube. The fuel rods, which are approximately four metres high (this depends on reactor power), are made of zirconium alloy tubes, also called cladding. Pellets measuring 8.2 mm in diameter, and made of uranium dioxide (UO₂) or a mix of uranium and plutonium oxides ((U,Pu)O₂) are stacked inside the rods, and make up the nuclear fuel. The fuel assemblies are partially renewed during scheduled reactor outages, which occur every 12 to 18 months.

The core is placed inside a carbon steel reactor vessel (Figure 1.3) which has a stainless steel liner and a head that is removed for refuelling operations.

Reactor coolant system and secondary systems (Figure 1.4)

The reactor coolant system removes the heat released in the reactor core through pressurised water circulating in the coolant loops. Each loop is connected to the reactor vessel and equipped with a pump (reactor coolant pump), which circulates the heated water in contact with the fuel assemblies towards heat exchangers (steam generators), where the reactor coolant transfers some of its energy to the secondary systems before it is returned to the core.

A tank (pressuriser) connected to a coolant loop allows the water to expand, due to its dilation, and controls the pressure (normal operating pressure = 155 bar) to maintain the heated water (in liquid form) at a temperature of more than 300°C in the reactor coolant system.

The secondary systems convert the thermal energy produced by the core into electricity. The (radioactive) water in the reactor coolant system transmits some of its heat to the (non-radioactive) water in the secondary systems in the steam generators; this forms steam, called secondary steam, which expands in a turbine coupled to a generator. On leaving the turbine, the steam is cooled in a condenser. The condenser tubes are cooled either using water drawn from a river or the sea (once-through cooling), or via a tertiary loop where water is cooled by air in cooling towers (closed loop).
The containment building (or reactor building) houses the reactor coolant system, part of the secondary systems, including the steam generators, and a number of safety and operations auxiliary systems.

The reactor building is composed essentially of a concrete cylinder, topped with a concrete dome (the roof of the building), forming a strong barrier built to leaktightness specifications. It prevents radioactive materials from escaping into the outside environment and protects the reactor against external hazards.

It is designed to withstand pressures reached during design-basis accidents (4 to 5 bar absolute) and remain leaktight under these conditions. The concrete walls rest on a concrete foundation raft which forms the base of the building.

### Main auxiliary systems and engineered safeguard systems

(Figure 1.5)

The auxiliary systems contribute to basic safety functions (controlling core neutron reactivity, removing heat from the reactor coolant system, containing radioactive materials and protecting people and the environment from ionising radiation) both during normal operation at power and when the reactor is shut down or restarted.

The main systems concerned are:

- The chemical and volume control system (CVCS), which:
  - adjusts the boron concentration in the water in the reactor coolant system by adding demineralised or borated water according to variations in reactor power;
  - adjusts the water inventory in the reactor coolant system according to temperature variations;
  - maintains the water quality in the reactor coolant system by injecting chemical substances to reduce the corrosion product content of the water;
- the residual heat removal system (RHRS), which, during reactor shutdown, removes the residual heat produced by the fuel assemblies in the reactor vessel and prevents the temperature of the water in the reactor coolant system from rising.
- The function of the engineered safeguard systems is to control accident situations and limit their consequences, in particular radioactive release to the environment. The main engineered safeguard systems are:
  - the safety injection system (SIS), which injects borated water into the reactor core, in particular in the event of a loss of coolant accident, to halt nuclear reactions and maintain an adequate water inventory in the reactor coolant system;
  - the containment spray system (CSS) which, in the event of an accident leading to a significant increase in pressure in the reactor building, reduces the pressure and thus maintains containment integrity. The system is also used to remove radioactive aerosols that may be released into this containment;
  - the steam generator emergency feedwater system (EFWS), which cools the water in the reactor coolant system if the main feedwater system (MFWS) is unavailable.

### Other systems

Other reactor safety-related systems include:

- the component cooling water system (CCWS), which cools some of the safety-related equipment in the CVCS, SIS, CSS and RHRS and ventilation systems;
- the essential service water system (ESWS), which cools the CCWS via the heat sink;
- the fuel pool cooling and purification system (FPCPS), the functions of which include removing the residual heat from the fuel assemblies stored in the spent fuel pool;
- the ventilation systems, which play a critical role in the containment of radioactive materials by placing the rooms under varying degrees of negative pressure and filtering aerosols prior to release;
- fire protection systems;
- the instrumentation and control (I&C) system and electrical systems.

![Figure 1.5](image)
Overall assessment of safety and radiation protection performance of nuclear power plants in operation

Operating safety: main trends
Radiation protection: main trends

Although the safety of a reactor relies primarily on its design and quality of construction, the conditions under which it is operated play a crucial part in guaranteeing satisfactory standards of safety and radiation protection at all times.

IRSN’s assessment of radiation protection and safety performance at EDF nuclear power plants is based on the analysis of a vast quantity of data obtained from operating these reactors. Data relative to events and incidents affecting French - and even foreign - nuclear facilities provide one of the richest sources of operating experience feedback. For an overall appraisal of operating safety and radiation protection, IRSN has developed tools and methods for analysing operating experience feedback including, in particular, its own indicators (see IRSN's 2007 Public Report, page 10).

These help to identify both general and reactor-specific trends or deviations in radiation protection and safety performance. The two following chapters present the main lessons that IRSN has drawn from its overall assessment of radiation protection and safety performance for the year 2014.
Operating safety: main trends

The number of significant events reported relating to safety at EDF nuclear power plants dropped in 2014 (down by 8%). Early detection by EDF teams of anomalies may have contributed to the fact that none of the events reported had a very significant impact on safety.

IRSN ensured that all significant events reported were immediately followed up by appropriate corrective action and in-depth analysis by the licensee to identify any additional corrective action required.

It pointed out the drop in the number of significant events relating to maintenance non-quality, which reflects the organisational improvements made by the licensee regarding maintenance and equipment changes. Nevertheless, half the significant safety events reported by EDF were caused by maintenance errors.

More effective control of these activities therefore remains a priority for EDF.

Licensees of basic nuclear installations must report to ASN, the French Nuclear Safety Authority, any significant events relating to safety, radiation protection, the environment, and transport of radioactive materials within forty-eight hours of detection.

The term “significant safety event” refers to events with a potentially significant impact on facility safety.

The term “significant radiation protection events” refers to ionising radiation exposure events that present a potential threat to the health of exposed persons.

Significant environmental events and transport-related events involving radioactive materials are beyond the scope of this report.

Significant events are analysed as part of the general review of NPP operating experience. Such events are subject to detailed analysis by the licensee and lead to the definition and subsequent implementation of appropriate measures to prevent their recurrence at the plant in question or at any other facility. EDF reports significant events not only for reasons of transparency, but also to allow stakeholders in the nuclear sector to share operating experience feedback.

EDF reports 646 significant safety events in 2014, with an average of 11 reported for each reactor in 2014, compared with 2013.

Number of significant safety events: what is the real significance of this indicator?

For IRSN, the number of significant events is not in itself a quantifying measure of good operating practices, and variations in this number cannot be directly interpreted as a rise or fall in safety or radiation protection levels compared with previous years. Significant safety events do, however, point to issues that need to be analysed and understood for the purpose of identifying appropriate strategies for improving safety at facilities during operation.

The ten criteria for reporting significant safety events (SSE)

- **SSE 1** Automatic reactor trip
- **SSE 2** Activation of an engineered safeguard system
- **SSE 3** Non-compliance with operating technical specifications (OTS)
- **SSE 4** Internal or external hazard
- **SSE 5** Malicious act (or attempt) potentially affecting facility safety
- **SSE 6** Transition to fallback state in accordance with operating technical specifications or emergency operating procedures in response to unexpected plant behaviour
- **SSE 7** Event that caused or could cause multiple failures
- **SSE 8** Event or anomaly specific to the main primary or main secondary cooling system, or pressure vessels in systems connected to them, resulting or potentially resulting in an operating condition that was not considered during design or is not covered by existing operating procedures
- **SSE 9** Design, manufacturing, installation or operating fault concerning functional systems and equipment not covered by criterion 8, resulting or possibly resulting in an operating condition that was not considered at the design stage and is not covered by design basis conditions and existing operating procedures
- **SSE 10** Any other event likely to affect the safety of the facility and considered significant by the licensee or ASN.
The drop in the number of significant safety events reported in 2013 was confirmed in 2014.

1. The numbers of significant safety events shown in Figure 2.1 for 2010, 2011 and 2012 differ from those given in the 2012 Public Report. This is because there was some confusion regarding certain significant safety and radiation protection events, which led to numbers of significant safety events being slightly overestimated in the 2012 Public Report.

2. Owing to an error in interpretation, the number of significant safety events involving a 1300 MW reactor outage in 2013 was slightly overestimated in the 2013 Public Report.

The International Nuclear Event Scale (INES) is used to classify safety-related events at nuclear facilities according to seven levels. Level 0 events are classified as deviations.

The number of automatic and manual reactor trips remained stable

The annual number of reactor trips (Figure 2.3) (automatic or manual) should not be interpreted as an indicator whose changes directly reflect changes in facility safety levels. A reactor trip is a planned response to certain deviations in parameters and serves to return the facility to a safe state. Nonetheless, if a reactor trip occurs while the reactor is at power, it can lead to a thermal-hydraulic transient inside the reactor, which exerts stress on certain mechanical components and can lead to large amounts of effluent. Automatic reactor trips are the more frequent and in 2014 accounted for 77 out of 55 trips. In addition, some reactor trips are a sign of equipment anomalies or a lack of familiarity with operating procedures. In 2007, EDF initiated steps to address this issue and has since succeeded in bringing the number of automatic reactor trips to just under one per year.
Overall assessment of safety and radiation protection performance of nuclear power plants in operation

IRSN's Position on Safety and Radiation Protection at Nuclear Power Plants in France in 2014

Stable number of deviations from the authorised operating domain

The number of deviations from the authorised operating domain remained stable in 2014. In 2014, 52 significant safety events (compared with 49 in 2013) concerned an inadvertent overshoot of assigned physical parameter limits in the authorised operating domain (Figure 2.4). That represents an average of 0.9 significant safety events per reactor and per year. It should be noted that deviations from the authorised operating domain were brief and half of them were detected and corrected in less than four minutes.

Example of a deviation from the authorised operating domain

On 3 June 2014, reactor 1 at the Tricastin nuclear power plant was being restarted. This involved switching from the steam generator emergency feedwater system to the main feedwater system. During this operation, an operator setting error caused too high a flow of water inside the steam generators. This in turn caused the average temperature of the reactor coolant to fall below the authorised limits defined in the operating technical specifications (OTS).

Increase in the number of fallback initiations not performed

The annual number of fallback initiations (Figure 2.5) reflects the considerable impact of unforeseen operating problems that require the plant operator to shut down a reactor in accordance with operating technical specifications to guarantee satisfactory safety levels.

