REPORT

IRSN'S POSITION ON SAFETY AND RADIATION PROTECTION AT NUCLEAR POWER PLANTS IN FRANCE IN 2013
Cover Photo / General view of the Cruas-Meysse Nuclear Power Plant on the Rhone River.
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ENHANCING NUCLEAR SAFETY IN FRANCE AND ABROAD
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IRSN, the Institute for Radiation Protection and Nuclear Safety, is the French national public expert on research and assessment of nuclear and radiological risks. IRSN was set up by Article 5 of Law 2001-398 of 9 May 2001 and its operation defined by Decree 2002-254 of 22 February 2002, modified on 7 April 2007 to take into account Law 2006-686 of 13 June 2006 on nuclear security and transparency. It is an independent industrial and commercial public establishment under the joint supervision of the ministers of defence, environment, industry, research and health.

It contributes to the implementation of public policies concerning nuclear safety and security, health and environmental protection against ionising radiation. As a research and expert appraisal organisation, it works together with all parties concerned by these policies while preserving its independence of judgement.

Operators are responsible for safety of their facilities. They must demonstrate relevance of technical and organisational solutions applied for this purpose (safety files and release impact studies).


IRSN assesses the files submitted by operators to the different competent authorities. It permanently analyzes plant operating experience feedback. It assesses exposure of man and the environment to radiation and proposes measures to protect the population in the event of an accident. Nuclear safety being largely science based, IRSN’s expertise capability is permanently enhanced through its research activities, usually developed in an international framework.

Local Information Committees (CLI) and the High Committee for Nuclear Transparency (HCTISN) gather the stakeholders concerned by nuclear facilities, and constitute leading bodies for access to information and monitoring of safety and security, health and environmental protection issues.
IRSN’S KEY FIELDS OF COMPETENCE: R&D AND OPERATIONAL ASSESSMENT

IN 2013 1,790 STAFF MEMBERS AT 11 SITES AND A BUDGET OF €307 MILLION

› Nuclear safety and security, Reactors, Fuel cycle, Waste management, Transport, Radioactive sources
› Radiological protection of people (including patients)
› Environmental protection and monitoring
› Nuclear & radiological emergency management and operational intervention capability
› Training and education
› Information management and interaction with stakeholders and the public

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IRsn’s Position on Safety and Radiation Protection at Nuclear Power Plants in France, 2013
IRSN dedicates significant resources to a continuous technical watch on the state of safety of nuclear power plants in France and the purpose of this annual report is to inform stakeholders and the public at large by providing IRSN’s point on view on safety and radiation protection at these facilities. The new format of this report is intended to be more educational and facilitate reading and understanding of concrete safety and radiation protection issues associated with the operation of nuclear power plants.

While this report does not find any notable change in significant events from the previous report in 2012, it confirms the increase in the number of significant radiation protection events observed since 2010. IRSN notes that the overwhelming majority of events that occurred in 2013 had no significant impact on plant safety and no consequences for the health of workers and the public. Although the analysis confirms that EDF’s efforts appear to be successful in handling the main causes of certain types of events (periodic tests, etc.) that had been increasing in recent years, vigilance over organizational and human aspects is still necessary during a period of significant staff turnover.

In the report, IRSN also provides its analysis of several events and anomalies that it considered the most significant in 2013. In particular, they concern deviations that may jeopardize seismic resistance of safety-related equipment, whose number has increased in recent years.

IRSN also provides an overview of several assessments that led or will lead EDF to implement modifications to its reactors to improve the level of safety. In particular, it concerns protecting nuclear plants from internal and external hazards, which is one of the important topics of the third ten-yearly review of 1300 MW reactors.

Finally, IRSN continues to pay particular attention to EDF’s efforts to improve reactor safety as a result of stress tests performed after the disaster at the Fukushima-Daiichi Nuclear Power Plant in Japan on 11 March 2011.

Hoping that this report supplies the information you require, I await your comments as part of our continuous improvement effort.
KEY EVENTS
2013
The 58 pressurised water reactors (PWRs) in the EDF nuclear power plant fleet, which are located on 19 sites, were commissioned between 1977 and 1999. These reactors are grouped into series, each of which combines reactors with the same power capacity and standardised design. New technological developments have been introduced over the course of their design and implementation, thereby explaining the different “types” of reactors in each series. Everyone involved must be constantly attentive to safety and radiation protection, as these can never be taken for granted. They are being continuously improved and must always be a top priority, as the plant operator remains responsible for reactor safety under all circumstances. For IRSN, progress is always the result of careful examination and consideration of national and international operating experience, on the one hand, and new scientific knowledge through research, on the other hand. It must adopt this approach in order to ensure, for example, that equipment ageing is not a potential factor in the safety level of reactors. Every year since 2008, IRSN has published its position on the safety and radiation protection at nuclear power plants (NPPs) in France, emphasising the main changes since previous analyses in order to highlight improvements and areas for improvement. These IRSN reports are intended to inform the public of the risks associated with NPP use in France and to help to answer their concerns regarding nuclear power. As the results of the survey conducted in France by IRSN and the BVA* institute in 2013 on the safety risks associated with radioactivity show, considerable fears regarding a serious NPP accident still remain, although they can be seen to have decreased slightly (8 points down on 2011) in the replies to the question “In your opinion, could a nuclear power plant accident as serious as Fukushima occur in France?” As in previous years, this report stating the opinion of IRSN regarding safety and radiation protection at NPPs in France in 2013 is not intended to be exhaustive. It indicates the elements that IRSN has judged the most significant for the year in question and of which it feels that the public should be informed.

(*) http://www.irsn.fr/FR/IRSN/Publications/barometre/Pages/default.aspx

For the French, the Fukushima and Chernobyl accidents (38% and 23.2%, respectively) are the two most frightening catastrophic events, much more so than other, non-nuclear, catastrophic events (for further information: » IRSN Barometer 2014)
IRSN’s examination of the NPP fleet in 2013 revealed a slight reduction in the number of significant safety-related events compared with 2012.

Variation in the number of safety-related events reported between 2009 and 2013.

Its analysis has revealed a significant drop in the number of events linked with the implementation of periodic inspections and tests; this must be linked with the new operating procedures set up by EDF to improve the drafting of the periodic inspection and testing rules.

However, its analysis of certain types of events, particularly those linked with deviations from reactor operating parameters, has shown that EDF must pay particular attention to maintaining the expertise of its employees in the current context of massive personnel renewal. IRSN is carefully monitoring the increase in the number of short-lived deviations from operating parameters (half of the events of this type are detected and corrected within six minutes).

Lastly, IRSN has observed an increase in the number of safety-related equipment failures in reactors in 2013. This increase is mainly due to compliance defects in the seismic-qualified valves present in many systems. These defects have various causes including equipment corrosion or mechanical fatigue, anchorage non-compliance and loose or missing screws.

Variation in the number of radiation protection events reported between 2009 and 2013.

IRSN’s analysis revealed a slight increase in the number of unauthorised access events. EDF has solved this issue by setting up an action plan for deployment on all sites to improve the handling of radiological risks during work in radiologically-controlled areas, notably by preparing the activities in these areas more thoroughly.
Similarly, the industrial radiography operations mainly performed to check the condition of welds in pipework by means of a radioactive source resulted in an increase in the number of radiation protection events in 2013. Although EDF has set up actions to improve the preparation, coordination, anticipation and monitoring of these operations, IRSN feels that EDF should pay particular attention to the conditions under which they are performed (scheduling, etc.).

IRSN's analysis also showed that the efforts of EDF since 2009 to improve personnel compliance with the radiation protection rules must be maintained and reinforced, as the number of related significant events is increasing. Most of these events involve personnel not wearing a dosimeter when in an RCA.

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

For IRSN, the number of significant safety-related events does not itself serve as a quantifying measure of good operating practices, and variations in this number cannot be directly associated with a variation in safety level, which may be better or worse than before. Significant safety-related events are, however, indicative of issues that need to be analysed and understood with a view to identifying relevant strategies for improving plant safety and radiation protection during operation.

The facilities and their operating methods are not fixed in time. Various modifications due to concerns about safety, radiation protection, availability and cost lead to technical or organisational changes. The French Environment Code requires French plant operators to perform a safety review of their facilities every ten years.

What does a 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 handle any compliance gaps detected.
› A safety review intended to bring the safety level of the oldest reactors up to that of the most recent ones where possible; the safety review may lead EDF to revise its reference documents.
› The deployment of improvements resulting from the safety review.

This review is intended to improve the safety of a facility throughout its service life. The protection of NPPs against internal and external hazards forms an important part of the safety review of PWRs. Although the related risks were taken into account when the NPPs were designed, periodic reviews are essential to incorporate the latest knowledge and operating experience. The third ten-yearly review of the 1300 MW reactors is currently being conducted. In it, EDF has notably examined certain hazards such as tornadoes, wind-generated projectiles, frazil ice, drifting oil slicks, and explosions on the site concerned but outside the nuclear island. The IRSN assessment of the studies conducted by EDF on this subject has highlighted the considerable progress made in its analysis of the risks associated with the hazards.

Frazil ice: a natural weather hazard consisting of the formation of ice crystals in water. It occurs when the water temperature is below its melting point.
such as studying the phenomena and assessing facility vulnerabilities. As a result of these studies, EDF will set up additional facility monitoring and protective measures in the coming years.

**CONTAINMENT OF RADIOACTIVE MATERIALS…**

Within a reactor, radioactive materials are contained by placing successive “barriers” between them and people or the environment; these “barriers” include the fuel rod cladding and the containment building, for example.

**Fuel rod cladding**

The reactor core is composed of fuel assemblies. Each assembly consists of rods in which fuel pellets contained in metal tubes called “cladding” are stacked. This cladding is the first “barrier” intended to limit the release of radioactive materials into the primary coolant system.

During its stay in the reactor, the fuel rod cladding used in the fuel assemblies, which is made of zircaloy 4 (a zirconium-based metal alloy containing tin), corrodes in contact with the water in the reactor coolant system, potentially weakening the cladding in an accident.

As a result, EDF plans to gradually replace the zircaloy 4 with an alloy that is less sensitive to corrosion. Until the cladding is completely replaced in all the reactors concerned (the last ones are scheduled for replacement in 2020), IRSN considers that measures to restrict the operation of the reactors are required.

**The containment building**

The containment building of a PWR is also a “barrier” intended to limit the release of radioactive materials into the environment.

In this respect, it is essential to ensure the leaktightness of the containment buildings (by checking their leak rate) and to monitor ageing. They are monitored while the reactor is in operation, during scheduled reactor outages and ten-yearly safety reviews.

The containments of 1300 MW and 1450 MW reactors include two concrete walls: an inner wall constituting the “internal containment” and an outer wall constituting the “external containment”, separated by an “inter-containment space”.

The containment leak rate, which is checked during the “containment pressure test”, is in reality the “internal containment” leak rate.

As a result, EDF performs a “containment pressure test” every ten years to check the leaktightness of every containment and assess its mechanical behaviour. In the case of some reactors, EDF has carried out re-cladding work to improve the leaktightness of their containments.

**Containment pressure test:** in order to check its leak rate, the containment is “inflated” with air at its design pressure (~ 5 times atmospheric pressure) by around a dozen compressors for three days.

The quantity of air that can escape from the containment is determined by calculating the variation of the air mass contained in the containment for different pressure values.

In view of the results of these tests, IRSN has judged that the containment buildings of the 1300 MW and 1450 MW reactors are currently robust and capable of performing their containment function. It should also be emphasised that, in addition to its monitoring and re-cladding work programmes, EDF is continuing to develop new techniques for improving containment leaktightness, beyond the design pressure.
EDF must periodically shut down its reactors (every 12–18 months) to replace spent fuel assemblies by new assemblies, perform equipment testing, maintenance and servicing operations, check that the equipment and the facility is operating correctly by conducting periodic inspections and tests, etc.

In response to the risks associated with the operations conducted during these outages, EDF has, for some years, significantly developed its organisation responsible for preparing and monitoring the maintenance actions performed during the outages in order to improve their supervision. These outages pose risks to safety and radiation protection, due to the number and variety of operations performed in a very limited time. In this respect, IRSN’s analysis of operating experience has shown that many of the events reported each year by EDF occur during the considerable work performed in scheduled outages. IRSN has assessed the effectiveness of the safety and radiation protection management measures chosen by EDF during the outages. In this context, IRSN has conducted interviews and examined in detail the activities performed during three scheduled reactor shutdowns. It has found that the outage preparation conditions could be degraded (causing additional work in the preventive maintenance programmes, delays built up in the outage work campaigns of previous years, etc.).

In addition, other organisational or working method changes have occurred, and the cumulative effect of these changes has a considerable impact on the teams at the sites.

Lastly, IRSN found that EDF should pay particular attention to the equilibrium between the workload and the competent resources available, in view of its chosen aim extending the service life of its NPPs beyond 40 years, which would undoubtedly increase the maintenance workloads.

Reactor outage safety and radiation protection management at the time of the changes.
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IRsn’s Position on Safety and Radiation Protection at Nuclear Power Plants in France, 2013
FRENCH NUCLEAR POWER PLANT FLEET
1 FRENCH NUCLEAR POWER PLANT FLEET

INTRODUCTION

The nuclear power plants (NPPs) currently in operation in France include a total of 58 pressurized water reactors (PWRs), referred to as "second generation," by comparison to the European Pressurized Water Reactor (EPR) under construction, referred to as "third generation". One specificity of the French NPP fleet is its standardization, with many technically similar reactors located at 19 nuclear facility sites (Figure 1.1). Each site includes two to six PWRs. The nuclear reactor fleet is composed of three series, based on the electrical power supplied:

- The 34 reactors in the 900 MW series include six CP0-series reactors (two at Fessenheim and four at Bugey), and 28 CPY-series reactors (four at Tricastin, six at Gravelines, four at Dampierre, four at Blayais, four at Chinon, four at Cruas and two at Saint Laurent).
- The 20 reactors in the 1300 MW series are subdivided into two trains, the reactors in the P4 train (four at Paluel, two at Saint Alban and two at Flamanville) and the reactors 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 main components of the PWRs operating in France are presented in relatively generic and simplified fashion later in this chapter to provide the basis for understanding this report.

![Fig.1.1 / Status of PWRs in Metropolitan France.](image-url)
Broadly speaking, a nuclear reactor includes two parts (Figure 1.2): the "nuclear island", in which nuclear fission produces heat, and the "conventional island", where that heat is transformed into electric current, which also includes the facility's normal cooling system.

Nuclear island
The nuclear island includes primarily:
› the reactor building (RB), which contains the reactor and the entire reactor coolant system, as well as part of the system ensuring the reactor's operation and safety;
› the fuel building (FB), where the facilities for storing and handling new fuel (waiting to be loaded in the reactor) and irradiated fuel (waiting to be transferred to the reprocessing plant) are located;
› the safeguard auxiliary and electrical equipment building that houses the main safeguard systems in the lower part and the electrical equipment (control room and service rooms, power sources, and reactor instrumentation and control) in the upper part;
› the nuclear auxiliary building (NAB), which houses the auxiliary systems required for normal reactor operation;
› two buildings, separated physically, which each house a diesel generator (emergency electrical power supplies);
› an operations building.

Conventional island
The conventional island equipment supplies electricity to the transmission system from the steam produced in the nuclear island. The conventional island includes:
› the turbine hall, which houses the turbine generator set (it transforms the steam produced in the nuclear island into electricity), and its auxiliaries;
› the pumping station, which cools the installation via the heat sink, watercourse or sea (once-through cooling);
› a cooling tower, if the unit is refrigerated in a closed loop.

