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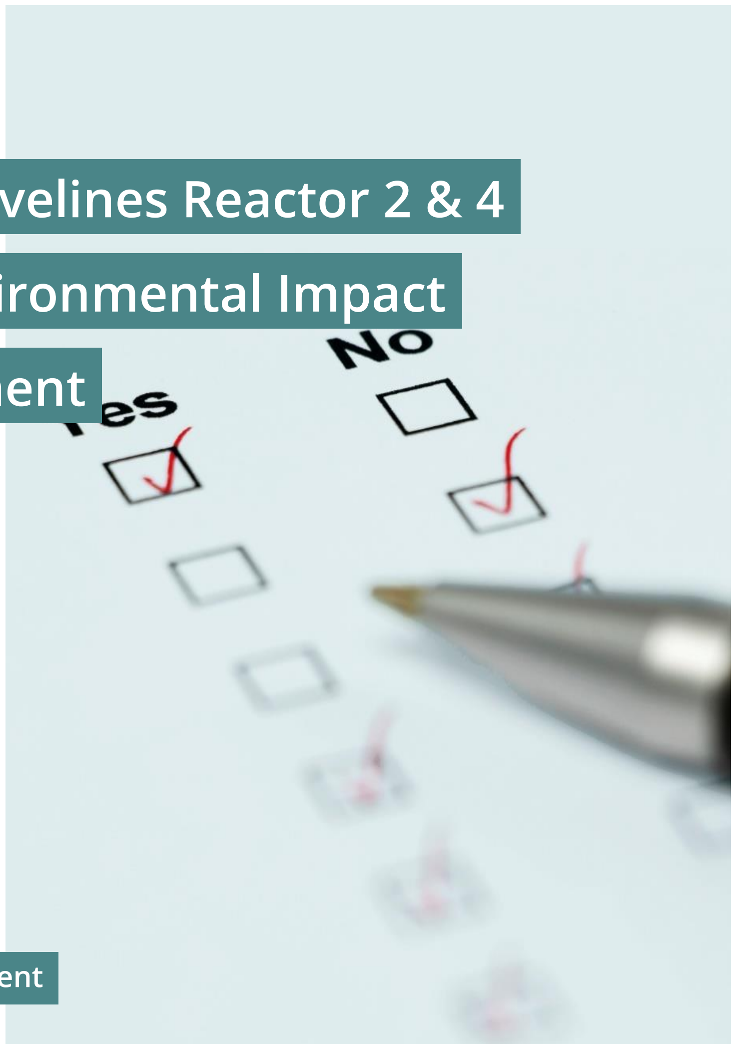
NPP Gravelines Reactor 2 & 4 LTO Environmental Impact Assessment



Expert Statement

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Expert Statement

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Agriculture and Forestry, Climate
and Environmental Protection,
Regions and Water Management
Republic of Austria

REPORT
REP-1014

VIENNA 2025

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Title photograph © iStockphoto.com/imagestock

Contracting authority Federal Ministry of Agriculture and Forestry, Climate and Environmental Protection, Regions and Water Management; Directorate General VI – Environment and Climate; Directorate VI/8 General Coordination of Nuclear Affairs

Publications For further information about the publications of the Umweltbundesamt please go to: https://www.umweltbundesamt.at/

Imprint

Owner and Editor: Umweltbundesamt GmbH
Spittelauer Laende 5, 1090 Vienna/Austria

This publication is only available in electronic format at https://www.umweltbundesamt.at/.

ISBN 978-3-99004-861-0

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SUMMARY

The Gravelines NPP consists of six pressurized water reactors with a capacity of 900 MWe each. These reactors were commissioned between 1980 and 1985. France notified the 4th Periodic Safety Review ("Public consultation procedure on the 4th safety review report") of the Gravelines nuclear power plant (reactor 2 and 4), which is to be considered as a lifetime extension in accordance with the UNECE Espoo Convention on Environmental Impact Assessment (EIA) in a Transboundary Context. The competent authority is the French department of the Préfecture du Nord. The project applicant is Électricité de France (EDF).

Austria is participating in this transboundary EIA, as significant impacts of an accident cannot be excluded. The aim of Austria's participation in the process is to give recommendations to minimize, and in the best case eliminate, possible significant adverse impacts on Austria.

Procedure

The operating authorization of French nuclear power plants is not limited in time. However, every ten years, French NPPs are subject to a Periodic Safety Review (PSR). The fourth PSR plays a special role, as it marks the regulatory process for the Long-Term Operation (LTO) of an NPP beyond 40 years. The French PSR framework mandates a comprehensive safety assessment in two phases: generic and plant specific.

For the 4th PSR of the 900 MWe nuclear power plants, EDF has set as a general guideline the objective of achieving the nuclear safety targets of the latest generation of reactors, whose reference reactor for EDF is the EPR-Flamanville 3. This guideline has been confirmed by the ASNR. The generic phase ended with the publication of the ASNR's opinion on February 23, 2021, which contained general regulations that had previously been the subject of a public consultation. (ASN 2021) Once the generic phase is complete, inspections of all 32 reactors at the 900 MWe nuclear power plants should follow over a period of approximately ten years (from 2019 to 2031).

There is a high degree of public involvement in the process of the life-time extension of the French NPP fleet. However, an EIA procedure according to the EIA Directive is not performed.

Long-Term operation and operational experience

Based on the information provided in the EIA documents, it can be concluded that a comprehensive aging management program was implemented to ensure operation. This is also indicated by the results of the first Topical Peer Review (TPR) as set out in Article 8e of Directive 2014/87/EURATOM. However, addressing the problems associated with the aging of structures, systems, and components (SSCs) poses a major challenge for the plant, which has been in operation for more than 40 years. Since most SSCs were originally designed for a nominal

operating lifetime of 40 years, the 4th PSR can be considered the necessary approval to operate the nuclear power plant beyond its original design life. Therefore, the 4th PSR requires a more detailed consideration of aging management. The EIA documents do not clearly indicate whether there has been a comprehensive expansion of the scope of aging management compared to the 3rd PSR. Only a few examples of preventive component replacement are presented. As far as is known, ASNR proposed expanding the scope of aging management during the generic phase of the 5th PSR. This should also be performed for the 4th PSR.

In the framework of the generic phase of the 5th PSR of 900 MWe reactors, the ASNR requires EDF to define, by December 31, 2025, the strategy for taking into account the findings from the discovery of stress corrosion cracking and, more generally, the risk of unexpected degradation of components in the primary and main secondary circuits through the checks required by the additional inspection and maintenance programs. The cause of the cracks, inter-crystalline stress corrosion, is a well-known corrosion phenomenon, but it was not expected in the relevant areas and therefore the pipes were not inspected for it either. This means that the aging management concept for components in the primary and main secondary circuits is called into question.

The ASNR's proposal during the generic phase of the 5th PSR to extend aging management beyond the 4th PSR is supported. As proposed by the ASNR, the focus must be on components that are necessary for controlling accident situations. However, the scope of the program “qualification of materials under accident conditions” in the 4th PSR is very limited for Gravelines 2 and 4.

The evaluation of safety-related incidents over the last five years revealed a high number of safety-related incidents that were classified as INES level 1. In addition to the events regarding deficiencies in earthquake protection, a number of events that compromised safety also occurred in both reactors. The reason for the large number of safety-related events could be a lack of safety culture combined with a large number of age-related events. Also noteworthy were the incidents involving contamination of workers and the blockage of the ultimate heat sink (UHS) by jellyfish.

External hazards

The EIA documents provide information on the hazards considered in the safety demonstration for the units 2 and 4 of Gravelines and on measures already implemented or decided to be implemented to strengthen the resistance of the reactors with respect to external hazards. For most hazards, however, methods, data and assumptions used in the hazard assessment are not specified. It remains particularly unclear if design basis events with exceedance frequencies not higher than 10^{-4} per annum have been determined for all external hazards as required by WENRA 2021, and how Design Extension Conditions (DEC) are addressed.

Non-conformity with WENRA Reference Levels is observed for earthquakes. The Design Basis Earthquake (DBE) is based on deterministic analyses which are no longer state of the art. A comparison of the currently deterministically determined DBE with a Probabilistic Safety Hazard Assessment (PSHA) based DBE at a mean return interval of 10,000 years is missing, although a PSHA was performed to determine the seismic SND value, which is relevant for the design of the hardened safety core. PSHA revealed a ground acceleration of 0,41 g for the seismic level for the hardened safety core (SND) with an average return period of 20,000 years. Documents do not state a ground motion value characterizing the 10,000 years earthquake which shall be used to define the seismic design basis of an existing NPP according to WENRA (2021). It is therefore unclear if the deterministically derived seismic design basis value for the Gravelines reactors, the SMS=0,2 g, envelopes the ground motion value of a PSHA-derived design basis earthquake with an average recurrence interval of 10,000 years. The high value for the SND=0,41 g suggests that this may be not the case. It therefore remains to be demonstrated that the seismic resistance of all SSCs important to safety is sufficient to conservatively ensure the fundamental safety functions for a DBE with an average recurrence interval of 10,000 years.

The implementation of the Hardened Safety Core, one of the most important measures to provide protection against external hazards, is still pending. ASNR prescribes the implementation to be completed in 2029. The fact that implementation is only scheduled 17 years after the fundamental decision by ASNR (ASN 2012) is remarkable given that WENRA requires the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). This suggests that the announced implementation schedules do not comply with the WENRA requirement.

Terrorist attacks and acts of sabotage can have a significant impact on nuclear facilities and cause serious accidents. Nevertheless, they are only mentioned in very general terms in the EIA documents submitted. Similar EIA reports have covered such events to a certain extent. Even if precautions against sabotage and terrorist attacks cannot be discussed in detail for reasons of confidentiality, the necessary legal requirements should be set out in the EIA documents.

Information regarding the issue of terror attacks would be of great interest, considering the far-reaching consequences of potential attacks. In particular, the EIA documents should include information on the requirements for the design against the targeted crash of a commercial aircraft. This topic is particularly important, because the reactor building as well as the spent fuel building of the Gravelines NPP is vulnerable against airplane crashes. It is important to mention that the EPR's 1.8-meter-thick outer reinforced concrete shell is designed to withstand the impact of a large passenger aircraft. However, the wall thickness at the Gravelines NPP is less than 1.0 m. Furthermore, the increasing availability and performance of drones is raising the potential threat to nuclear facilities. A recent assessment of the nuclear security in the France points to shortcomings compared to necessary requirements for nuclear security in regard to "security culture", "cybersecurity" and "protection against insider threats".

Safety aspect of accident without core-melt and spent fuel pools

The analysis utilizes both Deterministic Safety Analysis (DSA) and Probabilistic Safety Analysis (PSA) to re-evaluate operational transients, Design Basis Accidents (DBA), and Design Extension Conditions (DEC).

Significant safety enhancements have been implemented or are planned to reduce radiological consequences and improve defense-in-depth across the plant. An Augmented Ultimate Heat Sink Connection was implemented by diversifying the connection of the Steam Generator (SG) Auxiliary Feedwater System (ASG) to the Fire Fighting Water Reservoir; this secures long-term heat removal capability during accident sequences involving loss of normal and emergency feedwater. For thermal-hydraulic control, the capacity of the Main Steam Line Safety and Relief Valves (GCT-a) was uprated (PNPE1141) to accelerate the Reactor Coolant System (RCS) cooldown and depressurization following various transients. Furthermore, a lower permissible concentration of Iodine-131 (I-131) in the RCS water was enforced to reduce the potential radiological source term during accidents.

Regarding the Spent Fuel Pool (SFP), its integrity is supported by the implementation of mobile cooling capabilities (PTR bis), which align with post-Fukushima requirements for diverse, long-term cooling. The water supply to the SFP was strengthened, and the installation of flame arrestors in the SFP building is planned to prevent fire propagation. Finally, two key requirements set by the ASNR are currently outstanding: the validation of the Critical Heat Flux (CHF) correlation for deformed fuel assemblies (Study-B) and the final integration of findings regarding the Fuel Assembly Grid Buckling Limit (Study-D).

