IRSN's Position on Safety and Radiation Protection at Nuclear Power Plants in France, 2012

IRSN report DG/2013-00005-EN
Enhancing Nuclear Safety, Security and Radiation Protection

The Institute for Radiological Protection and Nuclear Safety set up by law 2001-398 of 9 May 2001, is the French national public expert in nuclear and radiological risks. IRSN contributes to the implementation of public policies concerning nuclear safety and security, health and environmental protection against ionizing radiation. As a research and expert appraisal organisation, IRSN works together with all the parties concerned by these policies while preserving its independence of judgment.

THE FRENCH ORGANISATION FOR NUCLEAR SAFETY, SECURITY AND RADIATION PROTECTION

- 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 analyses 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 (HCITSN) gather the stakeholders concerned by nuclear facilities and constitutes leading bodies for access to information and monitoring of safety and security, health and environmental protection issues.

IRSN KEY FIELDS OF COMPETENCE - R&D AND OPERATIONAL EXPERTISE CAPABILITY

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

IRSN key numbers
1,786 persons
1,200 researchers and experts
ME\$21 (2010 budget)
Performed using data transmitted by EDF to ASN and IRSN after each minor and major event, our analysis helps further safety by drawing lessons from the most significant events and noting trends that help guide IRSN’s studies and research.

Jacques Repussard
IRSN Director-General

IRSN devotes considerable resources to its ongoing technical safety monitoring of France’s 58 nuclear power plants.

After working a year to consolidate the data, IRSN is once again publishing an annual summary of its monitoring effort and providing its independent point of view on all progress and problems concerning safety and radiation protection encountered in the French nuclear power plant fleet in 2012.

There was a noteworthy increase in the number of significant events, the great majority of which had only a slight impact on facility safety. IRSN notes however that the constantly increasing number of EDF teams involved in detecting deviations contributes to this increase and the maintenance of a high level of safety in the French nuclear power plant fleet during a period of rapid transition to a new generation of engineers and operations and maintenance technicians.

In this report, IRSN is committed to providing its analysis of several events that it found to be the most noteworthy and of “generalised” anomalies, i.e., found in several reactors in the same series or across several series.

Nuclear power reactors in France are regularly modified to improve safety, particularly during ten-year safety reviews, which include new safety requirements and related changes. The March 2011 accident at the Fukushima Daiichi nuclear power plant in Japan led to a reassessment of the robustness of nuclear facilities in France and of preparedness for extreme situations that are highly unlikely but plausible and not taken into account in the design of facilities. In this context, IRSN presents the results of its analysis in 2012 of EDF’s proposed “hardened safety core” measures for equipment and organisation.

We hope you find this report useful and look forward to your feedback to help us improve it for the future.
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INTRODUCTION AND OVERVIEW

This report presents IRSN's overall assessment of the radiation protection and safety performance of currently operating EDF nuclear power plants for the year 2012. The report consists of three parts. As in previous years, it does not aim to be exhaustive, but rather to focus on those aspects that IRSN considers most significant for the year 2012.

The first part of the report presents the main trends that emerge from IRSN's overall assessment of the radiation protection and safety performance of currently operating nuclear power plants for the year 2012. The second part presents a number of events, incidents or faults reported in 2012, among those deemed particularly significant by IRSN. The third part of the report focuses on safety-related issues which have been the subject of extensive analysis and review by IRSN, including significant changes or modifications in NPP design or operation to address safety concerns and, in some cases, economic concerns.

Achieving optimal safety and radiation protection performance requires constant vigilance on the part of all those involved and must remain an absolute priority, with the operator assuming full responsibility for continuously improving the safety of its facilities. IRSN considers that such continuous improvement is achieved first and foremost through careful analysis and consideration of national and international operating experience, along with new scientific knowledge gained from research.

IRSN notes the absence in 2012 of incidents with a potentially significant impact on nuclear power plant safety, the surrounding environment or nearby communities. However, the year 2012 has witnessed an increase in the total number of significant safety-related events. One of the main trends observed is an increased percentage of reported events associated with the detection of nonconformances affecting several types of reactors. This increase can mainly be explained by EDF's implementation of an improved nonconformance detection and handling procedure (presented in the last part of this report) which led to the identification, in 2012, of various nonconformances presumably present for several years but previously undetected. It is also worth noting that inconsistencies among the number of significant safety-related events reported per nuclear power plant have decreased progressively between 2009 and 2012.

The number of events associated with periodic testing remains stable since 2011, whereas the number associated with inadequate equipment maintenance or modification actions has steadily increased over the past several years (308 reported in 2011, 395 in 2012), in a context of large-scale personnel renewal. Since the majority of maintenance activities are subcontracted, in 2012 EDF undertook an overhaul of its subcontractor monitoring procedures, with operational implementation in the various sites scheduled for completion by 2013.

With regard to radiation protection, IRSN notes that the effective dose received by the majority of exposed workers over a period of 12 consecutive months is below the annual public radiation dose limit (1 mSv). No individual worker received a radiation dose of 16 to 20 mSv (regulatory dose limit) over a period of 12 consecutive months (as opposed to 2 workers in 2011, 3 in 2010, and 10 in 2009). This could be a beneficial effect of lowering the early-warning threshold level from 16 to 14 mSv, but this assumption has yet to be confirmed for the year 2013. The annual collective dose to workers has slightly decreased and is close to the 2010 value (0.67 H.Sv per reactor in 2012, as opposed to 0.71 H.Sv per reactor in 2011 and 0.62 H.Sv per reactor in 2010). The year 2012 has
also witnessed an increase in the number of significant radiation protection-related events reported for EDF nuclear power plants. This increase mainly concerns events associated with gamma radiography inspections and failure to comply with dosimeter badge-wearing requirements or periodic inspection deadlines for mobile radiation protection devices. Further progress therefore remains to be achieved in these areas. On the other hand, the number of events associated with risk assessment errors during work preparation remains stable since 2010.

Faults may occur with nuclear power plant equipment or reactor monitoring systems. Given the standardisation of EDF nuclear power plant reactors, such faults may affect an entire reactor series or even the entire reactor fleet. Certain types of faults may have a significant or potentially significant adverse impact on reactor safety and are therefore subject to analysis and follow-up action by IRSN. A few examples deemed particularly significant by IRSN are presented in the second part of this report, including a discussion of several incidents reported in 2012 involving the detection of migrating foreign bodies inside equipment.

French nuclear reactors are subject to modifications throughout their operating lives, particularly with a view to ensuring continuous safety improvement. Most of these modifications are the result of studies conducted within the framework of ten-yearly safety reviews, leading to the definition and implementation of new safety requirements and associated changes. Certain issues analysed in parallel with said safety reviews may also lead to changes. This is the case, for example, with the new guidelines document for the protection of basic nuclear installations against external flooding (prepared as part of the response to a nationwide storm that caused partial flooding of the Blayais site in late 1999) or the intense heatwave baseline requirements document (based on lessons learned from the drought period of 2003). The relevance of such an approach has been clearly demonstrated during the definition of NPP protection measures against natural hazards exceeding design-basis levels (‘hardened safety core’), as part of the post-FUKUSHIMA review process. These issues have been the subject of detailed analysis by IRSN in 2012.
OVERALL ASSESSMENT OF SAFETY AND RADIATION PROTECTION PERFORMANCE OF EDF NUCLEAR POWER PLANTS IN OPERATION

The way to manage reactors' operation is a determining factor for ensuring continuously optimal radiation protection and safety performance. IRSN's assessment of the safety and radiation protection performance of currently operating EDF nuclear power plants is based on the analysis of various data obtained through continuous monitoring of reactor operation. Data obtained from events and incidents affecting national as well as foreign nuclear facilities constitute one of the main sources of applicable operating experience. In order to produce an overall assessment of the safety and radiation protection performance of currently operating nuclear power plants, IRSN has developed a number of operating experience analysis tools, methods and indicators (see IRSN public report 2007, in French, page 10). These techniques significantly contribute to the identification of both general and reactor-specific trends and deviations in safety and radiation protection performance. The two sections that follow present the main lessons to be drawn from IRSN's overall assessment of safety and radiation protection performance for the year 2012.
Safety performance: Main trends

Although the year 2012 has witnessed an increase in the number of significant safety-related events (approximately +13% as compared to 2011), the vast majority of events reported had a very low effective impact on nuclear power plant safety and were handled without issues. The ever-increasing efforts of EDF teams to detect nonconformances at both the site-specific and national level may partly explain the observed increase.

To be noted in particular is the increasing number of significant safety-related events associated with transient operating conditions (on average, approximately 1 per reactor in 2012, as compared to 0.8 in 2011), despite an observed decrease in the number of deviations from plant operating parameters. This trend, together with the increasing number of inadequate maintenance actions, shows the difficulties faced by EDF in maintaining the optimal level of technical expertise needed to perform and monitor maintenance activities in a context of large-scale personnel renewal.

Significant safety-related events

Operators of nuclear installations must report all safety, radiation protection, environmental and transport-related events to the Nuclear Safety Authority (ASN) within 48 hours after detection. The term ‘significant safety-related events’ (ESS) is used herein to refer to events with a potentially significant impact on nuclear power plant safety. The term ‘significant radiation protection-related events’ (ESR) is used herein to refer to ionising radiation exposure events posing a potential threat to the health of exposed workers. Significant environmental-related events (ESE) and transport-related events (EST) are beyond the scope of this report.

Significant events are investigated as part of the general review of operating experience from currently operating nuclear power plants. Such events are subject to detailed analysis by the operator upon detection, leading to the definition and subsequent implementation of appropriate measures to prevent them from reoccurring. Significant events are reported for reasons of transparency, and also to allow operating experience to be shared among nuclear entities and organisations. They are therefore subject to review by IRSN for the purpose of identifying valuable lessons at the national or even international level.
The year 2012 has been marked by an increase in the number of significant safety-related events reported by French nuclear power plant operators. On average, approximately 13 significant safety-related events were reported per reactor in 2012, as compared to 11 in 2010 and 2011, and 13 in 2009. This increase amounts to approximately 40% for Level 1 events, and approximately 10% for Level 0 events. The main factors possibly responsible for this increase are discussed below. In 2012, a total of 103 significant safety-related events were reported as Level 1 events, and only one of these events was reclassified as Level 2 by the Nuclear Safety Authority (ASN).

The significant safety-related event reclassified as Level 2 corresponds to the reported absence of siphon breakers in the spent fuel pool cooling system piping of two Cattenom NPP reactors. It is addressed in detail in the present report (see page 32).

It should be noted that inconsistencies among the number of significant safety-related events reported per reactor have decreased progressively between 2009 and 2012. However, the Civaux nuclear power plant, which is the most recent among those currently operating, shows a different trend (observed since 2009), with 18 significant safety-related events reported for each of its two reactors in 2012. As a result, in 2013, ASN chose the Civaux site to conduct a thorough review of good operating practices, with technical support from IRSN.

The International Nuclear Event Scale (INES) is used to classify safety-related events in nuclear power plants according to 7 levels, with Level 0 events corresponding to deviations.

Relevance of number of significant safety-related events reported: For IRSN, the number of significant safety-related events reported does not serve as a quantifying measure of good operating practices, and variations in this number cannot be directly associated with a variation in safety level. These events are indicative of issues to be interpreted and investigated with a view to identifying relevant strategies for improving plant safety and operation.
On the other hand, in 2012, the operators of the Penly and Saint-Laurent B nuclear power plants reported the lowest number of events (on average, approximately 7.5 significant safety-related events per reactor, which represents a significant decrease compared to 2011).

One of the main trends observed in 2012 is the increasing share of significant safety-related events associated with the detection of nonconformances, which amounted to 13% of all significant safety-related events reported in 2012, as compared to 7% in 2011. Most of these events were 'generic' in that they involved common-mode faults affecting several reactors. This increase can mainly be explained by the recent implementation of the improved nonconformance detection and handling procedure presented in this report (see page 71). For example, in 2012, this procedure led to the identification of various nonconformances that had previously remained undetected for several years.

**Decrease in number of deviations from authorised operating conditions**

In 2012, 30 significant safety-related events concerned unintentional deviations from authorised operating parameters (as compared to 50 in 2011), which amounts to an average of 0.55 per reactor. The observed decrease is a clear indication of the increased vigilance exercised by nuclear power plant operators in monitoring and controlling plant operating parameters.

Indeed, the year 2012 has witnessed the lowest number of reported events of this type since 2009. Another trend worth noting is the steady decrease in the duration of deviations from authorised operating parameters. In 2012, 90% of such deviations were detected and corrected in less than 15 minutes, which is indicative of an improved responsiveness on the part of operating teams since 2010.

In most cases, these deviations correspond to a brief overshoot/undershoot of primary coolant pressure and temperature limits. Temperature’s overshoots remain very limited (approximately 2°C) and occur for the most part when the reactor is in power operation. Pressure overshoots occur for the most part during delicate reactor shutdown procedures involving manual pressure control.
High number of significant safety-related events during management of transient operating conditions

The management of transient operating conditions poses a high risk of deviation from authorised operating parameters and requires good knowledge of equipment availability.

In 2012, the number of significant safety-related events directly associated with the management of transient operating conditions amounted to an average of one per reactor (as compared to 0.8 in 2011). This mainly concerns delicate restart procedures, particularly power buildup sequences to return the reactor to power operation state after a scheduled maintenance outage.

The analysis of significant safety-related events experienced during management of transient operating conditions reveals organisational insufficiencies in three main areas:

- proper understanding of relevant physical phenomena and associated operating principles. Specific measures such as the implementation of improved preparation procedures, the constant presence of experienced operators among shift personnel or the organisation of simulator training sessions should contribute to ensuring good understanding of relevant physical phenomena and proper implementation of associated operating procedures. Certain operating contingencies and work schedule constraints (for example, during holiday periods) may prevent the implementation of such measures. Nevertheless, particular note should be given to the significant efforts currently planned to improve operating practices within the scope of the annual training plan for 2014;

- proper preparation and implementation of operating manuals. The operating manuals available in the control room must provide operating teams with proper guidance, risk-related warnings and rapid diagnostic procedures. However, in certain cases, these documents may not serve their intended purpose, either due to incompleteness or due to improper interpretation by operating teams;

- robust and efficient organisation of operating teams under all circumstances. In practice, the distribution of work tasks among operating personnel may sometimes be adjusted on a real-time basis in response to operating contingencies or significant workloads, particularly during maintenance outages. Under such circumstances, plant monitoring personnel may lack the necessary perspective for effective team supervision and plant management.

In most cases, it is the combination of several factors which leads to the occurrence of a significant safety-related event during management of transient operating conditions.

Example of deviation from authorised operating parameters during transient operating conditions

On 23 December 2012, the Cruas NPP Unit 3 operating team initiated a power buildup sequence as per the power grid load management plan. Two successive primary coolant dilution actions were implemented to ensure slow power buildup. This resulted in a deviation of in-core power distribution parameters from specified values, which was corrected in less than 10 minutes through implementation of additional actions.

In this example, as in many others, transient operating conditions were particularly unfavourable, with power build up strongly anticipated by the power grid. During the transient, operators in the main control room lost access to the computerised plant control system (system to help the operating team). Moreover, they failed to correctly interpret observed physical phenomena and were unable to anticipate the imminent deviation from authorised operating conditions prior to initial alarm activation.

This event has been presented to all EDF operating teams during plant control training sessions organised as part of the operating experience review programme.
Monitoring of equipment performance

Increase in number of equipment malfunctions

The year 2012 has been marked by an increase in the number of reported equipment malfunctions. The number of electrical equipment malfunctions has steadily increased over the past several years. IRSN has therefore carried out a power source performance and reliability analysis whose conclusions will be presented to ASN experts in 2014. Also to be noted is the increase in the number of auxiliary feedwater system malfunctions.

Effectiveness of nonconformance detection methods

The percentage breakdown of methods used to detect nonconformances leading to significant safety-related events has remained stable between 2011 and 2012. Operating teams play an essential role by performing alarm acknowledgement tasks, daily monitoring of plant operating parameters and routine plant inspection rounds. Alarm activation is the most important detection method.

Operator inspections of equipment-related tasks are classified by IRSN according to two independent levels:

- Level 1 corresponds to inspections performed during execution of equipment-related tasks (personnel self-check, technical inspection, etc.)
- Level 2 corresponds to inspections performed after execution of equipment-related tasks (documentation reviews, etc.)

Level 1 and Level 2 inspections

All equipment is subject to the degradation of certain physical properties, even when used as intended by design. However, in addition to this inevitable degradation, damage may be caused by non-compliant use or intrusive/inadequate actions. All activities involving nuclear power plant equipment are therefore subject to a series of independent Level 1 and Level 2 inspections to ensure rapid detection and correction of malfunctions.
The share of Level 2 inspections in the detection of nonconformances exceeds that of Level 1 inspections, since nonconformances which are rapidly detected and immediately corrected (Level 1) do not need to be systematically reported as significant safety-related events. As a result, such nonconformances are not considered in detail during operating experience reviews.