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**Fig.1.2** / General presentation of a pressurized water reactor (1300 or 1450 MW) and its main systems.
The reactor core is composed 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 (their height varies based on the power of the reactor), are composed of zirconium alloy (or zircaloy) tubes, also referred to as cladding. Pellets measuring 8.2 mm in diameter, composed of uranium dioxide (UO₂) or a mix of uranium and plutonium oxides ((U, Pu)O₂), which constitute the nuclear fuel, are stacked inside the rods. The fuel is partially renewed during scheduled outages, which occur between every 12 and 18 months.

A carbon steel reactor vessel clad in a stainless steel “skin” sits inside the core, with a head that is removed for refuelling operations (Figure 1.3).

Reactor coolant system and secondary systems (Figure 1.4)

The reactor coolant system removes the heat released in the reactor core by circulating pressurized water, referred to as the primary coolant, in the coolant loops. Each loop, connected to the reactor vessel, is equipped with a pump (reactor coolant pump), which circulates the heated water in contact with the fuel assemblies towards the heat exchangers (steam generators), in which the primary coolant transfers its energy to the secondary systems before returning to the core.

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

The secondary systems convert the thermal energy produced by the core into electricity. The water in the reactor coolant system (radioactive) transmits its heat to the water in the secondary systems (non-radioactive) in the steam generators; the secondary steam is held in the facility’s steam turbine, coupled to the alternator (Figure 1.4).

When the steam leaves the turbine, it passes into a condenser that is cooled by river water, stream water, seawater (once-through cooling) or cooling towers in which the water is cooled on contact with air (closed loop).
Containment

The containment (or reactor building) houses the reactor coolant system, part of the secondary systems, including the steam generators, and certain safety and operations auxiliaries. Broadly speaking, the reactor building is composed of a concrete cylinder, topped with a concrete dome (the roof of the building) forming a resistant shell that meets leak tightness specifications; it ensures that radioactive materials are contained in relation to the external environment and protects the reactor against external hazards. It is designed to resist pressure reached during accidents incorporated into the design (4 to 5 bar absolute) and maintain their seal under these circumstances. The concrete walls rest on a concrete foundation raft, which constitutes the base of the building.

Main auxiliary systems and safeguard systems

The auxiliary systems contribute, both during normal operation at power and during reactor outages and restarts, to performing the safety functions (controlling the neutron reactivity of the core, removing the heat from the reactor coolant system and residual heat from fuel, containing radioactive materials and protecting people and the environment from ionising radiation). Specifically, this includes:

- the chemical and volume control system (CVCS), which:
  - adjusts the boron concentration in the water in the reactor coolant system by supplying demineralized water or borated water based on variations in the reactor core power,
  - adjusts the water level in the reactor coolant system based on temperature variations,
  - maintains the water quality in the reactor coolant system by reducing its content of corrosion products through injection of chemical substances;
- the residual heat removal system (RHRS), which during a reactor outage, removes the residual heat produced by the fuel in the reactor vessel and prevents the water in the reactor coolant system from heating due to the presence of fuel in the core.

The role of the safeguard systems is to control accident situations and limit their consequences. The main safeguard systems are:

- the safety injection system (SIS), which injects borated water into the reactor core in the event of a loss of coolant accident to halt the nuclear reaction and maintain the water inventory in the reactor coolant system;
- the containment spray system (CSS) which, in the event of a significant increase in pressure in the reactor building, reduces that pressure and thus maintains containment integrity. This system is also used to reduce radioactive aerosols that may be released into this containment;
- the steam generator emergency feedwater system (EFWS), which maintains the water level in the steam generator secondary side and thus cools the water in the reactor coolant system in the event that the normal feedwater system is unavailable.

Other systems

The other major reactor safety systems include:

- the component cooling water system (CCWS), which cools some of the reactor-safety critical 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), which removes the residual heat from the fuel assemblies stored in the spent fuel pool;
- the ventilation systems, which play a critical role in containing radioactive materials by depressurizing the rooms and filtering releases;
- fire suppression systems;
- the instrumentation and control system and electrical systems.
OVERALL ASSESSMENT OF SAFETY AND RADIOLOGICAL PROTECTION PERFORMANCE OF THE NUCLEAR POWER PLANT FLEET
The level of reactor operational safety is a determining factor for ensuring continuously optimal radiation protection and safety performance. IRSN’s assessment of radiation protection and safety performance at EDF’s NPPs is based on analysis of a large amount of data obtained through continuous monitoring of reactor operation. Data relative to events and incidents affecting national as well as foreign nuclear facilities form one of the key sources of feedback from which lessons can be learned. For an overall perspective on operating safety and radiation protection, IRSN has developed tools and methods for analysing operating experience, including indicators it has established itself (see IRSN’s 2007 Public Report, page 10*).

These tools contribute to the identification of both general and reactor-specific trends or deviations in radiation protection and safety performance. The two sections that follow present the main lessons to be drawn from IRSN’s overall assessment of radiation protection and safety performance for the year 2013.

OPERATING SAFETY: MAIN TRENDS

In 2013, the number of significant safety-related events fell slightly compared to 2012 (approximately 5% lower), but this number was still higher than in 2011. The efforts made by all EDF teams, in particular in detecting any deviations as quickly as possible, have helped ensure that no events with a serious impact on safety occurred. IRSN has verified that any significant events have entailed immediate and appropriate corrective action and in-depth analysis by the plant operator.

IRSN notes the reduction in the number of significant events related to periodic testing, which demonstrates improvements in the organisation of such tests on the part of the plant operator. Nonetheless, EDF’s objective remains to improve skills, including maintenance and operating skills, particularly with regard to events related to failures in reactor control. IRSN carried out an in-depth study on how to improve skills at NPPs, notably recommending improving support for new staff through mentoring by experienced personnel.

Plant operators of basic nuclear facilities are required to report all safety, radiation protection, environmental and transport-related events to ASN, the French Nuclear Safety Authority, within forty-eight hours of detection. The term “significant safety-related event” is used herein to refer to events with a potentially significant impact on NPP safety. The term “significant radiation protection events” is used herein to refer to ionising radiation exposure events posing a potential threat to the health of exposed workers. “Significant environment-related events” and “transport-related events” are beyond the scope of this report.

Significant events are analysed as part of the general review of operating experience from NPPs.

### Significant safety-related event (SSE) reporting criteria

| SSE 1 | Automatic reactor trip |
| SSE 2 | Activation of safeguard system |
| SSE 3 | Non-compliance with technical operating specifications |
| SSE 4 | Internal or external hazard |
| SSE 5 | Malicious act (or attempt) potentially affecting NPP safety |
| SSE 6 | Transition to fallback state as per technical operating specifications or emergency operating procedures in response to unexpected operating behaviour |
| SSE 7 | Event causing or with the potential to cause multiple failures |
| SSE 8 | Event or fault specific to the main primary or secondary cooling system (or pressure vessel components connected thereto), resulting or potentially resulting in operating conditions not included in the design basis or 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 operating conditions not included in the design basis or existing operating procedures |
| SSE 10 | Other events potentially impacting NPP safety and deemed significant by the operator or by ASN |

Such events are subject to detailed analysis by the plant operator upon detection, leading to the definition and subsequent implementation of appropriate measures to prevent them from reoccurring. EDF must report significant events for reasons of transparency, and also to allow operating experience to be shared among nuclear entities and organisations. Significant events therefore give rise to discussions with EDF and are examined by IRSN for the purpose of identifying valuable lessons at national and even international level.

A slight reduction in the number of significant safety-related events

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

For IRSN, the number of significant safety-related events reported does not itself serve as a quantifying measure of good operating practices, and variations in this number cannot be directly associated with a variation in safety level. Significant safety-related events are, however, indicative of issues that need to be analysed and understood with a view to identifying relevant strategies for improving plant safety and radiation protection during operation.

In 2013, 699 significant safety-related events were reported by EDF: an average of 12 significant safety-related events were thus reported for each NPP in 2013, compared to just over 12.5 in 2012 and around 11 in 2011. There was therefore a slight reduction in the number of significant safety-related events, following on from 2012 in which the number rose compared to 2011. Overall, over the last five years, IRSN has not observed any substantial change in the number of significant safety-related events.

(1) IRSN data and coding. Note also that the numbers of significant safety-related events taken into account for 2009, 2010, 2011 and 2012, shown in Figure 2.1, differ from those given in the Public Report for 2012. This is because there was some confusion regarding certain significant safety-related events and significant radiation protection events which resulted in a slight overestimation of the numbers of significant safety-related events given in the earlier Public Report.
reported. In 2012, EDF deployed a new method for analysing every significant safety-related event in greater depth, with a view to identifying lessons to be learned in terms of identifying the causes and defining related corrective actions. Nonetheless, the efficiency of this new method has yet to be confirmed; it is possible that it could result in a reduction in the number of significant safety-related events in the coming years.

Of the significant safety-related events reported in 2013, 85 were classified as Level 1 events on the INES, however, unlike the preceding two years, no Level 2 events were reported.

The annual number of reactor trips remained stable

The number of reactor trips should not be interpreted as an indicator for which there is any direct link between changes in said indicator and changes in a facility's safety level. Reactor trip is the planned automatic control systems' response to any deviation of a parameter, in order to return the facility to a safe state.

Nonetheless, if this occurs while the reactor is in power generation, automatic shutdown can cause a thermo-hydraulic transient inside the reactor, exerting stress on certain mechanical components and generating large amounts of effluent. In addition, some reactor trips reveal anomalies in the equipment or a lack of familiarity with operating procedures. Since 2007, in a bid to deal with this, EDF has initiated actions enabling it to keep the average number of trips per reactor to just under one a year. IRSN nonetheless believes that EDF must remain particularly vigilant with regard to activities relative to which the risk of reactor trip has been identified.

Increase in the annual number of deviations from authorised operating domain

Authorised operating domain includes various operating states ranging from power operation to shutdown, each associated with a set of technical specifications defining all applicable operating requirements and parameters (pressure, temperature, boron concentration, water level, etc.) and all essential equipment needed to maintain the reactor in a safe state as per safety demonstration criteria. It is strictly forbidden for operators to voluntarily deviate from authorised operating mode without meeting applicable requirements for changing the reactor state. In the event of inadvertent deviation, the operator must take all necessary measures to return the reactor to its initial state (or to achieve a correct state) as early as possible.

Following a reduction for two consecutive years, the number of deviations from authorised operating domain rose by nearly 50% in 2013 compared to 2012. In 2013, 49 significant safety-related events involved unintentional deviations from authorised operating domain (compared to 30 in 2012). This means an average 0.8 significant safety-related events per reactor per year. In the coming years, IRSN will pay particular care to this subject. In addition, it should be noted that the duration of deviations from authorised operating domain remains short. Half of such events are detected and corrected in less than six minutes.
In most cases, deviations from authorised operating domain involve a brief overshoot/undershoot of primary coolant pressure and temperature limits. Also, over 50% of deviations from authorised operating domain are caused by operating errors, during the sensitive phases of manual control of the reactor.

Example of a deviation from authorised operating domain:

On 16 June 2013, reactor 1 at the Chooz B NPP was shut down at the request of the national electricity distribution network operators. During shutdown operations, a deviation from authorised operating domain (drop in reactor coolant temperature to a value below the limit given in the Operating Technical Specifications, or OTS) occurred, lasting for a little under three minutes.

In this example, the deviation from authorised operating domain was due to a combination of factors: faults in how the operating team was organised, in the preparation of the shutdown and in the instruction used by the operating team.

**Example of a fallback initiation required but not performed:**

During the night of 2 to 3 July 2013, an operator in the control room of reactor 3 at the Paluel NPP detected four failures in the instrumentation and control (I&C) system. This operator’s initial diagnosis was erroneous, based on an error in the interpretation of the information provided by a technician present in the rooms housing the defective equipment. The automation expert on call was contacted at his home to discuss the urgency of corrective action. Without requesting any further information, the expert confirmed the initial diagnosis and put off corrective action until the following morning. Because of this error, the operating team did not apply the instructions in the OTS requiring fallback. The next morning, the automation experts detected the error and fallback mode was immediately initiated.

To prevent a repeat of such an event, the plant operator updated the instruction relative to dealing with malfunctions detected by the operating team to provide for exhaustive reporting of all information required for diagnosis and for discussions with automation experts. In addition, automation experts are reminded that they must, in compliance with their diagnostics guide, ask for all the information required to establish their own diagnosis. In addition to these two corrective measures, IRSN notes that EDF has not planned any other corrective action relative to problems in analysing malfunctions. The malfunctions in the I&C system entailed a loss of information crucial to safety in the control room; this information is necessary for deciding whether or not to initiate emergency operating procedures.

**Reduction in the annual number of fallback initiations as stipulated in the OTS but not performed**

The annual number of **fallback initiations** demonstrates the considerable impact of problems that may arise during operating and require the plant operator to shut down a reactor in application of the OTS in order to maintain satisfactory safety levels.

Following a sharp drop in 2011, IRSN observes that the number of fallback initiations performed remained stable between 2012 and 2013. When fallback is required but not performed, this is a noncompliance with the OTS. This may be due to a number of different reasons (erroneous diagnosis of the deviation detected, overshooting the period of time allowed for restoring compliance or conflict between safety and availability).

The annual number of **fallback initiations required but not performed** has fallen: there were four in 2013 compared to ten in 2012 and seven in 2011. The four fallback initiations required but not performed in 2013 were the result of erroneous or delayed diagnosis, entailing noncompliance with the OTS. These errors in diagnosis had various causes: human error in identifying malfunctions, incorrect risk analysis following equipment malfunction or a failure in organisation.

**Fig. 2.4** / Numbers of fallback initiations and fallback initiations required but not performed between 2008 and 2013.
In 2013, malfunctions affecting either the safety injection and chemical and volume control systems (SIS/CVCS), or the heat sink, were reported as significant safety-related events by EDF. The number of such events has increased, by 48% and 44% respectively compared to 2012. These increases are largely due to the high number of events reported by all NPPs related to conformity defects in the locking mechanism of the screws of seismic-qualified valves used in many systems, including the SIS and CVCS and heat sink. These defects are caused by maintenance non-quality. A programme to bring the defective valves into compliance is currently, in 2014, being implemented by EDF.

With regard to steam generator emergency feedwater systems (EFWS) and containment spray systems (CSS), the annual number of malfunctions remained stable between 2012 and 2013. Lastly, the annual number of redundant electrical system failures has fallen significantly (the number of failures more than halved) in 2013 compared to 2012, whereas this number had significantly increased in 2012 compared to 2011. In 2012, it would seem that various generic anomalies led the NPPs to report more significant safety-related events than in previous years. IRSN has found that these anomalies mainly include compliance defects relative to the seismic resistance of certain equipment, together with circuit breaker malfunctions causing the unavailability of the emergency distribution switchboards.

**Increase in the annual number of malfunctions affecting safety-related equipment**

Periodic tests are conducted to regularly check the availability of safety-related equipment. The definition of the periodic test schedule (including the frequency of each individual test), the test conditions and the criteria to be met as set out in the general operating rules are all crucial, as is full and effective compliance with this schedule on the part of plant operators. Prior to 2007, the documents setting out the periodic test rules were drawn up on the initiative of individual plants. New standard operating procedures have been drawn up as part of the operating practices and procedures standardisation project with a view to reducing plant workloads and the risks of inconsistent documentation among plants. Since 2007, a “pilot” plant was tasked with drawing up standardised periodic test operating procedures. These were then validated by a plant other than the pilot plant before being disseminated to all plants operating reactors with the same power rating. This validation process is intended to detect possible errors prior to implementation at all relevant sites. After an initial ‘test’ period of drawing up and implementing new operating procedures (2008-2009), this new organisational structure seems to be bearing fruit.

Since 2010, there has been a reduction in the annual number of significant safety-related events reported due to errors in the preparation of periodic test rules and procedures contained in operating manuals, which may be explained by the improved quality of standardised operating procedures. The annual number of significant safety-related events has decreased from 34 to 12 in the space of four years (2010-2013), i.e. to a third of the number of this type of significant safety-related event.