Safety aspects of core melt accidents

Severe accidents (SA) involving core meltdown were not taken into account in the design of the French 900 MWe reactors. However, as a result of previous PSRs, facilities and measures for SA management have been implemented. According to the ASNR, the objective of the fourth PSA for the 900 MWe reactors is to bring the safety level of the reactor closer to that of the EPR in Flamanville, a third-generation reactor. In third-generation reactors, features to mitigate the effects of core melt accidents are already implemented in the design; these cannot be fully transferred to second-generation reactors such as Gravelines 2 and 4. The EIA documents do not contain a systematic comparison between the safety level of the 900 MWe reactors and the safety level of the EPR in order to identify the remaining gaps.

The modifications planned as part of the 4th PSR in the event of a core-melt accident focus on heat removal from the containment without opening the filtered pressure relief system and on stabilizing and cooling the corium on the basement.

Based on current knowledge, a failure of the containment cannot be ruled out after the modification to stabilize and cool the molten core has been implemented. On the one hand, not all important modifications have been implemented yet, and on the other hand, it is not possible to assess whether the modifications (especially the reinforcement of the basement) are sufficient based on the available information.

The planned modifications for heat removal without using the filtered pressure relief system in the event of a core-melt accident have not yet been fully implemented. In addition, the reinforcement of the filtered pressure relief system (U5 system) against severe earthquakes has not yet been carried out. This means that even after completion of all Phase A measures of the 4th PSR, a core-melt accident with a major release of radioactive substances is still possible at Gravelines 2 and 4. The EIA documents do not provide a complete overview of which of the planned modifications meet the ASNR requirements published at the end of the generic phase of the 4th PSR. Most of the measures are not scheduled to be implemented until the end of phase B and the supplementary phase (2029). The EIA documents do not indicate whether this schedule will be adhered to.

Radiological impact of accidents / Transboundary effects

The EIA documents address events and accident sequences corresponding to three categories of design-basis accidents, as well as an additional category representing beyond design-basis events, including core-melt and fuel element storage pool scenarios.

The analysis of radiological consequences presented in the report lacks sufficient technical detail. Essential information required for independent verification, such as radionuclide inventories, source-term assumptions, release fractions, and the methodology for dispersion modelling, is not provided. Consequently, both the transparency and reproducibility of the radiological impact assessment are limited.

The EIA documents indicate that, for design-basis accidents, the radiological consequences are expected to remain below national reference levels and do not give rise to transboundary risks. For beyond design-basis accidents, including scenarios involving core melt, the report acknowledges the potential for long-range impacts, but lacks sufficient technical detail to allow independent verification of these findings. The report does not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances greater than 5 km after 7 days and within 1 km after one year. Additionally, the assessment omits information on ground deposition, despite its significance for evaluating long-term radiological impacts and potential contamination of the food chain.

Modelling of atmospheric dispersion and deposition conducted by the expert team demonstrate that, under certain meteorological conditions, a severe accident at Gravelines 2 and 4 could lead to ground deposition of Cs-137 in Austria above the national screening threshold of 650 Bq/m². Although the study does not assess the probability of such conditions, the results indicate that transboundary impacts greater than those implied in the EIA documents cannot be excluded.

Overall, the EIA documents provide an assessment of radiological consequences without providing complete information on assessment methodology and underlying data to support the claims, particularly for severe accidents with potential transboundary effects. More detailed source-term information, dispersion modelling inputs, and food-chain contamination assessments would be needed to fully evaluate the potential impact on Austria and to support the claims made in the EIA documents.

Assessment of the time frame

The timeframe for completing all measures under the 4th PSR (5 years after the release of the PSR report = 2029/2030) is not uncommon. However, as the period following the 4th PSR corresponds with the start of Long-Term Operation (LTO), some of the specific measures require special attention. It is important that the agreed implementation period is not extended. A lack of financial resources or the known problems with supply chain availability, including human resources, could affect the implementation period. It is particularly noteworthy that important safety modifications listed as part of the 4th PSR were already considered necessary as part of the EU stress test (2012), and their implementation had been agreed upon.

ZUSAMMENFASSUNG

Das Kernkraftwerk Gravelines besteht aus sechs Druckwasserreaktoren mit einer Leistung von jeweils 900 MWe. Diese Reaktoren wurden zwischen 1980 und 1985 in Betrieb genommen. Frankreich hat die vierte Periodische Sicherheitsüberprüfung („Öffentliches Anhörungsverfahren zum vierten Bericht zur Sicherheitsüberprüfung“) des Kernkraftwerks Gravelines (Reaktor 2 und 4) notifiziert, die als Laufzeitverlängerung gemäß der UNECE Espoo Konvention über die Umweltverträglichkeitsprüfung (UVP) im grenzüberschreitenden Rahmen zu betrachten ist. Die zuständige Behörde ist das französische Departement „Préfecture du Nord“. Die Antragstellerin des Projekts ist Électricité de France (EDF).

Österreich beteiligt sich an dieser grenzüberschreitenden UVP, da erhebliche Auswirkungen eines Unfalls nicht ausgeschlossen werden können. Ziel der Beteiligung Österreichs an diesem Verfahren ist es, Empfehlungen zur Minimierung und im besten Fall zur Vermeidung möglicher erheblicher nachteiliger Auswirkungen auf Österreich zugeben.

Verfahren

Die Betriebsgenehmigung für französische Kernkraftwerke ist zeitlich nicht begrenzt. Alle zehn Jahre werden französische Kernkraftwerke jedoch einer Periodischen Sicherheitsüberprüfung (PSÜ) unterzogen. Die vierte PSÜ spielt eine besondere Rolle, da sie den Genehmigungsprozess für den Langzeitbetrieb (Long-Term Operation, LTO) eines Kernkraftwerks über 40 Jahre hinaus markiert. Der französische PSÜ-Rahmen schreibt eine umfassende Sicherheitsbewertung in zwei Phasen vor: eine generische und eine anlagenspezifische Phase.

Für die 4. PSÜ der 900-MWe-Kernkraftwerke hat EDF als allgemeine Leitlinie das Ziel festgelegt, die nuklearen Sicherheitsziele der neuesten Reaktorgeneration zu erreichen, deren Referenzreaktor für EDF der EPR-Flamanville 3 ist. Diese Leitlinie wurde von der ASNR bestätigt. Die generische Phase endete mit der Veröffentlichung der Stellungnahme der ASNR am 23. Februar 2021, die allgemeine Vorschriften enthielt, die zuvor Gegenstand einer öffentlichen Konsultation gewesen waren. (ASN 2021) Nach Abschluss der generischen Phase sollen über einen Zeitraum von etwa zehn Jahren (von 2019 bis 2031) Inspektionen aller 32 Reaktoren der 900-MWe-Kernkraftwerke durchgeführt werden.

Die Öffentlichkeit ist in hohem Maße in den Prozess der Laufzeitverlängerung der französischen Kernkraftwerke eingebunden. Ein UVP-Verfahren gemäß der UVP-Richtlinie wird jedoch nicht durchgeführt.

Langzeitbetrieb und Betriebserfahrung

Auf der Grundlage der in den UVP-Unterlagen enthaltenen Informationen kann der Schluss gezogen werden, dass ein umfassendes Alterungsmanagementprogramm zur Gewährleistung des Betriebs durchgeführt wurde. Dies geht auch aus den Ergebnissen der ersten Topical Peer-Review (TPR) gemäß Artikel 8e der Richtlinie 2014/87/EURATOM hervor. Das Management der mit der Alterung von Strukturen, Systemen und Komponenten (SSCs) verbundenen Probleme stellt

jedoch eine große Herausforderung für das Kernkraftwerk dar, das seit mehr als 40 Jahren in Betrieb ist. Da die meisten SSCs ursprünglich für eine nominelle Betriebsdauer von 40 Jahren ausgelegt waren, kann die 4. PSÜ als die erforderliche Genehmigung für den Betrieb des Kernkraftwerks über seine ursprüngliche Auslegungsdauer hinaus angesehen werden. Daher erfordert die 4. PSÜ eine detailliertere Betrachtung des Alterungsmanagements. Aus den UVP-Unterlagen geht nicht eindeutig hervor, ob der Umfang des Alterungsmanagements im Vergleich zur 3. PSÜ umfassend erweitert wurde. Es werden nur wenige Beispiele für den vorbeugenden Austausch von Komponenten angeführt. Soweit bekannt, hat die ASNR vorgeschlagen, den Umfang des Alterungsmanagements während der generischen Phase der 5. PSÜ zu erweitern. Dies sollte auch für die 4. PSÜ durchgeführt werden.

Im Rahmen der generischen Phase der 5. PSÜ für 900-MWe-Reaktoren verlangt die ASNR von EDF, bis zum 31. Dezember 2025 eine Strategie zu definieren, um die Erkenntnisse aus der Entdeckung von Spannungsrisskorrosion und allgemeiner, des Risikos einer unerwarteten Degradierung der Komponenten im Primär- und Sekundärkreislauf durch Kontrollen im Rahmen der zusätzlichen Inspektions- und Wartungsprogramme zu berücksichtigen. Die Ursache der Risse, interkristalline Spannungsrisskorrosion, ist ein bekanntes Korrosionsphänomen, das jedoch in den betreffenden Bereichen nicht zu erwarten war und diese daher auch nicht darauf untersucht wurden. Damit wird das Alterungsmanagementkonzept für Komponenten im Primär- und Hauptsekundärkreislauf in Frage gestellt wird.

Der Vorschlag der ASNR das Alterungsmanagement während der 5. PSÜ gegenüber jenem der 4. PSÜ zu erweitern, wird unterstützt. Wie von der ASNR vorgeschlagen, muss der Schwerpunkt auf Komponenten liegen, die für die Beherrschung von Unfallsituationen notwendig sind. Der Umfang des Programms „Qualifizierung von Werkstoffen unter Unfallbedingungen" in der 4. PSÜ ist für Gravelines 2 und 4 jedoch sehr begrenzt.

Die Auswertung sicherheitsrelevanter Ereignisse der letzten fünf Jahre ergab eine hohe Anzahl sicherheitsrelevanter Ereignisse, die als INES-Stufe 1 Ereignisse eingestuft wurden. Neben den Ereignissen im Zusammenhang mit Mängeln beim Erdbebenschutz kam es in beiden Reaktoren auch zu einer Reihe von Ereignissen, die die Sicherheit beeinträchtigten. Der Grund für die große Anzahl sicherheitsrelevanter Ereignisse könnte ein Mangel an Sicherheitskultur in Verbindung mit einer großen Anzahl altersbedingter Ereignisse sein. Bemerkenswert waren auch die Vorfälle mit Kontamination von Mitarbeitern und die Blockierung der Wärmesenke durch Quallen.

Externe Gefahren

Die EIA-Dokumente enthalten Informationen zu den Gefahren, die bei den Sicherheitsnachweisen für die Blöcke 2 und 4 von Gravelines berücksichtigt wurden, sowie zu den bereits umgesetzten oder beschlossenen Maßnahmen zur Stärkung der Robustheit der Reaktoren gegenüber externen Gefahren. Für die meisten Gefahren werden jedoch die bei der Gefahrenbewertung verwendeten

Methoden, Daten und Annahmen nicht angegeben. Es bleibt insbesondere unklar, ob für alle externen Gefahren, wie von WENRA (2021) gefordert, Auslegungsereignisse mit einer Eintrittshäufigkeit von nicht mehr als 10^{-4} pro Jahr festgelegt wurden und wie mit den erweiterten Auslegungsbedingungen (DEC) umgegangen wird.