The non-negligible share of ‘vigilance actions’ in the detection of nonconformances is worth nothing. The effectiveness of this detection method (independent of organisational structure) is indicative of the safety culture of plant personnel (e.g. daily reporting of observed nonconformances, regardless of direct responsibility). This leads to considering the complementarity of organised monitoring activities, on the one hand, and vigilance actions performed individually by all plant personnel, on the other hand.

**Decrease in number of significant safety-related events due to non-compliance with periodic testing requirements**

Periodic tests are conducted to regularly monitor the availability of safety-related equipment. The definition of a periodic test plan (including individual test criteria, conditions and frequency as per general operating rules) is absolutely essential, as is full and effective compliance on the part of operators.

The period from 2009 to 2012 has witnessed a decrease in the number of significant safety-related events due to errors in the preparation of test rules and procedures contained in operating manuals. Prior to 2007, the preparation of these documents was left up to the initiative of individual sites. Since 2007, new standard operating procedures are prepared by a pilot site (operating procedure standardisation process) so as to reduce site-specific workloads and risks of inconsistent documentation among sites. These documents are validated (by a site other than the pilot site) prior to distribution to all sites equipped with the same type of reactor. This validation process is intended to detect possible errors prior to implementation in all relevant sites. After an initial period of preparation and implementation of new operating procedures (2008-2009), this new organisational structure began to bear fruit in 2010.

Moreover, given the significant number of periodic tests to be performed on a nuclear reactor (several tens of thousands), with testing frequencies ranging from daily to ten-yearly, the new organisational structure shows a certain robustness in terms of timely completion of periodic tests (see figure 5), although there is still room for improvement (particularly with regard to risks of schedule drifts further to unexpected events).

Like any equipment-related activity, periodic testing poses a risk of equipment degradation due to inadequate action by relevant personnel. A decrease in the number of such inadequate actions has been observed between 2011 and 2012, which is indicative of improved control of periodic test rules and procedures.
Steady increase in number of inadequate equipment maintenance or modification actions

Like previous years, the year 2012 has been marked by a very large increase in the number of significant safety-related events associated with inadequate equipment maintenance or modification actions (308 in 2011, 395 in 2012).

Maintenance activities comprise all actions needed to maintain or restore equipment in or to a specified state so as to perform a given function.

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

**Corrective maintenance** comprises all actions needed to restore correct operation of malfunctioning equipment.

Analyses of these events frequently note a lack of competence. Faced with the retirement of large numbers of experienced personnel, operators need to train new personnel.

In this context, site managers should increase their efforts to monitor maintenance activities during scheduling, preparation or execution phases, particularly in cases where real-time detection of inadequate maintenance actions is not possible. Since the majority of maintenance activities are subcontracted, in 2012 EDF undertook an overhaul of its subcontractor monitoring procedures, with operational implementation in the various sites scheduled for completion by late 2013.

**Example of inadequate action during preventive maintenance**

On 28 May 2012, during a scheduled refuelling outage of the Cruas NPP Unit 4 reactor, the operator conducted an internal inspection of a safety injection valve as per the preventive maintenance plan. In accordance with standard post-maintenance practices, this valve underwent both intrinsic requalification (verification of non-modification of equipment performance) and functional requalification (verification of proper equipment operation in current configuration).

Intrinsic requalification test results demonstrated the valve’s compliance with applicable requirements, whereas functional requalification test results led to the identification of valve leakage due to an assembly error not detected during intrinsic requalification testing. The analysis of this event revealed a lack of knowledge of valve-specific technology (not widely implemented in currently operating plants) on the part of subcontractor personnel, not compensated by the requalification test procedure. Moreover, Level 1 and Level 2 inspections were equally ineffective in detecting the nonconformance. In order to prevent such nonconformances from reoccurring, the operator has undertaken a modification of the requalification test procedure and has requested that the subcontractor improves its level of competence.
Fault recovery performance

Faults detected in nuclear installations need to be handled very quickly so as to minimise their impact. In certain cases, fault recovery is performed automatically, with threshold overshoot events (e.g. high feedwater level in steam generators) or specific reactor operating conditions (e.g. turbine shutdown when nuclear power output exceeds rated output by 30%) automatically triggering reactor protection functions (turbine trip, reactor scram).

Since 2011, approximately 7% of significant safety-related events are associated with reactor scrams. Although a reactor scram is an expected automatic response to bring the reactor to a safe shutdown state, it should be noted that when it occurs while the reactor is in power operation, it can cause significant mechanical stresses on certain reactor components and generate large amounts of effluents. Therefore, since 2007, EDF has implemented specific corrective actions that have significantly reduced the number of reactor scrams.

Faults that do not automatically trigger reactor protection functions need to be handled by the operator. As soon as a fault has been clearly identified, its severity level is evaluated. In the vast majority of cases, the operator disposes of a certain amount of time to address the fault (amount of time specified as per general operating rules or determined directly by the operator). The fault-recovery actions most frequently implemented in nuclear installations are corrective maintenance or operating actions performed by operating teams to restore compliant operation. Unlike operating actions (which can be completed in a few hours), equipment maintenance actions require extensive preparation and may take several days to complete. The number of significant safety-related events requiring the implementation of maintenance actions to return the reactor to a safe state has increased significantly between 2011 and 2012, whereas the number requiring the implementation of operating actions has decreased over the same period.

It should be noted that, in cases where the fault handling time specified by general operating rules cannot be observed, these same rules may require the beginning of reactor shutdown (i.e. transition from one operating state to another). Fallback mode procedures amounted to 3 to 4% of fault-recovery actions in 2012.

Figure 6: Percentage breakdown of fault-recovery actions in 2011 and 2012

Fault recovery actions are intended to return the reactor to a safe state as per safety demonstration criteria. For this purpose, the operator may either perform direct actions to restore equipment availability (maintenance or operating actions) or bring the reactor to an operating state where equipment availability is not required for maintaining a safe state as per safety demonstration criteria (fallback mode, reactor scram).
Radiation protection performance: Main trends

The year 2012 has witnessed an increase in the number of significant radiation protection-related events reported for EDF nuclear power plants. In particular, the assessment conducted by IRSN indicates an increase in the number of events associated with gamma radiography inspections and personal and collective dosimetry management. Despite the encouraging results of EDF efforts for ensuring improved personal and collective dosimetry management, the implementation of best radiation protection practices by all relevant personnel remains a major area for improvement.

Significant radiation protection-related events

Regulations regarding the protection of workers exposed to ionising radiation require basic nuclear installation operators to report significant radiation protection-related events to the Nuclear Safety Authority (ASN) as per reporting criteria (see table below).

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<th>Significant radiation protection-related event (ESR) reporting criteria</th>
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EDF examines the context, causes and radiological impact of all reported events and implements corrective actions to prevent such events from reoccurring. The results of these analyses are submitted to ASN and IRSN experts. IRSN uses this information to assess proposed measures, follow-up on reported events and identify overall trends for currently operating plants.

The year 2012 has witnessed an increase in the number of significant radiation protection-related events reported by EDF (112 in 2012, as opposed to 97 in 2011 and 86 in 2010). These events are classified according to event type in the chart shown below. The main lessons learned are described further below.

Operations conducted at the bottom of the refuelling cavity

Effective and equivalent doses

The effective dose is used to estimate whole-body radiation exposure based on body tissue sensitivity and exposure type (alpha, beta, gamma, and neutron). The equivalent dose is used to estimate the radiation exposure of individual organs. Effective and equivalent doses are expressed in Sieverts (Sv).

Regulatory dose limits:
- Effective dose limit for members of the public: 1 mSv/year (excluding natural and medical radiation exposure).
- Effective dose limit for exposed workers (over a period of 12 consecutive months): 20 mSv

Equivalent dose:
- Extremities (hands, forearms, feet and ankles): 500 mSv
- Skin: 500 mSv
- Cystalline: 150 mSv

Among the significant radiation protection-related events reported in 2012, two have been classified as Level 1 according to INES criteria. These events involved localised head-region exposure to radiation levels exceeding 50% of the regulatory annual dose limit (500 mSv). They occurred during removal of sealed, ventilated protective clothing used to perform activities at the bottom of the refuelling cavity. The first case concerned a worker directly exposed during removal of protective clothing. The second case involved indirect exposure to a non-radiation protected phone located near the changing room area and used by subcontractor personnel after radiation exposure. Exposed workers were immediately admitted by medical services to remove radioactive particles. Degowning operations pose a high risk of contamination transfer and must be carried out in well-delimited areas as per specific procedures. The analysis of these two events revealed inadequate degowning conditions (cramped spaces with insufficient radiation shielding).
Decrease in number of unauthorised access events in radiation-controlled areas

In accordance with applicable regulations, nuclear power plant radiation protection personnel are responsible for implementing a radiological zoning scheme based on measured dose equivalent rates and a three-colour classification system. The largest share of unauthorised access events corresponds to ‘orange areas’ (radiation-controlled areas with dose equivalent rates possibly exceeding 2 mSv/h), despite the decrease observed in 2012. Such events may result in worker exposure, annual dose limit overshoots or unauthorised access to orange areas.

Fixed-term and temporary workers are assigned a specific status. According to Article D.4154-1 of the French Labour Code, such workers are not authorised to work in orange areas. A number of significant radiation protection-related events are due to non-compliance with this regulation for various reasons (inadequate work preparation, non-identification of orange areas near workstations, etc.).

EDF has identified two main areas for possible improvement: on the one hand, the implementation of an overall strategy for improved identification, mitigation and protection of ‘radiation hotspots’ and, on the other hand, the reinforcement of preparatory work for activities potentially leading to radiation exposure of temporary workers. In addition, EDF has implemented a number of measures to deny such workers access to orange areas (in particular, since early 2010, the alarm threshold of dosimeters worn by such workers has been reduced from 2 to 1.6 mSv/h).

Moreover, requirements associated with exceptional working conditions in ‘red areas’ have been strengthened in accordance with applicable regulations. Events involving non-compliance with such requirements (potentially causing significant impact) remain limited in number, although not completely eliminated (less than 5 events recorded per year since 2008).

New increase in number of events associated with gamma radiography inspections

Gamma radiography requires the use of strong radiation sources. Nonconformances may therefore result in significant worker exposure.

IRSN notes that the number of events associated with gamma radiography inspections has increased in 2012, whereas it appeared to remain stable during previous years.
The largest share corresponds to gamma radiography inspections in turbine hall areas. The potential for interference with other onsite activities is extremely high, particularly in a context of nonconformance handling procedures with on-the-fly update of work schedules. For example, in one event, ultrasonic testing personnel inadvertently entered a restricted area reserved for X-ray inspections. Following the identification of valve weld defects, a series of inspections were conducted, each scheduled on-the-fly to save time. Due to the limited preparation time required, established inspection procedures were not fully observed, all personnel were not correctly notified, and the restricted area was extended excessively, making it more difficult to monitor. Collateral activities were not detected and exposed personnel were not alerted by audio signals, as they were wearing ear plugs. This example illustrates the need to remain attentive to established work schedules and to the coordination of collateral activities in case of contingencies.

**Absence of variation in the number of events associated with inadequate radiation protection assessment**

The number of significant radiation protection-related events associated with risk assessment errors during work preparation remains stable since 2010 and amounts to approximately 13% of reported events.

Prior to work execution, a forecast dosimetry assessment is prepared in order to identify potential risks and implement suitable operational measures. This rigorous preparation process is formalised in a radiation work procedure document intended for relevant personnel. Nevertheless, several cases of inconsistencies between said document and actual radiological conditions have been reported in 2012. Such inconsistencies can lead to underestimation of radiological risks and subsequent implementation of insufficient or inadequate radiation protection measures. The assessment therefore needs to be conducted in a rigorous manner (to detect potential hotspots), taking into account the same configuration as during work execution (to ensure consistency with actual radiological conditions).

**Worker dosimetry monitoring**

Individual dosimetry monitoring is essential for ensuring effective radiation protection of workers exposed to ionising radiation. It provides whole-body exposure measurements used to demonstrate compliance with regulatory dose limits and actively contributes to the implementation of ALARA principles by monitoring individual and collective doses so as to detect non-compliances as early as possible during work execution.

All workers accessing radiation-controlled areas are required to wear passive and operational dosimeters. The year 2012 has been marked by an increase in the number of non-compliances with this basic rule. Nevertheless, most individuals quickly notice the omission on their own.

Mobile radiation protection devices are also very useful for measuring plant and equipment exposure levels during and after work execution. These devices are periodically inspected to ensure proper operation. The year 2012 has witnessed a significant increase in the number of events associated with failure to comply with periodic inspection deadlines. One site operator detected a total of 54 such non-compliances in 2012. This is indicative of a lack of organisation in the local management of radiation protection and measurement systems needed to protect personnel from abnormal exposure to ionising radiation.
Every year, EDF carries out an assessment of radiation doses to EDF and subcontractor personnel based on operational dosimetry results. The year 2011 had been marked by an increase in the annual collective dose, whereas 2012 has witnessed a decrease back to the level observed in 2010 (0.67 H.Sv per reactor in 2012, as opposed to 0.71 H.Sv per reactor in 2011 and 0.62 H.Sv per reactor in 2010), attributable to a decrease in number of maintenance operations as compared to 2011.

Regarding individual doses, IRSN notes that the effective dose received by the majority of exposed workers over a period of 12 consecutive months is below the annual public radiation dose limit (1 mSv). Moreover, no individual worker received a radiation dose of 16 to 20 mSv (regulatory dose limit) over a period of 12 consecutive months (as opposed to 2 workers in 2011). This could be a beneficial effect of lowering the early-warning threshold level from 16 to 14 mSv, but this has yet to be confirmed for the year 2013.

The analysis of passive dosimetry results for the year 2012 shows that 90% of EDF personnel received individual doses below 1 mSv, with the average individual dose amounting to 0.29 mSv (as compared to 0.30 mSv in 2011). The internal radiation exposure incurred by nuclear power plant employees is mainly monitored by whole-body radiometric examinations. The number of EDF and subcontractor employees who underwent whole-body radiometric examinations in 2012 has increased by 13% (as compared to 2011), with a total of 174,270 examinations conducted within the scope of routine medical surveillance procedures, and 6,309 for individual follow-up purposes. A total of 0.3% of these examinations confirmed internal exposure.

*Personal dosimetry* comprises external and internal dosimetry.

*External dosimetry* involves measuring the doses received by a person exposed to a field of radiation (X-rays, gamma, beta, neutrons) 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 have an audio and visual alarm that alerts workers if they are in a field of radiation that exceeds certain thresholds.

*Internal dosimetry* measures the dose received as a result of incorporating (inhaling or ingesting) radioactive substances. This type of dosimetry involves whole body radiation measurements (direct measurement of internal contamination) and radiotoxicological tests.
EVENTS, INCIDENTS AND ANOMALIES

No incident that occurred in 2012 in an EDF reactor represented a serious risk for the facility, the environment or people. This chapter presents events and anomalies that IRSN judges significant and which concern the discovery of loose parts inside the coolant systems, the detection of deviations relating to equipment fixings, non-compliance dating from the facilities’ commissioning, and line-up errors.

Despite the actions that EDF has taken for some years, several events linked with the presence of loose parts in the systems occurred in 2012.

A screw head belonging to a reactor coolant pump component in Chooz B2 was discovered under a fuel assembly when the reactor core was unloaded, and this screw damage caused a generic event for the 1450 MWe reactors.

At Cruas 3, the reactor coolant system’s acoustic detection system detected the presence of two SG nozzle dam fixing parts that could have reached the reactor core and blocked the movement of the control rod assemblies.

Lastly, a piece of piping jammed in the discharge line of a turbine-driven pump at Bugey 4 could have resulted in insufficient water reaching the steam generators.

Following the Fukushima accident, the international WANO (World Association of Nuclear Operators) association has recommended checking the compliance of siphon breakers in the discharge lines of pool cooling systems. It was during these inspections that the Cattenom plant’s operator noticed that two of its facilities have never had siphon breakers since they were constructed and had to report the only incident classified as an INES level 2 event in 2012.

In a reactor core, the chain reaction must be controlled at all times. For this, the neutron flux in the core is constantly monitored by the measuring systems installed near to the core. The “field of vision” of these systems is relatively limited, however, sometimes preventing a representative flux measurement in some core zones, as was the case when a fuel loading error occurred at Dampierre. EDF’s planned installation of a second measuring system of a different design on a coolant system will provide a direct and rapid measurement of the reactor coolant system’s boron concentration, thereby offering a robust measure for preventing a criticality accident occurring.