**Reduction in the number of significant safety-related events due to noncompliance with periodic testing requirements**

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**Figure 2.5** / Annual number of failures involving safety-related equipment between 2011 and 2013.

**Figure 2.6** / Trends in the number of significant safety-related events due to non-compliance with periodic test rules or testing frequencies between 2007 and 2013.
2013 also saw a reduction in the number of significant safety-related events reported due to noncompliance with Periodic Test frequency, for the fourth year running. Given the huge number of periodic tests that must be performed on a reactor (several tens of thousands), with testing frequencies ranging from daily to ten-yearly, the new organisational structure implemented by the plants with regard to periodic tests is proving to be effective, even in the event of delays in the test schedule further to unexpected events.

**Percentage of system alignment errors remains relatively high**

System alignment consists, for example, in opening or closing valves and switching equipment on or off with a view to creating a circuit suitable for performing the functions required for a specific operating state. System alignment may be necessary in order to perform maintenance work, or to test a system to ensure its availability, or to change the reactor state. This is one of the most common operations at an NPP, and is performed tens of thousands of times each year at facilities in France.

Significant safety-related events reported further to errors related to system alignment operations accounted for 5% of significant safety-related events in 2013. The number did fall slightly in 2013 (36 significant safety-related events) compared to 2012 (44 significant safety-related events). Alignment operations nonetheless still account for around forty situations which could have resulted in the unavailability of reactor safety-related systems, or even of safeguard systems, in spite of several local and national action plans. Although some of the valves in question can be operated from the control room, the majority of system alignment tasks must be performed by technicians in the rooms containing the equipment, which makes it difficult for the operating team in the control room to monitor and immediately test these actions. Because of this, the reliability of alignment actions depends, among other things, on their traceability and the quality of implementation. The most common errors include errors in the choice of valve to operate, failure to set the valve correctly and operations that do not comply with the operating documents.

One of the difficulties in system alignment tasks is due to the fact that the representation of the state of the facility available to the operating technicians is fragmented and is not updated in real time. In addition, alignment tasks are more complex than it appears from the mechanical drawings because of the diversity of equipment and technology involved, the location of equipment at different levels of the facility and the rooms to be crossed through. As a result, the “lines of defence” formed by the preparations for a alignment operation and constant communication between the technicians and the operating team play a determining role in ensuring safety for alignment tasks. Furthermore, when the reactor is in shutdown mode, these “lines of defence” are even more at risk when the operators in the control room are less available and when there is any change in the schedule of scheduled alignment tasks (see article on “managing safety and radiation protection during reactor shutdown” in this report).

**Reduction in the number of errors committed within the general framework of maintenance operations, but a rise in the number of poorly mastered technical tasks during maintenance operations**

In 2013, there was a reduction in the number of inappropriate actions committed during one of the stages in a maintenance task (preparing the task, performing the technical task on the equipment, testing, etc.). However, this trend does not apply to the number of significant safety-related events caused by a poorly mastered technical task during maintenance, known as a maintenance non-quality: 146 significant safety-related events in 2013 compared to 107 significant safety-related events in 2012. This pattern has been observed for several years.

The increase in the number of maintenance non-qualities has, in particular, been related to difficulties encountered by EDF in maintaining skills in a context of a high number of employees taking retirement. At a meeting of the Advisory Committee for Nuclear Reactors devoted to feedback from operating experience, this was the subject of a recommendation to EDF made by IRSN, to the effect that EDF should pursue its efforts, in particular by promoting on-the-job mentoring schemes.

In addition, since the majority of maintenance operations are outsourced to subcontractors, EDF undertook an overhaul of its subcontractor monitoring procedures in 2012, with operational implementation organised at the end of 2013 at various facilities. More generally, at the request of the ASN, a more in-depth technical review is underway at IRSN regarding EDF’s control over subcontracted activities. IRSN notes that, of all the significant safety-related events related to maintenance operations, the majority (44%) involved preventive maintenance operations.

![Fig. 2.7 / Annual number of maintenance non-quality events during equipment maintenance or modification activities between 2011 et 2013.](image)
**Example of inappropriate action during a preventive maintenance operation:**

On 13 October 2013, EDF detected that a valve in the Emergency Feedwater System of reactor 1 at the Chinon B NPP failed to respond to a ‘Close’ command. Following investigation, it appeared that an installation error in a new type of seal was made on the valve in question, during an operation carried out as part of the basic preventive maintenance schedule, on 24 July 2013.

The subcontractor’s personnel did not know the specifications relative to installation of this new type of seal. In spite of the fact that there were no installation instructions, which should have been supplied by the manufacturer, the subcontractor’s personnel did not refer to the existing operating procedure because, in the past, they had committed errors when using other operating documentation. Furthermore, the subcontractor’s technical inspector and the EDF supervisor had backed the choice of installation procedure without referring to the instructions or any other source of information even though they were also unfamiliar with the installation specifications for this new type of seal. To prevent such an event reoccurring, EDF has undertaken to improve training for EDF personnel on the installation of this type of seal and the use of operating instructions. Nonetheless, EDF is not dealing with the effectiveness of its subcontractor monitoring procedures.

Corrective maintenance operations are carried out in two stages: diagnosis followed by corrective action. The significant safety-related events related to these activities are mainly caused by EDF automation experts. The latter are specialised personnel who are usually called upon by the operating teams to troubleshoot any anomalies in facility operating and the tasks they perform are usually complex. Automation experts perform inappropriate actions both during the diagnosis stage and when carrying out the actual corrective action, thereby causing unavailability of safety-related systems. In 2013, the significant safety-related events caused by corrective activities were mainly due to inappropriate preparation and risk analysis related to operations. Such errors are largely due to ineffective coordination between the automation experts and the operating teams and a lack of knowledge of how automated systems work. Insofar as an operating anomaly impacts facility availability or safety, troubleshooting activities are often implicitly related to the constraints of an emergency situation. Knowledge of troubleshooting activities and of the facilities, together with the quality of documentation explaining related safety issues, are essential especially given that the personnel involved are under pressure to act swiftly.

**Example of erroneous action during diagnosis of a failure that led to switching to emergency operating mode:**

On 4 November 2013, to diagnose the cause of a failure in an electrical board in reactor 3 at the Paluel NPP, the automation experts disconnected an electronic module used to establish the mean temperature of the reactor coolant. This action, deemed by the automation experts to be without any specific risk, was not discussed with the operating team. Disconnecting the module caused the mean temperature, read in the control room, of the reactor coolant to rise, triggering an alarm. This is a criterion for switching to emergency operating procedures, so the operating team, unaware of the maintenance activity taking place, applied the relevant instruction. When they heard the general information message issued by the operating team, the automation experts realised the consequences of their action and reconnected the module. This event reveals an obvious fault in the automation experts’ preparation of their operation, as well as shortfalls in their knowledge of the facility which meant that they did not fully understand the related risks.

**Effective detection systems**

The speed with which an anomaly is detected is crucial in order to correct the malfunction as quickly as possible and limit, as far as possible, the actual and potential consequences of the anomaly. This is why there are several systems for detecting anomalies: alarms, periodic tests, inspection rounds and monitoring, etc. It is in fact essential to have a large number of different monitoring and detection methods in order to ensure the facility’s compliance with requirements. The operating teams play an essential role in detecting any anomaly that may occur. These teams are in charge of dealing with alarms and of day-to-day facility monitoring.

Figure 2.8 shows that, in 2013, as in the preceding years, the majority of anomalies (nearly a third of significant safety-related events reported annually)
were detected thanks to the activation of alarms. In addition to technical and organisational methods specifically dedicated to detecting anomalies (alarms, periodic tests, rounds, etc.), the term “vigilance”, in Figure 2.8, refers to the personnel’s capacities to constantly draw on their knowledge of the facility, their activities and the related risks. The ability to remain vigilant for any anomaly is an effective method for detecting every kind of anomaly. Such “vigilance” is manifested by a questioning attitude and information-sharing between personnel when an anomaly is detected and results in an objective view of the activity to be performed and awareness of its potential direct and indirect consequences relative to safety.

**Example of an anomaly detected thanks to team vigilance:**

On 10 April 2013, spent fuel assemblies were removed from reactor 2 at the Nogent-sur-Seine NPP. The fuel assemblies were, initially, loaded into a leaktight package filled with gas. During this operation, EDF General Services personnel observed the formation of ice on the cylinder containing the gas used. This unusual phenomenon led these operators to question what was happening and then call an expert at the Risk Prevention Section to analyse the situation. The expert found that the gas cylinder used was a cylinder of argon instead of helium, i.e. not in compliance with the reference requirements. In this example, it was thanks to the vigilance of the personnel that the injection of an inappropriate gas was detected. The interpretation of an abnormal physical phenomenon, in this case the formation of ice on the cylinder, led the EDF General Services personnel to inquire about the activity in progress which could otherwise have resulted in compromising the cooling and containment of the fuel.

**General: breakdown of significant radiation protection events reported**

The regulations relative to protecting workers from the hazards of ionising radiation require nuclear facility operators to notify ASN of any significant radiation protection events. Such events are reported as per the reporting criteria defined by ASN (see table below).

In 2013, the events reported mainly involved access conditions, with 40% of significant radiation protection events reported as per Criterion 7, and 37% of significant radiation protection events reported as per Criterion 10. Events meeting Criterion 3 and Criterion 6 each accounted for nearly 6% of all notifications. These figures are similar to those for 2012.

**The 10 significant radiation protection event (SRPE) reporting criteria**

| SRPE 1 | Noncompliance with regulatory annual individual dose limit requirements, or unexpected situation with potential to cause such non compliance under reasonably representative conditions, regardless of exposure type (including bodily exposure). |
| SRPE 2 | Unexpected situation leading to a 25% overshoot of a regulatory annual individual dose limit value, regardless of exposure type (including bodily exposure). |
| SRPE 3 | Any significant noncompliance with radiological cleanliness standards, e.g. presence of radiation sources exceeding 1 MBq outside radiation-controlled areas, or detection of radiation-contaminated clothing (> 10 kBq) by site entrance/exit radiation monitors or in the course of whole-body radiometric examinations. |
| SRPE 4 | Any activity (operation, task, modification, inspection, etc.) posing a significant radiological risk, conducted without radiation protection assessment (justification, optimisation, mitigation) or without exhaustive consideration of such assessment. |
| SRPE 5 | A malicious act or attempted malicious act liable to impact the protection from ionising radiation of workers or members of the public. |
| SRPE 6 | An abnormal situation affecting a sealed or unsealed source with an activity level higher than the exemption limits. |
| SRPE 7 | Signalling fault or noncompliance with technical conditions for access to restricted or prohibited areas (orange/red areas or gamma radiography inspection areas). |
| SRPE 8 | Non-compensated malfunction of collective radiation monitoring systems. |
| SRPE 9 | Failure to meet inspection deadlines for radiation monitoring equipment, by more than a month in case of fixed collective radiation monitoring systems (1-month inspection frequency as per applicable regulations) and more than three months for other types of equipment (12 to 18-month inspection frequency as per applicable regulations and general operating rules). |
| SRPE 10 | Any other non compliance deemed significant by ASN or operator. |

**RADIATION PROTECTION: MAIN TRENDS**

Compared to 2012, 2013 saw an increase in the annual number of significant radiation protection events involving workers reported for EDF NPPs. IRSN’s analysis indicates an increase in the number of significant events related to anomalies in performing gamma radiography inspections, in radioactive source management and in contamination outside controlled areas.

EDF’s efforts to manage dosimetry for personnel must be pursued and improved upon since the number of events related to worker dosimetry is slightly higher than in 2012. IRSN’s analysis has also shown that further improvements must be made to practices regarding removing clothing after performing operations, and this applies at all plants.
With regard to other types, the numbers of significant radiation protection events reported were generally stable, at around 3% (while the remaining 8% involved Criteria 1, 2, 4 and 9). EDF analysed the circumstances and the causes of each of these events, together with their actual and potential radiological consequences. EDF then identified and implemented corrective measures to prevent such events from reoccurring. These analyses were submitted to ASN and IRSN. The information thus furnished has been used by IRSN to analyse trends for all NPPs in operation in France. To support its analysis of the trends, IRSN examined the significant radiation protection events and categorised them according to type, as illustrated in the graph below.

In particular, IRSN chose to examine the causes and corrective measures related to those types of event that occurred in the highest numbers or that resulted in the most serious actual or potential consequences.

In 2013, the number of significant radiation protection events reported by EDF increased (119 significant radiation protection events in 2013, compared to 112 in 2012 and 97 in 2011). This increase can be explained by three factors:

- errors committed during gamma radiography inspections accounted for a rise from 13 significant radiation protection events to 17;
- errors in radioactive source management accounted for a rise from 6 significant radiation protection events to 8;
- contamination outside controlled areas accounted for a rise from 2 significant radiation protection events to 6.

IRSN finds that the annual number of significant radiation protection events rated as level 1 or level 2 under the “INES for radiation protection events” has decreased from nine events in 2012 to three events in 2013 (two level 1 events and one level 2 event). In 2013, the level 2 significant radiation protection-related event reported occurred at the Blayais site; during this event, the body contamination of a worker employed by a subcontractor came...
close to the regulatory equivalent dose limit (of 500 mSv) for skin exposure. The worker was cleaning a heat exchanger by brushing, an operation which carries a high risk of contamination. For this operation, he was equipped with the appropriate personal protective equipment, i.e. an air-purifying respirator (cartridge gas mask) and self-contained breathing apparatus. Having completed his task, he removed his clothing alone in the airlock designed for this purpose, but he was unable to immediately monitor for any potential body contamination because there was no contamination metre in proximity to the airlock. On leaving the reactor building, he checked using the “hand/foot monitor”, which did not detect any contamination. It was only when he left the controlled area that the small object monitor detected contamination of his protective glasses and helmet. The radiation monitor at the exit from the controlled area then confirmed contamination to the worker’s neck. He was treated by the site medical team. It is estimated that the exposure period lasted 50 minutes (from the time he removed his clothing to the time he left the controlled area), implying a significant skin dose. According to EDF, the main cause of this event was the lack of a contamination metre for monitoring exposure, which would have reduced the exposure level. IRSN considers that, in addition to this, the stage of removing clothing was inadequately performed, and could have caused the contamination to migrate from the personal protective equipment to the worker’s neck. Following this event, workers have been reminded of the measures that must be complied with regarding monitoring checks on leaving worksites. EDF’s logistics department has also undertaken to define additional surveillance measures for activities entailing serious radiological hazards.

**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 risk of exposure, over a period of 12 consecutive months:

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

**Slight increase in the number of events occurring upon entering a controlled zone**

In accordance with applicable regulations, NPP radiation protection personnel are responsible for implementing a radiological zoning scheme. This radiological zoning scheme implies marking out the different areas, based on measured dose equivalent rates, commonly known as “dose rates”, set using a radiation metre, and marked using a three-colour classification system. The main cause of the events reported involving unauthorised access to controlled areas was
non-observance of the technical conditions for access to “Orange Areas” (radiation-controlled areas with dose equivalent rates possibly exceeding 2 mSv/hr). These events may have resulted in worker exposure, causing, depending on the case, exposure in excess of the annual dose limit, or in workers entering an Orange Area when they are not authorised to do so.

Events related to unauthorised access to controlled areas include events that reveal shortfalls in the procedures relative to access to Orange Areas and to Red Areas (red area with dose equivalent rates possibly exceeding 100 mSv/hr).

Given the implied risks, access to a Red Zone is subject to specific provisions (locked area, authorisation approved by the facility manager, etc.). Noncompliance with these provisions can have serious consequences. For several years, the annual number of such events has been less than 5, and fell between 2012 and 2013. Noncompliance with the conditions applying to Orange Area access accounts for approximately 40% of the significant radiation protection events reported. This type of deviation can result in exposing workers to unforeseen doses, liable to be in excess of the applicable annual dose limit, or in allowing workers to enter an Orange Area when they are not authorised to do so. In accordance with Article D.4154-1 of the French Labour Code, personnel on fixed-term and temporary contracts, hereinafter called “non-permanent workers”, are not authorised to work in Orange Areas.