The authorised operating domain includes various operating modes ranging from reactor shutdown to power operation. Each operating mode is associated with technical operating specifications that define all the operating requirements and limits to be observed (pressure, temperature, boron concentration, water level, etc.) and all the essential equipment required to maintain the reactor in a safe state in accordance with safety demonstration criteria.

It is strictly forbidden for operators to deliberately deviate from a reactor’s current authorised operating mode without meeting the applicable requirements for changing the reactor state. In the event of inadvertent deviation from an operating mode, the operator must take all necessary measures to return the reactor to its initial state or to return to a correct situation as soon as possible.

Figure 2.4
Trend in the number of inadvertent overshoots of physical parameters between 2010 and 2014

Figure 2.5
Number of fallback initiations and fallback initiations required but not performed between 2010 and 2014.
Overall assessment of safety and radiation protection performance of nuclear power plants in operation

IRSN's Position on Safety and Radiation Protection at Nuclear Power Plants in France in 2014

Figure 2.6
Number of failures involving safety-related equipment between 2011 and 2014

Feedback initiation
Any failure or sign of malfunction affecting safety-related equipment is detected by monitoring carried out while the reactor is in operation. The operating technical specifications require the plant operator to return the reactor to a safer state (fallback state) than when the anomaly was detected, depending on the seriousness of the situation. Feedback initiation is the first of the operations designed to bring the reactor to the fallback state. It is preceded by a period called the “initiation period” which enables the plant operator to either resolve the anomaly or implement palliative measures to maintain the reactor in its initial state, or prepare fallback if the anomaly cannot be resolved or compensated for within this period.

Following a sharp drop in 2011, the number of feedback initiations performed in accordance with specifications remained stable between 2012 and 2014. Failure to perform a required feedback initiation is a failure to comply with the OTS and may be due to a number of reasons: mistaken diagnosis of the deviation detected, overshooting the time limit for restoring compliance, or a conflict between safety and availability.

The annual number of reactor feedback initiations required but not performed rose from four in 2013 to eight in 2014. The eight feedback initiations not performed in 2014 were due to mistaken or late diagnosis, leading to non-compliance with the OTS. The diagnosis errors or delays in question had various causes: difficulties in identifying malfunctions, incorrect risk analysis following an equipment malfunction or an organisational failure.

Example of a feedback initiation required but not performed
On 7 June 2014, a test carried out by the plant operator of the Saint-Alban NPP to check closure times of main steam supply system valves revealed that the closure times were not satisfactory. In such an event, the operating technical specifications stipulate that the equipment in question must be brought into compliance, or a fallback to a safe reactor state initiated, within eight hours. In the instance described here, the valves were immediately brought into compliance and their performance, and their closure times in particular, verified. The following morning, however, the safety engineer observed errors in valve closure times. This showed that the problem detected the day before was still unresolved and that the correct procedure - reactor fallback initiation within eight hours - had not been observed.

Fewer equipment failures
The number of safety-related equipment failures (Figure 2.6) was generally lower in 2014 than in 2013. In particular, this drop was observed in the safety injection and chemical and volume control systems (SIS/CVCS) and in the heat sink, where the number of failures fell by 56% and 46% respectively compared with 2013. The steam generator emergency feedwater system (EFWS) was also concerned, although to a lesser extent, with a 21% drop. This drop can largely be explained by the correction of the many compliance gaps in the restraints on seismic-qualified valves reported at all plants in 2013. These defects were found in many systems, including the SIS, CVCS and EFWS, as well as the heat sink, and were due to maintenance non-quality.

Lastly, an increase was observed in the annual number of containment spray system (CSS) failures (nine in 2014, up from five in 2013). This increase was due in particular to cleaning problems detected in CSS risers at several plants. Loose parts and other material such as adhesive tape was found in the risers, which could lead to obstructions. IRSN recommended that EDF should check the cleanliness of these risers during the next refuelling outages in all 900 MW and 1300 MW reactors that might be affected. This problem does not concern 1450 MW reactors.

Frequency of periodic tests less well-observed
The definition of the periodic test schedule (in particular the frequency and test conditions of each test) and the criteria to be met as set out in the general operating rules are all crucial. Since 2007, periodic test procedures have been prepared as part of the project to standardise maintenance practices and methods (PHPM). See IRSN’s 2013 Public Report, page 28. After an initial “test” period for the new operating procedures, the new approach has proved successful and the number of significant safety events due to periodic tests has fallen by 50% since the implementation of the 2013 new test schedule.
Overall assessment of safety and radiation protection performance of nuclear power plants in operation

IRSN’s Position on Safety and Radiation Protection at Nuclear Power Plants in France in 2014

Details of significant safety events (Figure 2.7) fell between 2010 and 2013 and was stable between 2013 and 2014. In addition, there was an increase in the number of significant safety events due to non-compliance with the periodic test frequency (27 in 2014 compared with 20 in 2013) (Figure 2.7); this was perhaps due to less careful periodic test planning.

However, given the many periodic tests to be performed on a reactor (several tens of thousands per year), with frequencies ranging from every day to every ten years, the total number of significant safety events remained low.

**Rise in the number of system alignment errors**

The number of significant safety events reported following a system alignment error rose by nearly 20% (from 36 to 43) between 2013 and 2014, despite several local and national action plans. These events could have led to safety-related systems - or even engineered safeguard systems - becoming unavailable. Most system alignment tasks must be carried out by technicians in the rooms housing the equipment, which makes it difficult for the operating team in the control to monitor and immediately check these actions. For this reason, the reliability of alignment actions depends in part on their traceability and the quality of implementation. The most common errors include errors in the choice of valve to be operated, failure to set the valve correctly and operations that do not comply with the operating documents.

One of the difficulties in system alignment is that the representation of the state of the facility available to the technicians concerned is fragmented and not updated. In addition, alignment tasks are more complex than mechanical drawings indicate, because of the diversity of equipment and technology, the location of equipment at different levels of the facility and the rooms to be crossed. The "lines of defence", in other words, the preparations for an alignment operation and constant communication between the technicians and the operating team, play a vital role in ensuring the successful completion of alignment tasks.

**Errors during maintenance or equipment changes**

The downward trend in the number of inappropriate actions on equipment during maintenance or equipment changes (Figure 2.8) already observed in 2013 was confirmed in 2014. In particular, the number of significant safety events due to a poor technical grasp of a maintenance task - or maintenance non-quality - also fell sharply (down by 40%). However, poorly performed maintenance operations remained the cause of half the significant safety events reported by EDF. A rising trend in the annual number of events relating to maintenance non-quality had already been observed in 2013 (27 in 2014 compared with 20 in 2013) (Figure 2.8): this was perhaps due to less careful periodic test planning.

**System alignment consists, for example, in opening or closing valves and switching equipment on or off to create a circuit suitable for performing the functions required in a specific operating state. System alignment may be necessary in order to perform maintenance work, test a system to ensure its availability, or change the reactor state. It is carried out tens of thousands of times each year at facilities in France.**

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been observed in the years prior to 2013. This increase had been related to difficulties encountered by EDF in maintaining skills at a time when a large number of employees were retiring. As most maintenance operations are outsourced to subcontractors, EDF began an overhaul of its subcontractor monitoring procedures in 2012. Implementation began at various facilities at the end of 2013. Maintenance is mainly carried out during reactor outages, which EDF uses as an opportunity to repair and inspect some types of equipment. Outage periods are consequently extremely busy and require very careful preparation to ensure optimum maintenance work and equipment change planning.

**Example of inappropriate action during preventive maintenance**

On 3 November 2014, EDF observed a failure in a fire detector on reactor 2 at the Nogent-sur-Seine nuclear power plant. After diagnosis by an electrician, a subcontractor was called in to carry out the repair work. The subcontractor, having replaced the faulty detector, wished to check that the fire signal was correctly transmitted to the control room, without the fire damper actually closing. This required setting the detector to its “test configuration”. In fact, the fire damper associated with the detector closed during the test, causing the control room ventilation and iodine filtration system to shut down. The maintenance documents given to the subcontractor for the job did not match the specific features of the detector and the repairer drew on his personal experience which, in this case, was not suited to the job. To prevent any recurrence of a similar error during the work on the fan power supply mentioned above. In this example, the field operator detected the anomaly before the temperature in the electrical rooms reached the alarm threshold defined in the operating technical specifications. If the temperature threshold had been exceeded, the electrical equipment in the rooms could have suffered damage and possibly a failure.

Rapid detection of anomalies is crucial in order to correct the fault quickly and mitigate the actual and potential consequences of the anomaly as far as possible. That is why there are many systems for detecting anomalies (Figure 2.9), such as alarms, periodic tests, inspection rounds and monitoring. The operating teams play an essential role in detecting any anomalies that might occur. They are in charge of responding to alarms, which reveal most anomalies (a third of the significant safety events reported every year) and of day-to-day monitoring of the facility. Inspection rounds are also instrumental in detecting many anomalies.

**Effective anomaly detection systems**

**Example of an anomaly detected during an inspection round**

During a refuelling outage at the Cruas-Meysse nuclear power plant in 2014, work was carried out to make a change to the electrical power supply of the fans in the electrical rooms. The job was considered completed on 9 May 2014. However, on 13 June 2014, a field operator noticed a high temperature during an inspection round. The ensuing investigation revealed that the fan was turning in the wrong direction. This was due to
The annual number of significant radiation protection events involving workers reported for EDF nuclear power plants fell in 2014 compared with 2013. In particular, the number of significant events involving gamma radiography inspections was halved. Once again, the increased number of significant events reported was due to non-compliance with access conditions for orange and red radiologically controlled areas. IRSN’s analysis pointed to an increase in the number of significant events related to shortcomings in personnel dosimetry. EDF should continue and amplify the efforts it has made in this area since 2009. IRSN also highlighted two particularly significant surface contamination events as a result of which the operators concerned exceeded one quarter of the regulatory annual exposure limit.

Breakdown of significant radiation protection events

The regulations on protecting workers from the hazards of ionising radiation require the licensees of basic nuclear installations to notify the French Nuclear Safety Authority (ASN) of any significant radiation protection events (SRPEs). Such events are reported according to criteria defined by ASN.

The events reported in 2014 based on ASN criteria (Figure 2.10) mainly concerned criteria 10 (other significant non-compliance) and 7 (non-compliance with access conditions for radiologically controlled areas), which respectively accounted for 45% and 35% of significant radiation protection events. Criterion 3 (events related to radiological cleanliness) concerned 8% of all events reported. The number of significant radiation protection events for each of the other criteria (2, 4, 6 and 9) remained stable overall at around 3%.

EDF analysed the circumstances and causes of each of the events reported, together with their actual and potential radiological consequences. It then identified and set up corrective action to avoid any recurrence of the events. The results of its analyses are sent to ASN and IRSN.