Insufficient torquing of an oil system flange fixing screw in a Penly 2 reactor coolant pump caused an oil leak followed by fire outbreaks that damaged the pump.
The combination of a valve being incorrectly positioned following maintenance work and a second valve resulted in 140 m$^3$ of reactor coolant being discharged inside the Cruas 4 reactor building, although none reached the outside.

During inspections conducted in Belgium on the Doel nuclear power plant’s reactor 3 and the Tihange nuclear power plant’s reactor 2 in the summer of 2012, defects were found in both reactor vessels’ walls. These were previously-undetected manufacturing defects. This discovery naturally led IRSN to consider the possibility of similar defects existing in the reactor vessel walls of French nuclear power plants.
Presence of loose parts inside the systems

The presence of a loose part inside equipment or a system can affect a nuclear facility’s safety and radiation protection. Despite the actions taken by EDF, several events of this type still occurred in 2012. As a result of its analysis, IRSN has specified the associated risks and shown the need for EDF to step up its action plan.

The presence of loose parts or foreign material inside the facilities’ systems can have various consequences, including damage to the following:

- the first or second containment barrier with, for example, the appearance of leaktightness defects in the fuel cladding or damage to steam generator (SG) tubes;
- the control of reactivity with the blocking of a control rod assembly;
- the radiological cleanliness of the systems, due to the activation of particles from the foreign material.

Activation: any material receiving a neutron flux captures neutrons, making part of its nucleuses radioactive. This phenomenon is called “neutron activation”.

Feedback has shown that, as soon as reactors have been commissioned, foreign material from various sources and of all types (solid or liquid) and forms have been accidentally introduced into the systems.

The foreign material identified to date accidentally entered the systems:

- during maintenance and operation:
  - process waste (metal shavings, welding rods, joint compound, turnings, etc.)
  - maintenance waste (adhesive tape, vinyl, cable sheaths, cloths, pieces of pipe, etc.)
  - fixing elements (screws, washers, bolts, rivets, pins, screw heads, etc.)
  - tools (spanners, screwdrivers, etc.)
  - resins, lubricants and liquids representing a potential chemical hazard;
- during fuel handling: pieces of fins or grid assemblies;
- following equipment failures resulting in parts breaking or becoming loose (ball bearings, floodlight fragments, etc.);
- due to human negligence: miscellaneous forgotten or dropped objects (badges, pens, torches, dosimeters, camera batteries, etc.).

Measures such as anti-debris meshes have been designed to trap certain loose parts; in addition, acoustic systems simplify their detection.
EDF conducted an action plan in 2008, reinforcing its requirements in terms of equipment and system cleanliness in order to avoid the entry of loose parts. These requirements have been stated in an internal directive enacting the “FME” initiative implementing countermeasures based on the prevention, early detection and retrieval of foreign material. According to this directive, a loose part’s retrieval is a matter of priority even if it is difficult to reach. If extracting the part is found to be impossible, the resulting level of danger is analysed in order to determine the acceptability of the reactor operating when the foreign material is present.

By implementing the directive, EDF has been able to reduce the number of events linked with the presence of loose parts. EDF has therefore integrated measures intended to limit the risk of foreign material entering the systems into its working practices. EDF has also increased its integration of feedback and its organisation and supervision of risky maintenance work. Despite this initiative, several important events including the three described below occurred in 2012.

**Screw head discovered under a fuel assembly foot (Chooz B2 — 26 February 2012)**

During operations to unload the fuel from the Chooz B2 reactor in order to shut it down for maintenance, a suction adapter screw head from a reactor coolant pump and debris consisting of locking cup fragments were found under an assembly leg. As the event is described elsewhere in this report (see page 28 of this report), only the safety issues posed by the presence of this loose part are discussed below.

The screw head and its cup moved through the reactor coolant system. They caused impacts in the reactor coolant pumps and on the reactor vessel as well as on the reactor vessel’s internal equipment. The video inspections showed impact marks on a reactor coolant pump impeller vane ①, on the impeller ring ② and on a bottom head penetration flange ③ (see Figure 1, below).

![Figure 1: Locations of impacts on the equipment](image)

The screw head’s impact marks, which can be seen in the photographs, are small and shallow. The hydraulic performance of the reactor coolant pumps was unaffected and EDF did not judge that the impacts had harmed the reactor vessel and its internal structures.
Considering the size and location of the loose parts found, the conclusions of IRSN’s analysis were as follows:

- a reactor coolant pump rotor is unlikely to be blocked due to the presence of a reactor coolant pump screw;
- as the reactor vessel’s stainless steel liner has been impacted, the possibility of a crack appearing and perhaps spreading during the operating cycle cannot be excluded. The same is true of the vessel bottom head penetrations and their welds.

The analysis conducted by IRSN showed that the corrective actions (extracting the foreign material and replacing all of the reactor coolant pumps’ suction adapter and hydrostatic bearing screws) and monitoring actions introduced by EDF were inadequate. This is because neither monitoring of the reactor coolant pumps for vibrations nor monitoring of the reactor coolant system’s flow rate can detect loose screws. Furthermore, the acoustic monitoring of the coolant systems did not detect the loose parts passing through the reactor coolant system or the reactor vessel. Sample screws will therefore be inspected in the 900 MWe and 1300 MWe reactors’ reactor coolant pumps. EDF will modify the design of the 1450 MWe reactors’ coolant pump suction adapters in 2013; IRSN considers that the periodic maintenance program should also be revised.

Passive-seal plug — or SG nozzle dam — fixing elements discovered in the reactor coolant system (Cruas 3 — 14 September 2012)

During the reactor restart phase after a maintenance outage, the acoustic and vibratory detection system detected a significant noise in the channel head of the steam generator’s hot leg in the corresponding loop when reactor coolant pump 1 was commissioned.
After interrupting the reactor’s restart, the operator opened both channel heads (on the water’s “inlet” and “outlet” sides) of the steam generator in loop 1 and found two parts belonging to a fixing assembly for a passive-seal plug (SG nozzle dam). This nozzle dam had been fitted then removed during the outage.

Both parts found, a convex washer and a bolt head, were used to identify the defective plug. Upon examination, four fixing elements were found to be missing: despite a search, the other two missing elements—a bolt and a screw—could not be found.

EDF then assessed the level of danger on the basis of the following information:

- the detailed description of the missing elements, their initial locations and their origins;
- the possible route of these elements inside the systems;
- the possible consequences of their passage through or presence in the equipment;
- the possibilities of detecting these elements by the acoustic detection systems.

IRSN considers that these parts are unlikely to be in the systems connected to the reactor coolant system. Nevertheless, in order to cover this possibility and avoid the presence of loose parts in the reactor core and the blocking of a control rod assembly’s movement, the assemblies’ manoeuvrability was tested on a weekly basis for the first two months of operation after reloading as well as undergoing a mid-cycle control rod drop test. No problems were detected during these tests. EDF also introduced corrective actions to avoid SG nozzle dam fixing
parts failing, such as inventorying these parts after the dams are removed. IRSN judged that the actions introduced by EDF were satisfactory and the new passive-seal plug model could be used.

**Piece of tubing found in the discharge line from the steam generators’ turbine-driven auxiliary feedwater pump (Bugey 4 – 16 November 2012)**

An analysis of the results of the periodic testing of the steam generators’ auxiliary feedwater system (ASG) revealed flow rate imbalances. Various investigations (valve adjustments, draining of sensors, etc.), combined with radiographic inspections of the lines, revealed the presence of a loose part jammed in an elbow of steam generator 2’s water supply line at the discharge from the ASG turbine driven-pump. This foreign material was extracted from the system and seems to match a piece of tubing used in the assembly or maintenance of the systems.

The extracted tubing’s dimensions are as follows:
- diameter = 60 mm
- length = 312 mm

The foreign material’s size and type suggest that it remained jammed near to the place in which it was forgotten. The operator examined the elbow’s internal wall in order to detect any possible damage. The three steam generators’ water auxiliary lines were also inspected internally as far as the ASG adjustment valves to ensure that no other foreign material was present. These investigations did not reveal any other damage. Subsequent tests showed that the flow rates in the steam generators’ lines were balanced and stable again.

*Figure 6: Piece of tubing found*

Nevertheless, such an event can have significant consequences: IRSN considers that it would be difficult to guarantee an adequate water flow to the steam generators in the event of prolonged use of the ASG system if an accident occurs, due to the foreign material’s presence and its possible movement inside the lines.

**Conclusion**

The introduction in 2008 of the measures specified by EDF, including raising employee awareness of the “FME” initiative, has helped to greatly reduce the number of events involving foreign material.

Despite this initiative, several important events mainly due to equipment failures occurred in 2012.
Damage in the reactor coolant pump fixing screws of the 1450 MWe reactors

Various forms of damage have been observed in the internal component fixing screws of the 1450 MWe reactors’ coolant pumps. The associated risks consist in the creation of foreign material in the reactor coolant system of the reactor concerned and reactor coolant pump damage. These findings led EDF to question the design of these screws, their torquing method and the corrosion resistance of the materials used for all screws. As a result, the screw design has been modified, a screw replacement programme has been carried out and monitoring has been increased.

Origin of the reactor coolant pump fixing screw damage found in 2012

In February 2012, EDF discovered a screw head and locking-cup fragments in the protective anti-debris mesh at the leg of a core fuel assembly during a scheduled outage of the Chooz B plant’s reactor 2 (Figure 1).

These loose parts (Figure 2) have been identified as coming from a reactor coolant pump’s suction adapter fixing. The suction adapter is an internal pump component designed to channel the water towards the impeller (inlet channel) in the pump’s volute (see page 24 of this report for the article on “Loose parts”).

A locking cup is a thin metal part surrounding a screw head. Notches are made in the screw head and the housing (machined) into which the screw is inserted. Once the screw has been torqued, the cup is deformed by torquing so that it matches the shape of the notches. This prevents the screw from becoming loose and is referred to as a locking measure or a locking method.
In the case of the reactor coolant pumps of the 1450 MWe reactors, there are five screw- or stud-fixed internal equipment connections per pump (Figure 3), including that of the suction adapter.

Given the potentially generic nature of this event, IRSN recommended that the inspection be extended to cover the condition of screws and studs in the reactor coolant pumps of all 1450 MWe reactors (Chooz B and Civaux), that all plenum screws be replaced and that a sample of the screws be assessed.

This recommendation has also been extended to the screws equipping the pumps’ hydrostatic bearings, which are constructed from the same material as the suction adapter’s fixing screws.

A hydrostatic bearing is a system for positioning a rotating shaft; the shaft is positioned by the presence of a film of fluid injected under pressure through symmetrical nozzles around the shaft, thereby preventing it from moving off-centre.
The investigations conducted during these inspections revealed two types of problems:

- loosening of suction adapter screws, often accompanied by damage to part of the threading and the disappearance of the locking cup. This loosening can cause repeated contact between the screw heads and the pump’s impeller while rotating, resulting in the screws cracking due to mechanical fatigue, or even impacts ultimately causing screw heads to break. The loosening of these suction adapter screws is the result of a design defect in the screws and their locking method, as well as insufficient torquing;

- the presence of stress corrosion in many suction adapter and hydrostatic bearing screws made in stainless steel. For both types of screws, the grade of steel mentioned above (A286) does not comply with the design choice, as this grade was abandoned at the beginning of the '90s in favour of the 316L grade, which is less sensitive to stress corrosion. The discovery — in 2012 — of suction adapter screws manufactured in A286 grade steel therefore constitutes non-compliance that could not be explained by the available documentation. It should be noted that A286 grade steel is acceptable for the fixing screws of the pumps’ other internal components, however, as these screws are less sensitive to stress corrosion because they are subject to less mechanical stress.

Nevertheless, one case of corrosion-induced cracking in a hydrostatic bearing screw manufactured in 316L grade steel within a coolant pump in Civeaux reactor 2 was also found during an inspection. As the assessment did not reveal any defects in this screw’s manufacture, this finding leads us to question this steel grade’s resistance to stress corrosion.

**Risks associated with this damage**

The damage to the screws can result in the following:

- the creation of loose parts in the reactor vessel.
  
  The presence of a screw head in the reactor vessel could damage the stainless-steel liner of the reactor vessel’s internal wall through repeated impacts. The reactor vessel’s steel would then rapidly corrode as a result of the lining being pierced, which is unacceptable.

  In addition, if locking-cup fragments become jammed, this could alter the operation of the control rod drive mechanisms needed in order to control the nuclear reaction, for example;

- damage to the connection provided by the screws between the suction adapter and the pump’s volute.

  This could result in the suction adapter eventually becoming loose and possibly resting against the hydrostatic bearing. The pump’s flow rate would be reduced as a result, affecting the cooling of the reactor core as well as altering the reactor coolant pump assembly's behaviour, causing considerable vibration.
Measures taken by EDF to anticipate the generic risks of fixing damage

After replacing all non-compliant screws, EDF defined a strategy that enables it to anticipate the generic risks of fixing damage, based on the following three areas:

- redesigning the suction adapter screws’ heads and connection to the pressure vessel, and increasing the crimping torque;
- reassessing the risk of suction adapter and hydrostatic bearing fixing screws manufactured in 316L grade steel breaking as a result of stress corrosion;
- stepping up the fixing screw and stud monitoring program.

IRSN’s analysis

Firstly, IRSN judged it necessary for EDF to check that the screws and studs used to fix the suction adapters and hydrostatic bearings are indeed manufactured from 316L grade steel and not from A 286 grade steel. It also stated that a comprehensive inspection programme covering all screws used in the 1450 MWe reactors should therefore be set up; EDF did so in 2012 and 2013.

Secondly, IRSN studied the relevance and sufficiency of the measures proposed by EDF, notably regarding the design of the suction adapters’ fixing screws and their method of torquing and locking:

- IRSN considered that the proposed design measures were indeed likely to reduce the risk of fatigue-induced screw breakages or loosening. Increasing the bending radius of the transition radius between the head and the cylindrical part of the screw reduces the local concentrations of stresses and, therefore, the risk of breakage through vibration fatigue. These measures will be introduced in 2013 and EDF will inspect two reactor coolant pump assemblies in 2015 to check that they are effective in resolving the loosening problems.
- In IRSN’s view, there is a known risk of stress corrosion in hydrostatic bearing screws manufactured in 316L grade steel, due to the discovery of a case of a hydrostatic bearing screw with no manufacturing defects cracking as a result of stress corrosion.

As IRSN considers that the increased tightening torque and the reduction in the cylindrical part’s diameter encourages the appearance of stress corrosion due to the increased stresses, it judged that these measures needed to be accompanied by a review of the maintenance measures. EDF plans to reassess the periodic maintenance programme.

IRSN’s assessment can be viewed (in French) on the IRSN website at:

http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2013-00021-N4-EPR.aspx
Lack of siphon breakers in the coolant lines of the irradiated fuel assembly storage pools

At the design stage, it was planned to install a device named a “siphon breaker” in the fuel assembly storage pool’s coolant system line in order to avoid the pool being accidentally emptied through the siphon effect. This device consists of a circular orifice drilled in the upper area of the line’s immersed part. EDF has checked the compliance of the devices installed as part of the actions taken in response to feedback from the Fukushima accident. It then noticed that such a device was not installed in the storage pools’ coolant system lines of the Cattenom nuclear power plant’s no. 2 and no. 3 reactors.

Storage pool
Each of EDF’s nuclear power plants includes a fuel assembly storage pool mainly designed to do the following:

- provide the workers with biological protection against ionising radiation;
- remove the residual heat from the assemblies unloaded from the core before they are transferred to AREVA’s La Hague reprocessing plant or reloaded into the core;
- store the new fuel assemblies before they are loaded into the core.

In order to perform these functions, an adequate level of water must always be maintained above the fuel assemblies. Depending on the reactor’s type, the pool depth varies between 12 m and 13 m, as the height of a fuel assembly is approximately 4 m.

Residual heat
Within a nuclear reactor, the nuclear fission chain reaction mainly results in the production of heat energy and fission products.

After the chain reaction stops, the fuel still releases some heat, named “residual heat”, which decreases over time. Shortly after it stops, the residual heat is mainly due to the fission products. This residual heat varies from 0.5% to 0.1% of the reactor’s rated power, between one day and a month after the reactor is shut down.

The residual heat must be removed in order to prevent the fuel assemblies overheating, potentially harming their integrity and resulting in radioactive discharges into the environment.
The water in the spent fuel pool is continuously cooled by a cooling and treatment system. The water is sucked out from the upper part of the pool before being cooled in heat exchangers at the bottom of the pool, near the base of the fuel assembly storage racks.