According to EDF:

- 61% of unauthorised Orange Area access events are detected using dosimeter alarms;
- 44% of Level 7 significant radiation protection events involve non-permanent workers;
- Approximately 30% of Level 7 significant radiation protection events involve access to Orange Areas as a result of inadequate planning of the activity to be performed.

To reduce the number of events related to unauthorised access to Orange or Red Areas, EDF is pursuing its actions in the following areas:

- preventing measures at ‘radiation hotspots’;
- setting lower dosimeter thresholds;
- raising the audio volume of dosimeter alarms;
- improving planning for those activities mainly effected by such events and liable to entail exposure of non-permanent workers;
- analysing the organisational and human factors involved in such events.

IRSN believes that these actions are satisfactory in theory, but that relevant action plans should be implemented at the sites. At a meeting of the Advisory Committee for Nuclear Reactors devoted to feedback from operating experience, EDF undertook to follow this advice.

Increase in the annual number of events related to gamma radiography inspections

EDF carries out gamma radiography inspections during maintenance operations mostly performed during reactor outage. Given the workload involved in a gamma radiography inspection, the high activity levels of the radioactive sources used and the often difficult work conditions, the level of radiation protection-related hazard involved in gamma radiography is high. In theory, these inspections are preferably performed at night, when fewer people are present. In addition, to manage the risks, the operator takes steps to avoid any other tasks being performed at the same time as and in proximity to gamma radiography inspections. Last, visible and continuous marking prohibiting access to the exclusion zone is set up and forms a line of defence.

The number of events related to gamma radiography inspections reported in 2013 has increased compared to 2012, as was also the case for 2012 compared to 2011; around fifteen cases are reported each year. Most of the events related to gamma radiography inspections occurred in the turbine hall (which is not in a controlled area). Analysis of the reported events shows that the primary causes are unintentional access due to inadequate marking or intentional access; the latter account for just over half of the significant radiation protection events related to gamma radiography inspections.

The measures implemented by EDF since 2009, combined with ‘conventional’ radiation protection regulations, do not, therefore, appear to be effective in light of the steady increase in the annual number of such events. (7 significant radiation protection events were related to gamma radiography inspections in 2009). EDF has carried out analysis of the organisational and human causes of the reported events. This reveals that, in around 80% of such significant radiation protection events, the contextual factors are worker fatigue, performance of an uncommon activity, stress due to tight schedules, heavy workload and working within unfamiliar teams of specialists. These contextual factors must not be ignored since highly experienced personnel were involved in 80% of significant radiation protection events caused by human factors. EDF has deployed measures, primarily aimed at improving planning, coordination, preparedness and supervision of gamma radiography inspections. In addition, EDF has specifically identified the causes and the personnel involved at the different stages of planning and performing gamma radiography inspections. IRSN nonetheless believes that certain causes, such as changes in worksite conditions, have not been dealt with in this
Slight increase in annual dose measurements for personnel

There was a significant increase in annual dose measurements for personnel in 2012 compared to 2011 and this remained steady between 2012 and 2013. Some 63% of this type of significant radiation protection-related event involved the failure to wear passive or operational dosimeters. It should be noted that, in 2013, 50% of cases where personnel neglected to wear passive or operational dosimeters occurred during emergency operations to assist a victim of an occupational accident; the other 50% involved forgetfulness in the locker room. Monitoring individual dosimetry for personnel is designed to measure whole-body received doses. In particular, this makes it possible to verify compliance with the dose limits defined by current regulations.

The individual dose for “gamma” radiation characterises external exposure of EDF and subcontractor personnel to ionising gamma radiation. This is monitored using passive and operational dosimeters. Furthermore, IRSN believes that the trend in the “number of workers who received an effective dose greater than 16 mSv over a period of 12 consecutive months” is a reliable indicator given the increase in the number of maintenance activities and the overall exposure limit of 20 mSv/year for workers. EDF lowered the threshold for monitoring this indicator by determining the current number of workers who received an individual dose higher than 14 mSv over a period of 12 consecutive months. This has led to lowering the alert threshold on electronic dosimeters to 14 mSv, thus improving monitoring of specialised occupational activities entailing the highest risk of exposure.

Under the regulations, all workers accessing radiation-controlled areas are required to wear passive and operational dosimeters. EDF has implemented a measure encouraging workers to check in front of a mirror that they have not forgotten anything prior to entering a controlled area (often referred to as the “Have you got everything?” measure): helmet, badge, dosimeters, etc. It is possible that, out of habit, workers forget this method of self-checking. It is nonetheless the case that, even with this “line of defence”, the number of cases where an (“operational” or “passive”) dosimeter has been forgotten increased in 2012 compared to 2011 and again in 2013 compared to 2012.

For certain types of operation, EDF requires workers to wear a passive dosimeter measuring the dose to the extremities (wristband dosimeters for the wrists and ring dosimeters for the fingers). As a result, the number of passive wristband or ring dosimeters made available by EDF has steadily increased at all NPPs in France.

Fig. 2.13 / Mirror for self-checking workers have all the equipment required to enter controlled area.

Fig. 2.14 / Ring dosimeter.

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 person. The dosimeters worn by workers are designed to show the dose to the whole body, either later, after reading at an approved laboratory (“passive dosimetry”) or in real time (“operational dosimetry”). Operational dosimeters used by EDF have an audio and visual alarm that alerts workers if they are in a field of radiation that exceeds certain thresholds pre-set 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 deferred whole body radiation measurements (direct measurement of internal contamination) and radiotoxicological tests.
In the last few years, including 2013, only 10% of all the wristband and ring dosimeters made available have detected an equivalent dose higher than 0.1 mSv. In order to further reduce the number of cases of radiation exposure, EDF Central Services have sent out a reminder to the plants of the recommendations that must be applied to prevent any risk of exposure to the extremities, the main hazardous work activities and what to do in the event that a highly radioactive object is detected.

Annual number of other types of event remains stable

Errors in “risk analysis” cause around 15% of significant radiation protection events reported each year. Notifications of significant radiation protection events related to “errors in radioactive source management” involve sources, usually low-level activity sources, most of which are contained in fire detectors and in radiation protection measuring devices.

There was a slight increase in the number of such events in 2013 compared to 2012, from 6 to 8 events, which remains low; nonetheless, IRSN would emphasise that the potential consequences of such an event can be serious.

Significant radiation protection events involving “training” solely concern failures to maintain up-to-date skills with regard to radiation protection training; the number of this type of event has remained very low over the last three years (fewer than 5 events per year).
OVERALL ASSESSMENT OF SAFETY AND RADIOLOGICAL PROTECTION PERFORMANCE OF THE NUCLEAR POWER PLANT FLEET
3
EVENTS, INCIDENTS AND ANOMALIES
3 EVENTS, INCIDENTS AND ANOMALIES

Understanding how and why an event occurs requires, first, thorough knowledge of the facts and context. This is required before analysing root causes, estimating actual and potential impact on the plant’s safety and, where relevant, the populations and the environment, and evaluating whether the corrective actions taken are appropriate to prevent a recurrence. These analyses are an essential IRSN activity in connection with monitoring power plant operations.

The events may have varying origins, ranging from human or organisational failures to equipment or design failures. The causes may also be external to the power plant, such as external hazards.

One characteristic of EDF’s fleet of pressurized water reactors is its standardisation. It consists of three reactor series. Each series includes similar reactors of the same power (900 MW, 1300 MW and 1450 MW). Beyond the economic aspect, standardisation offers operating advantages (including uniform operating reference standards, optimised maintenance and shared feedback).

Nonetheless, standardisation can also be a disadvantage if the plant operator discovers a failure or error that is liable to affect several or, even, all reactors. This then involves a “generic” anomaly. IRSN pays particular attention to detecting such anomalies and the measures EDF takes to address them. Some anomalies may require complex measures and take several years to be corrected.
CONTAMINATION OF COMPRESSED AIR PRODUCTION AND DISTRIBUTION SYSTEMS AT CRUAS 2

Several maintenance anomalies involving the compressed air production system equipment of reactor 2 at the Cruas power plant spread alumina powder to the air-operated valves that control the turbine steam supply and caused the valves to malfunction. EDF quickly brought the faulty equipment back into compliance. It also replaced the affected valve controller, referred to as a positioner, and the clogged filters in the compressed air production and distribution systems.

A specific in-service programme to monitor the air-operated valves in the reactor’s main systems was established and implemented until the next scheduled outage, during which these systems were checked and cleaned many times. IRSN reviewed EDF’s corrective and preventive actions taken after this incident.

Compressed air production and distribution systems

Two compressed air production and distribution systems produce the air needed for air-operated valves to function. The compressed air supplied must be of good quality; that is, dry and without particles. The first system produces compressed air using two redundant compressors. The air must be dried after compression. This task is performed by two pieces of equipment called “dryers”, in which the water molecules are captured by alumina pellets. The compressed air is filtered upstream and downstream of the dryers and in a buffer tank before it is introduced into the second system. That system distributes the compressed air to the air-operated valves via piping that is generally equipped with filters and storage capacity.

The main systems contributing to the functioning of a reactor or involved in a reactor outage or during incident or accident conditions are equipped with air-operated valves. The air-operated valves used include isolation valves, which are in open or closed position (isolating function), and “control” valves, which have a variable opening (regulating function). Their proper operation thus contributes to facility safety.

Origin and consequences of the event

In January 2013, while reactor 2 at the Cruas nuclear power plant was in production, two air-operated valves controlling the turbine generator’s steam supply and a compressor in the reactor’s compressed air production system malfunctioned, jamming in the open position. EDF quickly found that the presence of alumina powder had caused these malfunctions. It replaced the faulty equipment and brought the dryer, from which the alumina had escaped, back into compliance.

EDF conducted an in-depth analysis and determined that the alumina powder had entered the compressed air production and distribution systems during preventive maintenance performed on one of the dryers and on the filters located upstream and downstream of that dryer. The proper operating procedure for replacing the alumina in the dryer was not followed during these operations. This procedure requires a makeup 48 hours after placing alumina pellets in the dryers because the product shrinks. The makeup was not performed in this case, causing the alumina pellets to deteriorate and turn into powder and creating enough space for them to move as air flowed across the dryer. The friction produced as the pellets rubbed against one another eroded them. In addition, the filters protecting the dryer were...
not tightened adequately. The filter located upstream of the dryer was equipped with a cartridge smaller than normally provided. The alumina powder drawn by the suction of the compressed air related to the operation of the air-operated valves then spread throughout the compressed air distribution system via these two non-leaktight filters. EDF estimated that 20 litres of alumina powder thus entered the compressed air and distribution system. The granulometry of this powder was between 1 and 4 microns. If the compressed air systems are contaminated, the alumina powder accumulates in the inlet nozzle and the internal components of the valve positioner (Figure 3.2), obstructing the compressed air supply ducts and causing the valves to malfunction. The dimension of the nozzle orifice depends on the type of valve (their diameter is between 0.5 and 1.75 mm). Orifices of a smaller diameter are particularly sensitive to clogging from alumina powder.

![Image of compressed air intake and positioner](image)

Fig.3.2/Alumina powder clogs the compressed air intake of the positioner of an air-operated control valve.

These maintenance quality failures are due to a lack of rigour and inspections by workers as procedures are carried out. The analysis of the risks associated with replacing the alumina in the dryer was not taken into account fully in preparing for and carrying out the tasks. During its in-depth analysis of the event, EDF noted that the alumina moved along the pathways corresponding to the highest consumption of compressed air; that is, primarily in the supply lines of the control valves in continuous operation as, for example, the control of the steam generator feedwater supply or the turbine generator steam supply. The malfunction of the compressor of the compressed air production system during its no-load operation, which was revealed during a periodic test, had no actual impact on the facility. The malfunctions of the two valves controlling the steam flow rate supplying the turbine of the turbine generator had no impact on reactor safety. It did have consequences for the operation of the power plant because those malfunctions prevented the turbine generator from responding to changes in the power grid load. However, most of the air-operated valves were not protected by filters fine enough to capture the alumina particles. The contamination could thus have produced failures - possibly simultaneous - in other air-operated valves in safety-related reactor systems, particularly for the following functions:

- water spray in the pressurizer (this function reduces the pressure of the reactor coolant system if it experiences an abnormal increase);
- feedwater supply (normal and emergency) to the steam generators (this cools the reactor coolant system by producing steam);
- reactor cooling via the main steam relief train (this allows for removing the reactor's thermal power when it is in the outage phase and the secondary system is isolated).

Such malfunctions may lead to a reactor trip as a result of reaching the operating parameter limits or an incident condition, for example, if the residual heat removal system fails. They may also disrupt its safe outage if the steam generators’ emergency feedwater system fails.

Contamination of the compressed air systems at Cruas 2 thus led EDF to conduct an in-depth risk analysis and implement a significant action and inspection plan so that the reactor could remain in production until the scheduled outage following the incident.

**IRSN position**

Given the risks posed by this incident for the availability of systems required for reactor operation or outage and implementation of procedures in incident or accident conditions, IRSN conducted an analysis of this event. This analysis showed that EDF’s initial actions, which involved bringing the equipment contaminated by alumina powder back into conformity (cleaning the air compressor, replacing the alumina in the dryer and its upstream and downstream filters and replacing the positioner of the two faulty valves) and replacing the filters contaminated with alumina were not sufficient. IRSN concluded that supplemental actions were required, including cleaning all the piping in the compressed air production and distribution systems using air blasts and checking all the filters in the systems feeding the pneumatic actuators, which must operate for the reactor to function safely. In addition, IRSN determined that the air-operated valves that were not protected by a filter blocking the alumina particles should be checked and their positioner replaced if alumina is present. IRSN determined that inspections and possible replacement should be carried out as soon as possible for systems involved in a reactor outage, such as the steam generator emergency feedwater system and the residual heat removal system. It found that the inspections and replacements to be
performed on the air-operated valves of the other reactor safety-related systems should have been conducted by the time of the scheduled reactor outage following the incident.

**EDF’s corrective actions**

EDF quickly performed inspections on the filters of the compressed air distribution system. Filters that were clogged by alumina powder were replaced. Alumina was also present in the system’s storage tanks, which was purged. These interventions were performed with the reactor in operation and in accordance with the provisions of the **operating technical specifications**.

The operating technical specifications of a reactor define the configuration of the systems required in each of its operating modes. They specify the length of authorised outages for the systems’ main equipment and the appropriate operating procedure in the event that the length of this authorised outage is exceeded.

Given the potential risk that the control valves, such as those of the turbine generator’s steam supply or the condensate extraction system, could be affected by alumina powder again, EDF changed the reactor’s operating mode, excluding load following and **house load operation tests** until the reactor’s next scheduled outage. Those operations require the ability to control turbine generator steam supply.

**House load operation** refers to a situation in which the power plant is operated at reduced power. The electricity that the turbine generator produces is used exclusively to meet the reactor-specific needs (the power plant is then disconnected from the electricity transmission grid).

EDF conducted an analysis of the risks posed by the unavailability of reactor safety-related valves to ensure that the reactor could remain in production. This analysis primarily addressed reactor safety-related systems that come into play during a reactor outage or in an incident or accident condition. EDF evaluated the sensitivity of the valve positioners to clogging by alumina powder based on their technology. It estimated the impact of a valve malfunction resulting in loss of the control valve regulation function on the operation of these systems and on reactor safety by taking into account valve **fail-safe position upon loss of air**.