Bei Erdbeben wird eine Nichtkonformität mit den WENRA-Referenzlevel festgestellt. Das Auslegungserdbeben (DBE) basiert auf deterministischen Analysen, die nicht mehr dem neuesten Stand der Technik entsprechen. Ein Vergleich des derzeit deterministisch ermittelten DBE mit einem auf einer Probabilistischen Seismische Gefährdungsanalyse (PSHA) basierenden DBE bei einer mittleren Wiederkehrperiode von 10.000 Jahren fehlt, obwohl eine PSHA durchgeführt wurde, um den seismischen Wert für den Hardened Safety Core (SND) zu bestimmen, der für die Auslegung des Hardened Safety Core relevant ist. Die PSHA ergab eine Bodenbeschleunigung von 0,41 g für den SND mit einer mittleren Wiederkehrperiode von 20.000 Jahren. In den Unterlagen ist kein Wert für Bodenbewegung angegeben, der das 10.000-jährige Erdbeben charakterisiert, das gemäß WENRA (2021) zur Definition der seismischen Auslegungsgrundlage eines bestehenden Kernkraftwerks herangezogen werden soll. Es ist daher unklar, ob der deterministisch abgeleitete seismische Auslegungswert für die Reaktoren in Gravelines, SMS=0,2 g, den Bodenbewegungswert eines aus der PSHA abgeleiteten Auslegungserdbebens mit einer durchschnittlichen Wiederkehrperiode von 10.000 Jahren umfasst. Der hohe Wert für SND=0,41 g deutet darauf hin, dass dies möglicherweise nicht der Fall ist. Es bleibt daher noch zu zeigen, dass der Erdbebenschutz aller für die Sicherheit wichtigen SSCs ausreicht, um die grundlegenden Sicherheitsfunktionen für ein DBE mit einer durchschnittlichen Wiederkehrperiode von 10.000 Jahren konservativ zu gewährleisten.

Die Umsetzung des Hardened Safety Core, einer der wichtigsten Maßnahmen zum Schutz vor externen Gefahren, steht noch aus. Die ASNR schreibt vor, dass die Umsetzung bis 2029 abgeschlossen sein muss. Die Tatsache, dass die Umsetzung erst 17 Jahre nach der grundlegenden Entscheidung der ASNR (ASN 2012) geplant ist, ist bemerkenswert, da die WENRA die *„rechtzeitige Umsetzung der identifizierten, vernünftigerweise durchführbaren Sicherheitsverbesserungen"* verlangt (WENRA 2021, Referenzlevel A2.3). Dies deutet darauf hin, dass die angekündigten Umsetzungszeitpläne nicht den Anforderungen der WENRA entsprechen.

Terroranschläge und Sabotageakte können erhebliche Auswirkungen auf kerntechnische Anlagen haben und schwere Unfälle verursachen. Dennoch werden sie in den vorgelegten UVP-Unterlagen nur sehr allgemein erwähnt. Ähnliche UVP-Unterlagen haben solche Ereignisse bis zu einem gewissen Grad behandelt. Auch wenn Vorsorge gegen Sabotage und Terroranschläge aus Gründen der Vertraulichkeit nicht im Detail behandelt werden können, sollten die erforderlichen rechtlichen Anforderungen in den UVP-Unterlagen dargelegt werden.

Angesichts der weitreichenden Folgen potenzieller Anschläge wären Informationen zum Thema Terroranschläge von großem Interesse. Insbesondere sollten die UVP-Unterlagen Angaben zu den Anforderungen an die Auslegung gegen den gezielten Absturz eines Verkehrsflugzeugs enthalten. Dieses Thema ist besonders wichtig, da sowohl die Reaktorgebäude als auch das Gebäude für abgebrannte Brennelemente des Kernkraftwerks Gravelines durch Flugzeugabstürze gefährdet sind. Es ist wichtig zu erwähnen, dass die 1,8 m dicke äußere Stahlbetonhülle des EPR so ausgelegt ist, dass sie dem Aufprall eines großen Passagierflugzeugs standhält. Die Wandstärken im Kernkraftwerk Gravelines betragen jedoch weniger als 1,0 m. Darüber hinaus erhöhen die zunehmende Verfügbarkeit und Leistungsfähigkeit von Drohnen die potenzielle Bedrohung für kerntechnische Anlagen. Eine kürzlich durchgeführte Bewertung der nuklearen Sicherung in Frankreich weist auf Mängel im Vergleich zu den notwendigen Anforderungen an die nukleare Sicherung in Bezug auf die „Sicherungskultur“, die „Cybersicherheit“ und den „Schutz vor Insider-Bedrohungen“ hin.

Sicherheitsaspekte von Unfällen ohne Kernschmelze und im Brennelementlagerbecken

Die Analysen nutzen sowohl deterministische Sicherheitsanalysen als auch probabilistische Sicherheitsanalysen (PSA), um Betriebstransienten, Auslegungsstörfälle (DBA) und erweiterte Auslegungsbedingungen (DEC) neu zu bewerten.

Es wurden erhebliche Sicherheitsverbesserungen umgesetzt oder sind geplant, um die radiologischen Auswirkungen zu verringern und das gestaffelte Sicherheitskonzept im gesamten Kernkraftwerk zu verbessern. Durch die Diversifizierung der Verbindung des Hilfs-Speisewassersystems (ASG) des Dampferzeugers (SG) mit dem Löschwasserreservoir wurde eine verbesserte Verbindung zur Wärmesenke geschaffen, die eine langfristige Wärmeabfuhr bei Unfällen mit Ausfall der normalen und Not-Speisewasserversorgung gewährleistet. Zur thermohydraulischen Kontrolle wurde die Kapazität der Sicherheits- und Überdruckventile der Hauptdampfleitung (GCT-a) erhöht (PNPE1141), um die Abkühlung und Druckentlastung des Reaktorkühlsystems (RCS) nach verschiedenen Transienten zu beschleunigen. Darüber hinaus wurde eine niedrigere zulässige Konzentration von Jod-131 (I-131) im RCS-Wasser vorgeschrieben, um die potenziellen radiologischen Auswirkungen bei Unfällen zu reduzieren.

Die Integrität des Lagerbeckens für abgebrannte Brennelemente (SFP) wird durch die Implementierung mobiler Kühlkapazitäten (PTR bis) verbessert, die den Anforderungen nach Fukushima für eine diversitäre, langfristige Kühlung entsprechen. Die Wasserversorgung des SFP wurde verbessert, und die Installation von Flammensperren im SFP-Gebäude ist geplant, um eine Ausbreitung von Bränden zu verhindern. Schließlich sind derzeit noch zwei wichtige Anforderungen der ASNR offen: die Validierung der Korrelation des kritischen Wärmeflusses für deformierte Brennelemente (Studie B) und die endgültige Integration der Ergebnisse bezüglich des mechanischen Verhaltens der Brennelemente (Studie D).

Sicherheitsaspekte von Kernschmelzunfällen

Schwere Unfälle (SA) mit Kernschmelze wurden bei der Auslegung der französischen 900-MWe-Reaktoren nicht berücksichtigt. Als Ergebnis früherer PSÜ wurden jedoch Einrichtungen und Maßnahmen für das SA-Management implementiert. Laut ASNR besteht das Ziel der vierten PSÜ für die 900-MWe-Reaktoren darin, das Sicherheitsniveau der Reaktoren an das des EPR in Flamanville, einem Reaktor der dritten Generation, anzunähern. In Reaktoren der dritten Generation sind bereits Einrichtungen zur Minderung der Auswirkungen von Kernschmelzunfällen in der Auslegung integriert; diese können nicht vollständig auf Reaktoren der zweiten Generation, wie Gravelines 2 und 4, übertragen werden. Die UVP-Unterlagen enthalten keinen systematischen Vergleich zwischen dem Sicherheitsniveau der 900-MWe-Reaktoren und dem Sicherheitsniveau des EPR, um die verbleibenden Lücken zu ermitteln.

Die im Rahmen des 4. PSÜ geplanten Modifikationen für den Fall eines Kernschmelzunfalls konzentrieren sich auf die Wärmeabfuhr aus dem Sicherheitsbehälter ohne Öffnung des gefilterten Druckentlastungssystems sowie auf die Stabilisierung und Kühlung des Coriums auf dem Fundament.

Nach dem aktuellen Kenntnisstand kann ein Versagen des Sicherheitsbehälters nach Umsetzung der Modifikation zur Stabilisierung und Kühlung des geschmolzenen Kerns nicht ausgeschlossen werden. Zum einen sind noch nicht alle wichtigen Modifikationen umgesetzt, zum anderen lässt sich anhand der vorliegenden Informationen nicht beurteilen, ob die Modifikationenausreichend sind.

Die geplanten Modifikationen zur Wärmeabfuhr ohne Einsatz des gefilterten Druckentlastungssystems im Falle eines Kernschmelzunfalls sind noch nicht vollständig umgesetzt. Darüber hinaus wurde die Verstärkung des gefilterten Druckentlastungssystems (U5-System) gegen schwere Erdbeben noch nicht durchgeführt. Das bedeutet, dass auch nach Abschluss aller Maßnahmen der Phase A der 4. PSÜ ein Kernschmelzunfall mit einer erheblichen Freisetzung radioaktiver Stoffe in Gravelines 2 und 4 weiterhin möglich ist. Die UVP-Unterlagen geben keinen vollständigen Überblick darüber, welche der geplanten Modifikationen den am Ende der generischen Phase der 4. PSÜ veröffentlichten Anforderungen der ASNR entsprechen. Die meisten Maßnahmen sollen erst am Ende der Phase B und der ergänzenden Phase (2029) umgesetzt werden. Aus den UVP-Unterlagen geht nicht hervor, ob dieser Zeitplan eingehalten wird.

Radiologische Auswirkungen von Unfällen / Grenzüberschreitende Auswirkungen

Die UVP-Unterlagen befassen sich mit Ereignissen und Unfallabläufen, die drei Kategorien von Auslegungsstörfällen entsprechen, sowie einer zusätzlichen Kategorie, die auslegungsüberschreitende Unfälle umfasst, darunter Szenarien mit Kernschmelze und im Brennelementlagerbecken.

Die dargestellte Analyse der radiologischen Folgen weist keine ausreichenden technischen Details auf. Wesentliche Informationen, die für eine unabhängige

Überprüfung erforderlich sind, wie Radionuklidinventare, Annahmen zum Quellterm, Freisetzungsanteile und die Methodik für die Ausbreitungsmodellierung, werden nicht bereitgestellt. Folglich sind sowohl die Transparenz als auch die Reproduzierbarkeit der Bewertung der radiologischen Auswirkungen begrenzt.

Aus den UVP-Unterlagen geht hervor, dass bei Auslegungsstörfällen die radiologischen Folgen voraussichtlich unter den nationalen Referenzwerten bleiben und keine grenzüberschreitenden Auswirkungen verursachen. Bei auslegungsüberschreitenden Unfällen, einschließlich Szenarien mit Kernschmelze, räumt der Bericht zwar die Möglichkeit weitreichender Auswirkungen ein, enthält jedoch keine ausreichenden technischen Details, um eine unabhängige Überprüfung dieser Ergebnisse zu ermöglichen. Der Bericht enthält keine quantitativen Analysen, die die Aussage untermauern, dass die Kontamination von Lebensmitteln nach sieben Tagen in Entfernungen von mehr als 5 km und nach einem Jahr in Entfernungen von weniger als 1 km unter den EU-Grenzwerten bleiben würde. Darüber hinaus fehlen in der Bewertung Informationen zur Bodenkontamination, obwohl diese für die Bewertung der langfristigen radiologischen Auswirkungen und der potenziellen Kontamination der Nahrungskette von Bedeutung sind.

Die vom Expert:innenteam durchgeführten Modellierungen der atmosphärischen Ausbreitung und Bodenkontamination zeigen, dass unter bestimmten meteorologischen Bedingungen ein schwerer Unfall in Gravelines 2 und 4 zu einer Bodenkontamination von Cs-137 in Österreich führen könnte, die über dem nationalen Schwellenwert von 650 Bq/m^2 liegt. Obwohl die Studie die Wahrscheinlichkeit solcher Bedingungen nicht bewertet, deuten die Ergebnisse darauf hin, dass grenzüberschreitende Auswirkungen, die über die in den UVP-Unterlagen dargestellten hinausgehen, nicht ausgeschlossen werden können.