In the event of a leak, breach or line-up error in the cooling and treatment system’s line, siphoning may occur in the discharge line concerned, draining the pool and partially or totally uncovering the fuel assemblies. Passive “siphon breakers” were incorporated into the design to interrupt the siphon effect. These consist of circular orifices, the diameter of which is approximately 1/10 of that of the pipe, drilled into the upper area of the immersed part of the pipe concerned.

Following the Fukushima accident, the international WANO (World Association of Nuclear Operators) association recommended checking the compliance of the siphon breakers provided for in the discharge lines of the fuel pool cooling and treatment system. During the inspections conducted at the end of 2011, Cattenom’s operator noticed the lack of siphon breakers in the lines submerged in the pools of the no. 2 and no. 3 reactors (no holes had been drilled in these lines). Inspections of EDF’s other reactors did not reveal any other coolant systems without a siphon breaker.
**Reasons for the lack of “siphon breakers” at Cattenom**

The “siphon breakers” did indeed appear on the design and production plans. The deviation therefore occurred at the time of construction. Because the event took place more than 25 years ago, its exact cause is difficult to determine. EDF has nevertheless offered the following explanation: during construction, the coolant system was tested under pressure following its completion; a hole should then have been drilled in the line, but this was not done. Furthermore, the final acceptance testing carried out to ensure the facility’s compliance with the plans, which took place before the facility was commissioned, did not detect the deviation.

During an examination conducted by IRSN in 2002 concerning the safety of the storage of irradiated fuel assemblies in pools, EDF committed to performing, within six months, periodic inspections to ensure that the “siphon breakers” are not blocked. EDF has indeed incorporated a periodic inspection (every three years) to check that the “siphon breakers” on the fuel pool cooling and treatment system lines are not blocked into its preventive maintenance programmes (end of 2009 for the 900 MWe reactors, August 2010 for the 1300 MWe reactors and April 2011 for the 1450 MWe reactors). However, these inspections have not yet been carried out at the Cattenom site, which contains four 1300 MWe reactors.

**Corrective actions**

The compliance inspections carried out in the pools following the discovery of the absence of siphon breakers in two reactors at the Cattenom plant showed that, in the case of the 900 MWe, 1300 MWe and 1450 MWe reactors, the siphon breaker diameters meet the specifications (20 mm), except for the Golfech 1 (17 mm) and Belleville 1 (15 mm) reactors. Siphon breaker compliance inspections have also been carried out in the reactor buildings’ pools; these have not revealed any deviation.

The “siphon breakers” on the fuel pool cooling and treatment systems’ discharge lines of the Cattenom nuclear power plant’s no. 2 and no. 3 reactors were drilled by specialist divers within approximately a month of the deviation being detected, with the reactor in operation. The “siphon breakers” of the Golfech 1 and Belleville 1 reactors’ pools were also made compliant approximately a month after the deviation was detected.

**Importance for safety**

A line-up error or breach in a system connected to the storage pool could cause an uncontrolled drop in the water level. To limit this drop, the pool’s cooling pumps are then stopped. If the leak is lower than the pool, a siphoning effect could result and the level would then continue to drop.

If there is no “siphon breaker” and the control operators do not respond to the problem, the draining of the pool would result in the stored fuel assemblies being exposed then damaged and lead to radioactive discharges into the environment.

**Adequacy of the existing siphon breakers**

IRSN has asked EDF to show that the existing “siphon breakers” are adequate for more serious draining than that envisaged in the design phase. The studies conducted by EDF have shown that the existing diameter of the “siphon breakers” would not stop the draining begun by a guillotine break in the fuel pool cooling and treatment system’s main drain. As a result, EDF has committed to increasing the cross-sectional flow area of the “siphon breakers” for all reactors before March 2014.

In addition, as the possibility of a “siphon breaker” becoming blocked cannot be excluded, IRSN has recommended that EDF modify the design of the fuel pool cooling and treatment system’s discharge line, on the basis of functional diversification, for example.
Weakness in the means of monitoring the reactor cores

The nuclear chain reaction in a reactor core must be controlled at all times. This notably requires the monitoring of the neutron flux by means of measuring systems installed near to the core, including during reactor outages. However, this monitoring has malfunctioned during the core loading or unloading phases. EDF has therefore taken measures to solve this problem. IRSN’s analysis has concluded that these measures are inadequate and requested an equipment modification designed to prevent a criticality accident.

Why monitor the neutron flux during a reactor outage?

Even during a reactor outage, its core emits neutrons. These neutrons are due to the disintegration of the fission products present in the irradiated fuel assemblies and the presence of neutron sources required when starting the reactor. During an outage, the core must remain sub-critical; that is to say, the number of neutrons produced in this way must be less than the number of neutrons absorbed or escaping from the core. The neutrons are absorbed by the boron dissolved in the reactor coolant system’s water and by the control rod assemblies inserted into the fuel assemblies. As a result, the number of neutrons increases whenever the boron concentration decreases or control rod assemblies are withdrawn. If there are too many of these neutrons, this could trigger an uncontrolled nuclear chain reaction and cause a criticality accident. During a reactor outage, this accident could mainly result in workers located in the reactor building becoming irradiated and ultimately cause fuel fusion.

The neutron flux represents the number of neutrons passing through an area of one square metre in a second.

A criticality accident is the triggering of an uncontrolled nuclear chain reaction within an initially sub-critical medium.
How is the neutron flux monitored during a reactor outage?

Neutron detectors belonging to the “source range channels” (SRCs) positioned around the pressure vessel measure the number of neutrons escaping from the pressure vessel. If this number increases, the neutron flux within the reactor’s core also increases. When this flux increase exceeds a defined limit, an alarm warns the operator that an unexpected flux variation has occurred. This alarm must be triggered early enough to give the operator time to act before the criticality accident occurs. In this situation, the operator performs the actions defined in the procedures.

Initial doubts regarding the effectiveness of the source range channels

A reactor core contains fuel assemblies with different characteristics. A loading plan defines the fuel assemblies’ exact position in the core. This is designed so that the neutron flux is evenly distributed when the reactor is in operation. The loading plan must be respected in order to ensure that the chain reaction is always controlled irrespective of the reactor’s operating conditions.

In 2001, an error occurred while a fuel assembly was being loaded into a Dampierre reactor core, resulting in the position of 113 assemblies being offset. Subsequent calculations revealed that the neutron flux had increased considerably, although this had not been detected by the SRCs. In 2005, EDF explained why this increase was not detected by the SRCs.

Why can the SRCs not detect a neutron flux increase in all cases?

The SRCs are located outside the reactor vessel and their “field of vision” of the neutrons is relatively limited. If the number of neutrons increases considerably in some areas of the core, the SRCs may be relatively insensitive to the flux, as was the case when the loading error occurred at Dampierre in 2001. The flux measured by the SRCs is not always representative of the flux level inside the reactor core. As a result, the operator may be informed late that the neutron flux is increasing abnormally.
**Measures taken by EDF to compensate for the “SRC anomaly”**

When the reactor is shut down and the reactor vessel is open while containing fuel assemblies, the neutron flux increase may be due to the following:

- a loading error;
- a control rod assembly being withdrawn while the vessel head is being lifted;
- a “dilution” (reduction in the concentration) of the boron present in the reactor coolant system water.

EDF responded to the loading error issue in 2003 by setting up an organisation to prevent fuel assemblies being positioned wrongly. Nevertheless, it demonstrated that shuffling the fuel assembly positions could not result in criticality.

Concerning the control rod assembly withdrawal accident, the operator showed in 2003 that the boron concentration was sufficient to avoid a criticality accident occurring when the pressure vessel’s head was lifted.

Concerning boron “dilution”, the operator chose in 2005 to use the existing boron concentration measurement system, called a boron meter, to detect a decrease in the reactor coolant system’s boron concentration sufficiently early. This automatic monitoring system, which is connected to the nuclear sampling system, now performs a safety function. A chemist also measures the boron concentration manually every 90 minutes in case this automatic monitoring system fails.

**From IRSN’s assessment to the French Nuclear Safety Authority (ASN)’s decision**

The measures taken by EDF to avoid a criticality accident occurring were not brought into question by IRSN in 2004 regarding the loading error or control rod assembly withdrawal situations.

In 2011, on the other hand, IRSN’s examination of the corrective solution to the “SRC anomaly” concluded that the boron concentration measurement systems (continuous and manual) were not sufficiently reliable (see opinion (in French) on the IRSN site: [http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2011-00273-EDF-Palier-CPY-APR.aspx](http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2011-00273-EDF-Palier-CPY-APR.aspx)). Under these conditions, IRSN recommended that EDF study other diversified equipment independent of the current boron meter as rapidly as possible. As a result, ASN recommended in 2012 that EDF install an equipment modification to prevent a criticality accident occurring in the event of the boron contained in the reactor coolant system’s water becoming “diluted”. Until a robust solution is found, IRSN recommended that EDF take actions in the short term to guarantee an acceptable safety level regarding the criticality risk.

**Definitive processing solution proposed by EDF**

EDF proposed installing another, differently-designed, boron meter on a system for measuring the boron concentration directly and rapidly in the reactor coolant system’s water. EDF envisages installing the first of these second boron meters in 2017 and then generalising their installation on up to 13 reactors a year if industrially feasible.

Until this definitive modification is set up, EDF has, in accordance with IRSN’s recommendations, taken actions to reduce the “dilution” risk, improve the reliability of the existing boron meter and raise the operators’ awareness of the importance of the existing boron meter’s new role: performing a safety function.
Fire outbreaks in a reactor coolant pump room at Penly 2

In April 2012, it was noticed that an oil pump in a reactor coolant pump in Penly nuclear power plant’s reactor 2 had been operating unexpectedly for a prolonged period. This, combined with insufficient torquing of an oil system flange fixing screw, caused an oil leak followed by fire outbreaks that damaged the reactor coolant pump. A series of technical failures and human errors then resulted in the reactor coolant system leaking water that was collected in a tank for this purpose.

Following this incident, EDF immediately took corrective actions that IRSN examined in 2013 as part of the detailed analysis of the incident concerned.

Reactor coolant pumps
The reactor coolant pumps (RCPs) pump the water flowing through the coolant system (RCS) into the system’s different legs (hot leg + “U” leg + cold leg) and contribute to the reactor core’s cooling function by transferring heat energy from the core to the steam generators (SGs) and the secondary systems. During power, the reactor automatically shuts down if a reactor coolant pump stops unexpectedly or even slows down unexpectedly.

French pressurised water reactors are equipped with three or four reactor coolant pumps, depending on their power. The 900 MWe reactors have three reactor coolant pumps, whereas the 1300 MWe reactors (including those at Penly) and the 1450 MWe reactors have four reactor coolant pumps.
Reactor coolant pumps are machines whose shafts rotate vertically at approximately 1500 rpm. They consist of an electric motor that is lubricated by an oil system. This motor drives a shaft and an impeller that causes the water to move through the reactor coolant system. The seal between the pump’s drive shaft and the motor is provided by a system of three successive seals into which water is injected at high pressure against the flow to prevent any water leaking out from the coolant system (see Figure 3). Some of the water injected into the seals enters the coolant system; the rest is collected and sent to two special systems: the chemical and volume control system (CVCS) and the nuclear island vent and drain system (NIVDS). The seals are designed to operate under temperature conditions lower than those of the reactor coolant system when the reactor is operating. This is why they are protected from heat by means of two systems: the injection of cold water under high pressure against the flow, and a cooling device called a “thermal barrier” (see Figure 4 and Photograph 1), whose main role is to cool the water present in the seals if the injection of cold water under high pressure fails.

**Chemical and Volume Control System (CVCS):** this system is connected to the reactor coolant system and mainly used to maintain the quality of the coolant, adjust its volume and control its boric acid concentration.

**Nuclear Island Vent and Drain System (NIVDS):**
this system collects the liquid and gaseous waste that could cause radioactive contamination and which is produced by the nuclear systems and facilities.

![Figure 3: Block diagram of a reactor coolant pump’s seals](image-url)
**Reactor coolant pump seals**

The three successive seals of reactor coolant pumps must ensure that the coolant system is leaktight. In order to achieve this, cold water is injected into these seals to block the hot coolant (represented in green in the following diagrams). The three seals utilise three successive controlled leaks to reduce the pressure from 155 bar (the coolant system’s internal pressure) to atmospheric pressure.

**Description of seal 1**
- The profile of the faces is designed so that the clearance is constant \(e = 10 \mu m\), resulting in a controlled leak.
- This 10 \(\mu m\) clearance is obtained by balancing the pressures on the floating ring (self-stabilising).
- The faces must never come into contact.

**Description of seal 2**
- Seal 2 blocks the leak from seal 1.
- This is a standard rubbing face seal.
- In the event of seal 1 failing, seal 2 can hold for a sufficiently long time to allow the facilities to be shut down, thanks to a cooling system called a “thermal barrier”.

**Description of seal 3**
- Seal 3 blocks the leak from seal 2.
- This is a standard rubbing face seal.
- Water from the reactor boron and water makeup system is injected between the rubbing faces to avoid contaminated borated water flowing back from the leak from seal 2.

*Figure 4: Description of a RCP – Block diagram*  
*Photograph 1: Thermal barrier of a RCP*

*Figure 5: Description of the role of the three reactor coolant pump seals*
**RCP shaft lifting oil system**

An oil system lubricates each operating reactor coolant pump’s bearings (hydrodynamic conditions); an additional system for injecting oil under high pressure, called “lift oil”, provides an oil film on the reactor coolant pump’s thrust bearing (see Figure 4) before and during start-up, as well as when the motor is shut down (hydrostatic conditions).

During a reactor coolant pump’s start-up phase, its oil pump is started automatically so that the shaft can be lifted (to reduce torque). After this pump has been operating for approximately two minutes and thirty seconds, the reactor coolant pump’s motor is started up. The oil lift pump then stops after a delay of approximately one minute.

This pump is also used to lubricate the bearings while the reactor coolant pump slows down when the pump is shut down. In this case, it stops after a delay of fifteen minutes.

The oil pump’s operation is therefore associated with the reactor coolant pump’s startup and shutdown sequences.

**Timeline of the incident**

During the night of 3–4 April 2012, while Penly nuclear power plant’s reactor 2 was in operation, reactor coolant pump 2’s oil lift pump started up unexpectedly. The operators in the control room did not notice that the pump had started up; as a result, the pump worked continuously until the reactor coolant pump shut down around midday on 5 April.

Before the reactor coolant pump automatically shut down, an alarm indicating that reactor coolant pump 1’s lubricant oil reservoir level was low was triggered in the control room. Two minutes later, “fire” alarms activated by the sensors located in this reactor coolant pump’s room were triggered in the control room (the oil leak near to the very hot components (approximately 300°C) caused fires to break out). Twenty minutes later, this reactor coolant pump’s shutdown sequence was automatically triggered by a high temperature alarm from one of the motor’s shaft bearings, due to the loss of much of the motor’s oil. The reactor coolant pump’s shutdown was automatically followed by that of the reactor itself. The operators controlled the reactor in accordance with the emergency operating procedures.

As soon as the external emergency services had been called, the internal fire-fighting teams were mobilised. The conventional On-site Emergency Plan was activated at the beginning of the afternoon. EDF’s response teams and external firemen entered the reactor building several times in order to extinguish the fire outbreaks with extinguishers and take various preventive and monitoring actions.

While reactor coolant pump 1 was shutting down, the controlled bleed-off line from seal 1 was automatically isolated due to excessive bleed-off, probably as the result of the damage to seal 1 (see Figure 3 — “Leak from seal 1 to CVCS system” and the description of seal 1 in Figure 5). Cold water continued to be injected into the seals of the four reactor coolant pumps, however, notably reactor coolant pump 1 which had been shut down, where the pressure of the injected water was transferred to seal 2 as a result. Around 6 PM, EDF, fearing that the shut-down reactor coolant pump’s seal 2 would not withstand the pressure, decided to reopen the controlled bleed-off line from seal 1 to the CVCS system in order to relieve seal 2. The valve on the controlled bleed-off line was no longer internally leaktight, however, probably as a result of particles from the damaged seal 1.
The unsuccessful attempt to reopen this valve then caused coolant water collected in the purpose-designed nuclear island vent and drain system (NIVDS)’s tank to leak.

As a result of the emergency operating procedures, the coolant system leak through seal 2 was gradually reduced by reducing the pressure in the reactor coolant system. Reactor 2 reached a stable shutdown state during the night of 5–6 April 2012. Shortly afterwards, the conventional On-Site Emergency Plan was lifted once it had been confirmed that the fire had been extinguished.

This incident, which has not affected the environment, was classified as an International Nuclear Event Scale (INES) level 1 event.