The ten criteria for reporting significant radiation protection events (SRPE)

| SRPE 1 | Non-compliance with regulatory annual individual dose limit requirements, or unexpected situation with potential to cause such non-compliance under plausible representative conditions, regardless of exposure type (including body contamination). |
| SRPE 2 | Unforeseen situation leading to a 25% overshoot of a regulatory annual individual dose limit value, regardless of exposure type (including body contamination). |
| SRPE 3 | Any significant non-compliance with radiological cleanliness standards, in particular the presence of contamination sources exceeding 1 MBq, outside radiologically controlled areas, or detection of contaminated clothing (> 10 kBq) by C3 monitors or during whole-body radiation dosimetry. |
| SRPE 4 | Any significant activity (operation, task, modification, inspection, etc.) posing a radiological risk, conducted without radiation protection assessment (justification, optimisation, mitigation) or exhaustive consideration of such assessment. |
| SRPE 5 | A malicious act or attempt liable to impact the protection of workers or members of the public from ionising radiation. |
| SRPE 6 | An abnormal situation affecting a sealed or unsealed source with an activity level higher than the exemption limits. |
| SRPE 7 | Signage error or failure to comply with technical conditions for access to or spending time in an area subject to special regulations or prohibited area (orange and red areas or gamma radiography inspection areas). |
| SRPE 8 | Uncompensated failure of collective radiation monitoring systems. |
| SRPE 9 | Inspection of a fixed collective radiation monitor more than a month late (regulatory inspection frequency of one month for fixed systems) or more than three months late for other types of monitor (when the inspection frequency defined in the general operating rules is between 12 and 60 months). |
| SRPE 10 | Any other deviation of significance to ASN or the licensee. |
Overall assessment of safety and radiation protection performance of nuclear power plants in operation

IRSN’s Position on Safety and Radiation Protection at Nuclear Power Plants in France in 2014

The information provided is used by IRSN to monitor trends for all NPPs in France. IRSN observed a slight downward trend in the number of significant radiation protection events reported by EDF (Figure 2.11) (110 in 2014, compared with 119 in 2013 and 112 in 2012). This decrease can be explained by four factors:

- a sharp drop in the number of events related to gamma radiography inspections;
- a drop in the number of events involving access to or time spent in orange areas;
- a drop in the number of events due to deviations regarding training and certification;
- fewer cases of contamination outside radiologically controlled areas.

However, a significant rise was observed in the number of events related to worker dosimetry, in particular failure to wear a dosimeter. The number of events relating to radiation-contaminated clothing and surfaces, radiological monitoring faults (i.e. concerning radiation portal monitors) and unauthorised access to red controlled areas was also slightly higher in 2014.

IRSN observed a decrease in the annual number of significant radiation protection events at INES Level 1 or 2. No Level 2 events were reported in 2014 and only three events were classified at Level 1.

**Personnel dosimetry: rising number of SRPEs**

The most important trend in 2014 compared with previous years was the number of significant radiation protection events related to personnel dosimetry, which saw a considerable increase, rising from 11 in 2013 to 18 in 2014. Moreover, 17 of these 18 events involved failure to wear a passive or operational dosimeter. These figures should, however, be set against the large number of people required to enter radiologically controlled zones. The figures would, however, seem to indicate that the “have you got everything?” measure has not produced the expected results so far (see page 35 of the IRSN 2013 Public Report).
It should be noted that the use of mobile radiation monitors in controlled areas depends on the radiological conditions of the job and is not systematic. This means that workers’ personal doses are only measured by passive and operational dosimetry. The use of dosimeters is thus the only way to ensure constant monitoring of worker exposure.

During the radiation protection events mentioned earlier, the workers concerned had not heard their dosimeter alarms. As part of its corrective measures, EDF has decided to increase the volume of existing dosimeter alarms to 85 dB, and to work on a prototype for an alarm dosimeter combining a vibration feature and audible alarm.

In addition, tests were carried out at the Dampierre-en-Burly NPP on lowering alarm thresholds by about 25% on operational dosimeters. The results showed that lowering dosimeter alarm thresholds led to improved personal dose monitoring. In 2014, EDF decided to extend this practice to all its nuclear power plants in 2015.

Event analysis also revealed difficulties in allowing for changes in radiological conditions between the preparation and performance of an operation. Discrepancies exist between measurement results given in the mapping tools used for prior analysis and the actual radiological conditions in which workers carry out their tasks. It has been found that these maps are not always updated before the work begins.

**Marked decrease in the number of SRPEs related to gamma radiography inspections**

The second most important point for 2014 was a marked decrease in the number of events related to gamma radiography inspections. After increasing in 2013, the number of events due to gamma radiography inspection (Figure 2.13) errors was halved in 2014, falling from 17 in 2013 to eight in 2014 for a more or less identical volume of activity. This trend is very encouraging given the radiological consequences that this type of event can have.

**High percentage of SRPEs related to risk analysis**

The number of significant radiation protection events related to risk analysis errors accounted for about 11% of events reported in 2014 regarding radiation protection. One such event at the Tricastin NPP on 18 August 2014 was classified at INES Level 1 and drew particular attention. During maintenance work, the outside contractor’s site manager was informed that two workers carrying out work at the bottom of the reactor cavity had exceeded their forecast dose limit. Work was interrupted and the two workers concerned were taken off the job.

One of them had received a dose equal to a quarter of the annual regulatory limit of 20 mSv. Several problems were detected:

- the work had been poorly prepared, causing an eightfold increase in the collective dose, due in particular to an underestimation of workers’ forecast exposure times, and to the fact that the dose equivalent rate at the workstation was higher than expected (the maintenance documents did not include the latest site mapping data);

**Personal dosimetry**

Personal dosimetry comprises external and internal dosimetry. **External dosimetry** involves measuring the doses received by a person exposed to a field of radiation generated by a source outside the individual in question. The dosimeters worn by workers are designed to show the dose to the whole body, either by having the dose read at an approved laboratory at a later date (“passive dosimetry”), or in real time (“operational dosimetry”). Operational dosimeters have an audible and visual alarm that alerts workers if they are in a field of radiation that exceeds certain thresholds preset to detect abnormal situations. **Internal dosimetry** measures the dose received as a result of incorporating (inhaling or ingesting) radioactive materials. This type of dosimetry involves whole-body radiation measurements (direct measurement of internal contamination) and radiotoxicological tests.

The results showed that lowering dosimeter alarm thresholds led to improved personal dose monitoring.

**Mapping**

Preparing work inside a radiologically controlled or supervised area involves making a radiological assessment to limit personnel exposure to ionising radiation. The radiological working environment (dose rates, smearable and fixed contamination) must be mapped out geometrically to determine the personal and collective radiation protection measures required for the job. This is referred to as mapping and serves to locate “hot spots” in the work area, show radiological measurement results (together with the related uncertainty), and take appropriate action to adapt the planned work to the radiological conditions.
Effective and equivalent doses

The effective dose is used to estimate whole-body radiation exposure. It factors in the sensitivity of the different types of body tissue as well as the specific type of radiation (alpha, beta, gamma, neutron). The radiation exposure of individual organs is called the equivalent dose. These doses are expressed in sieverts (Sv).

**REGULATORY DOSE LIMITS:**

The effective dose limit for members of the public is 1 mSv/year (excluding natural and medical radiation exposure).

Regulatory dose limits for workers at the greatest risk of exposure are as follows, over a period of 12 consecutive months:

<table>
<thead>
<tr>
<th>Effective dose (whole body)</th>
<th>20 mSv</th>
</tr>
</thead>
<tbody>
<tr>
<td>Equivalent dose</td>
<td></td>
</tr>
<tr>
<td>Extremities (hands, forearms, feet and ankles)</td>
<td>500 mSv</td>
</tr>
<tr>
<td>Skin</td>
<td>100 mSv</td>
</tr>
<tr>
<td>Crystalline lens</td>
<td>150 mSv</td>
</tr>
</tbody>
</table>

**Few, but significant, surface contamination events**

Few events related to surface contamination were reported in 2014, although two of the five events that did occur were classified at INES Level 1. The two events, which occurred at the Belleville-sur-Loire and Le Blayais nuclear power plants, are described below.

The event at the Belleville-sur-Loire plant, which occurred on 26 August 2014, involved contamination from a cobalt-60 particle while two workers employed by a contractor were cutting up used filters from a nuclear auxiliary building ventilation system. The work had begun on 18 August 2014. On 26 August 2014 one of the two workers, who had initially been assigned to work outside the containment airlock, wished to make his facilities radiologically cleaner in order to limit the exposure of workers to radiation.

**Other types of event**

Cross-category analysis by IRSN of all significant radiation protection events reported in 2014 showed no significant change in the number of events involving non-compliance with orange and red area access conditions, which still accounted for 35% of all events reported.

The number of significant radiation protection events reported due to contaminated clothing was very slightly higher than in 2013, up from three to five events. Significant radiation protection events regarding training only concerned non-compliance with the validity dates of radiation protection training: their number has fallen steadily for the past three years.
and highly contaminated areas are monitored after removing additional protective clothing.

EVEREST is designed to ensure that workers wearing normal blue working overalls and those wearing white overalls cannot pass each other inside the controlled area, which reduces the risk of spreading contamination. Personnel leaving a controlled area are therefore monitored in their normal working overalls and helmet by passing through new-generation C2 portal monitors. These monitors are equipped with beta and gamma detectors and can detect low-level clothing, body and internal contamination no matter where the contamination is located.

Operating experience feedback regarding body contamination detected by C2 portal monitors shows that the rate of confirmed detections/total number of C2 monitor checks at sites implementing the EVEREST measure is half that observed at sites where EVEREST is not implemented. In spite of this positive feedback, however, EDF should keep a close eye on the use of good practices associated with EVEREST, as evidenced by analysis of the event reported in 2014 by the Cattenom NPP, where this measure was in place.

At the time of the event, two contractors were carrying out two different operations on the heat exchanger of the chemical and volume control system of the reactor coolant system. The first contractor was responsible for brushing the heat exchanger welds. The second had the task of carrying out liquid penetrant tests on these welds. These jobs had to be carried out one after the other with orange area access authorisation and in compliance with “radiological working conditions with an identified radiological contamination risk and the necessary protective measures”.

A flow restrictor to ensure dynamic containment and a dressing/undressing airlock had been installed at the entrance to the work area. An air contamination monitor was also installed.

The workers wore ventilated protective suits.

They also had to carry out their work in turns. When brushing was completed, the first contractor’s team left the work area and the C2 monitor detected that a member of the team had been contaminated by radiation. When the second contractor’s team left the area later, the C2 monitor detected that they had also been contaminated. Four other people who had passed through the corridors near the work area and who had helped the working teams to undress were also contaminated. This led to internal contamination in ten of the workers, although the individual doses remained well below the dose recording threshold of 0.5 mSv.