EDF found that the air-operated valves belonging to the following systems were not sensitive to clogging by alumina powder:

- residual heat removal system;
- chemical and volume control system;
- containment spray system in the pressurizer;
- supply for the steam generator emergency feedwater pump and the emergency turbogenerator set (which ensures injection to the reactor coolant pump seals in the event of a total loss of power).

In addition, EDF inspections determined that certain valves susceptible to clogging, such as the steam generator emergency feedwater system valves, were not contaminated by alumina. It found that the isolation valves that did not operate after the compressed air systems were contaminated were probably clean and, as a result, available when first actuated, having noted that the reactor should have been able to be shut down and action then taken to repair the faulty valves.

In its conclusion, EDF found that the systems required for the reactor outage were capable of performing their function and that the reactor shutdown was not necessary. An in-service programme to monitor the air-operated valves of the main reactor safety-related systems was set up. It included regular inspections to ensure the proper functioning of the valves during periodic tests, particularly of those most susceptible to clogging, and regular verification of the availability of the compressed air systems (dryers and filters).

Cruas 2 was thus authorized to continue its operating cycle until the scheduled outage eight months later. During that outage, a significant programme of actions and inspections was implemented and most of the piping and air-operated valves of the reactor safety-related systems or those necessary for the reactor’s operation were cleaned. No malfunctions occurred in the valves in the reactor safety-related systems during the period leading up to the scheduled outage. However, during the cleaning performed during the reactor outage, alumina powder was found in many pipes, valves, and the filters protecting these valves, primarily in the steam control systems feeding the turbine generator set, the condensate extraction system and the steam generators’ feedwater return system, coming from the condensor.

In response to the causes of the event, EDF implemented a set of targeted actions at the Cruas power plant intended to prevent the recurrence of failures and inappropriate maintenance on the dryers and the filters of the compressed air production system. It revised the operating procedures and the risk analyses associated with those interventions to ensure that all workers carry out the procedures properly. These actions fall within the more
general framework of the action plan implemented in 2013 at Cruas 2 to ensure greater reliability of the quality of the work and reduce maintenance quality failures by the strengthening agents’ training. Progress is expected, particularly with regard to compliance with safety requirements and mastery of the interventions. IRSN will monitor the effectiveness of these actions over the next operating cycles at Cruas 2.

**COMPROMISE OF EARTHQUAKE RESISTANCE OF CERTAIN EQUIPMENT**

The safety functions of a nuclear power plant must be ensured in the event of earthquake. Earthquake resistance requirements are thus associated with the equipment that handles or is involved in these functions, starting with the design phase. Since 2011, EDF has discovered deviations that could compromise the earthquake resistance of certain reactor safety-related equipment. Given the possible consequences of these deviations, IRSN recommended that EDF deploy its maintenance programme sooner than scheduled and expand the inspections planned under this programme to other equipment.

**What compromises the earthquake resistance of certain equipment?**

The safety functions of a nuclear power plant must be ensured in the event of earthquake. Earthquake resistance requirements are thus associated with the equipment that handles or is involved in these functions, starting with the design phase. However, certain phenomena could affect its earthquake resistance. They include primarily:

- chemical or electrochemical phenomena, such as corrosion, which may weaken metal structures;
- mechanical fatigue of the structures due to repeated stress or deformation; by modifying the local mechanical properties of a material, fatigue may result in cracks and, possibly, the failure of the structure in an earthquake, for example (increased loads).

In addition, non-conformity of materials or structures may affect their earthquake resistance. This involves non-conformities of the anchoring of the materials (lack of anchors, inadequate number of anchor points and inadequate anchor design). Human errors committed during installation (at construction) or re-installation (as part of maintenance work or physical modifications) may also result in non-conformities.

Compromised earthquake resistance of equipment that should be resistant constitutes a deviation from the safety reference framework. Feedback shows that such deviations affect primarily piping, supports, anchors, tanks, valves and sensors.
What inspections are performed on equipment that must be resistant in the event of an earthquake?

To ensure that the equipment concerned remains earthquake-resistant over time, EDF carries out a basic preventive maintenance programme.

Preventive maintenance comprises all actions performed on equipment to prevent or reduce the probability of subsequent malfunction. These actions are planned in advance and integrated into maintenance plans.

These inspections may involve:
› visual inspection of the visible portion of the equipment;
› closed-circuit television inspection (using small cameras) in tight spaces;
› ultrasound inspections to detect any problems inside the material.

Ultrasound inspection is a non-destructive inspection method that uses ultrasound emitted by a probe placed on the surface of the equipment to be inspected. It can detect possible problems inside the equipment via the echoes sent to the probe.

Some inspections require removing thermal insulation to gain access to areas that are not visible. This can be a complex process. Deviations may also be detected during ASN and IRSN site visits.

Examples of deviations

The two examples described below, which were addressed in an IRSN review, illustrate the causes of deterioration in equipment earthquake resistance. They include:
› corrosion or mechanical fatigue of reactor safety-related equipment connected to the feed tank of the fuel pool cooling system of the reactor building and the fuel building (refueling water storage tank, RWST);
› anchor failures in the 1300 MW reactor fans.

RWST supports and piping connected to it

Detection of initial deviations at Gravelines:

During EDF’s post-Fukushima inspections in June and July 2011, highly corroded pipe supports were found in the RWST rooms at all the Gravelines reactors. EDF concluded that “all of the corroded supports required renovation” and that “the anchors needed to be inspected to confirm the absence of corrosion” in the RWST rooms at the six Gravelines reactors.

During the first quarter of 2011, EDF conducted a review of the supports and anchors located in the RWST room of reactor 2. At that time, a containment spray system support was found to be seriously damaged. EDF concluded that the piping it supported could not be guaranteed to be earthquake resistant. During the compliance work, EDF found a new problem: a crack in the shell on a pipe in the containment spray system (CSS) (Figure 3.3). EDF found that this crack was due to mechanical fatigue and replaced the plate.

In early September 2012, EDF also noted advanced corrosion on two supports of the CSS of reactor 3. The earthquake resistance of the supports and, thus, of the CSS could no longer be guaranteed. This deviation was observed at the end of a review of the supports conducted after scaffolding was erected and insulation was removed from certain piping. These operations were not anticipated as part of the basic preventive maintenance programme.

Fig. 3.3: CSS piping support.
Following an inspection it conducted in September 2012, ASN requested a review of the damage to the safety-related equipment in the RWST rooms, including the RWST rooms themselves, at the six Gravelines power plants. In late 2012, EDF thus undertook a programme of visual controls of all safety-related equipment at the RWST rooms of the six reactors. During these inspections, some of the supports were found to be corroded, while others were missing. The RWST systems for the various reactors were thus no longer earthquake resistant. After the inspections were performed, EDF brought all the safety-critical equipment in the RWST rooms into compliance. EDF then highlighted the fact that the Gravelines power plant, located on the coast, was more subject to corrosion than other plants.

**Generic observation regarding the deviations and extension of the controls to other nuclear power plants:**

During an ASN inspection in early 2013, the supports in the RWST room of reactor 4 at the Blayais power plant were also found to be corroded. This discovery led EDF to propose a programme of specific inspections, including insulation removal, for the supports near the RWST of Blayais 2 and 4. Many supports had to be brushed and repainted. In addition, one support had to be changed because of advanced corrosion. However, the corrosion observed in these supports did not compromise the pipes’ earthquake resistance. Furthermore, during an ASN inspection in April 2013 of reactor 1 at the Paluel 1300 MW power plant, traces of corrosion were observed on the wall of the RWST. The plant operator then indicated that visual inspections of the wall of the RWST were not being performed as they were no longer part of the basic preventive maintenance programme and the RWST of the 1300 MW reactors are located in enclosed (P4*) or “partially covered” premises (P4*). During that same inspection, corrosion was also seen on supports of the containment spray system located in the RWST room.

It should be emphasized that the three sites affected by these deviations (Gravelines, Blayais and Paluel) are on the coast, where the marine air has a more corrosive effect on steel. In July 2013, recurrent deviations due to corrosion were discovered. EDF thus strengthened the inspections on the condition of the supports and piping (which had been visual up to that point) by examining their surface on a test basis after removing the thermal insulation around the corroded supports of the pipes connected to the reactor coolant systems at all the 900 MW reactors (except Fessenheim and Bugey), as well as other systems throughout the entire French nuclear fleet. The objective was to achieve a zero point for the reactor coolant system equipment that is most vulnerable to corrosion. However, EDF had not planned any inspections of the RWST themselves, because it believed that the closed room protecting them would shelter them from bad weather.

**IRSN analysis:**

In light of the results of all the inspections, IRSN concluded that the deviations observed and cited above posed safety problems, as the RWST contained some of the water needed to remove the residual heat released by the irradiated fuel assemblies stored in the spent fuel pool and to cool the reactor core and feed the containment spray during certain accidents. These deviations could prove to be generic and conditions in the power plants in the nuclear fleet should thus be examined. Initially, EDF planned to limit inspections with thermal insulation removal to the coastal power plants. IRSN insisted that factors other than saline air could promote equipment corrosion and concluded that the inspections should not be limited to the coastal plants.

IRSN further emphasized that feedback on the RWST and the supports and piping located in these tanks’ rooms showed the poor condition of this equipment and the inadequacy of the inspections performed in connection with the basic preventive maintenance programme. These inspections are visual. They are often conducted from a considerable distance because scaffolding is not specified and without removal of thermal insulation from the insulated equipment. IRSN thus concluded that EDF should speed the deployment of the planned inspection programme and that its scope should thus be extended to the RWST and connected piping, and to their supports and anchors located in the closed rooms of the RWST.

For more information about the IRSN analysis, [click here**](http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2013-00291-EDF.aspx)

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(*) P4 = Paluel, Saint-Alban and Flamanville.
P4* = Belleville, Cattenom, Golfech, Nogent and Penly.

(**) http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2013-00291-EDF.aspx
EDF inspected the supports of the piping connected to the RWST on those 900 MW reactors whose RWST is in an uncovered room. At ASN’s request, EDF identified the supports that are particularly sensitive to corrosion. For the most sensitive supports, EDF extended the scope of inspections to the 1300 MW reactors, based on their geographic location (coastal or not) and the type of room where the RWST is located (covered or “partially covered”). EDF’s proposals do not fully address IRSN’s recommendations. Technical exchanges among ASN, EDF and IRSN are thus continuing.

**Fan anchors**

### Detection of deviations at Flamanville and Paluel

Deviations were discovered in the safety-related fan anchors (Figure 3.4) of the 1300 MW reactors of the Flamanville and Paluel NPPs.

![Fan anchor](image)

These involved primarily sizing problems concerning anchor bolts and expansion bolts (diameter less than provided for, bolts that are too short, preventing them from being secured (Figure 3.5)), inappropriate installation system (bolts instead of embedded anchors or combination of different types of mounting systems for a single fan), lack of tightening or inadequate tightening of anchoring components, such as bolts, and failure to comply with standard practices. In the event of an earthquake, these anchors might not be able to support the fans, which could then come loose and no longer operate properly.

The systems affected by these deviations included:
- the annulus ventilation system (AVS); in an accident situation, this system maintains negative pressure in this space relative to the external atmosphere; it also collects leaks from the internal containment into the space between containments to filter them before they are discharged via the stack;
- the auxiliary safeguard building ventilation system, used in accident situations to maintain under-pressure in contaminated rooms and thus prevent the spread of contamination.

The unavailability of these systems in the event of an earthquake could have major environmental consequences.

### IRSN analysis:

Given the number of nuclear reactors affected in 2013 (six reactors, two in Flamanville and four in Paluel), IRSN concluded that EDF should immediately ensure the earthquake resistance of all safety-related fans or those that could damage safety-related equipment for the entire nuclear power plant fleet. Since most of the fan anchor inspections and any potential compliance work had already been performed at the 900 MW reactors in connection with the compliance checks during the 10-year outages, IRSN determined that EDF should:

- accelerate implementation of the fan anchor inspections planned as part of preventive maintenance at the 1300 MW and 1450 MW reactors (inspections scheduled every 10 years) without waiting for their 10-year visits, in order to obtain a thorough assessment, as soon as possible, of the results of the inspections and establish a “point zero” reference for the status of the anchors;
- respond to the deviations observed based on their consequences for safety.

ASN addressed IRSN’s recommendations in the form of requests to the plant operator. In connection with the monitoring of refuelling outages, IRSN observed that the power plants involved had accelerated the deployment of the fan inspections planned as part of the basic preventive maintenance programme for the fan anchors. In addition, given the installation errors observed in recent years in connection with scheduled inspections and rounds conducted during refuelling outages, technical exchanges with EDF continued regarding inspections on the anchoring of additional equipment, including pumps, motors and electrical equipment. Following these exchanges, ASN requested that EDF provide a schedule of anchor inspections on this equipment. According to EDF, three-quarters of the inspections should be performed by year-end 2014.
Conclusion

IRSN noted the recurrence, in recent years, of deviations reported as potentially compromising the earthquake resistance of safety-related equipment. These deviations have highlighted failures to comply with standard practices regarding the installation of certain equipment and the general deterioration of other equipment, revealing inadequate inspections performed under the basic preventive maintenance programme. In addition, feedback shows that the coastal power plants are more affected by corrosion than others. This situation entails heightened inspections and maintenance at these plants. Given the generic nature of the deviations observed, which could affect the earthquake resistance of many pieces of equipment, IRSN’s assessment of the earthquake resistance of the supports and piping located in the refuelling water storage tank rooms concluded that EDF should accelerate the deployment of the inspection programmes and extend the scope of these inspections. EDF supplemented certain inspection programmes that are examined by IRSN.

Fuel Assembly Deformations at Nogent 2

The fuel assemblies that constitute the reactor core deformed under the stress placed on them when the reactor is in operation. If the deformation becomes too extensive, it can slow the drop of the rod cluster control assemblies (RCCA) or even prevent them from completing their drop. RCCA drop time is a critical assumption in safety studies. This situation, which affected reactor 2 at the Nogent power plant in 2013, led EDF to move up the reactor’s scheduled outage by three months.

Why do fuel assemblies become deformed?

Fig.3.6/Diagram of a fuel assembly.
A fuel assembly includes 264 rods that are approximately four metres high and approximately one centimetre in diameter. Eight or nine grids determine the spacing of the rods and hold them in place (Figure 3.6). The assembly also includes 24 guide tubes. The absorber rods of an RCCA are inserted into them. A guide tube is a metal tube made up of a continuous portion and a lower portion of a smaller diameter, called the dashpot, of approximately 70 cm. This allows for the hydraulic braking of the RCCA at end of drop.

Rod cluster control assembly (RCCA)

The RCCA, inserted partially or completely in the core, help to guide the reactor and quickly halt the nuclear reaction.

One-third of the fuel assemblies are furnished with an RCCA. The other fuel assemblies use clusters with a steel thimble plug assembly that have no role in terms of the reactivity of the reactor core.

There are gaps of several millimetres between the fuel assemblies. They may thus warp laterally when they are in the reactor under the cumulative effect of hydraulic and mechanical stresses, radiation and temperatures above 300°C.

Deformation of fuel assemblies: monitoring in operation and impact on safety

EDF monitors the deformation of fuel assemblies at control reactors, where deformation is most significant. This monitoring is conducted using DAMAC measurement tool (Figure 3.7), which allows the plant operator to monitor changes in the deformation of fuel assemblies during each unloading.

Fifteen years of experience feedback shows that the fuel assemblies present different kinds of lateral deformations (Figure 3.8).

Excessive deformation of the fuel assemblies may slow the insertion of the RCCA in the core, or prevent their full insertion in the dashpot. The maximum time for the RCCA drop and their full insertion during an automatic reactor outage are important assumptions in the safety demonstration. Tests periodically verify compliance with the assumptions chosen for the safety demonstration relating to the RCCA drop time. Regulation requires EDF to perform such tests at the beginning and end of cycle.