Insgesamt liefern die UVP-Unterlagen eine Bewertung der radiologischen Folgen, ohne jedoch vollständige Informationen über die Bewertungsmethodik und die zugrunde liegenden Daten zur Untermauerung der Aussagen zu liefern, insbesondere für schwere Unfälle mit potenziellen grenzüberschreitenden Auswirkungen. Um die potenziellen Auswirkungen auf Österreich vollständig bewerten und die Aussagen in den UVP-Unterlagen untermauern zu können, wären detailliertere Informationen zum Quellterm, Angaben zur Ausbreitungsmodellierung und zu Bewertungen der Kontamination der Nahrungskette erforderlich.

Bewertung des Zeitrahmens

Der Zeitrahmen für die Umsetzung aller Maßnahmen im Rahmen des 4. PSÜ (5 Jahre nach Veröffentlichung des PSÜ-Berichts = 2029/2030) ist nicht ungewöhnlich. Da der Zeitraum nach der 4. PSÜ jedoch mit dem Beginn des Langzeitbetriebs (LTO) übereinstimmt, erfordern einige der spezifischen Maßnahmen besondere Aufmerksamkeit. Es ist wichtig, dass der vereinbarte Umsetzungszeitraum nicht verlängert wird. Ein Mangel an finanziellen Ressourcen oder die bekannten Probleme mit der Verfügbarkeit der Lieferkette, einschließlich der Humanressourcen, könnten sich auf den Umsetzungszeitraum auswirken. Besonders bemerkenswert ist, dass wichtige Sicherheitsmodifikationen, die im Rahmen des 4. PSÜ aufgelistet sind, bereits im Rahmen des EU-Stresstests (2012) als notwendig erachtet wurden und deren Umsetzung vereinbart worden war.

RESUME

La centrale nucléaire de Gravelines comprend six réacteurs à eau pressurisée d'une puissance de 900 MWe chacun. Ces réacteurs ont été mis en service entre 1980 et 1985. La France a notifié le quatrième réexamen périodique (« Procédure de consultation publique sur le quatrième rapport du réexamen périodique ») de la centrale nucléaire de Gravelines (réacteurs 2 et 4), qui doit être considéré comme une prolongation de durée de vie conformément à la Convention d'Espoo de la CEE-ONU sur l'évaluation de l'impact sur l'environnement (EIE) dans un contexte transfrontalier. L'autorité compétente est le département français de la préfecture du Nord. Le demandeur du projet est Électricité de France (EDF).

L'Autriche participe à cette EIE transfrontalière, car des impacts significatifs d'un accident ne peuvent être exclus. L'objectif de la participation de l'Autriche à ce processus est de formuler des recommandations visant à minimiser, et dans le meilleur des cas à éliminer, les éventuels impacts négatifs significatifs sur l'Autriche.

Procédure

L'autorisation d'exploitation des centrales nucléaires françaises n'est pas limitée dans le temps. Cependant, tous les dix ans, les centrales nucléaires françaises sont soumises à un réexamen périodique (RP). Le quatrième RP joue un rôle particulier, car il définit le processus réglementaire pour l'exploitation à long terme (LTO) d'une centrale nucléaire au-delà de 40 ans. Le cadre français du RP impose une évaluation complète de la sûreté en deux phases : générique et spécifique à chaque centrale.

Pour le quatrième RP des centrales nucléaires de 900 MWe, EDF a fixé comme ligne directrice générale l'objectif d'atteindre le niveau de sûreté nucléaire des réacteurs de dernière génération, dont le réacteur de référence pour EDF est l'EPR-Flamanville 3. Cette ligne directrice a été confirmée par l'ASNR. La phase générique s'est achevée avec la publication de l'avis de l'ASNR le 23 février 2021, qui contenait des réglementations générales ayant fait précédemment l'objet d'une consultation publique. (ASN 2021) Une fois la phase générique terminée, les inspections des 32 réacteurs des centrales nucléaires de 900 MWe devraient être effectuées sur une période d'environ dix ans (de 2019 à 2031).

Le public est fortement impliqué dans le processus de prolongation de la durée de vie du parc nucléaire français. Néanmoins, une EIE conforme à la directive EIE n'est pas réalisée.

Exploitation à long terme et expérience opérationnelle

Sur la base des informations fournies dans les documents d'EIE, on peut conclure qu'un programme complet de gestion du vieillissement a été mis en œuvre pour garantir le fonctionnement. C'est également ce qu'indiquent les résultats du premier examen thématique par les pairs (Topical Peer Review- TPR)

prévu à l'article 8e de la directive 2014/87/ EURATOM. Cependant, la résolution des problèmes liés au vieillissement des structures, systèmes et composants (SSC) représente un défi majeur pour la centrale, qui est en service depuis plus de 40 ans. Étant donné que la plupart des SSC ont été initialement conçus pour une durée de vie nominale de 40 ans, le 4e RP peut être considéré comme l'autorisation nécessaire pour exploiter la centrale nucléaire au-delà de sa durée de vie initiale. Par conséquent, le 4e RP nécessite un examen plus approfondi de la gestion du vieillissement. Les documents d'EIE n'indiquent pas clairement s'il y a eu une extension complète du champ d'application de la gestion du vieillissement par rapport au 3e RP. Seuls quelques exemples de remplacement préventif de composants sont présentés. À notre connaissance, l'ASNR a proposé d'étendre la portée de la gestion du vieillissement pendant la phase générale du 5e RP. Cela devrait également être réalisé pour le 4e RP.

Dans le cadre de la phase générique du 5e RP des réacteurs de 900 MWe, l'ASNR demande à EDF de définir, d'ici le 31 décembre 2025, la stratégie visant à prendre en compte les conclusions tirées de la découverte de fissures de corrosion sous contrainte et, plus généralement, le risque de dégradation inattendue des composants des circuits primaire et secondaire principal à travers les contrôles requis par les programmes d'inspection et de maintenance supplémentaires. L'origine des fissures, la corrosion sous contrainte intercristalline, est un phénomène de corrosion bien connu, mais il n'était pas susceptible de se produire dans les zones concernées et les tuyaux n'ont donc pas été inspectés à cet effet. Cela signifie que le concept de gestion du vieillissement des composants des circuits primaire et secondaire principal est remis en question.

La proposition de l'ASNR, visant à étendre la gestion du vieillissement au-delà du 4e RP pendant la phase générale du 5e RP est soutenue. Comme le propose l'ASNR, l'accent doit être mis sur les composants nécessaires au contrôle des situations accidentelles. Cependant, la portée du programme « qualification des matériels aux conditions accidentelles » du 4e RP est très limitée pour Gravelines 2 et 4.

L'évaluation des incidents liés à la sûreté au cours des cinq dernières années a révélé un nombre élevé d'incidents classés au niveau 1 de l'échelle INES. Outre les événements liés à des lacunes en matière de protection contre les séismes, plusieurs événements compromettant la sûreté se sont également produits dans les deux réacteurs. Le nombre élevé d'événements liés à la sûreté pourrait s'expliquer par un manque de culture de sûreté combiné à un grand nombre d'événements liés à l'âge des installations. Il convient également de noter les incidents impliquant la contamination de travailleurs et le blocage du dissipateur thermique ultime par des méduses.

Risques externes

Les documents EIE fournissent des informations sur les risques pris en compte dans la démonstration de sûreté des unités 2 et 4 de Gravelines et sur les mesures déjà mises en œuvre ou dont la mise en œuvre a été décidée afin de renforcer la résistance des réacteurs face aux risques externes. Pour la plupart des risques, les méthodes, les données et hypothèses utilisées dans l'évaluation des risques ne sont cependant pas précisées. Il reste particulièrement difficile de savoir si des événements de référence dont la fréquence de dépassement n'est pas supérieure à 10^{-4} par an ont été déterminés pour tous les risques externes, comme l'exige la WENRA 2021, et comment les conditions d'extension de la conception (en anglais DEC) sont traitées.

Une non-conformité avec les niveaux de référence de la WENRA est observée pour les séismes. Le séisme de référence (SMS) est basé sur des analyses déterministes qui ne sont plus à la pointe de la technologie. Il manque une comparaison entre le SMS actuellement déterminé de manière déterministe et un SMS basé sur une évaluation probabiliste des risques pour la sûreté (PSHA) avec un intervalle de retour moyen de 10 000 ans, bien qu'une PSHA ait été réalisée pour déterminer la valeur SND sismique, qui est pertinente pour la conception du Noyau Dure. La PSHA a révélé une accélération du sol de 0,41 g pour le niveau sismique du Noyau Dur (SND) avec une période de retour moyenne de 20 000 ans. Les documents ne mentionnent pas de valeur de mouvement du sol caractérisant le séisme de 10 000 ans qui doit être utilisée pour définir la base de conception sismique d'une centrale nucléaire existante selon la WENRA (2021). Il n'est donc pas clair si la valeur de base de conception sismique dérivée de manière déterministe pour les réacteurs de Gravelines, le SMS = 0,2 g, englobe la valeur du mouvement du sol d'un séisme de base de conception dérivé de l'analyse PSHA avec une période de récurrence moyenne de 10 000 ans. La valeur élevée de SND = 0,41 g suggère que cela pourrait ne pas être le cas. Il reste donc à démontrer que la résistance sismique de tous les SSC importants pour la sûreté est suffisante pour garantir de manière conservatrice les fonctions de sûreté fondamentales pour un DBE avec un intervalle de récurrence moyen de 10 000 ans.

La mise en œuvre du Noyau Dur, l'une des mesures les plus importantes pour assurer la protection contre les aléas externes, est toujours en suspens. L'ASNR impose que la mise en œuvre soit achevée en 2029. Le fait que la mise en œuvre ne soit prévue que 17 ans après la décision fondamentale de l'ASNR (ASN 2012) est remarquable étant donné que la WENRA exige la « mise en œuvre en temps utile des améliorations raisonnablement réalisables en matière de sécurité qui ont été identifiées » (WENRA 2021, niveau de référence A2.3). Cela suggère que les calendriers de mise en œuvre annoncés ne sont pas conformes à l'exigence de la WENRA.

Les attentats terroristes et les actes de sabotage peuvent avoir un impact significatif sur les installations nucléaires et provoquer des accidents graves. Néanmoins, ils ne sont mentionnés qu'en termes très généraux dans les documents d'EIE soumis. Des rapports d'EIE similaires ont couvert ces événements dans une certaine mesure. Même si les précautions contre le sabotage et les attentats terroristes ne peuvent être discutées en détail pour des raisons de confidentialité, les exigences légales nécessaires devraient être énoncées dans les documents d'EIE.

Les informations relatives aux attentats terroristes seraient d'un grand intérêt, compte tenu des conséquences considérables que pourraient avoir de telles attaques. Les documents d'EIE devraient notamment inclure des informations sur les exigences en matière de conception visant à prévenir le crash ciblé d'un avion commercial. Ce sujet est particulièrement important, car le bâtiment du réacteur ainsi que le bâtiment de stockage du combustible usé de la centrale nucléaire de Gravelines sont vulnérables aux crashs d'avion. Il est important de mentionner que l'enveloppe extérieure en béton armé de 1,8 m d'épaisseur de l'EPR est conçue pour résister à l'impact d'un gros avion de ligne. Cependant, l'épaisseur des murs de la centrale nucléaire de Gravelines est inférieure à 1,0 m. En outre, la disponibilité et les performances croissantes des drones augmentent la menace potentielle pour les installations nucléaires. Une évaluation récente de la sécurité nucléaire en France met en évidence des lacunes par rapport aux exigences nécessaires en matière de sécurité nucléaire en ce qui concerne la « culture de la sécurité », la « cybersécurité » et la « protection contre les auteurs menaces internes ».

Aspects liés à la sûreté en cas d'accident sans fusion du cœuret Piscine d’entreposage du combustible usé

L'analyse utilise à la fois l'analyse déterministe de sûreté et l'analyse probabiliste de sûreté (EPS) pour réévaluer les transitoires opérationnels, les accidents de conception (en anglais DBA) et les conditions d'extension de conception (en anglais DEC).