Area contamination checks revealed surface contamination in the corridors adjoining the work area, reaching a maximum of 8 Bq/cm² compared with an authorised regulatory limit of 0.4 Bq/cm². In the rooms adjoining these corridors, contamination points exceeding 100 Bq/cm² (maximum of 647 Bq/cm²) were measured.

This event brought to light shortcomings in workers’ compliance with EVEREST dressing and undressing practices, and in confinement: dynamic confinement was not effective because the flow restrictor was not installed in the right position; the "aerosol" monitors were not operating and contamination remained undetected, resulting in its spread to the neighbouring rooms and corridors.

Flow restrictor and static/dynamic containment

A flow restrictor is a device used to ensure dynamic containment by maintaining negative pressure inside a work area. The dynamic containment system controls the direction and rate of air flow, ensuring that air flows towards areas with the highest contamination level. A static containment system consists of obstacles such as walls (for example, the walls of a room considered leaktight).
Analysing events and incidents is a vital part of IRSN’s monitoring of operating safety at nuclear power plants. It calls for good knowledge of the facts and the background to the event or incident. This is essential for analysing root causes, estimating the actual and potential impact on facility safety and, in some cases, on the population and environment, assessing the relevance of corrective action taken by the licensee, and considering improvements that could be made to avoid any recurrence of the event.

Events and incidents have various causes; they can be due to human or organisational errors, equipment failures or failures induced by design faults.

A specific feature of EDF’s nuclear power fleet is its standardisation. The fleet is divided into three reactor series according to reactor power - 900 MW, 1300 MW and 1450 MW. Standardisation has operating as well as economic advantages (same operating reference documentation, optimised maintenance, shared operating experience feedback, etc.). However, the drawback is that any detected failure or error is liable to affect several reactors or even the entire fleet. This is referred to as a “generic fault”. IRSN pays particularly close attention to early detection of such faults and to how EDF deals with them. The process for correcting some generic faults can be complex and take several years. Palliative measures can then be implemented to ensure a satisfactory level of safety in the interim period until the fault has been corrected.

Presence of loose parts and cladding defects at the Saint-Laurent-des-Eaux B plant

Internal flooding leading to electrical equipment unavailability

Compliance gap possibly causing overheating in the emergency turbine generator room
Presence of loose parts and cladding defects at the Saint-Laurent-des-Eaux B plant

In 2014, loose parts were detected in the reactor coolant system of reactor 2 at the Saint-Laurent-des-Eaux B nuclear power plant. The presence of loose parts in the reactor coolant system can cause damage to fuel rod cladding. This in turn can lead to the release of fission products in the reactor coolant system and jam the rod cluster control assemblies.

The event concerning the Saint-Laurent-des-Eaux B2 reactor

In February 2014, a significant increase in the $^{133}$Xe concentration and the $^{133}$Xe/$^{135}$Xe ratio was detected in the reactor coolant system. As a result, a fuel cladding failure was suspected in the reactor. In accordance with the operating technical specifications, these parameters were closely monitored until the scheduled reactor outage on 22 August 2014. Significant $^{133}$Xe and $^{131}$I release was observed in the reactor coolant system.

When the reactor was shut down and the reactor coolant system depressurised, significant $^{133}$Xe and $^{131}$I release was observed in the system due to the difference between the pressure inside the fuel rods (Figure 3.2) and the reactor coolant system pressure.

Before opening the reactor coolant system, EDF purified it for 22 days. The environmental (gas and liquid) release measured following this event remained below the regulatory limits, thanks in particular to the purification and treatment of gaseous and liquid effluents, as a result of which the release had very little impact on the environment.

The event did have a slight impact on personnel dosimetry, as the purification process called for additional tasks inside the reactor building. Personnel dose measurements, however, were below regulatory limits.

After investigating the incident, EDF found that the radioactive release to the reactor coolant system was due to two leaking fuel assemblies (Figure 3.1). Significant $^{133}$Xe and $^{131}$I release was observed in the reactor coolant system.

Xenon-133 ($^{133}$Xe) and xenon-135 ($^{135}$Xe) are radioactive isotopes of xenon, a gas produced during nuclear fission reactions. A rise in $^{133}$Xe concentration in the reactor coolant system and in the $^{133}$Xe/$^{135}$Xe ratio are indicative of fuel rod cladding leakage.

Iodine-131 ($^{131}$I) is a radioactive isotope of iodine, which is one of the chemical elements produced during nuclear fission reactions.

A fuel assembly in a 900 MW reactor consists of 264 fuel rods, each of which contains a stack of slightly enriched uranium dioxide pellets enclosed in cladding. Each rod is about four metres long and about one centimetre in diameter. Rod pitch is determined by grids, which also hold the rods in place. The rod cluster control assembly absorber rods are inserted into guide tubes. A debris filter is fitted to the bottom of each fuel assembly.

After investigating the incident, EDF found that the radioactive release to the reactor coolant system was due to two leaking fuel assemblies.
How are leaking fuel assemblies identified?

When a fuel cladding failure is suspected while the reactor is in operation, EDF performs nondestructive tests when the fuel assemblies are unloaded to locate the leak causing the $^{133}$Xe and $^{131}$I release. First, liquid penetrant testing is performed in the refuelling machine mast on all the fuel assemblies. When this testing has not clearly determined whether a fuel assembly is leak-tight, the “suspected” assembly undergoes another liquid penetrant test in the fuel building cell.

In 2014, leaks were suspected in 17 fuel assemblies unloaded from reactor 2 at the Saint-Laurent-des-Eaux B plant following liquid penetrant testing in the refuelling machine mast. Liquid penetrant testing in the fuel building cell revealed leaks in two of these 17 fuel assemblies. Visual inspection of the two leaking assemblies did not clearly reveal any defects in the fuel rods. Televisual inspections, however, confirmed the presence of loose parts in the fuel assemblies in question.

The televisual inspections revealed several loose parts under the bottom nozzles of six fuel assemblies (Figure 3.3), inside the two leaking fuel assemblies, at the bottom of the reactor vessel and in the vessel internals.

Where do loose parts come from?

In this particular event, investigation showed that the loose parts came from a Helicoflex® seal (Figure 3.4), about 52 mm...
Loose parts and other foreign materials are unwanted items of all types and shapes found inside reactor systems. They can find their way into systems during maintenance or other operating activities (metal chips, filings, adhesive tape, screws, washers, wrenches, screwdrivers, etc.), during fuel assembly handling, as a result of equipment failures that damage parts or cause them to come loose, or as a result of human negligence (objects left behind or dropped) (see the article on page 24 of the 2012 PWR Public Report).


**IRSN’s position**

IRSN took part in the “reactive” inspections that were carried out by ASN at the Saint-Laurent-des-Eaux B plant to check the implementation of corrective measures by EDF.

It also examined the potential impact of the remaining loose parts on safety, and found that:

- the parts in question were very light and unlikely to cause any damage to piping, channel heads and other parts of the reactor coolant system and auxiliary systems;
- the presence of the loose parts was not likely to induce any corrosion phenomena in the reactor coolant system or damage to steam generator tubes.

IRSN nonetheless considered that, despite the presence of the fuel assembly debris filters (Figure 3.5), some loose parts may have remained in the reactor core, with the possible risk of:

- rod cluster control assembly jamming, although this risk is lessened by the size of the loose parts in question compared with the debris filter mesh;
- fuel rod leakage recurring due to cladding wear.

**Despite the presence of the fuel assembly debris filters, some loose parts may have remained in the reactor core.**

**What measures are implemented by EDF?**

The two leaking fuel assemblies have been removed from the reactor core for analysis to identify the leaking rods, locate the defects and determine their exact cause. This analysis is in progress and the results will be available at the end of 2015.

During the following cycle, many tests were carried out to check the performance of any equipment likely to be affected by the presence of loose parts. Valves and similar devices in systems where loose parts might be found were controlled and tested during the reactor restart phase.

The safety injection system (SIS) is used in accident situations to inject water into the reactor coolant system to cool the reactor core.

The residual heat removal system (RHRS) is designed to ensure reactor core cooling when the reactor coolant temperature falls below 180°C.

Rod cluster control assemblies (RCCA)

Used to partially or completely lower the control rods into the fuel assemblies to control the reactor and rapidly halt the nuclear reaction.

**Figure 3.5**

Diagram of a fuel assembly debris filter

Mesh size: 3.3 mm x 3.3 mm or 2.74 mm x 2.74 mm
Internal flooding leading to electrical equipment downtime

EDF has implemented measures at its facilities to guard against the risk of internal flooding. These are aimed in particular at precluding the risk of simultaneous flooding of both redundant electrical trains in reactor safety-related systems. However, two recent events at the Fessenheim and Le Blayais plants, where water flowed through openings that were not watertight, highlighted certain weaknesses in nuclear reactor electrical rooms regarding the risks of internal flooding. EDF responded by setting up an action plan covering inspections of and repairs to the openings in all buildings at all its nuclear power plants in operation.

Internal flooding at nuclear power plants

The various buildings of a nuclear power plant house a large number of water pipes. A leaking pipe can cause flooding in a room (this is known as internal flooding) and the water could then spread to other rooms through grating and openings (Figure 3.6) that are not leaktight. In this event, water could splash or cover electrical and mechanical equipment and cause failures. Internal flooding affecting electrical equipment could also generate untimely command signals (such as automatic reactor trip signals, or safety injection or containment spraying start-up). Protective measures are in place to guard against the risk of internal flooding due to breaks in water pipes. The main protective measures implemented by EDF to ensure that internal flooding cannot simultaneously affect both redundant electrical trains (A and B) of a safety system or both redundant trains of a system involved in shutting down the reactor and maintaining it in a safe state include:

- Raising equipment in the rooms where it is located or installing protective structures;
- Installing redundant equipment belonging to different trains in physically separate rooms (physical separation);
- Raising equipment in the rooms where it is located or installing protective structures;
- Installing redundant equipment belonging to different trains in physically separate rooms (physical separation);
- Raising equipment in the rooms where it is located or installing protective structures;
- Installing redundant equipment belonging to different trains in physically separate rooms (physical separation);
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- Raising equipment in the rooms where it is located or installing protective structures;
- Installing redundant equipment belonging to different trains in physically separate rooms (physical separation);
- Raising equipment in the rooms where it is located or installing protective structures;
- Installing redundant equipment belonging to different trains in physically separate rooms (physical separation);
taking steps to contain any flooding in a single safety train (for example by putting up low walls on the floor to prevent water from spreading from one room to another, adding leaktight covering to some openings, etc.);
installing drainage systems in rooms exposed to the risk of flooding.

However, two recent events at the Fessenheim and Le Blayais plants highlighted certain problems regarding leakage through openings.