Principle of fuel assembly deformation inspections using DAMAC tool

EDF developed the DAMAC portable fuel assembly measurement tool, which uses ultrasound to measure the deflections (or lateral deformations) in the fuel assembly grids. This examination is performed when the fuel assemblies are unloaded, during their transfer to the storage pool. The deflection of a fuel assembly, expressed in millimetres, corresponds to the maximum lateral shift of the grids.

(3) A cycle is the period of reactor operation between two outages for partial fuel reloading. A cycle lasts from 12 - 18 months, depending on the reactor.
Figure 3.9 illustrates the effect of the deformation of a fuel assembly on the drop time of the related RCCA.

Different behaviours based on the reactor

Some of the 1300 MW reactors and one 1450 MW reactor are affected by this significant deformation, which can lead to RCCA drop problems. The 900 MW reactors are less sensitive because their fuel assemblies are not as tall (3.6 metres rather than 4.2 metres for the 1300 MW and 1450 MW reactors).

The major deviations in the extent of deformation across the different reactors result from the diverse parameters that are the source of the fuel assembly deformations. The predominant parameters identified are:
- the hydraulic loads exerted on the fuel assemblies by the water circulating in the reactor core;
- the mechanical loads applied by the springs used to hold the fuel assemblies in position;
- the design of the fuel assemblies and, specifically, of the guide tubes (thickness and material), which influences the rigidity of the assemblies;
- the fuel assembly residence time in the reactor, which depends on the management;
- the position of the fuel assemblies in the core: a fuel assembly placed in the centre of the core becomes deformed more than one placed at the periphery.

Chronology of events at Nogent 2

Reactor 2 of the Nogent power plant is one of the control reactors. In late 2012, at the end of the 18th cycle, the DAMAC measures revealed significant changes in the lateral deformation of the reactor’s fuel assemblies.

The situation led IRSN to recommend that EDF conduct supplemental tests to measure drop times, in addition to those planned for the start and end of the cycle, to better monitor changes in RCCA drop times during the following cycle. During this cycle, in 2013, certain times measured during the first supplemental test requested by IRSN reached the alarm thresholds set in a decision by ASN. Consequently, in implementation of this decision, the plant operator had to conduct monthly RCCA drop time tests. Findings of significant increases in drop times in certain clusters during the cycle and the incomplete insertion of five RCCAs in the dashpot ultimately led EDF to shut down the reactor on 22 February 2014, three months before the planned outage date.

Why did this happen?

The specific situation at Nogent 2 resulted from a shift to the GALICE management system. This management is characterized by a specific plan for positioning the fuel assemblies and fewer new fuel assemblies at each reload than under the GEMMES management system implemented at all the other 1300 MW reactors.

These two specific features explain the significant deterioration in the behaviour of the core. As a result of this situation, EDF permanently terminated implementation of the GALICE management system (for Nogent 2, the only reactor using that system).

(4) The main change between the GEMMES and GALICE fuel management systems is the increase in the authorised burnup rate, which rose from 52 GWtU to 62 GWtU (burnup is the energy released by the combustion of one unit of mass of nuclear fuel).
IRSN's position

IRSN precisely analyzed the measurements of the RCCA drop times taken monthly by EDF during the 19th cycle and the consequences of the degraded situation for reactor safety, based on the plant operator’s estimates:

› the RCCA drop times that should have been met in month N+1 using the feedback and measurements of the drop time in month N;
› the number of RCCA that could be blocked in the dashpot.

The analysis of this data allowed IRSN to make a decision as to the technical acceptability, on a temporary basis, of the reactor’s operation with, in the event of an automatic reactor shutdown, several RCCA remaining partially inserted in the dashpot of the guide tubes. This situation was not included in the demonstration presented in the safety report.

The detailed analysis of all the hypothetical accidents included in the reactor safety demonstration, taking into account the specific situation of reactor 2 of the Nogent power plant, led IRSN to recommend that EDF implement specific operating provisions until the end of the 19th cycle that would increase core sub-criticality in the event of an automatic reactor shutdown. IRSN further recommended monitoring the measurement of RCCA drop times during the 20th cycle, after completing 40-50% of the cycle.

EDF measures taken for the 20th cycle - Outlook

For the 20th cycle, EDF positioned the fuel assemblies to limit their deformation:

› fuel assemblies that showed excessive deformation were not reloaded;
› the least deformed fuel assemblies were selected to house the RCCA and the seriously deformed fuel assemblies were placed next to new fuel assemblies.

The reloading of new fuel assemblies is composed, for the first time at Nogent 2, of newly-designed fuel assemblies with thickened guide tubes of “advanced” zirconium alloy (includes niobium, iron and tin). The purpose of this design is to increase the rigidity of the fuel assemblies and thus limit deformations over the course of the cycle.

IRSN approved the introduction of this new fuel assembly design in 2012 and an initial reloading of this type of fuel assemblies is currently used in reactor 2 at the Chooz B power plant. However, the potential benefits of the design will not be fully measurable for several years, when all of these reactor cores (Nogent and Chooz B) are composed of such fuel assemblies designed to be more rigid. In the meantime, EDF is continuing its heightened monitoring of the fuel assemblies at Nogent 2.
4
SIGNIFICANT UPDATES
Changes and upgrades are made to France’s nuclear reactors throughout their operational life, mainly with the aim of continuously improving safety. Advances in scientific and technical knowledge, weaknesses detected or lessons learned from operational experience, changes in environment or regulations, and economic factors, are just some of the reasons for making changes either to power plants or operating procedures. It may take several years of research to define changes or upgrades precisely, before they can be implemented. Safety reviews, carried out regularly every ten years, are one of the main frameworks for promoting and implementing such changes. From research to implementation, the biggest changes can take years. IRSN spends this time examining the documentation relating to the different implementation stages of the change.
The containment of radioactive materials is one of the safety functions of nuclear reactors. The containment of a PWR constitutes a “barrier” designed to limit the discharge of radioactive materials into the environment from the reactor core. The impermeability of containments and monitoring of their ageing are therefore vital. Every ten years, EDF performs a test to check the impermeability of each reactor containment and to assess its mechanical behaviour. EDF’s report on these monitoring activities was examined by IRSN as part of its safety review of the 1300 MW and 1450 MW reactors associated with their ten-yearly inspections.

### Contains

In order to keep radioactive discharges into the environment to acceptable values, including in accident situations, physical “barriers” are placed between the radioactive materials on the one hand and the public and environment on the other. The fuel cladding and the reactor coolant system housing constitute two physical barriers. The containment of pressurized water reactors is another barrier. It must be able to resist damage such as earthquakes and withstand the large increases in internal pressure and temperature associated with accidents. Controlling the impermeability of the inner containment wall (see below) is vital if it is to perform its containment role.

Reactor containment design differs according to reactor type: the containments of 900 MW reactors are made of prestressed concrete with the inner face totally covered with an impermeable steel liner. The containment of 1300 MW and 1450 MW reactors has a double wall: an outer one made from reinforced concrete (the “outer containment wall”) and an inner one made from prestressed concrete (the “inner containment wall”) (Figure 4.1).

### Prestressing

Concrete is very resistant to compression but not to traction. To reduce the risk of cracking, it therefore has to stay permanently compressed, particularly in areas where traction stress could develop. Compression deliberately applied to concrete is known as “prestressing”. For the containment of nuclear power reactors, this prestressing is achieved using steel cables immersed in the concrete.

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**Fig. 4.1** / Containments of 900 MW, 1300 MW and 1450 MW reactors.
Leakage rate criteria

The construction permit for each 1300 MW or 1450 MW reactor stipulates that “the internal containment will in particular be designed to withstand, without loss of integrity, the stresses resulting from an accident involving the sudden failure around its complete circumference of a pipe in the reactor coolant system, with total separation of the ends. In the context of such an accident, the maximum leakage rate of the containment must be less than 1.5% per day of the mass of gas held within the containment.”

To control its leakage rate, each inner containment wall is tested every ten years to check its impermeability and ability to withstand a pressure. The containment is “inflated” with air for three days until it reaches its design pressure (around five times atmospheric pressure) using a fan. This test is used to measure the air leakage rate, i.e. the amount of air that escapes from the containment. This rate is calculated on the basis of the change in the mass of dry air in the containment at different pressures. The mass of dry air is calculated using the ideal gas law.

The leakage rate criterion used during the impermeability tests includes a provision for ageing to ensure that the containment continues to comply with the regulatory limit of 1.5% per day under accident conditions for the ten years following the test. This provision for ageing allows the effect on impermeability of the ageing of the inner containment wall to be taken into account.

Impermeability testing of inner containment walls

For each reactor, the plant operator carries out an impermeability test of the inner containment wall as soon as the reactor is built and performs another test before the first refuelling, approximately three years later. Impermeability tests are then performed during the ten-yearly inspections.

From the first tests of the double-wall containments, EDF found that it was necessary to improve the impermeability of some areas of the inside face of the inner containment wall, and so a composite coating (fibreglass fabric impregnated with an epoxy resin) was applied. Despite these coatings, some containments, classified as “sensitive”, continued to have relatively high leakage rates. The application of the coatings was therefore continued, enabling a satisfactory situation to be achieved. The second round of ten-yearly impermeability tests have so far been carried out on 18 containments in 1300 MW reactors, including some “sensitive” containments. All 18 containments meet the leakage rate criterion, and the impermeability of some has even improved since the initial ten-yearly test.

Mechanical behaviour testing of inner containment walls

The tests carried out are also used to check that the inner containment walls are behaving properly from a mechanical point of view, particularly as regards the reversibility of deformations. For the purpose of this check, the inner containment walls are fitted with a measuring device consisting of sensors (strain gauges and thermocouples) sunk into the walls and instruments (pendulums and invar wire) mounted on the outer face of the walls. The device is used to measure the deformation and movement of the containment during the impermeability tests, but also throughout the life of the structure. Deferred deformation is measured to monitor concrete shrinkage and creep, and, indirectly, the loss of tension of the cables in prestressed concrete.

The IRSN has extrapolated the deformation measured since the inner containment walls were built. The extrapolation of these measurements is used to evaluate the deferred deformation of the inner containment walls in 60 years’ time. Some of these containment walls have a higher measured deformation and their expected deformation is therefore greater. However, according to IRSN’s initial estimates, the amount of deformation that occurs between 30 and 60 years will be comparatively smaller than the deformation that has already occurred or is expected to occur up to 30 years. EDF has agreed to present its own assessment of the deferred deformation of the concrete in the inner containment walls.

In view of the deferred deformation measured or extrapolated and the reversibility of the deformation at the time of the ten-yearly tests, IRSN believes that the containments of the 1300 MW and 1450 MW reactors are robust and can fulfil their containment function.

Problems with concrete

Diagnosing the physical condition of the containment is an important part of the safety analysis of a reactor. The physical condition of the containment is checked by EDF as part of its regular scheduled monitoring. Some problems are monitored particularly closely because of their damaging effect on structures. Meanwhile, IRSN has begun research into the

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(5) Deferred deformation: deformation measured since construction of the containment was completed.

(6) Shrinkage: Reduction in volume of a material or soil due to it drying out.

(7) Creep: Irreversible increase in the original deformation of a material under constant long-term stress.

(8) Nuclear Engineering and Design – ISSN 0029-5493 – Volume 266 – January 2014 – “Coupling between mechanical and transfer properties and expansion due to delayed ettringite formation in a concrete of a nuclear power plant”.
degradation kinetics of concretes that are likely to develop swelling problems such as alkali reaction or delayed ettringite formation. This research has already shown that the presence of water greatly encourages the development of these problems, which manifest themselves as cracks on the surface of the structure and loss of strength. IRSN has therefore recommended that EDF analyse the risk of rainwater entering the foundation rafts and inner containment walls of 1300 MW and 1450 MW reactors and, where appropriate, present measures to reduce this risk.

Conclusion

The coatings applied to the inner containment walls of 1300 MW and 1450 MW reactors to improve their impermeability produced good results in the latest ten-yearly impermeability tests. Consequently, IRSN considers the containment walls of the 1300 MW and 1450 MW reactors to be in a satisfactory condition at present. However, as a result of IRSN’s analysis, EDF has agreed to do the following in addition to its monitoring programmes and coating work:

› to present its own analysis of the deferred deformation of the concrete of the inner containment walls;
› to continue to develop new techniques to improve the impermeability of the containment walls, which could be implemented in addition to the existing measures at the time of the next ten-yearly inspections;
› to carry out a detailed analysis of the structure concerned, whenever swelling is detected in concrete, to show that the degradation remains limited.

Lastly, EDF is going to implement a major programme of tests to quantify leakage in situations where pressure is higher than the design pressure, using a model representing the inner containment walls of the 1300 MW and 1450 MW reactors in its nuclear power plants.
The risks associated with hazards are analysed from the nuclear reactor design stage. However, as knowledge increases, partly as a result of previous operational experience, the list of hazards and their levels are reassessed during the ten-yearly safety reviews.

The significant advances introduced during the safety review associated with the third round of ten-yearly inspections of the 1300 MW reactors are explained below, particularly the inclusion of hazards not previously taken into consideration.

"Internal" and "external" hazards

There are several different types of hazard that can damage equipment and affect the safety of nuclear power plants. They are split into categories according to origin:

- "internal" hazards: the source of the hazard is inside the installation, e.g. a fire breaking out in a room, a flood caused by a tank rupturing;
- "external" natural hazards: e.g. very high temperatures (heatwaves), very strong winds, etc.;
- "external" hazards associated with activities taking place outside the installation, such as accidental explosions nearby.

Safety-related equipment must remain operational when a hazard occurs. It must therefore be:

- either protected by measures that prevent the hazard from affecting it;
- or designed to remain in operation even if it is affected by the hazard.

In the safety demonstration, the analysis of hazard-related risks has two phases: determination of the characteristics of hazards likely to occur on each site, followed by a check that the existing measures are sufficient to keep their consequences to acceptable levels. Where appropriate, new measures are defined and implemented.

As knowledge increases, partly as a result of previous operational experience, the list of hazards and their levels (e.g. maximum wind speeds) are reassessed during the ten-yearly safety reviews. Obviously when significant events occur (the 1999 storms throughout France, the 2003 heatwave in France, the 2011 Fukushima disaster), the associated risks are reviewed without waiting for the next ten-yearly safety review (see pages 56, 64 and 67 of the 2012 PWR report ).

What does the ten-yearly safety review involve?

The French Environment Code (Article L.593-18) requires operators of nuclear power plants in France to carry out a safety review of their installation(s). For nuclear power reactors, this review, combined with the ten-yearly inspection, consists of several parts:

- a review to check the installation’s compliance with the safety baseline and current regulations,
- a safety review aimed at bringing safety levels at the oldest reactors up to the standard of the most recent reactors, as far as possible,
- the implementation of improvements resulting from the safety review.

Following the safety review associated with the ten-yearly inspection of each reactor, the plant operator sends the ASN a report containing the conclusions of the review for the reactor.

During the safety reviews associated with the third round of ten-yearly inspections of 1300 MW reactors, referred to as "VD3-1300", EDF reviewed the risks posed by the hazards presented in the table below. Furthermore, for hazards or combinations of hazards of natural origin that could simultaneously affect all the reactors on a site, EDF looked at the power plants’ ability to manage situations caused
by a total loss of the heat sink or a loss of off-site power, or a combination of both. Management of these situations mainly relies on equipment (pumps, generators, etc.) and the availability of reserve supplies on the site (water for cooling the circuits, fuel oil for running the generators, etc.). Generally IRSN has taken the view that the changes proposed by EDF to manage the consequences of this kind of situation constitute progress in terms of safety. However, further information still has to be provided by EDF, notably on the handling of certain hazards not previously considered likely to cause a total loss of the heat sink or the loss of off-site power.