Des améliorations importantes en matière de sûreté ont été mises en œuvre ou sont prévues afin de réduire les conséquences radiologiques et d'améliorer la défense en profondeur dans l'ensemble de la centrale. Une connexion renforcée au dissipateur thermique ultime a été mise en place en diversifiant la connexion du Système d’alimentation auxiliaire en eau du générateur de vapeur (ASG) au réservoir d'eau d'extinction d'incendie ; cela garantit une capacité d'évacuation de la chaleur à long terme lors d'accidents impliquant une perte d'alimentation en eau normale et d'urgence. Pour le contrôle thermohydraulique, la capacité des vannes de sécurité et de décharge sur la conduite de vapeur principale (GCT-a) a été augmentée (PNPE1141) afin d'accélérer le refroidissement et la dépressurisation du système de refroidissement du réacteur à la suite de divers transitoires. En outre, une concentration admissible plus faible d'iode 131 (I-131) dans l'eau du système de refroidissement a été imposée afin de réduire l’activité des éventuels rejets radioactifs en cas d'accident.

En ce qui concerne la piscine d’entreposage du combustible usé (en anglais SFP), son intégrité est renforcée par la mise en place de capacités de refroidissement mobiles (PTR bis), conformément aux exigences post-Fukushima en matière de refroidissement diversifié et à long terme. L'alimentation en eau de la SFP a été renforcée et l'installation de pare-flammes dans le bâtiment de la SFP est prévue afin d'empêcher la propagation du feu. Enfin, deux exigences clés fixées par l'ASNR sont actuellement encore en suspens : la validité de la corrélation de flux critique en présence d’assemblages déformés latéralement (étude

B) et l'intégration finale des conclusions concernant le comportement mécanique des assemblages de combustible (étude D).

Aspects de sûreté des accidents de fusion du cœur

Les accidents graves (SA) impliquant une fusion du cœur n'ont pas été pris en compte dans la conception des réacteurs français de 900 MWe. Cependant, à la suite des examens périodiques de sûreté (RP) précédents, des installations et des mesures de gestion des SA ont été mises en place. Selon l'ASNR, l'objectif de la quatrième RP pour les réacteurs de 900 MWe est de rapprocher le niveau de sûreté du réacteur de celui de l'EPR de Flamanville, un réacteur de troisième génération. Dans les réacteurs de troisième génération, des dispositifs visant à atténuer les effets des accidents de fusion du cœur sont déjà intégrés dans la conception ; ceux-ci ne peuvent pas être entièrement transposés aux réacteurs de deuxième génération tels que Gravelines 2 et 4. Les documents d'EIE ne contiennent pas de comparaison systématique entre le niveau de sûreté des réacteurs de 900 MWe et celui de l'EPR afin d'identifier les écarts restants.

Les modifications prévues dans le cadre du 4e RP en cas d'accident de fusion du cœur se concentrent sur l'évacuation de la puissance résiduelle du cœur sans ouverture du dispositif de décompression et filtration de l'enceinte (dispositif dit U5) et sur la stabilisation du corium sur le radier du bâtiment réacteur par son étalement et son renoyage.

Sur la base des connaissances actuelles, une défaillance de l'enceinte de confinement ne peut être exclue après la mise en œuvre de la modification visant à stabiliser et à refroidir le cœur fondu. D'une part, les modifications importantes n'ont pas encore toutes été mises en œuvre et, d'autre part, il n'est pas possible d'évaluer si les modifications (en particulier le renforcement du bâtiment réacteur) sont suffisantes compte tenu des informations disponibles.

Les modifications prévues pour sur l'évacuation de la puissance résiduelle du cœur sans ouverture du dispositif de décompression et filtration de l'enceinte en cas d'accident de fusion du cœur n'ont pas encore été entièrement mises en œuvre. En outre, le renforcement du système de décompression filtré (système U5) contre les séismes violents n'a pas encore été réalisé. Cela signifie que même après l'achèvement de toutes les mesures de la phase A du 4e RP, un accident de fusion du cœur avec un rejet important de substances radioactives est toujours possible à Gravelines. Les documents d'EIE ne fournissent pas un aperçu complet des modifications prévues qui répondent aux exigences de l'ASNR publiées à la fin de la phase générique de la 4e RP. La plupart des mesures ne sont pas prévues avant la fin de la phase B et de la phase supplémentaire (2029). Les documents d'EIE n'indiquent pas si ce calendrier sera respecté.

Impact radiologique des accidents / Effets transfrontaliers

Les documents EIE traitent des événements et des séquences d'accidents correspondant à trois catégories d'accidents de base, ainsi qu'à une catégorie supplémentaire représentant des accidents dépassant les limites de conception, y compris des scénarios de fusion du cœur et de piscine d’entreposage du combustible usé.

L'analyse des conséquences radiologiques présentée dans le rapport manque de détails techniques suffisants. Les informations essentielles nécessaires pour une vérification indépendante, telles que les inventaires des radionucléides, les hypothèses relatives au terme source, les fractions de libération et la méthodologie de modélisation de la dispersion, ne sont pas fournies. Par conséquent, tant la transparence que la reproductibilité de l'évaluation de l'impact radiologique sont limitées.

Les documents EIE indiquent que, pour les accidents de référence, les conséquences radiologiques devraient rester inférieures aux niveaux de référence nationaux et ne pas entraîner de risques transfrontaliers. Pour les accidents dépassant les limites de conception, y compris les scénarios impliquant une fusion du cœur, le rapport reconnaît le potentiel d'impacts à longue distance, mais manque de détails techniques suffisants pour permettre une vérification indépendante de ces conclusions. Le rapport ne présente pas d'analyses quantitatives pour appuyer les affirmations selon lesquelles la contamination alimentaire resterait inférieure aux limites fixées par l'UE à des distances supérieures à 5 km après 7 jours et à moins de 1 km après un an. En outre, l'évaluation omet les informations sur les dépôts au sol, malgré leur importance pour l'évaluation des impacts radiologiques à long terme et de la contamination potentielle de la chaîne alimentaire.

La modélisation de la dispersion atmosphérique et déposition réalisée par l'équipe d'experts démontre que, dans certaines conditions météorologiques, un accident grave aux réacteurs 2 et 4 de Gravelines pourrait entraîner des contaminations du sol de Cs-137 en Autriche supérieurs au seuil national de 650 Bq/m². Bien que l'étude n'évalue pas la probabilité de telles conditions, les résultats indiquent que des impacts transfrontaliers supérieurs à ceux impliqués dans les documents d'EIE ne peuvent être exclus.

Dans l'ensemble, les documents d'EIE fournissent une évaluation des conséquences radiologiques sans donner d'informations complètes sur la méthodologie d'évaluation et les données sous-jacentes à l'appui des affirmations, en particulier pour les accidents graves ayant des effets transfrontaliers potentiels. Des informations plus détaillées sur le terme source, les données utilisées pour la modélisation de la dispersion et les évaluations de la contamination de la chaîne alimentaire seraient nécessaires pour évaluer pleinement l'impact potentiel sur l'Autriche et appuyer les affirmations contenues dans les documents d'EIE.

Évaluation du calendrier

Le calendrier de mise en œuvre de toutes les mesures du 4e RP (5 ans après la publication du rapport RP = 2029/2030) n'est pas inhabituel en principe. Cependant, comme la période suivant le 4e RP correspond au début de l'exploitation à long terme (LTO), certaines mesures spécifiques nécessitent une attention particulière. Il est important que la période de mise en œuvre convenue ne soit pas prolongée. Le manque de ressources financières ou les problèmes connus liés à la disponibilité de la chaîne d'approvisionnement, y compris les ressources humaines, pourraient avoir un impact sur la période de mise en œuvre. Il convient de noter en particulier que d'importantes modifications de sécurité figurant dans la liste des modifications du 4e RP avaient déjà été jugées nécessaires dans le cadre du test de résistance de l'UE (2012) et que leur mise en œuvre avait été convenue.

1 INTRODUCTION

The Gravelines NPP consists of six pressurized water reactors with a capacity of 900 MWe each. These reactors were commissioned between 1980 and 1985.

France notified the 4th Periodic Safety Review ("Public consultation procedure on the 4th safety review report") of the Gravelines nuclear power plant (reactor 2 and 4), which is to be considered as a lifetime extension in accordance with the UNECE Espoo Convention on Environmental Impact Assessment (EIA) in a Transboundary Context. The competent authority is the French department Préfecture du Nord. The project applicant is Électricité de France (EDF).

Austria is participating in this transboundary EIA, as significant impacts of an accident cannot be excluded. The aim of Austria's participation in the process is to give recommendations to minimize, and in the best case eliminate, possible significant adverse impacts on Austria.

The Austrian Federal Ministry of Agriculture and Foresty, Climate and Environmental Protection, Regions and Water Management commissioned the Environment Agency Austria to coordinate the assessment of the submitted EIA documents in the framework of an expert statement.

2 PROCEDURE

2.1 Treatment in the EIA documents

The operating authorization of French nuclear power plants (NPPs) is not limited in time. However, every ten years, French NPPs are subject to a Periodic Safety Review (PSR), known in France as the Réexamen Périodique de Sûreté.

While NPPs are continuously inspected, a PSR involves a comprehensive evaluation of the state of structures, systems, and components (SSCs). It serves two main functions: a Conformity Check to verify plant components match their required safety standards, and a Safety Reassessment that compares the plant against current norms. The review aims to demonstrate that safety requirements will be fulfilled for at least ten years following the approval of the PSR.

The fourth PSR plays a special role, as it marks the regulatory process for the Long-Term Operation (LTO) of an NPP beyond 40 years. Since most SSCs were originally designed with a nominal 40-year lifespan in mind, the 4th PSR can be viewed as the authorization required to operate the NPP beyond its initial design life. Therefore, the 4th PSR includes a closer look at aging management and LTO-specific issues.

Aging affects not only the physical SSCs but also the regulatory framework. The safety standards according to which the NPP was designed often become superseded by more modern, stricter standards. Feedback from severe accidents has consistently driven the evolution of these standards, raising the bar for NPP design. Consequently, one aspect of the 4th PSR is to identify deltas (gaps) between the current design basis of the NPP and the modern state-of-the-art. The process requires proposing measures for backfitting (safety up-grades) the NPP to minimize these deltas as far as reasonably achievable. EDF and the ASNR have agreed to benchmark the safety levels of the French NPPs undergoing their 4th PSR against the standards applied to the EPR Flamanville 3 reactor, which is considered the current state-of-the-art reference.

The French NPP fleet can be broadly divided into three classes of NPPs. NPPs in each class were commissioned close to each other in time and share largely similar technology.

900 MWe reactors (32 units):

- Timeline: Construction largely spanned from the early 1970s to the late 1980s.
- Sub-types: Divided into type CP0, type CP1, and type CP2. The CP0 units were the earliest to be commissioned followed by the larger CP1 and CP2 series (e.g., Tricastin, Gravelines, Chinon).

1300 MWe reactors (20 units):

- Timeline: Construction periods generally started in the late 1970s and continued into the late 1990s.

- Sub-types: Divided into type P4 and type P'4. Plants include Paluel, Cattenom, and Belleville.

1450 Mwe Reactors (4 units):

- Timeline: Represents the latest series, with construction starting around the mid-1980s and concluding around 2000.
- Sub-types: Designated as type N4. Plants are Chooz B and Civaux.

The subject of this report is the 900 MWe fleet. The 900 MWe fleet consists of 32 reactors of the CP type, which are 3-loop pressurized water reactors. This fleet includes three sub-types: CP0, CP1, and CP2 (with CP1 and CP2 often jointly referred to as CPY). While Fessenheim units 1 and 2 were permanently shut down, EDF is planning to extend the operational life of all the other units beyond forty years. (ASN 2022)

France is conducting the 4th PSR in two phases a generic and a specific phase. For the 4th PSR of the 900-MWe nuclear power plants, EDF has set as a general guideline the objective of achieving the nuclear safety targets of the latest generation of reactors, whose reference reactor for EDF is the EPR-Flamanville 3. This guideline has been confirmed by the ASNR. The generic phase ended with the publication of the ASNR's opinion on February 23, 2021, which contained general regulations that had previously been the subject of a public consultation. (ASN 2021)

Once the generic phase is complete, inspections of all 32 reactors at the 900 MWe nuclear power plants should follow over a period of approximately ten years (from 2019 to 2031). EDF submits a review report to the government and the ASNR. This is prepared after the ten-year reactor inspection, during which modifications and inspection and maintenance work are carried out. The following timeline shows the main stages of the 4th PSR for Gravelines 4.