Internal flooding at the Fessenheim nuclear power plant

In April 2014, too much make-up water was added to the tank in the intermediate demineralised water system tank of reactor 1 at the Fessenheim plant, causing water to flow out of the tank through its overflow. As the overflow pipe was blocked by corrosion residue, the water entered another system through another line and flowed out at elevation +15.50 m of the electrical building. It then spread to the lower levels of the building through openings that did not meet watertightness requirements.

Water spilling into the building at different elevations led to numerous isolation faults in electrical equipment and caused a failure on an electrical train used for the automatic reactor trip signal.

The automatic reactor trip function remained available, however, because of its redundant train, and the reactor could also have been tripped manually on both trains. This sequence of failures led EDF to shut down the reactor for about fifty days to locate all the electrical and electronic modules in contact with water and replace and requalify them. In addition, the overflow pipe, which was not checked for blockages, was unblocked and will now undergo periodic borescopic inspection.

Internal flooding at the Le Blayais nuclear power plant

In March 2014, EDF reported that inspections had shown that leaktight covering was missing on 50 openings on all four reactors at the Le Blayais plant. The inspections had been carried out following an event reported in September 2012, during which a large quantity of water had been discovered in the rooms of train A in the reactor 1 electrical building at the Le Blayais plant.

Water leaking out (Figure 3.7) from a break in a potable water system pipe passing through a service duct at elevation +15.50 m, seeped through openings that were not watertight and ran along some cables, damaging electrical cabinets located at elevation + 11.50 m and affecting the relays in train A of the reactor protection system. Train B remained available because the reactor could also have been shut down manually on both trains.

In this context, an opening is a gap in a wall or ceiling between two rooms for cable and piping penetrations. Openings located between two adjoining rooms, each housing redundant equipment that is part of a separate safety system from that in the other room, must be watertight.

Leaking openings

One of the openings located at elevation +15.50 m, which should have been leaktight, had a gap in the covering about 1 cm wide and 40 cm long. In spite of the different inspections performed during preventive maintenance, this defect had never been detected. The defect was remedied and closer monitoring set up.

Another opening located at elevation + 11.00 m of the electrical building was not watertight, despite work carried out in 1996 to seal all the openings. EDF pointed out that it had sometimes been difficult to properly seal off some areas because of the number of cables in the openings. It has, however, undertaken a study to improve working procedures to seal the openings in this floor.
Events, incidents and anomalies

In March 2014, EDF reported that inspections had shown that leaktight covering was missing on 50 openings concerning all four reactors at the Le Blayais plant.

EDF has undertaken to update its safety documentation, and inspect and correct any compliance gaps concerning the openings of all nuclear power plant buildings for its entire reactor fleet.

IRSN analysed the events mentioned above, taking into consideration their potential impact on facility safety. It found that the event at the Le Blayais nuclear power plant could be considered "generic" given that:
> the modification work at the root of the observed compliance gaps was carried out on all 900 and 1300 MW reactors;
> the database defining the requirements applicable to the openings is used for both these reactor series and is not exhaustive.

Following IRSN’s technical review, EDF defined an action plan aimed mainly at ensuring that the information in the database relating to openings subject to watertightness requirements was exhaustive and compliant, carrying out inspections of openings, and rectifying any compliance gaps detected. EDF will also update its maintenance programme. It has undertaken to update its safety documentation, inspect and correct any compliance gaps concerning the openings of all nuclear power plant buildings for its entire reactor fleet. All compliance gaps concerning the electrical buildings of 900 MW reactors should be eliminated by 2016. This work is scheduled to continue until 2018 for the other buildings of 900 MW reactors and for 1300 and 1450 MW reactor buildings.

IRSN found EDF’s action plan satisfactory and will keep track of its progress.

Conclusion

The two events that occurred at the Fessenheim and Le Blayais nuclear power plants revealed a certain weakness in nuclear reactor electrical buildings regarding the risks of internal flooding, despite the protective measures implemented to guard against them. The events also showed that the safety documentation setting out requirements relating to the watertightness of openings was incomplete, and highlighted the limits of the preventive maintenance programme. EDF therefore began taking action to improve its safety documentation and maintenance programmes to bring its facilities into compliance.

Compliance gap possibly causing overheating in the emergency turbine generator room

In 2014, EDF detected that the temperature in the emergency turbine generator room could become too high after several hours of uninterrupted operation. This was observed for all reactors in the fleet, regardless of series. Overheating in the room was due to operation of the actual equipment and its auxiliaries, and to the lack or poor performance of ventilation in the room. It led to the unavailability of this equipment, which is required in the event of an accident involving total loss of reactor electrical power (EDF grid and emergency diesel generators). IRSN undertook an in-depth analysis of the safety impact of this “compliance gap” to assess the relevance and adequacy of the compensatory measures taken and the corrective action planned by EDF to remedy the situation.

Total loss of reactor electrical power

The safety systems of a nuclear reactor required in the event of an accident - the safety injection system in particular - are connected to a 6.6 kV electrical power supply via two redundant switchboards called LHA and LHB. Each of these switchboards can be energised by one of two electrical mains power supplies called “off-site power supplies” (a main supply and an auxiliary supply) or, when these fail, by two emergency diesel generators (called “on-site power supplies”). The simultaneous failure of all four electrical power supplies (off-site and on-site), combined with the failure of the two switchboards - LHA and LHB - lead to a station blackout (called H3 from the name of the operating procedure to be applied in this situation). In this event, residual heat removal from the reactor calls for the use of emergency electrical power supplies. These include:

→ an emergency turbine generator (TAS LLS) used to supply water to the reactor coolant pump seals to avoid damage to them which could lead to a loss-of-coolant accident;
→ a diesel generator (for 900 MW-CPY reactors) or a gas turbine (for other reactor series), to supply electrical power to some safety systems. In the wake of the Fukushima accident, it is eventually planned to equip each reactor with a station blackout diesel generator (SBO DG), which will also be used for cooling and maintaining a sufficient water inventory in the reactor vessel.

Total loss of electrical power is one of the “beyond design-basis accident” situations.

Function of the emergency turbine generator

Each reactor is equipped with an emergency turbine generator (TAS LLS) that is driven by steam from the secondary systems. In the event of total loss of electrical power, this turbine generator provides power via a specific electrical distribution board (LLS switchboard) for:

→ instrumentation essential for reactor control,
→ control room lighting,
→ a pump called the test pump used to supply water to the reactor coolant pump seals.

There is one test pump for two reactors in the 900 MW series.

The emergency turbine generator starts up automatically so that reactor coolant pump seal injection via the test pump is carried out in less than two minutes. According to operating procedures, the turbine-driven pump (or pumps in the case of 1300 and 1450 MW reactors) then removes residual heat to reach the operating state where reactor coolant pump seal injection is no longer required (reactor coolant system pressure less than 45 bar). This state is reached in less than 24 hours.

Studies of risk reduction category A situations also consider the case of turbine-driven generator failure. For 900 MW and 1450 MW reactors, the generator or gas turbine (LHT) has enough electrical power to re-energise the safety injection system and make up for the loss of

Beyond design-basis accident situations are abnormal or emergency situations induced by multiple failures affecting equipment or systems that are critical for reactor safety. They were not taken into account at the initial reactor design stage, but were the subject of safety studies based on French and international operating experience feedback, in particular the Three Mile Island accident in the USA in 1979.


Reactor coolant pump seals: three seals prevent leakage between the pump and the rotating shaft of the reactor coolant motor-driven pumps (see the description of the three seals on page 41 of the 2012 PWR Public Report). Water must be supplied to these seals (“seal injection”) to ensure that they remain leaktight. This function is carried out by the chemical and volume control system (CVCS) under normal reactor operating conditions.

In the event of total loss of electrical power, the CVCS can no longer fulfil this function. The absence of reactor coolant pump seal injection can lead to a loss-of-coolant accident.

Loss of reactor coolant can lead to core melt within a few hours if no water is supplied to make up for the loss.

coolant through the break at the reactor coolant pump seals. For 1300 MW reactors, however, the gas turbine cannot generate enough power²: the operating procedures for these reactors have therefore been modified to minimise the need for gas turbine power to manage the situation and, in particular, to compensate for the loss of coolant.

A generic compliance gap

In May 2012, during a periodic test without ventilation, EDF observed that the temperature in the emergency turbine generator (TAS LLS) room of reactor 2 at the Fessenheim nuclear power plant was high. In particular, there was a risk of the maximum operating temperature of the most sensitive equipment in the LLS system being exceeded.

According to EDF, the overtemperature in the room was mainly due to the operation of the equipment installed there, especially the emergency turbine generator (Figure 3.8).

The initial thermal studies performed by EDF following this observation confirmed that the reactors in other series were also concerned.

When the emergency turbine generator is in operation, the temperature in the room where it is installed increases steadily and rapidly in all reactor series. According to EDF calculations, the temperature would rise to 70°C for an outside temperature of about 25°C (and in less than one hour for 900 MW-CPY and 1300 MW reactors). That means it would exceed the maximum operating temperatures (between 40°C and 60°C) of various electrical and mechanical equipment belonging to the LLS turbine generator system. As a result, equipment required in the event of an accident involving total loss of reactor electrical power (including reactor coolant pump seal injection) could become unavailable in the short term.

At the Le Bugey plant, high temperatures would also affect the turbine-driven pump of the steam generator emergency feedwater system, which is installed in the same room as the LLS emergency turbine generator.

An LLS emergency turbine generator failure could occur within a few hours, if not less, especially in 900 MW-CPY and 1300 MW reactors, where calculated temperatures are the highest. For this reason, EDF examined whether the provisions made at nuclear power plants in connection with risk reduction category A could also be affected by high temperatures.

It also defined corrective action to restore LLS system operation, along with compensatory measures pending the implementation of corrective action.

Action and measures defined by EDF

EDF has defined long-term corrective action aimed at ensuring the availability of the emergency turbine generator in the event of total loss of electrical power. Within this context, it made a number of changes (including hot piping insulation and installation of louvres in the doors of the room) to the reactors at Fessenheim, which lowered the temperature in the room to an acceptable level to ensure correct emergency turbine generator performance.

In addition, EDF made plans to install mechanical ventilation in the emergency turbine generator room to guarantee availability in the event of total loss of electrical power. This will, however, require significant material changes. These changes are scheduled for 2017-2021 for 900 MW-CPY reactors, 2018-2021 for the reactors at Le Bugey, and 2018-2022 for 1300 and 1450 MW reactors.

Pending corrective action, EDF has adopted the following compensatory measures:

- 1300 MW reactors: as the current complementary measure no longer functions owing to the temperature increase in the room, EDF has introduced a modification to the emergency turbine generator I&C to prevent the generator in question from starting up in the event of failure on the two emergency-supplied distribution switchboards (LHA and LHB), and to restore the complementary measure;
- reactors at the Le Bugey plant: EDF has introduced an operating procedure to be followed in the event of a total loss of electrical power. It consists in keeping the doors of the LLS emergency turbine generator room open to ensure that the

² The case of total loss of electrical power combined with a loss of reactor coolant was not factored into reactor design.
steam generator emergency feedwater turbine-driven pump, which is located in the same room, continues to operate until the gas turbine starts up.