EDF’s hazard reports are mostly very similar to those presented during the safety reviews recently carried out on reactors with different power ratings. However, a notable feature of the safety review associated with VD3-1300 is the examination for the first time of the procedures selected by EDF to deal with frazil ice, wind-generated projectiles, tornados, drifting oil slicks, and the new “explosions baseline”. Three of these subjects are examined in more detail below.

### Frazil ice (Figure 4.3)

Frazil ice, an external, climate-related hazard of natural origin, refers to the formation of ice crystals in water, which happens in particular weather conditions (when the water temperature is below the melting point of water...). These ice crystals can stick to anything in the water (active frazil) such as plants, grilles, filters, etc., or they can form plates of ice covering all or part of the surface of the water (passive frazil). Frazil ice can block the water intake of an NPP by clogging it up, preventing it from taking in the water necessary for cooling the reactors, and consequently leading to a loss of cooling water. As part of the safety review associated with VD3-1300, EDF analysed the ability of 1300 MW
NPPs to cope with frazil ice. It evaluated the arrangements in place at the power plants, which rely on certain physical measures (circulation of hot water at the entry to the pumping station) and organisational measures (alerts enabling protective devices to be installed before the phenomenon occurs).

EDF took the view that coastal installations would not be affected by frazil ice. IRSN felt that, given the risks from loss of cooling water, EDF should carry out historical and statistical research before concluding that frazil ice would not form at coastal sites.

For installations beside rivers likely to be affected by frazil ice, the existing protective measures or the measures planned by EDF in the context of the safety review make a general contribution to power plant safety. However, IRSN believes that it is important for the planned measures to be in operation well before the water temperature reaches freezing point. In particular this means improving monitoring systems to ensure the problem is detected sufficiently early to implement the planned measures and prevent the water intake from icing up. In addition, for sites where EDF was not planning to install hot water circulation, IRSN indicated that the planned measures were not sufficient and that EDF therefore needed to come up with further protective measures (to find out more about frazil ice, click here*).

Wind-generated projectiles and tornadoes (Figure 4.4)

During the design of the nuclear power plants currently in operation, the pressure effects of strong winds were taken into account in order to design the buildings to current standards. However, the effects associated with projectiles generated by strong winds were not examined. Yet safety-related equipment situated outside buildings can be damaged by these projectiles. EDF suggested a method for dealing with this and took appropriate steps to protect equipment identified as vulnerable from the effects of these projectiles.

IRSN feels that this method represents a significant advance. However, the characterisation of the speeds of strong winds (which depend on the geographical location of the site, any “tunnel effects” linked to the presence of buildings, height off the ground, etc.) merits further study.

IRSN also felt that EDF needed to continue its research into the types of projectiles likely to damage the safety-related equipment of a reactor (particularly by studying the effects of small projectiles) and that it also needed to look into the ability of the equipment to withstand projectiles generated by strong winds.

Meanwhile, tornadoes were not considered a likely source of damage when the nuclear power plants currently in operation were designed. Having observed high-intensity tornadoes in France, IRSN felt that this phenomenon should be addressed in the safety demonstration for nuclear power plants in France. EDF recently proposed a national “design basis tornado” and a method for defining appropriate measures to

*) http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2013-00252-EDF.aspx

A tornado is a very localised violent phenomenon. Besides having very similar effects to strong winds, tornadoes cause negative pressure that produces suction effects. The intensity of tornadoes is measured retrospectively using the Fujita scale, based on the damage they cause.
Internal explosions and explosions outside the nuclear island

During the safety review associated with VD3-1300, EDF applied its “safety requirements baseline for protection against internal explosions at non-EPR NPPs”. In accordance with this baseline, EDF carried out risk analyses covering the explosion risks linked to hydrogen pipes on the site and to the electrochlorination process at pumping stations on coastal sites.

The electrochlorination of seawater is a chemical process used to remove the risk of proliferation of the organisms and microorganisms in the water. The process thus prevents the formation of organic deposits in the reactor coolant system; it leads to the release of hydrogen due to the chemical decomposition of the water.

As a result of these analyses, EDF decided to strengthen the hydrogen pipes that run outside the nuclear island to ensure they can withstand seismic margin earthquakes and avoid damage caused by nearby high energy line breaks. As a result, EDF took the view that the risk of an explosive atmosphere forming could be ruled out in the service galleries and ducts. However, IRSN felt that EDF should explore the value of having a system for the early detection of abnormal hydrogen releases from pipes and for limiting the consequences of this. This IRSN recommendation was taken up by ASN in its follow-up letter to EDF.

IRSN also identified other explosion scenarios that EDF should study, for example the explosion of a hydrogen cloud from the generator or from a leak in another circuit, which could form under the turbine hall ceiling. Concerning the electrochlorination process, IRSN felt that EDF should demonstrate that there was no risk of a hydrogen explosion in the installation if the ventilation system for diluting the hydrogen released and the process for automatically shutting down the plant in the event of fire or lightning damage should simultaneously fail. EDF also plans to make a number of changes to the storage areas where the pressurized gases necessary for the reactor, particularly hydrogen, are kept. A major refit of these storage areas will be undertaken to improve their ability to resist external damage, with the reinforcement of the bunkers, the installation of metal guards to protect against projectiles, and the construction of fire walls, etc.

IRSN notes the scale of the changes planned by EDF. However, the demonstration that an explosion in the gas storage areas would have no significant consequences for the safety-class buildings on the site still needs to be completed, particularly as regards the mechanical behaviour of the structure of the buildings in which these explosions might occur.

For more information about internal explosions, click here.


Conclusion

Generally speaking, significant advances have been made with analysing the risks associated with hazards during the reviews conducted as part of the third round of ten-yearly inspections of the 1300 MW reactors. These concern the study of particular phenomena and the evaluation of the vulnerabilities of installations, but also the introduction of monitoring and protective measures. However, there is further work to be done by EDF in response to the issues raised by IRSN following its assessment process.
In 1998 a leak occurred in the residual heat removal system (RHRS) of reactor 1 at the Civaux power plant when the reactor was shut down. The leak was caused by cracking due to thermal fatigue in a stainless steel pipe elbow. EDF began a major assessment, analysis and research programme to understand the phenomenon that caused the incident. At the end of this programme, which ran for more than 10 years, IRSN examined EDF’s conclusions and delivered its own analysis in 2013. IRSN highlighted the fact that EDF’s analysis and research had provided a more detailed understanding of the phenomenon, but it also said that an accurate record should be kept of the operating times of circuits vulnerable to thermal fatigue where there were big differences in temperature, to enable appropriate checks to be introduced.

Action taken as a result of the 1998 incident at Civaux

On 12 May 1998, reactor 1 at the Civaux power plant was undergoing a normal shutdown when a 30 m³/hour leak appeared in the RHRS. This system is used to remove the residual power from the reactor core. This major leak was caused by a 180 mm long crack through the pipe at an elbow immediately below a mixing tee linking a pipe (or line) carrying hot water bypassing the heat exchanger with a pipe carrying cold water coming out of the heat exchanger (Figure 4.5). In the mixing area, the hot water was at a maximum temperature of 180°C and the cold water at a temperature of around 20°C.

Examinations of the RHRS circuits removed revealed evidence of thermal fatigue that was not taken into account at the design stage. This phenomenon causes multiple shallow cracks, crazing (cracking of the surface) or isolated cracks starting at the base of a weld bead (Figure 4.6). From 2000, on IRSN’s recommendation, EDF introduced a policy of regularly checking the replaced RHRS sections every 450 hours of operation at wide temperature.

- Fatigue: damage caused by the repeated application and removal a very large number of times of a stress, even if it is below the elastic limit of the material, which can cause significant damage or even breakage.
- Thermal fatigue: fatigue caused by repeated temperature changes that cause size variations. If the material cannot freely expand and contract, these size variations produce cyclical stresses that generate fatigue.

Ultrasonic testing is a non-destructive testing method that uses ultrasonic waves emitted by a probe on the surface of the material being tested to detect any defects inside the material, by means of echoes sent back to the probe.
differences. The 450-hour frequency was based on the first cracks observed in the RHRS circuits removed from reactor 2 at the Civaux power plant. The plant operator’s initial analyses could not fully explain the location and extent of the damage observed. Furthermore, the traditional method of analysing (mechanical) fatigue could neither predict nor explain the damage observed. According to the reports on numerous inspections carried out since, thermal fatigue in the mixing areas of RHRS circuits is a phenomenon that depends on many factors: the operating time at wide temperature differences, thermal and hydraulic factors (the speed and temperature of the fluid), the condition of the surface and the mechanical stresses in the different components. Because there are so many factors, understanding this type of fatigue is particularly complex.

Vulnerability of mixing areas to thermal fatigue

When France’s nuclear reactors were originally designed, thermal fatigue was not taken into account when designing mixing areas. The Civaux incident prompted EDF to introduce a method for evaluating the risk of thermal fatigue in mixing areas in pipes:

› initially, the mixing areas at risk were defined and identified. An area of stainless steel is considered to be “at risk” if the temperature difference between the hot fluid and the cold fluid is greater than 80°C;
› then the risk of these areas cracking was assessed. EDF’s decision was based on an indicator commonly used for fatigue, the usage factor. This is defined as the ratio between the number of stresses applied to a particular component and the maximum stress values indicated by the mechanical fatigue curve for the material that the component is made from. Mixing areas with a usage factor of more than 1 were considered “vulnerable to thermal fatigue”.

This method was applied to all circuits containing mixing areas, particularly certain pipes connected to the reactor coolant system, in order to introduce appropriate monitoring for the vulnerable areas identified.

Research and development on thermal fatigue

At the end of 1999, EDF and AREVA began a major research and development programme. The aim of this programme was to improve the knowledge and analysis of thermal fatigue, particularly in mixing areas, and to identify more clearly the conditions under which the degradation mode observed would arise, through an exhaustive survey of the “vulnerable” areas.

In parallel, to bolster its own expertise in this area, IRSN also carried out some research. In partnership with the CEA, IRSN conducted a study aimed at understanding the specific nature of the stresses involved in thermal fatigue, compared with the stresses commonly taken into consideration for mechanical fatigue. The study was conducted using an experimental device known as FAT3D (Figure 4.7), in which a pipe was subjected to thermal stresses: a network of cracks rapidly appeared and
spread, with some spreading right through the material. In particular it was established that the number of cycles it took to make the cracks appear in the test specimens was always less than the number predicted by the calculation made using the usual analysis methods for mechanical fatigue.

Tests on models representative of a mixing area

EDF had some results of tests performed since 1976 on models that were a geometric representation of the mixing areas and reproduced the flow in these areas. These model-based tests were primarily designed to provide the loads (average temperature ranges, heat transfer coefficients) for studies of mechanical behaviour related to operating transients. Following the thermal fatigue incident at Civaux, EDF completed its database by carrying out tests that were not restricted to the RHRS circuit but also covered other mixing areas in the reactor coolant system feed line (Figure 4.8).

For the different mixing areas, these later tests allowed:

- the temperature ranges and the coefficients for heat transfer between the fluid and the internal wall of the pipes to be evaluated;
- information to be gained about the sites of temperature fluctuations due to the mixing of a hot fluid with a cold fluid, based on the geometry and the ratios of the flow rates of the two fluids.

The model-based tests and the associated computational models gave a realistic assessment of the thermal loading affecting the pipes in the mixing areas.

Taking account of the results of the research and development and the model-based tests, the inspection method for mixing areas “vulnerable to thermal fatigue” used from the 2000s involved the scheduling of periodic checks of the identified areas, usually at the time of the ten-yearly inspections.

Change to the method for assessing mixing areas

Since 2008, EDF has felt that the method and means of monitoring it has introduced address the problem of thermal fatigue in mixing areas. EDF has adopted a more representative modelling of thermal transients based on the results of the model-based tests, and has replaced the conventional values used for heat transfer coefficients with values based on reality, taking account of the thermal and hydraulic conditions that actually exist in mixing areas. However, the results of the research have not made it possible to upgrade the thermomechanical method used for estimating mechanical stresses. Lastly, the research has led to the adoption of a lower threshold than before for the vulnerability of stainless steels to thermal fatigue: mixing areas where the temperature difference of the fluids is 50°C or more (rather than 80°C or more) are now

Fig. 4.7 / FAT3D thermal fatigue testing device.

Fig. 4.8 / Diagram of the feed line connected to the reactor coolant system.
Considered to be “vulnerable”. This change has prompted EDF to review its list of mixing areas that might be "vulnerable to thermal fatigue".

Representativeness of the usage factor of the mixing areas

Between 1999 and 2002, EDF appraised the sections of the RHRS circuits removed after the Civaux incident. The usage factors of these sections were also calculated on the date of removal; the usage factors were greater than 1. This confirms their vulnerability to thermal fatigue. However, no correlation could be established between the usage factor values and the dimensions of the cracks. In 2001, EDF decided to investigate the possibility that there were defects in a mixing area considered “vulnerable to thermal fatigue”: the branch pipe of the reactor coolant system feed line (Figure 4.8). For appraisal purposes, EDF took the opportunity provided by the replacement of the steam generators in reactor 1 of the Fessenheim power plant, after 24 years of operation, to remove a section of piping from the reactor coolant system, including the connection joint of the feed line. The inspection of this connection joint, which had a usage factor calculated at more than 1, did not reveal the beginnings of any cracking due to fatigue. Furthermore, ultrasonic testing of feed line connection joints carried out since 2004 on around 20 reactors has not detected any evidence of thermal fatigue.

In 2013, IRSN examined the evidence from around a decade of inspections of mixing areas, including the sections from the RHRS circuit and the branch connections of the feed line. IRSN felt that the usage factor was an indicator of the risk of cracking, but that it was not suitable for assessing the damage that could occur as a result of thermal fatigue. It was therefore important that, in addition to the ten-yearly inspections, the frequency of inspection of all mixing areas with a usage factor greater than 1 should be determined on the basis of the operating time at wide temperature differences. This was implemented by EDF in the case of the mixing tee in the RHRS circuit.

Counting operating time at wide temperature differences

In 2008, EDF introduced a system for metering on a daily basis the amount of time spent operating at wide temperature differences, for all mixing areas “vulnerable to thermal fatigue”, and set thresholds for this. However, it was not until 2013 that EDF specified what action should be taken when these thresholds were reached. If the operating time of a mixing area at wide temperature differences is greater than expected, EDF will either bring forward the check of the area normally scheduled to take place during the ten-yearly inspection or will justify the fact that its condition will not have changed using calculations.

In IRSN’s view, if a threshold for the time spent operating at wide temperature differences is exceeded, EDF should inspect the mixing area concerned without delay.

Conclusion

Following the incident in 1998 involving the RHRS circuit of reactor 1 at the Civaux power plant, IRSN has analysed the substantial amount of work done by EDF to establish the risks to its reactors from thermal fatigue. A better understanding of local thermal and hydraulic phenomena, developed as a result of numerous model-based tests and some computational modelling, has brought improvements to the method for identifying mixing areas “vulnerable to thermal fatigue”.

However, IRSN still has reservations about basing an assessment of the risk of damage to sections of the mixing areas solely on a calculation of the usage factor. In IRSN’s view, priority should be given to checking these sections on the basis of the time spent operating at wide temperature differences. IRSN has shared its analysis with the French Nuclear Safety Authority (ASN), which will specify conditions for in-service monitoring.
SAFETY AND RADIATION PROTECTION MANAGEMENT DURING REACTOR OUTAGES

In the last few years EDF has introduced some significant changes to the structures responsible for preparing for and monitoring maintenance carried out during scheduled reactor outages. At the ASN’s request, IRSN carried out interviews and detailed observations of the activities performed during three reactor outages in 2012, to evaluate the effectiveness of safety and radiation protection management measures.

According to IRSN, with the prospect of longer outages linked to possible reactor lifetime extensions, a reduction in tensions during reactor outages and a better balance between workload and resources are essential if risk management is to be improved. In addition, the many organisational changes agreed are having a marked effect on the work of staff, making it necessary for EDF’s overall change management strategy to take greater account of the knowledge of those involved in the work and the real-life difficulties they experience on the ground.