Figure 1: Main stages of the 4th PSR for Gravelines 4 (EIA-REPORT G4 P.1 2025)

Main stages of the 4th PSR for Gravelines 4

Source: Environment Agency Austria

umweltbundesamt

Public Involvement in the PSR

Several steps were taken to involve the public in the generic phase of the 4th PSR of the 900 MWe reactors. These steps were designed to inform the public, facilitate the understanding of complex safety issues, explain the ASNR requirements associated with the review, and gather the expectations and positions of the various contributors.

The ASNR involved the public as early as 2016 in the development of its position on the "major objectives" of the 4th PSR of the 900 MWe reactors. This approach was continued in the development of its generic resolution on the 4th periodic safety review in early 2021. (ASN 2021)

While the public involvement process had similarities to an EIA, France always emphasized that the process is not to be seen as an EIA following the EU EIA Directive. Instead, France requested the High Committee for Transparency and Information on Nuclear Safety (HCTINS) to organize the process. Public comments for the specific phase for the NPP Gravelines, for instance, are possible until December 2025.

2.2 Discussion

There is a high degree of public involvement in the process of the lifetime extension of the French NPP fleet. However, an EIA according to the EIA Directive is not performed.

2.3 Conclusion

Since all the important elements of an EIA are present in the process, it is difficult to see why the last step, to implement the consultation in the frame of an EIA process, has not been taken.

3 LONG-TERM OPERATION AND OPERATIONAL EXPERIENCE

3.1 Treatment in the EIA documents

Ageing and obsolescence control

The EIA-REPORTs G2/G4 P.2 (2025) deal with the Ageing Management. The approach to controlling aging and dealing with obsolescence is based on three sustainable operational processes:

- the process for controlling the aging of structures, systems, and components (SSCs), which is being continued in the 4th PSR,
- the process of inspection during operation and maintenance,
- the process for addressing the obsolescence of materials and spare parts.

It is stated that the method used is in line with international best practices and consistent with the approach recommended by the IAEA in its Safety Guide No. NS-G-2.12 *"Ageing Management for Nuclear Power* Plants." (EIA-REPORT G2/G4 P.2 2025)

The main measures taken or proposed by the operator in this area have two objectives:

1 Proof of functionality of non-replaceable components after 40 years:

The operational reliability of the **reactor pressure vessel** has been proven using a conservative deterministic approach (neutron physics, materials, mechanics, etc.).

The mechanical performance of the **containment** is continuously monitored by monitoring devices (e.g., deformation measurement). A pressure test of the containment is performed during each ten-year inspection. This test was carried out for Gravelines 4 from June 6, 2024 to July 1, 2024 with the results meeting expectations. This test was carried out on the containment at Gravelines 2 from December 15 to 18, 2023, with results in line with expectations.

2 Proof of the functionality of replaceable materials after 40 years, which would otherwise be either replaced or modernized.

Components whose performance may deteriorate due to aging and whose failure may have an impact on safety are documented and regularly inspected. In this context, inspections, checks, and maintenance work were carried out on the following SSCs during the 4th PSR: structures, control and monitoring systems, electrical cables, mechanical and electromechanical equipment, electrical equipment, and instrumentation.

Following completion of the aging control analysis of the SSCs of Gravelines 2 and 4, maintenance and control measures were carried out, along with modifications to ensure the continued suitability of those units for operation for a period of ten years after the 4th PSR shutdown.

Risk of obsolescence

Controlling the risk of component obsolescence is based in particular on monitoring the availability of spare parts, their procurement and, if necessary, ordering new identical or equivalent equipment. This equipment is then subjected to the same qualification tests as the original equipment. As part of the 4th PSR of the 900 MWe reactors, EDF plans, for example, to replace certain control and monitoring devices and certain components of switchboards.

Dossier of Suitability for Continued Operation" (DAPE)

The "Dossier of Suitability for Continued Operation" (DAPE) examines in detail the control of aging risks for a component or a structure. It describes the associated aging management program, including aspects such as in-service monitoring, regular and extraordinary maintenance, operating conditions, possible changes, supplementary studies, R&D programs, laboratory tests, particularly in the field of materials, quality assurance procedures, etc. The DAPEs are updated every five years. (EIA REPORT G2/G4 P.2 2025)

There are currently 12 DAPE for the following components for the 900 MWe reactors:

- Reactor pressure vessel,
- Internal core components,
- Steam generators,
- Primary piping,
- Pressurizer,
- Primary motor pump group,
- Auxiliary lines of the primary main circuit,
- Power cables,
- Electrical penetrations,
- Control system,
- Containment,
- Structures.

The studies conducted for the 900 MWe CPY reactors were adopted by the Gravelines nuclear power plant in order to decide on the management of the aging of structures, systems, and components (SSC) in units 2 and 4. The Gravelines nuclear power plant teams thus adopted the studies conducted at the generic level and identified any specific features related to the SSC of Units 2 and 4. On this basis, index 00 of the DAPE was created and submitted to the ASNR one year before the start of the ten-year inspection. During the ten-year inspection, the SSC underwent a series of maintenance operations, inspections, tests, non-destructive tests, or modifications. (EIA REPORT G2/G4 P.2 2025)

Program for Complementary Investigations (PIC)

The implementation of the Program for Complementary Investigations (PIC) is an approach that aims to confirm the absence of operational failures in areas that are not regularly inspected. As part of the 4th PSR, the following areas were selected for the PIC:

- mechanical equipment of the primary and secondary circuit,
- other mechanical equipment: piping, heat exchangers, pumps, valves,
- containment.

For Gravelines 2, a visual inspection of the weld seam at the bottom of the feedwater storage tank for the steam generators was carried out as part of the spot check. No deviations were found during this inspection. (EIA-REPORT G2 P.1 2025)

Without justification, it is stated that no checks are to be carried out for Gravelines 4 as part of the supplementary investigation program. (EIA-REPORT G4 P.1 2025)

Stress corrosion of the auxiliary lines

As part of the proceedings initiated at the end of 2021 concerning “stress corrosion” on the auxiliary lines of the main primary circuit, investigations on the various reactors have shown that 900 MWe reactors such as those at Gravelines are hardly susceptible, if at all, to this phenomenon. In consultation with ASNR, a strategy for dealing with the nuclear power plants and a corresponding inspection program were established. (EIA-REPORT G2/G4 P.1 2025)

Objectives for the “continued operation after 40 years” of the 4th PSR

The 4th PSR of the 900 MWe reactors provides for a comprehensive work program on the aging of the plants as part of the continued operation of the plants after 40 years. The approach is based on aging management and maintaining the qualification of materials under accident conditions.

Qualification of materials under accident conditions

The objective of the “qualification of materials under accident conditions” is to verify that the organizational provisions required to ensure the sustainability of the qualification are in place. Verification of the organizational provisions was carried out and 257 materials qualified under accident conditions were physically inspected in Gravelines 2, and 258 materials qualified under accident conditions were physically inspected in Gravelines 4. All checks required under this program were carried out. Anomalies were analyzed and/or corrected. (EIA-REPRT G2/G4 P.2 2025)

Maintaining qualification under accident conditions is subject to a procedure based on several verification methods, ranging from document analysis and sampling for testing to replacement. The result of this step-by-step and comprehensive procedure involves a considerable amount of work and makes it possible to guarantee the extension of the service life up to the 5th PSR.

The following two projects are proposed by EDF (EIA-REPORT G2/G4 P.3 2025):

- Ensuring the qualification under accident conditions of an activity measurement chain in the reactor building after more than 40 years of operation.
- Ensuring the qualification under accident conditions for distribution boxes and cabinets of the electrical components of the emergency power supply system that are more than 40 years old.

Safety relevant events

According to the EIA-REPORT G2/G4 P.1 (2025), between January 2013 and December 2022, the Gravelines NPP reported 60 significant events. None of these had any noticeable impact on the environment. Each time, corrective and preventive measures were implemented and their effectiveness was verified. This analysis of ten years of operating experience confirms that the management of significant events is correctly integrated into the Gravelines power plant's management system.

It is further explained that, at the time of publication of the EIA report, the Gravelines NPP has no specific safety-related events classified at Level 1 on the INES scale for which corrective measures are planned in accordance with the applicable regulations but have not yet been completed. (EIA-REPORT G2/G4 P.1 2025)

3.2 Discussion

As in any industrial plant, the quality of the materials used in a nuclear power plant deteriorates during operation, particularly as a result of physical aging[1]. Exposure to ionizing radiation, thermal and mechanical stresses, and corrosive, abrasive, and erosive processes cause the components to age. The consequences of the aging processes are embrittlement, hardening, creep, wall thickness reduction, crack formation and growth, fatigue, and changes in electrical and other physical properties.

[1] Physical aging refers to the process by which the physical properties of structures, systems, or components (SSCs) change over time or through use (WENRA 2014).

The damage mechanisms associated with these phenomena are largely known as individual effects, but their actual long-term effects and, above all, their interaction under collective loads are often unknown. It is also to be expected that additional, previously unknown damage mechanisms will occur during prolonged use.

In the case of active components such as pumps and valves, whose function depends on switching operations and external energy supply, a reduction in functionality generally becomes clearly noticeable over the course of their operating life. Replacement can often be carried out as part of regular maintenance work.

The aging of passive components is difficult to detect during use. With a few exceptions (e.g., large-scale corrosion or rusting through), the aging processes of metals take place at the level of the microscopic lattice structure and are not directly visible from the outside.

The aging or deterioration of materials leads to a decrease in the functionality of SSCs as the operating life of a plant increases. To maintain plant safety, it is very important to identify the effects of aging on SSCs and to take corrective measures before integrity or functionality is lost.

Based on the information provided in the EIA documents, it can be concluded that a comprehensive aging management program was implemented to ensure continued operation. This is also indicated by the results of the first Topical Peer Review (TPR) as set out in Article 8e of Directive 2014/87/EURATOM. The first TPR focused on the Overall Ageing Management Programs (OAMPs) and four thematic areas: electrical cables, concealed pipework, reactor pressure vessels and Calandria, and concrete containment structures and Pre-stressed Concrete Pressure Vessels. The French NPPs met for the evaluated area the "TPR expected level of performance" for the Ageing Management Program. This is the level of performance that should be reached to ensure consistent and acceptable management of ageing throughout Europe.

France has completed the implementation of all actions resulting from the follow-up of the first TPR. As a result, it issued its final report in June 2021, updating its National Action Plan (NAcP) published in September 2019. The 2019 report contained four actions for the NPP fleet. The findings issued from the self-assessment and the peer review concerned the OAMPs and concealed pipework. All actions were implemented and the NAcP is therefore closed.

However, addressing the problems associated with the aging of SSCs is a major challenge for the plant, which has already been in operation for more than 40 years.

Since most SSCs were originally designed with a nominal 40-years operation time in mind, the 4th PSR can be viewed as the authorization required to operate the NPP beyond its initial design life. Therefore, the 4th PSR includes a closer look at aging management. It becomes not clear from the EIA documents whether the comprehensive extension of the scope of the ageing management is performed compared to the 3rd PSR. There are only few examples for preventive exchange of components are considered.

The ASNR's proposal during the generic phase of the 5th PSR to extend aging management beyond 4th PSR is supported. As proposed by the ASNR, the focus must be on components that are necessary for controlling potential impacts. Because age-related effects can cause safety-relevant components to fail in the event of an external impact, which may be essential for successful accident management. (UMWELTBUNDESAMT 2024b)

Updating of regulatory reference documents for the primary and main secondary circuits

In the framework of the generic phase of the 5th PSR, ASNR requires EDF to prepare regulatory reference documents justifying the maintenance of the integrity of components in the primary and main secondary circuits. These documents serve as input data for preventive maintenance programs.