EDF considers that these compensatory measures should ensure that the safety impact of this compliance gap remains at an acceptable level until a definitive solution is found.

**IRSN's position**

IRSN carried out an assessment of the relevance and adequacy of the compensatory measures implemented by EDF, as well as the acceptability of corrective action and the time frame for eliminating the compliance gap definitively. It found that the compensatory measures identified and set up by EDF were inadequate. In particular, it considered that EDF should toughen its operating and/or maintenance requirements on all equipment used to compensate for the unavailability of the LLS emergency turbine generator and the steam generator emergency feedwater pump at Le Bugey during an H3 situation, until the compliance gap is eliminated. Furthermore, it recommended that EDF should, at the first reactor outage, eliminate all compliance gaps liable to undermine the reliability of on-site electrical power supplies or the emergency-supplied power distribution systems of reactors. Regarding the time frame for eliminating the compliance gap, IRSN recommended that EDF should take every step to guarantee an effective reactor fallback procedure by the end of 2018, in the event of an H3 situation induced by an earthquake.

EDF also defined corrective action to restore LLS system operation, along with compensatory measures pending the implementation of corrective action.

**Current compensatory measure**

In the event of failure on both emergency-supplied distribution boards (LHA and LHB), the LLS switchboard will be energised via a permanent switchboard which, although it has no backup, is available because it is connected to an off-site power supply (Figure 3.9).

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**Figure 3.9**

Diagram of off-site and on-site electrical power supplies and station blackout diesel generators for 900 MW reactors
Significant upgrades

Changes and upgrades are made to France’s nuclear reactors throughout their lifetime, with the aim of continuously improving safety.

Advances in technical and scientific knowledge, weaknesses detected and lessons learned from operating experience feedback, changes in the environment or regulations, together with economic factors are just some of the reasons behind the changes made to facilities or operating procedures.

Periodic safety reviews, carried out every ten years in accordance with Article L.593-18 of the Environment Code, provide an ideal opportunity for implementing these changes. They are associated with the preparation of long ten-yearly reactor outages, during which such work as replacing heavy equipment is carried out, and major changes are made to facilities.

It may take several years of research and exchange, during which IRSN examines the documentation submitted by EDF, before some changes and upgrades can be precisely defined and implemented. Others, however, must be implemented more rapidly according to a suitable schedule.
Renovation of the I&C system on 1300 MW reactors

French 1300 MW reactors, which were commissioned in the mid 1980s, were among the first in the world to use digital technology - in other words, computers - to perform the most important safety-related automatic actions. After some thirty years of operation, characterised by excellent operating experience feedback, the computers, which were designed exclusively for these reactors, are now obsolete and EDF has had to renovate them. There is still some controversy concerning the use of digital technology for nuclear safety functions and it is within this context that IRSN carried out a precise assessment of EDF’s proposals for the renovation.

Two essential I&C systems

Like EDF’s other nuclear power reactors, 1300 MW reactors are equipped with a “protection” system which is not used to operate the reactor, but constantly monitors numerous physical parameters, including temperatures, pressures and neutron flux. It thus ensures real-time detection of any drift in parameter values liable to lead to an incident or accident. If necessary, the protection system can shut down the reactor automatically by inserting neutron absorbers into the core to interrupt the chain reaction. It can also trigger other safety actions, such as starting up the safety injection system to ensure core cooling in the event of a loss-of-coolant accident.

For some of these actions, the protection system requires a measurement of the neutron flux produced by the reactor core. Real-time determination of this flux is based on information received from detectors placed around the core, and calls for highly specific calculations. These are carried out by a dedicated system called the nuclear instrumentation system. These two I&C systems are classified at the highest safety level and must consequently meet very strict requirements in terms of construction and operation right from design.

Renovation due to obsolescence

The protection system and the nuclear instrumentation system in use since the 1300 MW reactors were commissioned were designed using digital technology on embedded computers specially designed for these applications. These computers had become obsolete and were also unable to handle the additional functions required after the periodic safety review that will be carried out as part of the third ten-yearly outage for this reactor series.

EDF therefore decided to replace both these systems. In light of the excellent operating experience feedback regarding these computers, EDF opted to use digital technology again and to turn to the same constructor that built the existing computers. However, as these were designed some thirty years ago, which is a long time in the rapidly changing world of digital technology, the hardware and software of both systems had to be completely redeveloped.

Architecture and reliability of new equipment

In view of the importance for safety of the two systems mentioned above and the scale of the renovation work (a total of about fifty computers for both systems), EDF began technical talks with IRSN from the outset of the project in 2011.

When IRSN assesses an I&C system, the first questions it asks are: will the system be capable of performing the required functions, even in the event of a component failure, fire or flooding in the electrical room where it is installed, or if part of the system is unavailable due to maintenance operations?

These questions concern the architecture (or structure) of the system. By opting to renovate the systems on a computer-by-computer basis, EDF was able to preserve the architecture of each system. In this architecture, each computer was replicated by several others (up to four) that performed the same calculations, but were located in different rooms. The results obtained by the different computers were then consolidated according to a voting system to ensure system robustness with regard to the hazards mentioned above. IRSN’s assessment thus entailed verifying that the operating principles

An embedded computer is a computer that is specially designed for real-time processing of electrical signals from sensors (pressure, temperature, etc.), to which it responds by sending other electrical signals to display alarms, automatically shut down the reactor, or supply electrical power to components such as pumps or valves. This type of computer is not designed to interact directly with a human operator, but to trigger automatic actions that must often be carried out in less than a second. Consequently, it is not equipped with a keyboard or screen.
Significant upgrades

and construction details did not impair the robustness of the initial architecture. As well as considering robustness regarding the above hazards, reactor design takes into account certain very hypothetical scenarios involving the failure of the entire protection system. These situations are dealt with by another system using different technology. IRSN checked that the components used for the new computers were also different from those used by this system, which has not been renovated or replaced.

It also ensured that each piece of equipment replaced met the appropriate reliability requirements. As an illustration, the reliability typically required for a printed circuit board in an embedded computer (Figure 4.1) performing safety functions is one failure for eleven years of almost uninterrupted operation.

Renovated system software

Following its hardware assessment, IRSN turned to the assessment of digital technology issues in order to ensure that the new software on the renovated computers had been developed using the necessary means to preclude any errors.

This issue was at the heart of the renovation project as far as safety was concerned. The software programs used are relatively large, with several tens of thousands of lines of code. In addition, the redundant computers use an integral copy of the same software and some parts of the software (for example, the parts used for self-monitoring processing) are actually identical in all the replaced computers. A single logic error and the whole protection system could be unable to perform the safety functions required under certain circumstances. Although provisions are made for another system to take over should such a situation occur, every step should be taken to prevent this type of situation from arising.

While it is true that everyone has come across software bugs in everyday applications, which is why it is often thought that systems based on digital technology are too complex to be fault-free, the operating experience feedback from the software used for more than thirty years is very satisfactory and is comparable to embedded software used on board airliners. No safety function has ever been prevented by a protection system software error at a French nuclear power plant. Software can, in fact, be designed to suitable standards of quality subject to compliance with “drastic” rules to master all the complexities. An overview of these highly specific rules, issued in international standards and guides, is provided below, together with an account of IRSN’s assessment of their application in the renovation project for 1300 MW reactor I&C systems.

Firstly, this type of software must run according to a principle known as “determinism”. This means that it must perform a series of tasks - data acquisition, data processing, transmitting orders - in a loop and in a sequence determined during design. This principle also means that each data item used by the software must always be stored in the same memory location in the computer. It significantly limits design choices for writing the software, thus simplifying verification. For the 1300 MW reactor instrumentation and control systems, IRSN closely examined design documents to check compliance with this principle. It also verified compliance with other software design rules, including the provisions included for each system to perform “self-monitoring” and, in the event of a fault, to switch to a predefined safer configuration.

Secondly, the software must follow a very strict process, which is also set out in an international code. In particular, this process defines a series of steps where the software designers must complete each step before proceeding to the next. Another requirement is that an independent team - not the design team - should carry out a verification at the end of each step. This second team is responsible for “independent verification and validation”: verifying design, defining various test campaigns and demonstrating their ability to detect any faults, and performing these tests. If a fault is detected or a change made, the standard stipulates that the relevant part of the process should be started over, which implies checking all the related documents.

IRSN checked that the development process for the renovation of the I&C systems of 1300 MW reactors - and the dozens of documents produced by this process - complied with the above requirements.

The process also entails performing various test campaigns.
in which tests are carried out successively on each part of
the software, then on the whole software program of interconnec
ted computers. IRSN also assessed these campaigns, which lasted several
months. The most difficult part of this type of assessment involved
checking that the tests selected by the manufacturer were both
relevant and adequate, taking into account technical criteria, the
type of functions the software had to perform and the programming
language used.

In addition to verification tests and manual analyses, EDF car-
rried out an automated verification of all the programs on new
computers, using an abstract interpretation static program
analysis tool. It was the first time that this type of tool had been
put to such systematic and comprehensive use in the nuclear
industry. It provided mathematical proof that the software on
the new computers was free of certain types of fault that are hard
to detect by testing (division by a potentially zero number, access
to out-of-bound arrays, etc.) This proof complemented other verifi-
cation methods to confirm design quality, where the objective is zero
defects.

**Conclusion**

Renovating the I&C systems of 1300 MW reactors is a major
undertaking, both in terms of the number of reactors con-
cerned (twenty) and the scale and safety functions of the systems
being renovated. Several factors contributed to the effectiveness
of IRSN’s detailed assessment of this work, which led to the
acceptance of the two renovated systems:

- the computer manufacturer,
- the licensee EDF and IRSN

have built up thirty years of experience in the field of safety
software;

- the assessment began at a sufficiently early stage to take
into account the complexity of the renovations and to monitor
each stage of the work;

- the manufacturer and the licensee sought to master the
complexity of the software programs by reducing functionali-
ties to the strict minimum;

- the joint goal was to drive progress in the state of the art
through the use of novel verification techniques grounded
in sound mathematical bases, which boosted confidence in
the renovated system software.

IRSN submitted its opinion on the renovation of the I&C sys-
tems of 1300 MW reactors to ASN in November 2014 and its
findings were published ([read more](http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2014-00413-VD3-1300.aspx)). I&C renovation work was
scheduled to begin in 2015 on reactor 2 of the Paluel plant.

Abstract interpretation of a computer program means representing it by an “abstraction”, such as the interval of values that each
variable can have for all the possible executions of the program, and “interpreting” the program instructions accordingly.