How is a scheduled reactor outage organised?

Reactor outages are scheduled so that a reactor can be partially refuelled and so that thousands of preventive and corrective maintenance operations can be carried out on its equipment. These operations are performed by hundreds of EDF staff and subcontractors. Preparation, planning, execution and feedback are coordinated by several dozen EDF staff who, for several months, form an “outage project team”. To direct this outage project team, several people forming a management team are responsible for coordinating the preparation and execution of the outage and the provision of operational feedback. There are several phases to a scheduled outage, as shown in Figure 4.9.

What are the risks during the different phases of a reactor outage?

The nuclear fuel must continue to be cooled while the reactor is shut down. During the outage phases, maintenance is carried out in accordance with the schedule drawn up by the outage project planning unit. During maintenance work, some safety-related equipment may be rendered temporarily unavailable, which means that compensatory measures must be taken. The different activities can lead to errors, despite preventive measures and measures to ensure the reliability of the work. These kinds of errors can make the safety functions less reliable and can generate “latent defects”, which are detected during the requalification testing or when the equipment is first used during the operational phase, or not until the next outage. Nearly half of significant safety events are reported during reactor outage phases. In addition, the maintenance performed during these outages can lead to significant exposure of workers to ionising...
radiation (around 80% of the collective dose received each year). Controlling these risks depends on operational decisions made in real time, but also on decisions made in advance concerning the way the maintenance is organised (Figure 4.9).

Organisational change limited by constraints

For some years now, EDF has gradually been introducing a new organisational structure for better management of reactor outages. One notable change is an increase in the hours when the outage project management team is available because of a daily rotation provided by two teams. Relief cover is provided by staff from other outage project teams, who have to put on hold their outage preparations for other reactors on the site, with the consequence that management responsibility is passed to staff who did not prepare for the current outage and therefore do not understand all aspects of the decisions made during the preparations.

The new organisational structure was specifically examined by IRSN in 2012, partly because of the particular risks presented by the reactor outage phases, and partly because of the risks brought about by the organisational change. Organisational changes temporarily destabilise an existing organisation, potentially increasing its vulnerability. Such changes have contributed to industrial accidents, such as the disintegration of the space shuttle Columbia during its return flight to Earth in 2003. Following the meeting of the Advisory Committee for nuclear reactors on 24 April 2008, on managing safety in a competitive environment, EDF developed a national approach to designing and managing change, which included the performance of an organisational risk analysis. This was adapted locally to individual sites. In 2012, IRSN observed that this approach had encouraged the appropriation of the new organisational standard for managing outages at nuclear power plants. EDF’s central services had also gained new capabilities for accompanying change, in terms of consultancy skills and the sharing of best practices. This was a positive change which should be protected in the long term. However, because of all the constraints on the nuclear power plants, they were unable to implement all parts of the new organisational standard for managing outages and took a more cautious, pragmatic approach. The consequence of this is that the need for adaptation to local constraints causes big variations in the way outages are organised, and EDF must control these in order to stabilise the way roles and responsibilities are exercised.

Preparing for reactor outages: a phase requiring protection of resources

The planning of preventive maintenance over ten years by EDF’s central services in liaison with the nuclear power plants aims to ensure that there is a balance between the workload and the available resources. For each outage, the maintenance programme is then fixed by the outage project team in agreement with the central services several months before the outage begins, so that the thousands of maintenance operations to be performed can be properly prepared and scheduled.

An outage project team consists of representatives from different sections within the nuclear power plant (operation, maintenance, safety, radiation protection), who can explain the requirements associated with their role or function and also any regulatory requirements. Working meetings are held to take account of the different requirements, and the progress of the preparations is regularly monitored within the project outage team. Some of the feedback from previous projects is integrated at this stage (Figure 4.9).

IRSN believes that all of these measures help to ensure compliance with safety and radiation protection requirements. However, IRSN noticed, during the observations and interviews it conducted in 2012 for three reactor outages, that the conditions for preparing outages may not be so ideal. This is due to the addition of extra tasks to the preventive maintenance programmes (because of ageing equipment and the stricter requirements associated with this, etc.), the effects on resources of cumulative delays during outages in previous years, but also the availability demanded of the different sections, which are also having to cope with the operational requirements of the other reactors at the nuclear power plant.

Managing outages: anticipation, coordination, centralisation and quick reactions to unexpected problems

EDF has organised the management of outages in project mode in order to encourage cooperation between the different sections, particularly when it comes to managing unexpected organisational issues (e.g. lack of availability of competent personnel or spare parts) and equipment problems (e.g. a valve that cannot be inspected because it is blocked) that occur during maintenance operations.

IRSN’s observations in the field have shown that managers of outage projects are strongly motivated to resolve unforeseen problems in real time, sometimes at the expense of preparing for the activities to be performed in the days that follow. They have also shown that including representatives from the different sections, and personnel with recognised
Execution of maintenance during a reactor outage

The maintenance carried out during a reactor outage is performed by several hundred workers. Despite the outage schedule drawn up to organise the work to be done by these workers, and despite the links set up between the management team and the maintenance teams, coordination could be improved, on the one hand between the management team and the workers on the ground, and on the other between the different sections. Difficulties with monitoring contractors have also been observed, despite action taken by EDF over more than 10 years. EDF’s control of subcontracted activities is in the process of being evaluated by IRSN in 2014.

Reversal of the burden of proof

This is a phenomenon that was identified in discussions prior to critical decisions being made in advance of NASA space shuttle accidents. It consists of demanding proof that a situation is not safe, if doubts are expressed during discussions. A proper demonstration of safety, on the other hand, requires the plant operator to provide proof that measures guarantee safety. In other words, the benefit of the doubt should be given to safety.

An organisation and its people experiencing multiple changes and evolutions

The changes to the organisation of reactor outages are not the only changes introduced by EDF to improve its industrial performance. Other organisational changes and changes to working methods have been made that are having an impact on the work of EDF staff during reactor outages (Figure 4.10). Although, during the interviews it conducted, IRSN could see that cumulative changes had been spread out over time by the central services and nuclear power plant managers, the teams on the ground were still having to cope with difficult working conditions. The effects of interaction between changes to the way work is organised or to the perimeter of work of some personnel have not entirely been anticipated.

The reactor operation and maintenance environment is also changing significantly to take account of expertise in safety and radiation protection, in an outage project team improves the level to which requirements in these areas are taken into account in decision-making during an outage. However, in some specific situations, the exercise of responsibilities or the status of personnel with this expertise can lead to inappropriate confrontations (reversal of the burden of proof), which can lead to risks that need to be analysed and controlled. Similarly, EDF must do more to take account of radiation protection requirements when amending maintenance work scenarios.

Fig. 4.10 / Safety and radiation protection management during reactor outages in the midst of other changes.
of regulatory changes and ageing installations, under difficult circumstances in view of staff turnover, during a drive to be more competitive. In particular, because of the conjunction of a large number of retirements, delays in recruitment and the amount of time needed to train personnel, there is currently an individual and collective skills shortage. IRSN has noted that EDF is setting up a project entitled “Generation 2020”, chiefly to address these problems. In addition, the many organisational changes agreed are having an effect on the work of staff, making it necessary for EDF’s overall change management strategy to take more account of the skills of those involved in the work and the real-life difficulties they experience on the ground.

Conclusion

The reactor outages in 2012 and 2013 took longer than expected, disrupting the organisation in ways that could affect safety and radiation protection. With EDF’s decision to extend reactor lifetimes beyond 40 years, which would undoubtedly increase the amount of maintenance to be carried out, IRSN believes that EDF must find a balance between the workload and the skilled resources at its disposal, with adequate margins. EDF has drawn up an action plan with the aim of achieving the first improvements in reactor outages in 2014. Other measures will be taken in future years. For more details, see the factfile on managing safety and radiation protection during reactor outages by clicking here.


**Corrosion of the Zircaloy 4 Cladding in Fuel Assemblies**

The cladding of reactor fuel assemblies is the first containment barrier for the fission products. While it is in the reactor, the cladding corrodes and becomes brittle from contact with the water in the reactor coolant system. Zircaloy 4 cladding is particularly vulnerable to this, and IRSN believes that EDF should limit the conditions in which it is used or should change the materials used for fuel rod cladding.

Corrosion of the cladding of fuel assemblies

The cladding of reactor fuel assemblies is the first containment barrier for the fission products. During its time in the reactor—around four or five years—the cladding of the fuel rods corrodes from contact with the water in the reactor coolant system. The corrosion of the cladding involves:

- the formation of a layer of zirconium dioxide (zirconia) on the surface of the cladding in accordance with the oxidation reaction: \( \text{Zr} + 2\text{H}_2\text{O} \rightarrow \text{ZrO}_2 + 2\text{H}_2 \); once the zirconia layer has reached a certain thickness, localised peeling of the layer, known as spalling, can occur (Figure 4.11);
- the absorption by the cladding of some of the

![Fig. 4.11 / Areas of spalling in fuel rod cladding.](https://www.irsn.fr/FR/expertise/rapports_gp/gp-reacteurs/Pages/Synthese-Rapport-IRSN-Management-Surete-Radioprotection-Arret-Tranche.aspx)
hydrogen released during the oxidation reaction; the hydrogen in the cladding migrates towards the areas where there is spalling and precipitates in the form of hydrides, which make the cladding brittle locally.

The cladding of fuel assemblies has for many years been made from zircaloy 4, a zirconium-based metal alloy containing tin and other alloying elements. This alloy is very vulnerable to corrosion. Over time EDF has increased the amount of time fuel assemblies spend in reactors, which has led to an increase in the thickness of the zirconia layer on fuel rod cladding and has therefore increased the risk of spalling of this cladding. For this reason, EDF introduced some new alloys a few years ago, which are much more resistant to corrosion.

However, there are still fuel rods with zircaloy 4 cladding in reactors currently in operation (more than 80% of France’s reactors). Consequently, the effects of spalling should be covered by safety assessments, particularly as regards control rod ejection accidents.

Control rod ejection accidents

By means of their partial or total insertion into the reactor core, control rod assemblies are used to control the reactor and bring a rapid halt to the nuclear reaction. Control rod ejection accidents can occur because of a difference in pressure between the reactor coolant system (155 times atmospheric pressure in normal operation) and the atmosphere inside the reactor containment (atmospheric pressure). Ejection leads to a rapid temporary localised runaway of the nuclear reaction; the temperature of the fuel assemblies near the ejected control rod increases and the thermal expansion of the fuel pellets in the cladding places significant amounts of stress on the cladding. Safety assessments aim to demonstrate that the fuel rod cladding will not fail in these circumstances.

Lessons learned from the experimental programme run in the CABRI reactor

The experimental programme run in the CEA’s CABRI reactor by IRSN and several partners, from the 1990s until 2003, studied the behaviour of fuel rods in control rod ejection accidents. Several alloys were tested, including zircaloy 4. The tests carried out showed that embrittlement due to spalling could lead to the premature failure of the affected cladding during an accident of this kind (Figure 4.12) compared with cladding unaffected by spalling. However, the degree of embrittlement caused by spalling cannot be estimated and the risk of clad failure can therefore not be assessed.

Spalling of zircaloy 4 in French reactors

During the 2000s, EDF considered replacing the fuel assemblies with zircaloy 4 cladding with fuel assemblies with M5 alloy cladding (an alloy developed by Areva). For this reason, EDF did not carry out any detailed inspections of the condition of zircaloy 4 fuel rod cladding. However, manufacturing problems (see page 44 of the 2008 PWR public report) meant that using M5 alloy as a fuel rod cladding had to be postponed.

At the same time, a number of technical discussions were taking place between EDF and IRSN concerning the zirconia thickness from which spalling was likely to occur in a reactor. Based on the examination of fuel rods with zircaloy 4 cladding irradiated in reactors, in 2013 EDF reached the conclusion that, in some cases, indications of spalling in cladding could be seen in zirconia thicknesses above 80 µm. It also appeared to be the case that the cladding of a number of fuel rods in EDF reactors was showing signs of spalling.


IRSN therefore felt that EDF should take measures to protect against the risk of spalling. So in 1999 the French Nuclear Safety Authority (ASN) asked EDF to check that there was no spalling of the cladding of fuel rods in its reactors.
Conclusion

IRSN felt that operating restrictions should be placed on the reactors concerned in order to:

› limit the zirconia thickness in zircaloy 4 cladding (e.g. by reducing the amount of time the fuel rods spend in reactors);

› limit the consequences of a control rod ejection accident by running the reactors concerned with the control rod assemblies held in the highest possible position in the reactor core; the temperature reached in the fuel in the event of a control rod ejection accident increases with the depth of insertion into the core of the control rod assembly assumed to have been ejected.

At the ASN’s request, EDF took compensatory measures at reactors using fuel with zircaloy 4 cladding. These measures have been applied since July 2014. (For more about IRSN’s opinion, click here*).

EDF is also planning to replace all fuel assemblies with zircaloy 4 cladding with fuel assemblies using alloys less vulnerable to corrosion. Because the length of time fuel assemblies spend in a reactor is around four or five years, their replacement with fuel assemblies using new alloys for the fuel rod cladding should be completed by 2020 at all French nuclear reactors.

**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)

**1450 series**
See N4 series

**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 gramme of radium; 1 Ci = 3.7×10^{10} disintegrations per second, or 37 billion Bq (or 37 billion disintegrations per second)

**CCWS**
Component Cooling Water System

**Control valve**
Valve whose opening can be controlled (to regulate flow passing through it)

**Corrective maintenance**
All operations performed in order to restore the capability of failing equipment

**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 Dampierre-en-Burly, four at Blayais, four at Chinon, four at Cruas-Meysse and two at Saint-Laurent-des-Eaux)

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

**CVCS**
Chemical and Volume Control System

**DER**
Dose equivalent rate, often termed “dose rate”, stated in mSv/h
**Effluent**
Any liquid or gaseous fluid from the facility that may be released directly or indirectly into the receiving environment.

**EFWS**
Emergency Feedwater System

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

**ESWS**
Essential Service Water System

**Fallback state of a reactor**
Control operation consisting of bringing the reactor to a state that is safer than the initial state (in which an anomaly was detected, for example)

**FB**
Fuel Building

**FPCPS**
Fuel Pool Cooling and Purification System

**Fuel assembly burnup**
Burnup is the thermal energy released per unit of mass of a nuclear fuel

**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

**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 kilogramme 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

**International Nuclear Event Scale (INES)**
It applies to events occurring at nuclear facilities and defines seven levels of severity based on event consequences

**Isolation valve**
Valves that are either in open or closed position (isolation function)

**I&C**
Instrumentation and control

**LOCA**
Loss-of-coolant accident

**LP Turbine**
Low pressure turbine cylinder

**Maintenance**
All actions that consist of maintaining or restoring equipment in a specified state or capable of ensuring a specific service

**MCR**
Main Control Room

**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 or Penly; four, as at Bugey and Dampierre; or six, as at Gravelines)
Preventive maintenance
All operations performed on available equipment to avoid later failure or reduce the probability of such failure; these operations, organised in advance, are part of maintenance programmes.

Sievert (Sv)
Unit used to estimate biological effects produced by radiation on an exposed organism (a function of its nature and exposed organs). Since this unit is very large, a submultiple of the Sv, the millisievert (mSv), which equals 10^{-3} Sv or 1 thousandth of an Sv, is often used. The dose equivalent rate is likewise stated in millisieverts per hour (mSv/h). The Sv replaced the rem; 1 Sv = 100 rems.

SIS
Safety Injection System; safeguard system used in loss-of-coolant accident situations

System alignment
Configuration of a system to make it available for operation, e.g., by controlling valves and switching electrical equipment on or off.