**Source of the oil leak**

When reactor coolant pump 1 was examined, it was noticed that one of the lift pump’s oil system flange seals was out of its initial position (see Photograph 2). An inspection of the torquing of the failed flange’s four screws revealed that the screw closest to the leak had been insufficiently torqued. Once the line had been removed, the seal’s inspection showed that it was properly located in its groove, but that it had been sectioned at the insufficiently-torqued screw.

EDF’s hypothesis is that prolonged operation of the lift pump, combined with insufficient torquing of a screw, resulted in the seal deforming excessively and then breaking.

**Source of the damage to seal 1**

Shortly after midday, it was noticed that the controlled bleed-off rate from seal 1 had increased and its head loss had decreased, which is indicative of the seal opening considerably. When the reactor coolant pump was examined, it was confirmed that the seal had indeed been damaged. However, the exact cause of the damage has not yet been identified. IRSN is currently studying this question as part of its detailed analysis of the incident mentioned at the beginning of this article.

**EDF’s corrective actions taken**

Following the incident, reactor coolant pump 1 of Penly reactor 2 was completely replaced, with the exception of the volute, and the torquing of all reactor coolant pump oil system flange screws in both of the nuclear power plant’s reactors was inspected.

EDF then developed and conducted a partial requalification programme on the coolant system before Penly 2 was restarted. This programme defined the actions to take (visual inspections, liquid penetrant testing, metallographic replica testing, cleaning and fitting of the replacement parts) following the incident.

In addition, EDF inspected reactor coolant pump 1, the lift pump’s electrical cell and the valves located on the controlled bleed-off line of seal 1 for the four reactor coolant pumps in order to determine the exact causes of the equipment failures.

EDF also examined the possible effect of the pressure increase in the controlled bleed-off line of the failed reactor coolant pump’s seal 1 upon the three other reactor coolant pumps.

In addition, the torquing of the oil system’s flanges was checked during the outages for reloading the French nuclear power plant units in order to obtain feedback.

Lastly, EDF is studying the feasibility of a system in the control room to detect that a reactor coolant pump’s oil lift pump is operating.
Outline of IRSN’s detailed analysis of the incident

In 2013, IRSN will examine whether the actions taken by EDF were appropriate and sufficient in its detailed analysis of the incident. The initial information gathered regarding the causes, sequence of events and results of the incident shows that its analysis is highly useful, notably regarding the following:

- the Penly reactor 2’s fallback behaviour until it reached a safe shutdown state;
- the causes of the detected equipment failures, particularly those of reactor coolant pump 1’s seal 1;
- the personnel working conditions, including those of service providers and the external emergency services, in terms of radiation protection;
- the effectiveness of the fire detection systems and of the fire-fighting actions taken by EDF when the incident occurred;
- the event’s general aspects.

As part of its detailed analysis, IRSN is to discuss technical details both with the Penly plant personnel who managed the event and with EDF’s central departments. The objective of this detailed analysis is to learn all possible lessons from this event and share them with the safety authorities and the nuclear power plant operators in France and abroad.
Coolant discharge from the reactor coolant system in the Cruas reactor 4 building

In June 2012, two human errors resulted in the chemical and volume control system (CVCS) of the Cruas plant’s reactor 4 being pressurised and heating up, followed by several of the system’s safety valves being opened. A large quantity of radioactive water from the reactor coolant system was then discharged into the reactor building, although none was released outside the building. The reactor was shut down and EDF carried out the necessary work to clean up the reactor building, check the facility’s condition and replace the damaged equipment. Considering the potential consequences of such a sequence of events, IRSN examined the suitability and sufficiency of EDF’s actions.

The system affected by the event

The chemical and volume control system (CVCS) controls the volume and chemistry of the reactor coolant system water (“reactor coolant”).

When the reactor is operating normally, the system maintains a suitable water level in the pressuriser (a device that pressurises the reactor’s coolant) by draining or adding water.

The CVCS therefore includes a line for draining water from the reactor coolant system, called the “letdown line”, and a line for returning water into the reactor coolant system, called the “feed line”; an additional line called the “excess letdown line” can be used if the letdown line is unavailable.

A buffer tank “absorbs” the difference in flow rate between the feed and the letdown and adjusts the volume of water in the reactor coolant system. When this tank’s water level is too low, if it is disconnected from the letdown line for example, the feed line is automatically fed by another tank, the refuelling water storage tank (RWST).

The CVCS also filters and purifies the reactor coolant and is used to inject chemical products mainly intended to prevent corrosion in the reactor coolant system. It is protected from overpressure by means of safety valves.

The reactor coolant is pumped through the CVCS by three pumps called “charging pumps”, only one of which is in operation when the system is operating normally. Before it is filtered and purified, it is cooled by the component cooling system (CCS) by means of heat exchangers.
In the case of the 900 MWe reactors, the CVCS also injects water at high pressure at the reactor coolant pumps’ seals (which provide leaktightness between the reactor coolant pumps’ shaft and body).

**Timeline of the event**

On 3 June 2012, while the Cruas plant’s reactor 4 was starting up after its annual outage for maintenance and fuel reloading, the three-way valve was operated on the CVCS’s excess letdown line.

During the operation, the valve was set wrongly (the first error), resulting in the reactor coolant flow fraction being directed to the reactor waste collection system (NIVDS) instead of the CVCS.

This situation resulted in a volume of reactor coolant exceeding the amount authorised during normal operation being detected in the periodic inventory of reactor coolant leaks.

In application of the procedures, the shift crew then suspended the reactor’s start-up operations.

Diagnosing a problem in the letdown line, the operators isolated it, resulting in the water flow to the feed line being switched automatically to the refuelling water storage tank (RWST). At the same time, the operators activated the excess letdown line in order to keep the water level constant in the reactor coolant system.

However, due to a valve in the excess letdown line’s cooling system being closed (the second error), the line was not cooled by the component cooling system (CCS). The temperature and then the pressure in the CVCS and the NIVDS increased, resulting in several relief valves opening. The water that escaped through these valves discharged into a special tank, the pressurised relief tank located in the reactor building. After a time, its protective membranes ruptured, resulting in 140 m$^3$ of reactor coolant being discharged into the reactor building, although none was released outside the building.

Furthermore, a filter on the excess letdown line was damaged as a result of the temperature increase, resulting in debris spreading into the CVCS as far as the charging pump, which was currently in operation but had not been damaged. A flow limiter on the reactor coolant pumps’ seal bleed-off line had also been blocked, increasing the line’s pressure.

While the reactor was being shut down, the reactor coolant system’s cooling speed and the temperature difference between the two ends of the pressuriser surge line (which connects the pressuriser to the reactor coolant line) exceeded the limits set in order to avoid the cooling system being damaged.

Once the source of the reactor coolant leaks had been located, EDF reset the three-way valve correctly in the reactor building, thereby resuming the cooling of the excess letdown line. The feed and letdown lines were then started up again and the excess letdown line was shut down, stabilising the reactor safely as a result. EDF then carried out the inspections and corrective work needed in the facility.
EDF’s corrective actions

EDF has carried out a major inspection and maintenance programme on the CVCS equipment that was subjected to the pressure and temperature increase. Inspections were also conducted on the NIVDS and reactor coolant system through which the debris resulting from the filter damage, as well as on the systems affected by the reactor coolant discharged into the reactor building, consisting of more than a hundred items of equipment. In particular, EDF replaced both the charging pump exposed to the debris and the filters clogged by the debris. It carried out corrective actions on the reactor coolant pumps’ seals and checked the excess letdown line’s supports, bearing in mind the expansion due to the temperature variations to which it was subjected. Lastly, EDF cleaned up the reactor building.

EDF also performed an analysis called an “assessment of level of danger” in order to identify the debris’ type and its CVCS and reactor coolant system impact hazards, and conducted inspections to check that there was no potentially harmful debris in the lines and equipment of the affected systems.

A hydrotest was conducted on the tank used to collect NIVDS reactor waste, which had been subjected to a pressure approximately twice its design pressure, to check that it can withstand a higher pressure than its operating pressure. Liquid penetrant testing was conducted upon all of the tank’s welds. Dimensional inspections of the tank’s shell were also carried out to check that it is not deformed.

EDF also analysed the mechanical strength of the heat exchanger located on the excess letdown line, which was subjected to a thermal shock due to the failure of its coolant system. A CCTV inspection was conducted upon the internal part of the heat exchanger, which provides leaktightness between the coolant system and the system being cooled, as well as a hydrotest to check that the device is internally leaktight.

In addition, EDF conducted a mechanical analysis of the pressuriser’s surge line, which had been subjected to a considerably higher temperature difference than the maximum value authorised in the operating technical specifications as a result of the event. EDF therefore checked that the maximum stresses and deformation judged acceptable for this line had not been exceeded in this situation.

Lastly, EDF modified the three-way valve’s operating procedure in order to specify the operating instructions and organised educational exchanges with the response teams, in which they were reminded of the principles and measures to take in order to change these settings (risk analysis and requalification). EDF has also updated its “line-up” procedures, in other words, the correct positioning of the valves, and reminded its teams of the requirements applicable to these line-ups.
The conclusions of IRSN’s analysis

In its assessment (which can be viewed in French at http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2012-00357-EDF-Cruas4.aspx), IRSN has examined EDF’s analysis as well as the results of its inspections of the tank used to collect NIVDS waste and the excess letdown exchanger in order to check their leaktightness, as this equipment performs the role of containing the reactor coolant (which is radioactive). It also examined the results of the inspections and the repairs carried out on the equipment in the systems affected by the event.

In addition, bearing in mind the potential consequences of the thermal stresses to which the pressuriser’s surge line was subjected, IRSN paid particular attention to EDF’s analysis conducted to check the line’s mechanical strength.

IRSN judged that the results of the analyses, inspections and repairs carried out by EDF were acceptable. However, it emphasised the need both for the reactor building to be carefully cleaned up and for all items of equipment and anchoring points that have been sprayed or immersed during the event to be comprehensively inventoried. This is because the reactor coolant discharged in the reactor building is corrosive, particularly for certain grades of steel.

IRSN also recommended that the condition of the mastic sealing providing the reactor building’s leaktightness along the inter-building seal between the internal structures’ foundation raft and the containment or reactor building be checked. This is because any deterioration in this sealing could allow water to penetrate into the seal as far as the metal liner, which could then be corroded without it being visible, or even inspectable. Such corrosion, should it develop, could adversely affect the containment function.

Bearing in mind the event’s impact upon many items of equipment that are important in the reactor’s safety, IRSN had also emphasised that the facility’s corrective actions could not not be considered as standard maintenance operations; EDF performed a minute inspection of work’s quality and the associated requalification before the reactor was restarted. EDF’s sampling inspection of the seal’s sealing, that is to say at a certain number of points, did not reveal the presence of any water in the seal.
Defects discovered in the Doel 3 and Tihange 2 reactors’ vessel walls in Belgium

During outage inspections of the Doel nuclear power plant’s reactor no. 3 and the Tihange nuclear power plant’s reactor no. 2 in Belgium in the summer of 2012, their operator detected defects in both reactors’ vessel walls. The analysis of the results of these inspections revealed thousands of indications attributable to the presence of previously-undetected production defects. This discovery naturally led IRSN to consider the possibility of similar defects existing in the reactor vessel walls of French nuclear power plants.

The reactor vessel containing the reactor’s core is an essential element of the nuclear power plants and cannot be replaced. As a result, particularly stringent inspections and checks are performed on its design, production, acceptance and operational monitoring.

Reactor vessels are composed of welded steel parts; in the case of the French reactor pressure vessels, all welded joints are circumferential. The 900 MWe, 1300 MWe and 1450 MWe reactors are similar and have the same number of constituent parts. They include a cylindrical part consisting of two shells called “core shells”, with the exception of reactor 1 of the Fessenheim power plant (FSH 1) which has three.

The entire internal surface of each reactor vessel is covered with a stainless steel liner approximately 8 mm thick laid in two layers by welding. This liner protects the reactor vessel’s steel against corrosion.

The dimensions of the reactor vessels increase with the core’s power, as shown in the following table which summarises the main characteristics of the French reactor vessels’ core zone.

<table>
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<tr>
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<th>FSH 1</th>
<th>900 MWe</th>
<th>1300 MWe</th>
<th>1450 MWe</th>
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<td>Number of “core shells”</td>
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<tr>
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<td>2383</td>
<td>2370</td>
</tr>
<tr>
<td>Reactor vessel mass (t)</td>
<td>130</td>
<td>440</td>
<td>460</td>
<td></td>
</tr>
</tbody>
</table>

Geometrical characteristics of the French reactor vessels’ shells

Figure 1: Constituent parts of a reactor vessel
Manufacture

All of the French nuclear power plants’ reactor vessels were manufactured by Framatome from forged parts almost exclusively supplied by Creusot-Loire, which notably supplied all of the reactor vessels’ shells.

The reactor vessels’ shells are manufactured from ingots poured in a steel plant, by means of a complex process including various hot-shaping operations:

- Cropping — cutting off an ingot’s ends — removes the zones containing impurities,
- Drilling the ingot (in the case of a solid ingot),
- Drawing the blank on chucks,
- “Shaping” (increasing the internal diameter of a hollow cylindrical part).

During these operations, most of the impurities and the macrosegregation are eliminated. The forged parts then undergo a full ultrasonic inspection to find any other defects that could, where necessary, result in the part being scrapped. The different parts are then assembled by circumferential welding and all welds are inspected using two different processes (radiography and ultrasound).

The stainless steel liner is laid by welding to the internal wall of the different shells comprising the reactor vessel. The completed reactor vessel is then subjected to factory hydrotesting (the test is required by the French regulations).

**Defects detected in the Doel 3 and Tihange 2 reactor’s vessel walls**

During the summer of 2012, the entire internal surface of the Belgian Doel 3 and Tihange 2 reactor vessels’ core zone was inspected for the first time. These inspections were intended to find any underclad defects in the first 25 millimetres under the internal wall, the zone in which any underclad defects occur (see Figure 3). In the case of the French reactor vessels, these inspections are systematically conducted every ten years after the reactors are commissioned.

Multiple defect indications that did not correspond to underclad defects were detected and the plants’ operator decided to conduct supplementary investigations throughout the thickness of the reactor vessels. These investigations revealed approximately 8,000 indicators in the Doel 3 reactor vessel’s core zone and 2,000 indicators in the Tihange 2 reactor vessel’s core zone. It should be noted that these reactor vessels were manufactured and inspected between 1974 and 1978 in compliance with the requirements of the US design and construction code (ASME, 1974 edition).

The operator attributed these indicators to manufacturing defects due to the steel containing hydrogen from the metal’s production (H₂ flaws — see Figure 4). However, the documentation of the time does not mention the presence of such indicators, which should have been detected and noted in the manufacturing inspection reports.
The operator supposes that this situation is due to human error, which leaves the question of whether these defects have existed since manufacture unanswered.

It should be said that the manufacturer of the Doel 3 and Tihange 2 reactor vessels' shells no longer exists and did not manufacture shells for the French nuclear power plants' reactor vessels.

During the Doel 3 and Tihange 2 reactors shutdown the operator conducted studies between July and December 2012 to justify the reactor vessels’ suitability for use. At the end of 2012, it sent two summary reports containing all of the technical information on the subject to the Belgian Nuclear Safety Authority with the objective of obtaining approval for these reactors being restarted. The nuclear safety authority demanded further information and the restarting of the reactors was finally approved on 17 May 2013; the Doel 3 and Tihange 2 reactors were restarted at the beginning of June 2013 (http://fanc.fgov.be/fr/news/reacteurs-des-centrales-de-doel-3-et-tihange-2-bilan-du-redemarrage/625.aspx, in French).

Underclad defects can occur when the stainless steel liner is being welded if the thermal conditioning specified in the welding procedure is insufficient. These are planar defects perpendicular to the reactor vessel’s internal wall (see Figure 3), corresponding to a pathway dilation in its steel. These defects occur just below the liner.

H$_2$ flaws can appear when the local hydrogen level in the metal is too high; in general, they are linked with segregation zones. They take the form of multiple pathway dilations almost parallel to the reactor vessel’s internal wall (see Figure 4).
The reactor vessel inspections performed in France are intended to detect defects and monitor their operational development. These examinations complement the regulatory hydrotests conducted at 206 bar every ten years. Several types of inspections are performed on the core shells:

- a complete CCTV inspection of the reactor vessel’s internal surface, including the recording of the images, using onboard cameras on the operational inspection system: this examination has been conducted since the end of the ’70s as part of the pre-service inspection as well as during the first complete inspection no later than 30 months after the reactor cooling system’s first hydrotest and then during the ten-yearly inspections. It is intended to reveal visible surface damage (impacts, wear, metal scuffing, etc.);

- an ultrasonic examination of the shell welds: this is a volumetric examination of the entire thickness of the welds, concerning the laid metal as well as the adjacent zones of the base metal on both sides of the weld for a distance of 50 mm. The examination is designed not only to detect defects perpendicular to the reactor vessel’s internal wall, but also to detect defects parallel to the wall. This examination has been performed since the ’70s at the same intervals as the CCTV inspections;

- an examination of the core zone to find underclad defects: this qualified inspection has been performed in France since 1999, as part the ten-yearly reactor inspections; it concerns all welds and the entire internal wall of the shells in the most highly irradiated parts of the reactor vessel, over a thickness of 25 millimetres starting from the reactor vessel’s internal wall. This zone is most sensitive to irradiation and is the one in which underclad defects can occur. These inspections have revealed the presence of around thirty underclad defects in the French reactors.