For example, if an instruction adds the variables a and b, which, at this point in the program are in the intervals [3, 5] and [-1, 1] respectively, then the sum will definitely be in the interval [2, 6]. If this sum then appears in the denominator of a division, it proves
that there is no division by zero at this point. This technique is used to demonstrate, in mathematical terms, the absence of faults such as
division by zero, square root of a negative number or access to an array element outside its bounds.

It is extremely difficult to design algorithms for relevant abstract interpretation of industrial-scale computer program. Thus, the
interval-based abstraction mentioned above often proves ineffective; if a variable can only have the values -2 and +2, it is represented by
the interval [-2, +2], with the result that the tool will be unable to prove that division by this variable is still possible, although in fact
it is. Abstraction therefore provides a “safe approximation” of reality, in that it may in some cases fail to prove a true property, but
cannot declare that a false property is true.

The last twenty years, however, have seen significant innovations, especially in France, which have led to the design of software tools
capable of analysing industrial-scale computer programs. These tools largely contribute to increasing confidence in safety software
programs. IRSN’s own research helps to develop tools such as this and encourages
their use.
Risks induced by operation of the Dunkirk liquefied natural gas terminal on the Gravelines nuclear power plant

EDF subsidiary, Dunkerque LNG, has built a liquefied natural gas (LNG) terminal in an area belonging to the Port of Dunkirk Authority, some four kilometres from the Gravelines nuclear power plant. As the terminal significantly modifies the near-field environment of the plant, EDF analysed the risks that operation of the terminal will induce for Gravelines. At ASN’s request, IRSN examined the related documentation submitted by EDF.

A collision at sea between an LNG tanker and a large ship or another heavy object could cause a break in the LNG tank on board the tanker, resulting in LNG spillage in the sea. In its analysis, EDF calculated that, allowing for tides, sea depths and ship draughts, an LNG tanker that has gone adrift (following a collision for example) would run aground at least 3 km from the reactor 1 building of the Gravelines nuclear power plant. In the event of immediate ignition, the LNG will flash fire, which would give off a great deal of heat, with thermal effects potentially reaching the nuclear power plant by radiation; however, if ignition is not immediate (this is known as delayed ignition), the LNG could vapourise on contact with water or the ground and form an explosive mixture with air. The cloud formed could be driven by unfavourable winds towards the nuclear power plant. If it were to explode, it could induce thermal effects, such as flash fire (explosion characterised by short duration, high temperature and rapidly moving flame front), and pressure effects. A flash fire only produces intense thermal effects in the immediate vicinity of the cloud; on the other hand, pressure effects can spread and cause significant overpressure over relatively large distances.
of the LNG, EDF models based on the above minimum distance between the plant and the tanker, showed that the potential thermal effects of an LNG pool fire at sea would be unlikely to significantly undermine plant safety. IRSN’s analysis supported these findings. Regarding delayed ignition, EDF and IRSN models both indicated that in the event of unfavourable winds, the cloud formed could cover much - if not all - of the Gravelines site. Nonetheless, given the distance between the plant and the loss of containment, the concentration of explosive gas would gradually decrease as the cloud progressed and would only slightly exceed the lower explosive limit (LEL). EDF concluded that safety at the Gravelines plant would therefore be unaffected.

EDF undertook to:

1. take steps before the LNG terminal is commissioned to strengthen the heat flux resistance of the diesel generator support systems on the roofs of the diesel generator buildings;
2. demonstrate the mechanical strength of the RWST with regard to an external pressure of 70 mbar generated by the ignition of an LNG vapour cloud in its retention pit. These changes will be analysed by IRSN.

A collision at sea between an LNG tanker and a large ship or another heavy object could cause a break in the LNG tank on board the tanker, resulting in LNG spillage in the sea. Natural gas vapours are only explosive when mixed with an oxidant. In the case of interest here, the oxidant is the oxygen in the air. The lower explosive limit (LEL) is the minimum concentration of natural gas in air at which it is physically possible for the mixture to explode.

The Gravelines nuclear power plant has six reactors. The reactors are grouped together two by two (twin reactors) or pair of reactors. Some equipment required to perform basic safety functions in the event of total loss of electrical power is common to a pair of reactors and can therefore only be used by one reactor at a time.
Any loss of containment affecting equipment at the LNG terminal could lead to LNG spillage and ignition.

Risks induced by the shore facilities of the LNG terminal

The shore facilities include the articulated arm for unloading the LNG from the moored tanker, the stationary storage tanks, the regasification system and the pipes for conveying the natural gas to the distribution network.

Any loss of containment affecting equipment at the LNG terminal could lead to LNG spillage and ignition. The hazard analysis for the terminal considers the conservative scenario of LNG spillage from port facilities, in which the four unloading arms of a tanker are simultaneously torn off while transferring LNG. On the basis of its assessments, EDF considered that this scenario presented no risk for the Gravelines nuclear power plant. IRSN’s analysis reached the same conclusion.

Loss of containment in the storage tanks could be caused by an external hazard. In this respect, the hazard levels factored into the design of the LNG terminal (snow, earthquake, winds, extreme heat, lightning) are consistent with all the values considered in the design of the Gravelines nuclear power plant. IRSN’s analysis reached the same conclusion.

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Emergency response

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In its on-site emergency plan, EDF has not described the emergency response procedure in the event of an accident at the LNG terminal that could directly or indirectly affect the nuclear power plant. IRSN recommended that EDF should complete the emergency response measures in place at the Gravelines plant to cover the risk of a cloud of natural gas suddenly reaching the site. Furthermore, IRSN stressed that an accident affecting the LNG terminal (shore facilities or shipping) could also have an impact on APF facilities and lead to a simultaneous loss of containment in all the hydrocarbon tanks on the affected sites. The heat flux that might ensue could prevent any movement of personnel in the vicinity of the Gravelines emergency response building. IRSN therefore recommended that EDF should consider this scenario and adjust its emergency response procedures accordingly.

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Conclusion

IRSN noted that the combined thermal and overpressure effects of the explosion of a drifting gas cloud could, based on the conservative scenarios studied, induce the total loss of electrical power of one or more nuclear reactors, the loss of integrity of the refuelling water storage tank of one or more reactors, and cause damage to equipment essential for the cooling of the spent fuel storage pools. IRSN therefore considered that EDF should implement additional material measures to guard against the risks associated with these scenarios, and that it should complete the emergency response procedures in its on-site emergency plan (to find out more about IRSN’s position: click here).

In decision 2015-DC-0518 of 20 August 2015, ASN required that EDF should implement measures to guard against these risks and complete its emergency response procedures.
Arbitration inside EDF on significant event reports concerning Loire Valley nuclear power plants

One of the key tasks of the independent safety review team is to analyse malfunctions, non-compliances and incidents relating to operating safety at a nuclear power plant. This analysis is carried out independently from the operating safety team responsible for operating the facilities. If the independent safety review team and operating safety team fail to agree on the analysis of the events, the matter is taken to the plant manager for arbitration.

IRSN examined the arbitration decisions taken regarding Loire Valley plants concerning non-compliances in terms of reportability to the French Nuclear Safety Authority (ASN), and observed that the quality of independent safety review team analyses regarding Loire Valley sites had seen a significant improvement between 2011 and 2013. About 20% of the arbitration decisions ruled against the independent safety review, but without adequate substantiation in IRSN’s view.

Responsibilities for safety

The manager of a nuclear power plant (NPP) is responsible for the safety of the plant’s nuclear facilities, on behalf of EDF, the licensee. On the plant manager’s authority, the shift manager, who is part of the operating safety team, is responsible for the operation and safety of the facilities under his command during his shift. As part of these responsibilities, the shift manager performs a real-time assessment of reactor safety and, if necessary, asks for suitable corrective action to be taken regarding possible operating events.

While the shift manager is responsible for the real-time operational control and safety of the facility, a safety engineer carries out independent daily facility safety assessments. The plant manager then compares the shift manager’s and safety engineer’s points of view.

During an inspection concerning good operating practices in 2012, French Nuclear Safety Authority (ASN) observed that the independent safety review team at the Chinon B nuclear power plant did not receive sufficient attention and support from the safety and quality department and from management, particularly in light of the little consideration given by site decision makers to independent safety review team analyses and recommendations. Following this observation, ASN carried out inspections focusing on the consideration given to independent safety review team analyses, the points of view expressed by the shift manager and the safety engineer are compared and recorded in a summary report.

If the shift manager and safety engineer disagree as to whether or not a case of non-compliance with the safety documentation exists, or regarding the processing method to be adopted, including reportability to the French Nuclear Safety Authority (ASN), the plant manager takes an arbitration decision.

Background to the IRSN analysis

The operating safety team consists of the operations department and the various technical departments responsible for maintaining facilities (electricians, mechanics, automatic control specialists, valve specialists, sheet metal workers, etc.).

While the shift manager is responsible for the real-time operational control and safety of the facility, a safety engineer carries out independent daily facility safety assessments. The plant manager then compares the shift manager’s and safety engineer’s points of view.

Shift personnel

The “operations” department of a nuclear power plant is responsible for monitoring and operating a reactor. These tasks are carried out round the clock by three shifts. In each shift, there is a shift manager (in charge of supervising operating activities), a deputy shift manager (in charge of coordinating real-time operating activities), a “reactor” operator (in charge of operating the reactor coolant system and related systems), a “steam” operator (in charge of operating the secondary systems and related systems), and field operators responsible for local tasks required for facility operation.
Significant upgrades

The independent safety review team: members and tasks

The independent safety review team is made up of safety engineers from the whole site. Their work includes:

• carrying out real-time and delayed verifications of facility safety;
• performing analyses, independently of those carried out by the NPP operating teams, focusing on malfunctions observed or non-compliances and incidents concerning facility safety.

Non-compliance processing methods

The order of 7 February 2012, which lays down the general rules applicable to basic nuclear installations, stipulates that each non-compliance detected must be processed by the licensee.

Under these rules, EDF must determine whether or not a non-compliance is a significant safety event according to ASN criteria.

Significant safety events must be reported to ASN: this is what the order refers to as “reportability”.

Events which are not significant safety events are only subject to internal processing by EDF (report issued, internal event analysis, etc.).

Scope of the IRSN analysis

Based on the ASN guide to significant safety event reporting criteria, IRSN examined non-compliances concerned by management arbitration decisions at Loire Valley NPPs between 2011 and 2014. In all, IRSN examined 198 arbitration decisions made by plant managers at the Chinon B, Belleville-sur-Loire, Dampierre-en-Burly and Saint-Laurent-des-Eaux B sites.

About 20% of the arbitration decisions ruled against the independent safety review, but without adequate substantiation in IRSN’s view. The following diagrams (Figure 4.5) sum up the percentages for each Loire Valley site:

- non-compliances that gave rise to a significant safety event report with no need for arbitration (in blue);
- arbitration decisions for or against the independent safety review team, that IRSN finds lacking in adequate substantiation (in yellow);
- arbitration decisions for or against the independent safety review team, that IRSN finds acceptable (in green).