EDF states that, for 900 MWe reactors, the analysis of the phenomena caused by stress corrosion cracking on auxiliary lines does not call into question the loads used in the reference documents and does not provide any additional information that would need to be included in the update of these files. In the ASNR's view, EDF's conclusion is called into question by the results of inspections carried out since the discovery of stress corrosion cracking. For example, the discovery of fatigue cracks in welds where they were not expected shows that current methods for estimating fatigue risk are not suitable for effective prevention of this risk. The challenges arising from this observation are compounded by the prospect of continued operation of 900 MWe reactors, which is likely to lead to new degradation phenomena or new sensitive areas.

The ASNR therefore requires EDF (within the framework of the 5th PSR) to define, by December 31, 2025, the strategy for taking into account the findings from the discovery of stress corrosion cracking and, more generally, the risk of unexpected degradation of components in the primary and main secondary circuits through the checks required by the additional inspection program and maintenance programs. The ASNR's requirement is in line with the high safety relevance of these cracks. The cause of the cracks, inter-crystalline stress corrosion, is a well-known corrosion phenomenon, but it was not expected in the relevant areas and therefore the pipes were not inspected for it either. This means that the aging management concept for unexpected damage to components in the primary and main secondary circuits is called into question.

Evaluation of significant effects

As part of this expert statement, an evaluation of safety-related events in Gravelines 2 and 4 between 2020 and 2025 was carried out based on reports from the ASNR.[2]

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2 https://annual-report.asn.fr/controle/l-asnr-en-region/hauts-de-france/centrale-nucleaire-de-gravelines/avis-d-incidents

This evaluation of safety-related incidents over the last five years revealed a high number of safety-related incidents that were classified as INES level 1. In addition to the events listed below regarding deficiencies in earthquake protection, a number of events that compromised safety also occurred in both reactors. Many of these events were due to violations of the general operating rules (RGE). The RGEs are a collection of regulations approved by the ASNR that define the permissible operating range of the plant and the associated regulations for reactor operation. In particular, they specify the maximum repair times in the event that the systems required for reactor safety are unavailable. The reported events were preceded by component failures and repair/maintenance failures. In some cases, no lessons were learned from previous events. The reason for the large number of safety-related events could be a lack of safety culture combined with a large number of age-related events. Also noteworthy were the incidents involving contamination of workers and the blockage of the UHS by jellyfish. (see below)

- On July 24, 2025, the operator reported a significant safety event relating to the departure from the operating range authorized by the RGE for **reactor 2** due to the **lower pressure limit of the primary circuit being exceeded** during a periodic test. Given that the reactor had exceeded its authorized operating range, this event was classified as level 1 on the INES Scale.
- On May 13, 2025, the operator reported an event involving an **error in the calculation** of **neutron parameters**, which resulted in the incorrect configuration of reactor power measurement chains for several hours in **reactor 4**. Due to the incorrect configuration of the reactor for several hours, a situation not provided for in the *technical operating specifications*, this event was classified as level 1 on the INES scale.
- On January 12, 2024, EDF reported a significant safety event relating to the late detection of the **unavailability of the boron injection line** for **reactor 4**. On January 10, 2024, while reactor 4 was in operation, the operator closed two valves on the borated water make-up circuit in order to replace a filter on this circuit. However, the operator did not open the filter bypass line. In this configuration, the boron injection line was unavailable. This event did **affect the safety function** related to reactivity control, and given that it was detected late, it was classified as **level 1** on the **INES** scale.
- On August 31, 2022, EDF reported a significant safety event concerning the **unavailability of one of the two sprinkler system lines in the containment building** (EAS) of reactor 2. On August 29, 2022, while **reactor 2** was being restarted, a level sensor in an EAS collection sump located at the bottom of the reactor building indicated a value below the level required by the reactor's RGEs. This situation led to the unavailability of one EAS system channel while the reactor was in an operating range where this system was required. Given its late detection, this event was classified as level 1 on the INES scale.

- On August 30, 2022, EDF reported a significant safety event relating to failure to meet the repair deadline for the function of the **reactor protection system** in **reactor 2**. Due to non-compliance with *general operating rules*, this event was classified as level 1 on the INES scale.
- On August 10, 2022, EDF reported a significant safety event relating to the exceeding of the annual downtime for the **heat exchangers in the intermediate cooling circui**t of **reactor 4**. This event did **affect the safety function** related to reactor cooling. As the operator did not comply with the technical specifications for reactor operation, this event was classified as level 1 on the INES scale.
- On July 15, 2022, EDF reported a significant safety event concerning the **unavailability of both cold source lines for reactor 2** at the Gravelines nuclear power plant. On July 10, 2022, during the shutdown of reactor 2, the operator carried out cleaning operations on the cold source, in particular **to remove shellfish and algae** trapped in the filters. Despite cleaning line B, the flow rate continued to decrease until it reached an alarm threshold requiring the application of incidental instructions. A drop in flow rate was also observed on line A. This event did **affect the safety function related to reactor cooling.** As the unavailability of both cold source lines was not covered by the reactor's technical operating specifications, this event was classified as level 1 on the INES scale.
- On March 10, 2022, EDF reported a significant safety event relating to the **late detection of the unavailability of a pump in the volumetric and chemical control system** of the main primary circuit of **reactor 2**. This event **affected the safety function** related to reactor cooling. Due to the unavailability of the equipment concerned, combined with its late detection, this event was classified as level 1 on the INES scale.
- On November 9, 2021, EDF reported a significant safety event relating to **reactor 4** exceeding its authorized operating range under the RGEs, due to excessively low pressure in the main primary circuit. As the event **affected the safety functions** related to reactor containment and cooling, it was classified as a level 1 event on the INES scale.
- On September 9, 2021, EDF reported a significant safety event relating to **reactor 2** exceeding its authorized operating range under the RGEs, due to **excessively low pressure in the main primary circuit**. If the pressure had continued to drop, the primary pumps responsible for circulating the primary fluid could have been damaged. As the event affected the **safety functions** related to reactor containment and cooling, it was classified as a level 1 event on the INES scale.
- On August 18, 2021, EDF reported a significant safety event relating to non-compliance with RGEs concerning the u**navailability of the turbine bypass system** to the atmosphere of the three steam generators (SG) of **reactor 4**. On August 11, 2021, reactor 4 was shut down for maintenance and refueling. During an operating maneuver to confirm the opening of the isolation valves of the turbine bypass system to the atmosphere of the three steam generators, these valves were closed by mistake, rendering the system unavailable. Due to the **unavailability** of the equipment

concerned, combined with its **late detection**, it was classified as level 1 on the INES scale.

- On August 6, 2021, EDF reported a significant safety event relating to non-compliance with RGEs concerning the **unavailability of one of the two nuclear power measurement systems** in **reactor 4**. Reactor 4 was shut down for maintenance and refueling. During an operating maneuver, a control rod became stuck in the up position. Under these conditions, general operating rules require that all neutron flux measurement systems be available. On the same day EDF decided to recheck neutron flux measurement chains it during the reactor shutdown. To do this, it was necessary to generate the unavailability of this measurement chain and the associated alarm. **Due to non-compliance with RGEs,** this event was classified as level 1 on the INES scale**.**
- On July 28, 2021, EDF reported a significant safety event relating to non-compliance with RGE concerning the **unavailability of the ventilation system for the containment building** of **reactor 4**. Reactor 4 was being shut down for maintenance and refueling. The fuel was in the reactor pressure vessel and the operations to be carried out required the opening of an airlock in the reactor building's containment. An initial analysis of the event concluded that the pre-configuration of the system in preparation for the opening of the access airlock, was not performed in accordance with the local procedure, which takes into account feedback from a similar event that occurred in 2020 on reactor 1. It shows **a failure to take lessons learned** into account. The event was therefore classified as level 1 on the INES scale.
- On July 23, 2021, EDF reported a significant safety event relating to non-compliance with a compensatory measure linked to a temporary modification of the technical operating specifications during work on one of the **two cooling circuits** of reactor 4. On July 18, 2021, **reactor 4** was shut down for maintenance and refueling. Maintenance work was carried out during shut down on the circuit of one of the two redundant seawater supply lines, rendering it unavailable. The second line remained available. One compensatory measure stipulated that no work should be carried out simultaneously on the pumps of the available redundant line. But during a periodic test of the emergency sprinkler system, the workers had to disconnect a pump that was part of the available redundant system. Due to the non-compliance with a compensatory measure associated with a temporary modification of the technical operating specifications, this event was classified as level 1 on the INES scale.
- On December 15, 2020, the operator reported a significant safety event relating to the **unavailability of the emergency injection pump** for the seals of the primary pumps common to **reactors** 1 and **2**. On October 11, 2020, following a periodic test, the start-up time of the emergency pump was found to exceed one of the criteria required by the *RGEs*. After several checks and replacement of components without solving the issue of

the emergency pump, on December 2, 2020, the backup pump was finally declared unavailable. This event is classified as level 1 on the INES scale.

- On May 28, 2020, the operator reported **a risk of long-term loss of reactor cooling in the event of an external explosion**. On February 11, 2020, EDF informed of the discovery of a deviation affecting the motors of the filter drums of the cold source of the **six reactors at the Gravelines nuclear power plant**. In the event of an explosion near the nuclear power plant, this deviation could result in the loss of long-term cooling capabilities for the fuel of the six reactors. This deviation is due to a design error. Given the industrial environment of the Gravelines NPP, in particular the presence of the natural gas terminal in Dunkirk, the nuclear power plant must be able to withstand a high-intensity explosion of external origin. Taking into account, on the one hand, the low probability of a high-intensity explosion of external origin and, on the other hand, the fact that the **deviation affects the safety function related to the cooling of the six reactors**, the ASNR classifies this event as level 1 on the INES scale.
- On April 29, 2020, the operator reported a safety-related event involving non-compliance with the RGEs for **reactor 2** with regard to the availability of a **ventilation system for the intermediate cooling circuit rooms**. Due to the delayed detection of the deviation, the ASNR has classified this event as **level 1** on the **INES** scale.
- On March 23, 2020, the operator reported a safety-related event involving **non-compliance with a compensatory measure due to carbon segregation on a steam generator in reactor 2**. It is classified as level 1 on the INES scale.
- On March 17, 2020, the operator reported a significant safety event relating to the **partial unavailability of the reactor cooling system** during shutdown. On March 12, 2020, **reactor 2** was being shut down for maintenance due to an anomaly in the safety injection circuit. Following difficulties in starting up the reactor shutdown cooling system, EDF identified that a valve supplying air to the flow control valve for this system was closed. After an initial analysis, the operator believes that this valve had remained closed since the previous reactor shutdown in 2019. This event is classified as level 1 on the INES scale due to the duration of this deviation. A similar event occurred in 2017.
- On March 17, 2020, the operator reported a safety-related event in connection with **non-compliance with the RGEs** for **reactor 2** with regard to the **availability of the cooling circuit**. On March 12, 2020, reactor 2 was shut down for repairs due to a malfunction. On the same day, EDF discovered a water leak on a filter in one of the two branches of the emergency water circuit (SEC), rendering the branch in question unavailable. On March 15, 2020, with the reactor shutdown in effect, the technical operating specifications required that the SEC circuit be repaired within 24 hours. This repair could not be completed within the 24-hour

timeframe prescribed by the technical operating specifications. This event is classified as level 1 on the INES scale.

Deficiencies of the seismic resistance

In recent years, significant deficiencies in the seismic resistance of various components of the Gravelines 2 and 4 (and other 900 MWe reactors) have been identified. It cannot be ruled out that there are others, as to date unidentified, deficiencies. Deficiencies in earthquake protection are of particular interest for the Gravelines NPP, as there are doubts about the adequacy of its design with regard to earthquakes (see Chapter 4).