The core zone areas inspected by ultrasound are shown in Figure 5. This inspection is not intended to find deep H² flaws, but it has detected H² flaws within the top 25 millimetres.

Manufacturing defects in French reactors’ shells, and the impact of defects discovered in Belgian reactors’ shells

Despite the careful design and manufacture of the French reactor vessels, some defects could nevertheless occur during manufacture. The main defects consist of underclad defects and H² flaws. Smaller, microcrack defects can also occur.

![Design: the designer draws up a reactor vessel mechanical design justification file for all of the reactor’s operating situations considered in the dimensioning (including accidental situations). It aims to demonstrate the reactor vessel’s mechanical strength, taking into account the safety coefficients specified in the French regulations. In particular, it examines the risk of fast fractures: the core’s neutron radiation results in a growing reduction in the ductility (= the material’s ability to be deformed without breaking) of part of the reactor vessel shells’ steel. The ten-yearly inspections analysing the metal’s resistance to fast fractures are therefore highly important in demonstrating safety, as the reactors’ suitability for continued operation depends on their results.](image-url)
Only certain, known, reactor vessels are affected by underclad defects, as the welding process was improved following the defects’ discovery in 1979. No underclad defects have been discovered in the walls of reactor vessels manufactured in France since that date.

Approximately 30 underclad defects in eight reactor vessels have been identified in the core shell zones of French reactors. The pressure vessel of the Tricastin plant’s reactor 1 is the most seriously affected with around 20 underclad defects, and six reactor vessels have only one underclad defect. The largest defect is located in the wall of Tricastin 1’s reactor vessel and is 11 mm long, including measurement uncertainties. These defects are monitored during operational service by means of special periodic inspections; no changes have been seen to date. In addition, detailed mechanical analyses have enabled IRSN to conclude that these defects are not dangerous, and it has asked EDF to ensure that the water injected into the damaged reactor coolant system of Tricastin 1’s reactor vessel is maintained at a constant temperature of 20 °C in order to limit the amplitude of the “cold shock” on the vessel.

One of the risks when forging the shells is the appearance of H2 flaws. In order to avoid the appearance of such defects in French reactor vessels, the hydrogen level in the steel is checked when the metal is poured and the metal undergoes a special thermal process if necessary when forged, to reduce the level of hydrogen in the metal part. During the last 50 years, only a few parts intended for French nuclear power plants have occasionally had H2 flaws. They were scrapped without argument following the manufacturer’s inspections.

After the indicators present in the Doel 3 and Tihange 2 reactor vessels were discovered, EDF and the manufacturer checked the measures taken during the French reactor vessels’ manufacture, on the basis of the vessels’ manufacturing documentation. The available information does not lead us to suspect that the situation regarding the French reactor vessels may be similar to that of Doel 3 and Tihange 2, taking into account the manufacturing measures and inspections performed upon the French reactors’ pressure vessels from the outset. Nevertheless, the operational inspection of the French nuclear vessels’ core zone, which is conducted every ten years, is only carried out on a depth of 25 millimetres from the internal wall and so does not concern the entire thickness of the reactor vessels’ shells.

As a result, IRSN has recommended that EDF formalise the documentation analyses performed on the reactor vessels’ manufacturing and inspection procedures, as well as conducting, from 2013 onwards, ultrasonic inspections of the entire thickness of reactor vessel shells, similar to those conducted on the Doel 3 and Tihange 2 reactor vessels.

**Conclusion**

The available information does not lead to suppose that the French nuclear power plant reactor vessel walls contain manufacturing defects that are similar, in terms of numbers and size, to those discovered in the reactor vessels of Doel 3 and Tihange 2.

The ultrasonic inspections conducted by EDF on the entire thickness of reactor vessel shells, the first of which will be carried out in 2013, will provide an exhaustive check to ensure the absence of defects.
SIGNIFICANT UPGRADES

France's nuclear reactors are retrofitted and upgraded throughout their operating lives, particularly to improve their safety on a continuous basis. These changes are brought about by many factors, such as advances in scientific and technical knowledge, identified weaknesses, lessons learnt from operating experience feedback, environmental changes and new or revised regulations. Years of studies may be necessary before certain retrofits and upgrades are clearly defined and implemented.

In the wake of the Fukushima-Daiichi accident of 11 March 2011, many actions were taken to ensure that existing nuclear power reactors and organisations are robust enough to withstand extreme situations not considered in the design of these reactors. In France, the analyses conducted as part of complementary safety assessments confirmed that France's facilities were by and large robust against the natural hazards considered at each site. However, IRSN's analysis demonstrated the value of supplementing the protective measures for existing facilities with a set of means for withstanding natural hazards of a magnitude greater than those thus far considered. Called the 'Post-Fukushima hardened safety core', these means were analysed by IRSN in late 2012 based on the file provided by EDF.

A major retrofit consisted in strengthening the rafts of the Fessenheim reactors, which were deemed to be not thick enough to withstand a fuel-melting accident leading to a vessel piercing.

Generally speaking, most retrofits are carried out while reactors are shut down for inspection every ten years.

Nuclear reactors are designed to withstand weather conditions such as high winds and heat waves. The windstorm that swept across France in December 1999 and partially flooded the Blayais nuclear power plant prompted a review of external-flooding risks at all of France's nuclear power plants. This review led to the creation of a guide on protecting basic nuclear installations from external flooding. Likewise, the heat waves of 2003 and 2006 showed the need to better protect plants from the risks of heat waves and prompted EDF to define an approach. This approach was analysed by IRSN in 2012 for the 900 MWe plant units.

Deviations from a reactor's baseline state may occur during the design, construction or operation of a reactor. Known as 'compliance gaps' these deviations may invalidate the safety demonstration of a facility presented in the safety analysis report. They are detected, characterised and dealt with in accordance with a special process reviewed by
IRSN in 2010. This section presents three examples of generic compliance gaps. The first is faulty fire compartmentalisation of electrical penetrations in the containment. The second is a problem with the diesel engines of the emergency and station blackout generators of the 900 MWe reactors. The third is substandard seismic resistance of pneumatically controlled valves.

EDF reports events during which the availability of safety-related equipment has been compromised. These events have many causes, some of which are technical in origin while others are related to human intervention. In 2012 IRSN analysed a sampling of these events and focused in particular on events related to operations or maintenance activities. One method of improving the reliability of equipment is to factor in, right from the design phase of equipment or upgrades, the activities of the people who will be using and maintaining it.
Strengthening of nuclear facilities following the Fukushima-Daiichi accident

Following the disaster that struck Japan’s Fukushima-Daiichi nuclear power plant on 11 March 2011, many actions were taken around the world to ensure that existing nuclear facilities and organisations are robust enough to withstand extreme situations not considered in the design of these facilities. In France, studies conducted by operators in the spring of 2011 as part of complementary safety assessments confirmed that France’s facilities were by and large robust against the natural hazards considered for each site. However, IRSN’s analysis demonstrated the value of supplementing the protective measures for existing facilities with a set of means for withstanding natural hazards of a magnitude greater than those thus far considered. These means make up what is known as the ‘Post-Fukushima hardened safety core’. An initial proposal by EDF for the hardened safety core of nuclear power plants was analysed by IRSN in late 2012.

Factoring in natural hazards during the design of reactors
France’s nuclear power plants are designed to withstand various types of natural hazards (snow, wind, extreme temperatures, flooding, earthquakes, etc.). Each type is associated with a level (e.g., wind speed). The list of hazards and associated levels is periodically reviewed for each site to take into account operating experience feedback and changes in knowledge or the doctrine about hazards. These reviews take place at least once every 10 years, when safety reviews are conducted.

To protect a facility from natural hazards, equipment used to ensure fundamental safety functions must remain available and operational when subjected to design-basis hazard levels. This equipment is therefore:

- protected by measures that prevent the hazard from affecting it,
- designed to remain operational in the presence of the hazard.

In case of the Fukushima-Daiichi accident, the design-basis hazard levels were insufficient to protect the plant.

The Fukushima-Daiichi accident
On 11 March 2011, an earthquake exceeding the design-basis earthquake for the Fukushima-Daiichi plant triggered a tsunami that overtopped the plant’s seawalls. The incoming water flooded the plant’s buildings, knocking out electrical power to reactors 1 through 4. This loss of power was followed by a progressive loss of cooling for the cores and SFPs. The temperature of the cores of reactors 1 to 3 (reactor 4 had been defuelled previously) rose, triggering fuel meltdown and structural damage and causing the radioactive substances contained in the fuel assemblies to be released into the containments of the reactors. The containments were not designed to prevent these substances from escaping into the environment. As a result, extensive amounts of radioactivity were released into the environment, contaminating workers and the general public as well as large swaths of Japan.
Complementary safety assessments and stress tests

On 23 March 2011, France’s prime minister asked ASN to conduct a safety audit of France’s nuclear facilities. The audit’s initial conclusions, which focused primarily on nuclear power plants, were presented in late 2011. The audit examined the risks of flooding, earthquake, loss of electrical power and loss of cooling as well as operational management of accident situations. In its decision of 5 May 2011, ASN ordered that the managers of France’s nuclear power plants carry out complementary safety assessments (CSA) to determine how their facilities would have reacted during the Fukushima-Daiichi accident.

On 24 and 25 March 2011, the Council of the European Union declared that stress tests would be carried out the nuclear power reactors in the EU’s Member States.

In September 2011, the operators of France’s main nuclear facilities (EDF, Areva, CEA and ILL) submitted their reports on the robustness of their facilities in relation to the various aforementioned situations to ASN. In EDF’s case, it submitted 19 self-assessment reports, one for each of its sites.

To make a comprehensive assessment of the robustness of the facilities, IRSN had to look at a very wide range of technical aspects. It presented the results of its assessment to two advisory committees, one for nuclear reactors (GPR) and one for laboratories and plants (GPU). Both of these committees reported to ASN’s director general.

IRSN’s analysis confirmed that France’s nuclear power reactors were by and large capable of withstanding the various natural hazards considered. However, this conclusion assumed that the facilities were in compliance with their safety baselines (see article on the handling of compliance gaps on page 70 of this report).

In addition, IRSN recommended that EDF define a set of requirements and additional measures to ensure fundamental safety functions, including during hazards considerably greater than design-basis hazards. These measures would make up what is referred to as a ‘CSA hardened safety core’.

In their opinion to ASN, the advisory committees confirmed their belief that it was necessary for operators of nuclear power plants to have robust means of dealing with these hazards in order to prevent core meltdown in the event of long-term total loss of heat sink or long-term total loss of electrical power at more than one facility on a site. They also stressed the importance of defining a set of means for mitigating releases following a severe accident in the event of hazards greater than those in the current design baseline.

Regarding the irradiated fuel assembly storage pools, the advisory committees considered it essential that EDF define and implement, as quickly as possible, strict measures to prevent fuel assemblies from being uncovered during their handling and storage in these pools.

Lastly, IRSN stressed that the emergency-response organisation and means must remain operational for hazard levels much greater than design-basis levels as well as radioactive or toxic ambient conditions resulting from a severe accident affecting several facilities on a site.

Regarding the EU stress tests, the robustness of France’s power plants was reviewed by European peers (a limited number of experts appointed by the safety regulators of various European countries or their technical advisers). In the first quarter of 2012 IRSN took part in reviewing the reports of stress tests conducted at power plants in other European countries. It also acted as a technical adviser to ASN regarding matters relating to France and its power plants.

This culminated in the publication, in May 2012, of reports by the European Nuclear Safety Regulators Group (ENSREG) for each country. These reports list the best practices implemented by each country and contain a number of recommendations. The report for France found the definition and deployment of a hardened safety core for France’s nuclear power plants to be an example of a good practice. ENSREG’s report on France can be downloaded at http://www.ensreg.eu/sites/default/files/Country%20Report%20FR%20Final.pdf
Content of the hardened safety cores

Following the opinion of the advisory committees, which was based on the operators' files and IRSN's analysis of these files (IRSN's report is available in French at http://www.irsn.fr/FR/expertise/rapports_gp/gp-reacteurs/Pages/Rapport-IRSN-ECS.aspx), ASN ordered EDF to define, by 30 June 2012, the content of the hardened safety cores for its nuclear power plants as well as the natural-hazard levels used to design these hardened safety cores (new equipment) or check their robustness (existing equipment).

EDF presented its equipment proposals for the hardened safety cores of its operational reactors in late June 2012. IRSN presented its analysis to the GPR in December of the same year.

In the summer of 2012, the operators of the other nuclear facilities (Areva, CEA and ILL) also submitted their hardened safety core proposals and the associated requirements to ASN. These proposals were the subject of a technical review by IRSN and an opinion of the advisory committee in charge of laboratories and plants (GPU) in April 2013 (the LUDD and RR public report is available at http://www.irsn.fr/Ludd-2011-2012-EN).

IRSN considered that EDF's proposed flood levels were in line with ASN's order. It was also satisfied with the idea of the Nuclear Rapid Response Force coming to the aid of accident-stricken sites.

As regards the risks associated with dewatering of assemblies in the fuel storage pools in the reactor building (while open) and in the fuel building, it pointed out the current lack of technical measures for mitigating radioactive releases in the event of dewatering of the fuel assemblies. IRSN found EDF's hardened safety core proposals for these pools to be overall satisfactory. However, it considered that the demonstrations of the structural resistance to extreme hazards of the spent-fuel vault and the shutdown of drainage by siphoning should be especially robust.

Lastly, EDF undertook to set up dedicated I&C and electrical distribution systems for the hardened safety cores and, wherever possible, to make these systems independent of the existing resources. IRSN considered that this should greatly increase the robustness of the harden safety cores.

Nuclear Rapid Response Force (FARN)
The operating experience feedback from the Fukushima-Daiichi accident prompted EDF to set up a Nuclear Rapid Response Force (FARN) to assist sites in managing severe accident situations liable to occur at France's nuclear power plants.

FARN's mission is to assist accident-stricken sites by providing them with response teams, equipment (lighting, air compressors, pumps, etc.) and resources (fuel oil, water, etc.).
However, in the conclusions of its analysis, IRSN considered it necessary that EDF supplements its proposals:

- by combining the hardened safety cores with a target to mitigate releases of short-lived fission products. This target comes on top of the EDF's safety target regarding long-term land contamination;
- by adopting, in the hardened safety cores, measures to prevent core meltdown triggered by extreme natural hazards and, should a meltdown occur, mitigate the consequences of partial or complete core meltdown. These measures must be supplemented by the means needed to manage the emergency. These means must also form part of the hardened safety core;
- by factoring in, for the hardened safety cores, seismic levels significantly greater than those used in the design of the facilities;
- by defining requirements (in terms of design, manufacture and operations monitoring) to guarantee a high level of confidence in the ability of the hardened safety cores to carry out their functions.

The GPR found that EDF's proposal for the hardened safety cores and associated requirements was not fully in line with the opinion it issued in late 2011. It recommended that EDF supplements its proposal with measures for preventing core meltdown with a high level of confidence and measures for significantly mitigating the radiological consequences of a severe accident. It also requested that EDF updates its file accordingly. It considered that the requirements for guaranteeing the ability of the hardened safety core to perform its functions in extreme-hazard situations had to be specified or revised.

At the end of the review, ASN announced that in 2013 it would draft requirements for EDF's nuclear power plants and that these requirements would supplement those issued in June 2012.

To learn more, read IRSN's dossier on CSAs, in French:
The foundation rafts of the two reactors at the Fessenheim plant have the distinctive feature of being the thinnest of all other French reactors in operation. During the third ten-yearly safety review of the plant's 900 MWe reactors, IRSN and the advisory committee for nuclear reactors looked at how they would react in the event of a core meltdown (severe accident) in which the molten fuel would pierce the reactor vessels and flow onto the concrete foundation rafts. At the end of the review, ASN ordered EDF to reinforce the rafts of the plant's reactors. It issued a deadline of June 2013 for reactor 1 and December 2013 for reactor 2. IRSN reviewed in 2012 the suitability of the reinforcements proposed by EDF.