Conclusion of the IRSN analysis

The licensee sent ASN a report providing a summary of independent safety review team activities and its action plan. The independent safety review team and operating safety team agreed that more than half the cases of non-compliances analysed by the Loire Valley nuclear power plants were to be reported to ASN. In cases where the independent safety review team and operating safety team disagreed, IRSN found that the arbitration decisions made by the plant managers concerned were correct in half such cases. In almost all other cases, the plant managers’ decisions supported the operating safety team’s position. IRSN also found that in such cases, the plant managers’ decisions were not adequately substantiated. On this point, EDF informed ASN that measures would soon be implemented to improve plant managers’ arbitration skills.

Example of an arbitration decision examined by IRSN

One arbitration decision examined concerned damage to a temperature sensor cable of a charging pump in the reactor chemical and volume control system (CVCS), while the reactor was in production. This event, caused by an error when opening a cable raceway, led to the reactor coolant system letdown line being shut down seven times. The independent safety review team considered that a significant safety event should be reported according to criterion 3 of the ASN guide given the number of times the letdown line was shut down and restarted without identifying the cause of the malfunction, the failure to apply the required procedure defined in the

Belleville-sur-Loire

57 arbitration decisions analysed
53 significant safety events
with no arbitration

Chinon B

83 arbitration decisions analysed
59 significant safety events
with no arbitration

Figure 4.5: Summary of arbitration decisions for each Loire Valley nuclear site
EDF safety documentation, and the failure to comply with inspection procedures after opening a cable raceway. Based on these arguments, IRSN found it justified to report a significant safety event. However, this was not the position of the plant manager, who considered that the letdown line had always been available, in spite of the inadvertent shutdowns. IRSN did not agree.

In addition, it observed that the quality of the analyses performed by the independent safety review teams of Loire Valley NPPs had improved over the period examined. However, as the managers of the NPPs concerned failed to take sufficient account of these analyses, EDF was unable, in IRSN’s opinion, to fully benefit from this improvement in terms of feedback, for example, by defining corrective action to prevent similar events from occurring at other nuclear power plants.

In order to complete its assessment, IRSN decided to examine arbitration decisions made by the NPP managers in the Rhone Valley in preparation for the ASN inspections.

About 20% of the arbitration decisions ruled against the independent safety review, but without adequate substantiation in IRSN’s view.

- Dampierre-en-Burly: 63% of 64 significant safety events with no arbitration
- Saint-Laurent-des-Eaux B: 64% of 38 significant safety events with no arbitration
- Total for Loire Valley: 52% of 214 significant safety events with no arbitration

<table>
<thead>
<tr>
<th>Plant</th>
<th>Arbitration vs. Independent Safety Review</th>
<th>No Arbitration</th>
<th>Total Analysis</th>
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</thead>
<tbody>
<tr>
<td>Dampierre-en-Burly</td>
<td>63%</td>
<td>20%</td>
<td>17%</td>
</tr>
<tr>
<td>Saint-Laurent-des-Eaux B</td>
<td>64%</td>
<td>22%</td>
<td>14%</td>
</tr>
<tr>
<td>Total for Loire Valley</td>
<td>52%</td>
<td>21%</td>
<td>27%</td>
</tr>
</tbody>
</table>
**Glossary**

**1300 series**
Includes twenty 1300 MW reactors commissioned between 1984 and 1993 (of which eight are “P4”: four at Paluel, two at Saint-Alban and two at Flamanville, and twelve are “P’4”: two at Belleville-sur-Loire, four at Cattenom, two at Golfech, two at Nogent-sur-Seine and two at Penly).

**Accident or incident**
Any unforeseen event during normal operation that is likely to have an impact on public security, safety and health or on the natural environment. An accident has potentially more serious consequences than an incident.

**Activity**
Number of spontaneous disintegrations - or decays - occurring in atomic nuclei per unit time. The unit of activity is the becquerel (Bq).

**APF**
Appontements pétroliers des Flandres (crude oil storage terminal).

**ASN**
Autorité de sûreté nucléaire, The French Nuclear Safety Authority (for civil nuclear activities).

**Automatic reactor trip**
This occurs when the control rods are lowered into the reactor core in response to an automatic control signal.

**Becquerel (Bq)**
Unit of radioactivity, 1 Bq = 1 disintegration per second. The unit is very small and measurements often use a multiple of the Bq such as the megabecquerel (MBq) = 10^6 Bq = 1 million Bq. The Bq replaced the curie (Ci) which represents the activity of 1 gram of radium; 1 Ci = 3.7 x 10^10 disintegrations per second, or 37 billion Bq (or 37 billion disintegrations per second).

**BL**
Electrical building.

**CCWS**
Component Cooling Water System.

**Containment**
Methods or physical structures designed to prevent or control the release and dispersion of radioactive materials.

**Corrective maintenance**
All the operations undertaken to restore failed equipment to correct working order.

**CP0 series**
Includes six 900 MW reactors commissioned between 1977 and 1979 (two at Fessenheim and four at Bugey).

**CPY series**
Includes twenty-eight 900 MW reactors commissioned between 1980 and 1987 (four at Tricastin, six at Gravelines, four at Le Blayais, four at Le Blayais, four at Le Blayais, four at Le Blayais, four at Le Blayais).

**CSS**
Containment Spray System; a safeguard system used in accident situations.

**CVCS**
Chemical and Volume Control System.

**Dose Equivalent Rate**
Absorbed dose rate, weighted in terms of its biological effects by the appropriate quality factor for the type of radiation concerned. It is measured in millisieverts per hour (mSv/h).

**Dose rate**
Radiation intensity (energy absorbed by matter per unit mass and time). It is measured in grays per second (Gy/s).

**Dynamic containment**
Measures taken to control the direction and rate of air flow and ensure that air flows towards areas with the highest contamination level.

**E**

**EDF**
Electricité de France (the French national electric utility).

**EFWS**
Emergency Feedwater System.

**EPR**
European pressurised water reactor (1650 MW reactor; one is currently under construction at Flamanville).

**ESWS**
Essential Service Water System.

**Event relevant to safety**
An event concerning safety that is reported by a licensee but does not meet the criteria defined by the nuclear safety authority.

**EVEREST**
French acronym for “evolving towards entry without a standard suit”: an EDF initiative that is part of its strategy to restore the radiological cleanliness of its facilities.

**Exposure**
The fact of being exposed to ionising radiation (external exposure if the source is located outside the body, internal exposure if the source is located inside the body).

**F**

**FB**
Fuel Building.

**Fission**
The splitting of an atom’s nucleus as a result of bombardment by neutrons. During this reaction, neutrons and ionising radiation are emitted and a great amount of heat is released.

**FPCS**
Fuel Pool Cooling and Purification System.

**Fuel assembly**
Bundle of fuel rods assembled in a metal structure, used in nuclear reactors.

**Fuel cycle**
A fuel cycle is the period of reactor operation between two partial refuelling outages. A cycle covers 12 to 18 months depending on the reactor and type of fuel management.
**GOR**
General Operating Rules. Applicable to nuclear reactor operation. They set out in operational terms the hypotheses and conclusions of safety studies in the safety analysis report, and define the limits and conditions for operating the facility.

**Gray (Gy)**
Unit used to express the quantity of radiation absorbed by the human body in terms of energy deposited by particles or radiation in matter. 1 Gy = 1 joule per kilogram of irradiated matter. It is the unit of absorbed dose. The gray replaced the rad; 1 Gy = 100 rads

**HP turbine**
High-pressure turbine cylinder

**INB**
Basic nuclear installation. PWRs are classed as INBs

**Independent safety review team**
Responsible for analysing malfunctions, non-compliances and incidents relating to operating safety at a nuclear power plant, independently from the operating safety team

**International Nuclear Event Scale (INES)**
Scale intended to facilitate understanding by the media and the general public of the importance for safety of nuclear incidents and accidents. It defines seven severity levels according to the consequences of these events: Levels 1 to 3 are "incidents". Levels 4 to 7 are "accidents", and "deviations" are classified below the scale at Level 0.

**Ionising radiation**
Electromagnetic waves (gamma) or particles (alpha, beta, neutrons) emitted when radionuclides disintegrate and, in the process, produce ions that penetrate matter

**Irradiation**
Deliberate or accidental exposure of an organism, material or body to ionising radiation.

**IRSN**
French Institute for Radiological Protection and Nuclear Safety

**LEL**
Lower Explosive Limit

**Licensee**
Person or organisation having overall responsibility for the operation and safety of a basic nuclear installation. For example, EDF is the licensee for French pressurised water reactors (PWRs)

**LNG**
Liquefied natural gas

**LP turbine**
Low-pressure turbine cylinder

**Maintenance**
All the operations undertaken to maintain or restore equipment in or to a specified condition or to allow it to perform a given function

**Manual reactor trip**
Shutting down the reactor manually

**MFWS**
Main Feedwater System

**MW**
The megawatt is the unit used to measure the electrical energy that a nuclear power plant provides to the electrical grid

**N4 series**
Includes four 1450 MW reactors commissioned between 2000 and 2002 (two at Chooz and two at Civaux)

**NAB**
Nuclear Auxiliary Building

**NPP**
Nuclear Power Plant, which may include several reactors (two, as at Fessenheim and Civaux; four, as at Bugey and Cattenom; or six, as at Gravelines)

**Nuclear fuel**
Fissile material (i.e. capable of undergoing a fission reaction) used in a reactor to develop a nuclear chain reaction

**Off-site emergency plan**
Emergency response plan concerning measures taken by the public authorities to protect the population in the event of a radiological emergency situation outside the site

**On-site emergency plan**
Emergency plan defining the organisation at the plant, as well as the special measures to be set up in the event of an accident situation affecting the facilities.

**Operating experience feedback**
Feedback produced on a given subject over a defined period of time

**Operating safety team**
Responsible for operating nuclear reactors

**OTS**
Operating Technical Specifications. Part of the General Operating Rules (GOR), these define the normal and degraded operating domains of the facility, specifying the permissible variations in controlled parameters and how long it is acceptable for equipment required in the event of incident or accident to remain unavailable.

**Preventive maintenance**
All the operations undertaken on available equipment to avoid any subsequent failure or reduce the probability of failure. These operations are planned in advance and incorporated in maintenance programmes

**Prohibited area (red area)**
Radiologically prohibited area in which the dose equivalent rate is likely to exceed 100 mSv/h
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<td><strong>PWR</strong></td>
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<td><strong>R</strong></td>
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<td><strong>Presence of radioactive materials at the surface of or inside a given medium. For humans, contamination may be external (on the skin) or internal (through inhalation or ingestion)</strong></td>
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<td><strong>RB</strong></td>
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<td><strong>RCA</strong></td>
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<td><strong>RCS</strong></td>
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<td><strong>Reactor fallback</strong></td>
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