**Why reinforce the foundation rafts of the reactors at the Fessenheim plant?**

For many years, IRSN has looked closely at preventing severe accidents and tightening measures for mitigating the consequences of such accidents at reactors in operation. French reactors are required to undergo safety reviews once every 10 years. These reviews provide the ideal opportunity to discuss improvements to be put in place.


Whereas reactors built after 1977 sit on foundation rafts measuring between 2.5 and 4.7 m thick, the foundation rafts under the reactors at the Fessenheim plant measured only 1.5 metres thick. IRSN realised that, in the event of a severe accident leading to a vessel piercing by a mixture of molten materials — consisting primarily of fuel and steel (known as corium) — the foundation rafts at the Fessenheim plant could in turn be breached within less than 24 hours.

EDF's Fessenheim nuclear power plant is located in northeastern France, at the border with Germany. It has two pressurised water reactors that were commissioned in 1977. Each PWR generates 900 MWe. They were the first PWRs of such capacity built in France.
With this in mind, and at the end of the third safety review, ASN ordered that the plant’s foundation rafts be reinforced to make them more resistant to a core meltdown with vessel piercing. It issued a deadline of late June 2013 for reactor 1 and late 2013 for reactor 2.

**Description of the upgrade**

EDF’s solution consisted in increasing the available corium spreading area and the thickness of the portion of the foundation raft under this area.

To carry out its solution, EDF chose to:

- create, in a room adjacent to the reactor pit, an additional corium spreading area bounded by low walls;
- connect this area to the reactor pit by a discharge channel that, for radiation-protection purposes, is usually closed off by a concrete melt plug;
- increase the thickness of the foundation raft of the reactor pit and the spreading area by 50 cm.

In the event of an accident with core meltdown, the corium would build up in the bottom of the reactor pit, melt the plug by eroding it from its sides, and flow into the additional spreading area.

As water would disrupt the spreading of the corium, gates have been placed on the low walls to prevent water entering the additional spreading area until the melt plug decomposes.

**Figure 1: General view of the upgrade**

**Figure 2: Gate and door leading to the spreading area**
What are the benefits in terms of extending the raft failure time?

IRSN analysed the upgrade and more specifically the corium-concrete interactions. It identified two types of situation:

1. those in which the corium does not come into contact with the water before spreading across the entire surface of the reactor pit and the additional spreading area. These are known as ‘dry’ situations;
2. those in which the corium comes into contact with the water in the reactor pit before the vessel fails. These are known as ‘wet’ situations.

IRSN considered that EDF’s proposed upgrade would provide a real benefit in terms of extending the raft failure time during a core meltdown accident for dry situations. According to EDF, in the worst-case dry situation, the upgrade would increase the raft failure time by at least around two days. IRSN’s assessments yielded the same order of magnitude and indicated that this time savings could be increased by voluntarily flooding the corium once it was fully spread out.

However, IRSN considered that in wet situations the water might substantially disrupt the various steps leading up to the full spreading of the corium. More specifically, this water could:

- delay the opening of the melt plug, thereby resulting in more substantial vertical destruction of the concrete in the reactor pit;
- prematurely solidify part of the corium and reduce, or even seal off, the discharge channel opening;
- limit the spread of the corium.

According to IRSN, if the corium is not allowed to fully spread, the presence of water in the reactor pit prior to vessel piercing could limit the benefits of increasing the raft thickness. Therefore, in order to maximise the benefit of the upgrade, IRSN considered that EDF should take measures to prevent wet situations, particularly since water in the reactor pit prior to vessel piercing could lead to a risk of explosion from the interaction of the molten corium with the water.

What were the constraints of the upgrade?

Two constraints had to be taken into account:

- worker exposure to ionising radiation (radiation from the vessel);
- risk of damage to equipment in the reactor pit.

EDF presented the main radiation-protection measures it adopted to mitigate worker exposure during work on thickening the rafts. It announced that these measures reduced the collective dose, initially estimated at 280 H.mSv, to a projected collective dose of about 90 H.mSv. IRSN considered that the matter of radiation protection had been adequately addressed during the study phases and particularly in the choice of methods. For example, the choice of injecting concrete from outside the reactor pit, meant that workers spent less time in the pit. Tests conducted on a life-size mock-up of the reactor pit made it possible to accurately calculate work times and how to proceed.

Regarding the second constraint, IRSN considered that the measures adopted by EDF mitigated the risks of damaging the in-core instrumentation system, whose measurement tubes are located near the areas where the work was conducted.
**IRSN’s conclusions**

IRSN concluded its opinion by considering that the upgrade proposed by EDF would provide a real benefit in terms of increasing the raft failure time in the event of a core meltdown accident leading to vessel piercing (IRSN’s opinion is available in French for downloading at [http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2012-00519-EDF-Fessenheim.aspx](http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2012-00519-EDF-Fessenheim.aspx)). Nevertheless, it stressed that measures for preventing the presence of water in the reactor pit prior to vessel piercing and for completely flooding the fully spread corium in the spreading area next to the reactor pit would further increase the benefit of the upgrade.

EDF completed the upgrade of both its reactors at the Fessenheim plant in 2013.
Guide on protecting nuclear facilities against external flooding

Principles for protecting pressurised water reactors from external flooding were defined in a fundamental safety rule (RFS I.2.e) in 1984. The windstorm of late December 1999 and the partial flooding of the Blayais plant caused by it highlighted the need to revise these principles. A guide applicable to all basic nuclear installations was drafted to replace the aforementioned fundamental safety rule. The guide was based on proposals made by a working group consisting of representatives of ASN, IRSN and French nuclear operators as well as experts in hydrology, hydraulics and meteorology. It contains a set of recommendations for defining and characterising external flooding as well as the protective measures to be implemented.

Principles for protecting pressurised water reactors from external flooding were defined in fundamental safety rule RFS I.2.e. This rule was published on 12 April 1984 by ASN (then known as the Service central de sûreté des installations nucléaires, or central department for the safety of nuclear facilities).

The windstorm of late December 1999 and the partial flooding of the Blayais nuclear power plant caused by it (see IRSN’s report in French at http://www.irsn.fr/FR/Actualites_presse/Communiques_et_dossiers_de_presse/Pages/inondation_centrale_Blayais_0999.aspx) prompted France’s nuclear operators as well as ASN and IRSN to conduct a sweeping review of the measures used to protect facilities against external flooding. This review led operators, and EDF in particular, to propose upgrades to their facilities. After being analysed by IRSN and approved by ASN, these upgrades were deployed at each facility. (see ‘Protecting nuclear power plants from external flooding’ in French on page 44 of IRSN’s 2007 public report).

This review also highlighted the need to revise fundamental safety rule RFS I.2.e. To that end, a working group consisting of representatives of ASN, IRSN and French nuclear operators as well as experts in hydrology, hydraulics and meteorology was formed. Their several years of work culminated in the drafting of a guide applicable to all basic nuclear installations and which replaced RFS I.2.e.

The guide defines external flooding as “flooding from any single or multiple cause (rain, cresting rivers, storms, broken pipes, etc.) located outside the structures, areas and buildings of basic nuclear installations containing systems or components requiring protection. External flooding therefore covers flooding caused outside the bounds of a basic nuclear installation as well as some types of flooding caused within the bounds of said installation.”

The guide contains a set of recommendations for defining and characterising external flooding from which nuclear facilities must be protected as well as the protective measures to be implemented.
Risks caused by external flooding

External flooding (of a river, from rainfall, etc.) poses various risks for nuclear facilities and can simultaneously affect all the nuclear facilities on a site:

- the safety of facilities, particularly the nuclear-island buildings for nuclear power plants, can be compromised if the basemat under the facilities is submerged and water enters rooms containing equipment essential to the safety of these facilities (e.g., motors of reactor-coolant pumps...);
- some types of flooding (cresting rivers, dam failure) may carry debris of types (branches, leaves, etc.) that can clog water inlets and thus hinder a facility's ability to draw in the water needed for its cooling systems;
- submerged electrical stations and fallen power lines and transmission towers (especially if flooding is accompanied by violent winds) can disrupt or cut off power to facilities;
- blocked roads can make it difficult for emergency-response teams and equipment to reach a facility and can affect means of communication.

Many different phenomena and situations

The working group first identified all the phenomena that can cause external flooding. It limited its selection to those liable to affect sites in France with nuclear facilities.

It then drew up a state-of-the-art of methods used to determine the characteristics of 'extreme events' (e.g., once-in-a-millennium-floods) caused by the adopted flooding 'phenomena' ('cresting river' in this particular case). It looked very closely at the causal dependencies between various phenomena, the uncertainties associated with the data and methods, and the influence of changes in weather patterns.

This work allowed it to define 11 reference situations for external flooding hazards (or reference flood situations, RFS) that nuclear facilities must be protected against as well as recommend methods for characterising these flood situations.
A RFS is developed from an event or a combination of events whose characteristics may be overestimated. The adopted approach consists in aiming for, by order of magnitude, an estimated annual probability of exceedance of each RFS of 1 for 10,000, taking into account uncertainties inherent to its determination.

The main facility protection principles recommended by fundamental safety rule RFS I.2.e were supplemented in the guide to more broadly cover the possible consequences of external flooding on basic nuclear installations. Such consequences include site isolation and unavailability of support functions (offsite power, offsite emergency resources).

After a broad consultation phase, the draft version of the guide was presented before the advisory committees for nuclear reactors (GPR) and laboratories and plants (GPU). IRSN's analysis report is available in French at: http://www.irsn.fr/FR/expertise/rapports_gp/gp-reacteurs/Pages/Guide-inondations_GPR-GPU_24052012.aspx.

The guide was amended to take into account the opinion of the advisory committees. It was published by ASN in early 2013 and is a source of reference for protecting nuclear facilities from external flooding.
Protection of nuclear power plants from extreme temperatures

The particularly hot summers of 2003 and 2006 prompted EDF to take another look at protecting its nuclear power plants during heat waves. It implemented a 'heat wave protection' project to ensure that its facilities will continue to safely generate electricity during extremely hot periods. In 2007, 2008 and 2012 IRSN analysed the various aspects of EDF's approach for checking whether equipment essential to the safety of its 900 MWe reactors would continue to operate properly during a heat wave. EDF implements the same approach for its 1300 MWe and 1450 MWe reactors.

The consequences of heat waves on nuclear power plants include:

- lowering their generating output or shutting them down in order to:
  - comply with general operating rules specifying requirements such as maximum allowable temperatures in certain rooms;
  - mitigate thermal discharges, especially when river-water temperatures are already high;
- damage to, or even failure of safety-related equipment.

**Verifying the temperature resistance of equipment**

As part of its 'heat wave protection' project, EDF used thermal calculations of temperatures reached inside its facilities to verify whether all safety-related equipment could operate at higher-than-expected temperatures and whether this equipment would continue to perform sufficiently at the maximum possible indoor temperature. More specifically, EDF compared the maximum allowable temperatures for this equipment with the calculated maximum temperatures liable to be reached in the rooms containing this equipment and taking into consideration the possible outdoor temperatures.

IRSN reviewed the data used by EDF to calculate the maximum indoor temperature, to wit:

- the maximum possible outdoor air temperatures between now and 2030;
- the maximum temperatures of the cooling water (river water, seawater, etc.) possible between now and 2030;
- plant states (at power or in shutdown) that lead to the highest temperatures in plant rooms and systems;
- scenarios that lead to the highest temperatures in plant rooms and systems (during normal operations, during an incident or during an accident such as total loss of power, etc.);
- piping running through plant’s rooms and contributing to the exchange of heat with the ambient air in these rooms;

These maximum temperatures were determined using an extrapolation method that takes into account both the high temperatures recorded since 1970 and climate changes predicted to occur between now and 2030.
• equipment that raises the heat in the plant's rooms: pumps, engines, compressors, electrical transformers, etc.

By way of example, the simplified diagram below depicts the case of a safety-related electrical cabinet located in a room where the ambient temperature is raised by heat from:

• the outdoor air (1);
• a pump (4);
• piping (5) whose own temperature is increased by:
  o water from a river warmed by the Sun (2);
  o 'hot' rooms the piping runs through (3);
  o the flow of the river water inside the pump (4).
**Actions for improvement**

Based on the results of the calculated maximum temperatures possible inside its facilities, EDF identified which equipment could be affected by high ambient temperatures. It initiated some twenty changes to make its facilities less vulnerable to heat waves.

A few examples of the upgrades planned or already carried out by EDF include:

- increased performance of the heat exchangers between river water or seawater and the water used to cool safety-related equipment;
- replacement of the thermostatic valves on the lubrication line of the safety injection pumps (see the "High ambient temperature for the safety injection pumps" article in French on page 48 of IRSN's 2008 public report);
- replacement of, and increase in, the performance of the chillers;
- establishment of shutdown procedures for non-safety-related equipment whose operation is likely to raise indoor temperatures.

**IRSN's opinion**

It is IRSN's view that the upgrades planned or already carried out by EDF shows considerably decrease the vulnerability of its facilities to heat waves.

However, IRSN considers that EDF has not satisfactorily demonstrated that the margins between the maximum operating temperatures of the equipment and the calculated maximum indoor temperatures are sufficient. It is therefore impossible to conclude that the equipment will correctly operate under all situations of extreme heat. IRSN therefore views it necessary that EDF justify the available margins by taking into account in particular uncertainties about the data and assumptions used for the thermal calculations. If sufficient margins are not provided, EDF may be required to make additional upgrades (see the IRSN's opinion submitted to ASN in French at [http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2012-00353-EDF-palier-CPY.aspx](http://www.irsn.fr/FR/expertise/avis/avis-reacteurs/Pages/Avis-IRSN-2012-00353-EDF-palier-CPY.aspx)).
Nuclear facilities and cold snaps

Just as it assessed EDF’s heat-wave protection approach, in the early 1990s IRSN began reviewing EDF’s baseline for protecting its facilities from periods of extreme cold.

Until the early 1980s, EDF’s 900 MWe and 1300 MWe reactors were designed for a design-basis cold-weather temperature of -15°C. At the time, temperatures below this level were not factored in.

In 1985-1986 and 1986-1987, France experienced harsh winters that created a number of frost-related incidents that affected outdoor systems in particular. Outdoor temperatures dropped well below the design-basis temperatures of EDF’s facilities, prompting EDF to take protective measures. Some sites even recorded outdoor temperatures of -33°C for periods of up to six hours. As a result, in 1986 EDF established a ‘cold snap protection’ approach at all its facilities around France. This approach was first applied to the design of EDF’s 1450 MWe reactors and implemented at its 900 MWe and 1300 MWe reactors during their safety reviews. Studies conducted by EDF and analyses carried out by IRSN resulted in a number of equipment upgrades (installation of additional heating equipment or reduction in ventilation flow rates in certain rooms, improved cold-weather insulation of pumping-station equipment, etc.) and organisational changes (operating instructions to be carried out in extremely cold weather between October and April).

France has experienced several cold snaps since those of the winters of 1985-1986 and 1986-1987. The most recent cold snap occurred in February 2012 and led to significant events at six plants in France. However, thanks to protective measures, these events had no real consequences on these facilities. Inadequate equipment maintenance and poorly scheduled maintenance just two causes of these events.

Given the possible consequences on the safety of reactors and other basic nuclear installations, IRSN closely analyses event causes and corrective actions implemented by operators to prevent such events from recurring. (read the article from the LUDD + RR 2011-2012 report at http://www.irsn.fr/Ludd-2011-2012-EN)
Management of compliance gaps

Deviations from a reactor’s baseline state may occur during the design, construction or operation of a reactor. Known as ‘compliance gaps’ these deviations may invalidate the safety demonstration of a facility presented in the safety analysis report. EDF implements a specific process for managing compliance gaps. IRSN’s review of this process and its participation in ASN-led inspections on how EDF implements its process have allowed EDF to significantly improve how it manages compliance gaps. IRSN also closely monitors how well EDF assesses the impact of these gaps on safety and the methods it uses to manage them.

State of facilities
EDF is required to demonstrate that its nuclear facilities are safe at each stage of their lifespans. This demonstration is predicated on compliance with a set of safety requirements making up the current safety baseline. In addition, it requires that these facilities comply with their design baseline. However, during construction or following operating or maintenance activities, the actual state of a facility may differ from the required state. For example, some types of equipment must continue operating after an earthquake or accident, that is, at severe conditions of ambient temperature, pressure and irradiation. Thus, prior to being put into service, this equipment is subjected to qualification to demonstrate that it is able to operate under such conditions. An item of equipment that fails qualification may no longer be able to fulfil the functions on which the safety demonstration is based. Maintaining a satisfactory safety level thus means that compliance gaps must be identified, analysed and corrected by plant operators. In EDF’s case, it has defined a process for managing gaps at all its facilities in France.

Compliance gaps – a category of their own
Checks and inspections conducted in the 2000s as part of the second ten-yearly reviews of the 900 MWe reactors revealed a large number of gaps with their design baseline. By their nature, these gaps are often generic, that is, they affect several reactors across different sites. This complicates, and in some cases delays, their management. In 2001 EDF began referring to them as ‘compliance gaps’ and set up a dedicated management method (see description below) allowing compliance deadlines to be adjusted to the impact these gaps have on safety.

%During each ten-yearly safety review, France’s reactors are subjected to:

- a **compliance review** that consists in verifying the reactors’ compliance with their applicable safety baseline;
- a **safety review**, i.e., an update to their baseline that takes into account international best practices, developments in knowledge, updates to facilities’ risk assessments or disadvantages, lessons learned through operating experience feedback, and rules applicable to similar facilities.
EDF defines compliance gaps as “deviations from the design baseline that justifies the safety level of facilities.”

Gaps affecting pressure equipment on reactor coolant systems and secondary systems (already covered by a specific process), safety, radiation protection, the environment, facility operation and the associated operations baseline are not considered compliance gaps.

Compliance gaps are generally caused by:

- weaknesses inherent to a facility (design, manufacture, assembly);
- maintenance operations;
- retrofits and upgrades;
- equipment ageing;
- anomalies in studies supporting the safety demonstration (see the ‘Study anomalies in the safety demonstration’ article on page 27 of IRSN’s 2011 public report);
- changes in the baseline related, for example, to better understanding of physical phenomena.

EDF’s compliance gap management process

EDF’s process is based on its compliance gap management policy (published on 5 July 2001) and consists of the following four steps:

**Detection - Discovery of the gap**
Gaps detected by facilities or engineering centres and identified during an initial analysis as potential compliance gaps are notified to ASN and IRSN by letter. If a gap is potentially generic, it is handled by EDF’s national engineering services. Depending on the severity or emergency, compensatory measures may be decided at this stage.

**Characterisation of the gap**
Characterisation consists in assessing the safety consequences of the gap and determining whether it is generic. It allows the operator to determine the urgency of the management strategy based on the risk posed by the gap. If, at the end of this stage, the safety consequences are found to be significant, a safety-significant event is reported to ASN.

**Definition of the management strategy**
Depending on the characterisation results, EDF defines a management strategy that may consist in:
- maintaining the facility as is;
- defining compensatory measures to make up for the compliance gap until permanent measures can be implemented;
- defining measures and a deadline for remediating the gap while maintaining the facility in its operating state;
- placing the affected reactor in a safe state if the potential safety consequences are deemed unacceptable.

**Implementation of corrective actions**
The facility is brought back into compliance. Depending on the safety impact, this is accomplished immediately, during outages or ten-yearly safety reviews.

IRSN has assessed EDF's process for managing compliance gaps as well as how it has managed certain gaps based on their potential consequences.

In 2010 IRSN reviewed EDF’s compliance gap management process and found that it was well in line with regulatory requirements. That said, it stated that EDF should maintain an up-to-date and thorough report of gaps that occur at its reactors so that more relevant safety analyses could be carried out, particularly in the case of multiple
compliance gaps. IRSN also considered that EDF should include compliance gaps in the reports of temporary modifications to general operating rules (GOR) that it may be required to send to ASN.

In April 2011 EDF issued interim measure No. 320 and requested the managers of its facilities to identify all compliance gaps at their own facilities by 1 July 2011 and to maintain this list up to date. IRSN and ASN drew up a joint list, which they regularly update.

In 2012 IRSN accompanied ASN as it inspected the proper enforcement of interim measure No. 320 and management of compliance gaps in France's nuclear power plants. Poor understanding of the concept of compliance gaps and deficiencies in the process for detecting these gaps were observed during the first visits. Noteworthy improvements were observed during subsequent visits. These improvements were due to the extensive assistance provided to the facilities by EDF's national engineering services throughout 2012.

Nevertheless, an analysis of the safety impact of an accumulation of compliance gaps had yet to be conducted. As a result, in 2012 EDF's national engineering services created an approach for analysing the safety consequences of an accumulation of compliance gaps. This approach was documented in a guide reviewed by IRSN in 2013. The services also plan to create, in 2013, a new version of the guideline that provides facility managers with recommendations on implementing the actions required by this approach.

When a compliance gap is reported by EDF, IRSN looks at how EDF handled the gap at each stage of its dedicated management process. In particular, IRSN assesses EDF's analysis of the safety consequences, the effectiveness of the corrective measures taken and the acceptability of the proposed compliance deadlines. Some compliance gaps are found by IRSN during ASN inspections or during reviews of outages for purposes such as refuelling.

Examples of compliance gaps

Three examples of how compliance gaps are managed are provided on the following pages and relate to:

- poor fire compartmentalisation of the electrical penetrations in the containment;
- a problem on the diesel engines of the emergency and station blackout generators of the 900 MWe reactors;
- poor earthquake resistance of pneumatically controlled valves.
Poor fire compartmentalisation of the electrical penetrations

The requirements in EDF’s fire-protection baseline call electrical wire conduits that end at the outer surfaces of wall penetrations and do not continue into the fire-resistant material used to fill the penetrations. In the event of fire, the gaps between the cable and protective conduits can facilitate the spread of smoke.

Between 2001 and 2008, work was performed as part of a project to upgrade fire protection at EDF’s plants. One of the aims of this work was to remove the conduits inside the penetrations. However, in July 2010, specialised EDF workers discovered that the conduits on the Cattenom site extended through the penetrations located between the control room of reactor 2 and the floor below. Inspections immediately carried out at the site’s four reactors revealed a hundred or so such deviations and prompted the Cattenom site to report a safety-significant event on 6 August 2010.

As soon as the first deviation was discovered on the Cattenom site, IRSN concluded that it was potentially generic and recommended that inspections be carried out at all the plants.

Initial inspections carried out in the other plants revealed similar deviations. Consequently, on 20 December 2010 EDF’s central services informed ASN of the discovery of a generic compliance gap (see the above description of the process used by EDF to manage compliance gaps). Characterisation of the deviation showed a total of 1170 non-compliant penetrations at 13 plants, with the consequence being potential propagation of flames in the event of fire, particularly in the control room. On 7 June 2012, EDF officially reported a generic compliance gap. Subsequent to IRSN’s recommendations, ASN requested that the penetrations be quickly brought into compliance. Upgrading was completed in late 2012.

The aim of fire compartmentalisation is to limit the spread of fire. More specifically, it relates to a set of measures taken to prevent fire from spreading from or into a space.
IRSN's guide on analysing fire hazards in basic nuclear installations (INB)


The approach described in the guide includes a step for checking the robustness of the safety demonstration, assuming failure of fire-protection measures, such as those used to limit the spread of flames.

Problem on the diesel engines of the emergency and station blackout generators of the 900 MWe reactors

In 2008 and 2009 EDF noted premature wear on the link-rod bearings of several diesel engines of the emergency generators at its 900 MWe reactors. As this wear could cause the diesel engines to malfunction, the damaged bearings were replaced by second-generation bearings.

*Exploded view of a piston showing the location of the connecting rod (in blue)*
However, in October 2010 two diesel engines equipped with these new bearings malfunctioned. Inspections conducted by EDF on two other diesel engines revealed abnormal wear of the bearings. EDF pre-emptively took heightened monitoring and maintenance measures to keep the engines at an acceptable level of reliability. IRSN had issued a first opinion on these measures to ASN in 2011 (see the ‘Problem on the diesel engines of the emergency and station blackout generators of the 900 MWe reactors’ article in French on page 42 of IRSN’s 2010 public report).

However, since April 2011, the second-generation bearings on five engines have had to be replaced for reasons of premature wear. On one of these engines, the bearings have even been replaced three times. Faced with the fact that the second-generation bearings are not a lasting solution, EDF conducted in-depth investigations throughout 2011. These investigations led it to conclude that the premature wear of the bearings might have been caused by insufficient lubrication caused by a geometrical difference. In 2012 EDF and its industrial partners therefore defined a new bearing, known as ‘2bis’, which eliminated the geometrical differences between the second-generation bearings and the original bearings.

In a second opinion issued in early 2013, IRSN considered that, in the light of the results of the qualification tests conducted, 2bis bearings appeared to solve the issue. However, it gave recommendations for the qualification of these bearings and the monitoring of their operation.

**Poor earthquake resistance of pneumatically controlled valves**

Certain pneumatically controlled valves (actuated by compressed air) must remain functional after an earthquake in order to bring the reactor back to a safe state. As a result, they are subjected to a qualification programme to ensure that they operate in such case. However, most of these valves are connected to the compressed air supply by hoses. In order for the valves to be qualified, these hoses must be fitted according to specific rules. If they are not fitted according to these rules and an earthquake occurs, they could break and prevent the valve from operating.

Several incorrectly fitted hoses were found during the outage of reactor 2 at the Cruas plant in 2011. The cause was the presence of T-shaped fittings on the compressed air supply hose of some valve actuators used to isolate the containment in the event of an accident. A check revealed that the procedures for fitting these hoses did not give any information on the rules regarding the set-up of the hoses. A similar deviation was subsequently found at the Civaux plant.

Considering that the deviation could be generic in nature, IRSN recommended in 2012 that ASN request EDF do the following at all its plants:

- inspect the compressed air supply hose on all seismic-qualified pneumatically controlled valves at the start of outages;
- replace non-compliant hoses that could prevent the pneumatically controlled valves from operating following an earthquake.

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Hoses must be sufficiently long and a minimum bend radius must be maintained to prevent them from being overly bent. Such hoses do not allow compressive movement or extension along their axis. Twisted hoses may break or cause their fittings to come loose. Hoses must be fitted such that they do not rub against surfaces or become tangled with other hoses.
As many non-compliant hoses were found, EDF wanted to define which valves were to be brought into compliance first. This led to many discussions with IRSN. Ultimately, on 31 May 2012 EDF asked the managers of its facilities to bring into compliance, starting with the first outage on 1 July 2012, non-compliant hoses that could compromise the correct operation of the pneumatically controlled valves needed to shut down the reactor following an earthquake compounded by failure of the offsite power supplies, this being the severest situation. The other valves will be brought into compliance during the next partial visit or the next ten-yearly review.
Contribution of people and organisations to equipment reliability

EDF regularly reports events during which the availability of safety-related equipment has been compromised. These events have many causes, some of which are technical in origin while others are related to human intervention. In 2012 IRSN analysed a sampling of these events and focused in particular on events related to operations or maintenance activities. One method of improving the reliability of equipment is to factor in, right from the design phase of equipment or upgrades, the activities of the people who will be using and maintaining it.

It is common practice in facility safety analyses to distinguish between the ‘technical’ reliability of equipment and ‘human’ reliability and to deal with each separately. While this distinction is often useful, a more comprehensive approach encompassing both types of reliability may result in more effective and lasting measures. The technical reliability of equipment is strongly tied to the human activities of final assembly, periodic testing and inspection and maintenance. While improving the reliability of these human activities helps to improve technical reliability, their reliability depends in part on equipment design. Some design choices can further complicate maintenance and operations activities — such as by imposing inconvenient working environments — and thus contribute to decreasing the reliability of these activities and the equipment in question.

Administrative lockouts

Each year, around 50,000 administrative lockouts are placed and removed at EDF’s plants. Between 2006 and 2011, 125 reports of safety-significant events (SSE) were related to administrative lockouts.

An analysis conducted by IRSN revealed that the design of the administratively locked-out equipment was partially responsible for one-quarter of these events.

Administrative lockout: a safety procedure used to ensure that safety-related equipment is locked in a given position and cannot be inadvertently actuated. This is accomplished using physical means such as padlocks, chains and seals. Administrative lockouts are placed on equipment whose position cannot be known from the control room. This is often the case for mechanical components such as valves. Very rarely does it entail electrical equipment (circuit breakers) or I&C equipment.
Some items of equipment do not have physical markings indicating their positions, making it difficult or even impossible to know, for example, whether they are open or closed. Before locking out equipment, maintenance workers may therefore have to actuate the equipment in order to determine its position or refer to the position the device is supposed to be in after being actuated during previous maintenance operations. This complicates administrative lockouts, increases the likelihood of placing equipment in the wrong position and makes it difficult to verify their position.

In addition to measures to train workers or organise the placement and removal of administrative lockouts, technical upgrades — such as installing position sensors on equipment, relaying position information to the control room, improving indications of valve positions and fitting actuators on equipment — can be a way to avoid equipment positioning issues.

The sequence of actions to be performed in order to actuate equipment is also hampered by difficulties. These actions may have to be carried out in a different order than that required for analogue devices. Twenty-odd devices at EDF’s 900 MW_e reactors are plagued by such difficulties. For example, the technology of the priming valve on the emergency turbo-generator (known as LLS) makes it hard for workers to lock it out in the closed position. Seven SSEs reported between 2006 and 2011 related to the unavailability of this valve. EDF plans to replace this valve by a more standard model.

**Maintenance activities**

Maintenance activities involved in the SSEs analysed consist of valve or pump maintenance and adjustments.

An analysis by IRSN revealed that problems with adjusting some valves were the cause of many SSEs, such as that which occurred at the Cruas 4 plant on 3 June 2012 and which led to a water spill in the reactor building (see the article about this event on page 44 of this report). Adjustments can sometimes be made difficult by a valve’s design and not all valves are adjusted in the same way. This can lead maintenance teams (mechanical, automation, valves) to make incorrect adjustments that can cause devices to either close only partially or not move at all when energised.

Equipment design features have also facilitated or failed to prevent the occurrence of a number of post-maintenance reassembly errors. These include incorrect assembly of the inner component of a valve on the oil line of a charging pump at Tricastin 3 on 24 April 2006, an orifice plate fitted in reverse on the auxiliary cooling system at Cattenom 2 on 19 July 2011, and orifice plates fitted in reverse on the auxiliary feedwater system of the steam generators at Dampierre 1 on 14 August 2012).

Such errors could be prevented by adequate foolproofing systems or physical markings that allow maintenance operators to more easily identify the direction of assembly of equipment. To prevent these events from occurring, EDF places importance on informing maintenance workers about these errors and the use of checklists during
maintenance operations. That said, retrofitting this equipment would virtually eliminate the possibility of these errors occurring. Until then, some events may recur.

Lubrication operations are a regular source of SSEs. A mixture of lubricants in the valve actuators at Nogent 2 on 13 July 2008, the use of the wrong grease-gun cartridge at Civaux 2 on 29 October 2008, the use of a non-used type of lubricant at Golfech 2 on 12 November 2008, and mixed lubricants at Blayais 3 and 4 on 25 August 2010, are a few examples. These inadequate means of lubrication and mixtures compromised the qualification or performance of the lubricated equipment (motor-operated valves or motor-pump assemblies). Given the risks involved and the safety significance of some of these events, EDF proposed to retrofit its equipment. This proposal was accepted by ASN in 2009. This retrofit is expected to be completed in 2019. Until then, EDF is implementing remedial measures (such as inspections and signage) to reduce the risk of these errors recurring. In 2012 a portion of the equipment at risk of lubrication errors was yet to be retrofitted and IRSN continued to follow the issue closely.

**Conclusion**

IRSN's analysis of EDF's operating experience feedback shows that some maintenance or operations activities can affect the reliability of safety-related equipment or systems. The risk of error during these activities can be reduced by even simple technical measures (signage, foolproofing systems, etc.).

The possibilities for improving administrative lockouts were the subject of scrutiny and recommendations by IRSN in 2012, particularly during operating experience feedback reviews and the ten-yearly safety reviews of EDF's reactors. EDF's experiments and proposed improvement measures were analysed by IRSN, which found them to be satisfactory.

More generally, France’s power plants could be made safer through better consideration of the constraints posed by equipment design on maintenance and operations activities. Operating experience feedback must systematically, thoroughly and meticulously be taken into account when designing equipment. Doing so can lead to retrofits that reduce sources of error.

Lastly, IRSN stresses that many of the aforementioned operations and maintenance activities require techniques and experience that are not explicitly set out in the operating documents provided to workers. This knowledge is an integral part of best industry practices and techniques. Experience in these techniques may be compromised if knowledgeable staff is replaced. Improving the design of equipment can therefore help to limit the effects of this.
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EDF: Pages 24, 25, 26, 27, 28, 29, 30, 38, 39, 40, 42, 49, 60 (Figure 2), 73, 74 (diagram), 76

IRSN : Pages 1 (foreword), 7, 9-15 (Figures 1 to 6), 17, 18, 33, 36, 47, 48, 50, 51, 59, 65, 67, 69, 74

Areva: Page 35

ASN: Page 60 (Figure 1)