- On April 1, 2025, the operator reported a significant safety event concerning a compliance deviation calling into question the **seismic resistance of cabinets containing level detectors for the pool cooling tank**. In March 2024, during inspections carried out in preparation for a future modification to a level detector, a discrepancy was found in a cabinet in **reactor 2**. Additional checks carried out in January 2025 showed that this discrepancy also affected the two cabinets in **reactor 4.** Given the potential consequences of this non-compliance, the event is classified as Level 1 on the INES scale.
- On February 10, 2025, the operator of the Gravelines nuclear power plant reported a significant safety event concerning a compliance deviation calling into question the **seismic resistance of certain sections of piping in the emergency raw water circuits** of the **Gravelines** 1, **2**, 3, and 6. This discrepancy follows the discovery in December 2024 of contact between certain sections of piping and the civil engineering structures or their supports for the two lines of this circuit of reactor 1. Additional checks showed that this also affects one channel in each of reactors 2, 3, and 6. Given the potential consequences of this non-compliance, the event is classified as **level 1** on the **INES** scale.
- On January 13, 2025, the operator of the Gravelines nuclear power plant reported a significant safety event concerning a compliance deviation calling into question the **seismic resistance of electrical cable trays** for **all reactors** at the Gravelines nuclear power plant. This deviation follows the ASNR's observation, during an inspection of reactor 5 during the 2024 maintenance shutdown, of corrosion on two cable tray supports. After analysis of the deviation by EDF's engineering services in October 2024, it was found that the cable trays' resistance to the safety-rated earthquake had not been demonstrated. Additional checks were carried out on **all reactors** in November 2024 and showed that this deviation also affected the other reactors on the site. Given the potential consequences of this non-compliance, the event, initially classified as level 0, has been reclassified by the ASNR to **level 1** on the **INES** scale.
- On May 13, 2024, EDF reported a significant safety event concerning defects in the **civil engineering anchoring of certain safety-critical equipment**. These defects concern among others **Gravelines 2**. As part of its facility inspections, EDF checks the compliance of anchors with the

civil engineering of equipment that is important for safety. These discrepancies date back to the construction of the reactors and could have compromised the integrity of the supported equipment in the event of an earthquake. Given its potential consequences for these reactors, this event is classified as level 1 on the INES scale

- On March 24, 2023, EDF reported new seismic resistance defects in the **electrical sources** of its nuclear power plants. These defects were detected during inspections carried out in 2022 and early 2023, following the ASNR's decision on February 19, 2019, requiring verification of the compliance of these systems. The inspections carried out since 2019 had already detected several discrepancies. At the end of 2019, beginning of 2020, and in the summer of 2022, EDF reported a significant safety event concerning the detection of earthquake resistance defects in certain equipment contributing to the operation of **diesel-powered emergency generators** in several of its reactors, also in **Gravelines 2**. The new faults detected concern the emergency diesel generators and relate to incorrect assembly of elastomer pipe fittings and corrosion on certain sections of piping or their supports. The event is classified as level 1 on the INES scale.
- On September 29, 2020, EDF reported a safety-related event concerning the inadequate earthquake resistance of the **heat exchangers** in the intercooling system of the 900 MWe reactor, including **Gravelines 2 and 4**. Given its potential consequences, this event is classified at level 1 on the INES scale for the 19 reactors concerned.
- On January 31, 2020, EDF reported a safety-related incident involving the risk of collision between **switch cabinets and relay housings** in 900 MWe reactors at nuclear power plants, including **Gravelines 2 and 4**. Given the potential consequences for the safety of the reactors concerned in the event of an earthquake, this event is classified as level 1 on the INES scale.

Several contaminations of workers

It should also be noted that there were five incidents of employee contamination above the permissible levels during maintenance work. This accumulation may also be an indication of deficiencies in the safety culture as well as a large number of damages in the reactor 2.

- On March 18, 2025, during welding work on a pipe located in the reactor building, a worker was contaminated on the head. (reactor 2)
- On May 23, 2024, a worker was contaminated during a waste sorting activity in the nuclear auxiliary building shared by reactors 1 and 2.
- On July 21, 2022, a worker was contaminated on the back of the neck, elbow, and knee. Traces of contamination on the ankles and traces of internal contamination were also detected. (reactor 2)
- On July 16, 2022, during a valve repair operation, a worker was contaminated on the neck and face. (reactor 2)

- On July 16, 2022, during a decontamination operation at the site, a worker was contaminated on the arm. (reactor 2)

Due to the fact that a quarter of the annual regulatory exposure limit for a worker were exceeded, these events were classified as level 1 on the INES Scale.

Automatic shutdowns of reactors 2, 3, 4, and 6 (Jelly fish clogging)

The operator informed of the automatic shutdowns of reactors 2, 3, 4, and 6 on August 10, 2025.[3] These shutdowns were caused by a massive influx of jellyfish into the filter drums of the seawater pumping stations used for the reactor cooling circuits. The station has two independent lines, protected by various devices that block natural or man-made elements present in seawater. The filter drums and their washing system are designed to block small floating or suspended particles using filter meshes measuring a few millimeters. (Each channel also includes a system of fixed and movable screens upstream of the filter drums designed to stop large objects. The inlet to the feed channel is protected by structures designed to retain hydrocarbons and larger floating objects.) (see chapter 4) This event was classified as level 0 on the INES Scale (INES).

Due to this incident, ASNR conducted an inspection on August 13, 2025. ASNR concluded: In particular, the sequence of actions taken to return the reactors to a safe state in order to prevent automatic shutdown is not in line with the kinetics of the phenomenon, and monitoring of marine biodiversity that may be causing clogging no longer appears appropriate. Jellyfish monitoring is carried out on an opportunistic basis, but over a period that does not correspond to their peak proliferation. ANSR requests to conduct an inventory of marine biodiversity that may be at risk of damage due to clogging of the UHS, define appropriate monitoring for new potential threats, and integrate predictive data into operating procedures. ASNR also stated: In addition, the clogging of filter drum No. 1 in reactor 2, regardless of the presence of jellyfish, raises questions about its effectiveness and the maintenance conditions during the last reactor shutdown in the first half of 2025. (ASNR 2025a)

3.3 Conclusions

Based on the information provided in the EIA documents, it can be concluded that a comprehensive aging management program was implemented to ensure operation. This is also indicated by the results of the first Topical Peer Review (TPR) as set out in Article 8e of Directive 2014/87/EURATOM. However, addressing the problems associated with the aging of SSCs poses a major challenge for the plant, which has been in operation for more than 40 years.

[3] Reactors 1 and 5 were shut down for maintenance

Since most SSCs were originally designed for a nominal operating lifetime of 40 years, the 4th PSR can be considered the necessary approval to operate the nuclear power plant beyond its original design life. Therefore, the 4th PSR requires a more detailed consideration of aging management. The EIA documents do not clearly indicate whether there has been a comprehensive expansion of the scope of aging management compared to the 3rd PSR. Only a few examples of preventive component replacement are presented. As far as is known, ASNR proposed expanding the scope of aging management during the generic phase of 5th PSR. This should also be performed for the 4th PSR.

In the framework of the generic phase of the 5th PSR of the 900 MWe reactors, the ASNR requires EDF to define, by December 31, 2025, the strategy for taking into account the findings from the discovery of stress corrosion cracking and, more generally, the risk of unexpected degradation of components in the primary and main secondary circuits through the checks required by the additional inspection and maintenance programs. The cause of the cracks, inter-crystalline stress corrosion, is a well-known corrosion phenomenon, but it was not expected in the relevant areas and therefore the pipes were not inspected for it either. This means that the aging management concept for components in the primary and main secondary circuits is called into question.

The ASNR's proposal during the generic phase of the 5th PSR to extend aging management beyond 4th PSR is supported. As proposed by the ASNR, the focus must be on components that are necessary for controlling accident situations. However, the scope of the program “qualification of materials under accident conditions” in the 4th PSR is very limited for Gravelines 2 and 4.

- The justification that no checks are to be carried out for Gravelines 4 as part of the Program for Complementary Investigations (PIC) should be provided.
- The evaluation of safety-related incidents over the last five years revealed a high number of safety-related incidents that were classified as INES level 1. In addition to the events regarding deficiencies in earthquake protection, a number of events that compromised safety also occurred in both reactors. The reason for the large number of safety-related events could be a lack of safety culture combined with a large number of age-related events. Also noteworthy were the incidents involving contamination of workers and the blockage of the ultimate heat sink (UHS) by jellyfish.
- In-depth investigations on components relevant for preventing external events to affect the nuclear safety of the plant should be carried out, in particular concerning those components of the original systems that connect the newly installed “hardened safety core” and systems for mitigating the effects of core-melt accidents.
- A complete analysis of the causes of the cracks in the auxiliary line due to stress corrosion cracking should be carried out and taken into account in order to take preventive protective measures against such damage and its effects already within the framework of the 4th PSR.

- The modification of the ageing management for the secondary and primary circuit components to detect unexpected degradation should be considered.
- A systematic ageing control of the components safety relevant concerning the resistance with regard to earthquakes should be considered.

4 EXTERNAL HAZARDS

4.1 Treatment in the EIA documents

EIA-REPORT G2/G4 P1 (2025, p. 32-37) provides a general overview of the external hazard types considered in the LTE process. The list accounts for the requirements stipulated by ASNR (ASN 2021) for the 4th PSR of the French 900 MWe reactor fleet. The following external hazards (natural or human-made) are of concern: earthquakes, extreme weather or climatic conditions (flooding, snow, heat wave, drought, extreme cold, high wind, tornado), influences from rivers (ice drift, icing, siltation, oil spills, silting, low water), lightning and electromagnetic interference, fire, industrial hazards (explosion, release of hazardous substances), aircraft crash, and malicious acts. The EIA documents note that studies on external hazards take into account the international standards set by WENRA. It is also stated that *“the use of the "Noyau Dur" [hardened safety core] to handle extreme events (earthquakes, floods, etc.) exceeding previously assumed values helps to meet these requirements”* (EIA-REPORT G.1 2025, p. 33).

Hazard assessment

Earthquake: The seismic design base for the NPPs of the 900 MWe fleet is deterministically derived from the maximum observed historical earthquake (SMHV[4]) increased by one degree of intensity giving the so-called maximum safety earthquake (SMA[5]) which is linked to a reference spectrum (SSD). Both determine the seismic design basis of the plant and are reassessed in PSR. Following the Fukushima Daiichi accident in 2011, a new seismic level (SND[6]) was defined (EIA-REPORT G2/G4 P.1 2025, p. 36). The SND is required to (i) envelope the ground motion of an earthquake with a recurrence interval of 20,000 years, based on probabilistic seismic hazard assessment, (ii) envelope the SMS increased by 50%, and (iii) take site effects into account.

EIA-REPORT G2 P.2 2025 (p. 216-226) states that the seismic hazard was re-assessed during the 4th PSR according to RFS 2001-01[7] and based on updated seismological findings (seismic-tectonic zoning, characterization of faults, etc.) and the historical seismicity data of the SisFrance 2012 database. The re-assessment led to new seismic ground motion spectra applicable to the 4th PSR. Two earthquakes define the 4th PSR SMS of Gravelines 2 and 4: The spectral frequencies envelope of the 4th PSR SMS is covered either by the 3rd PSR SMS for frequencies above 1.5 Hz, or by the design spectrum for frequencies below 1.5

4 SMHV: Séisme Majoré Historiquement Vraisemblable – Maximal plausible historical earthquake

5 SMS: Séisme Majoré de Sécurité – Maximum safety earthquake, equivalent to design basis earthquake

6 SND: Séisme Noyau Dur – Seismic level for the hardened safety core

7 Règle fondamentale de sûreté - RFS 2001-01 of 31st May 2001 concerning the determination of the seismic risk for the safety of surface basic nuclear installations

Hz. It is concluded from the EIA documents that the new low-frequency accelerations exceed the ones of the 3rd PSR[8]. This led EDF to analyze structures exhibiting low-frequency behavior. EDF concluded that the water intake structures, the machine room and equipment in the pumping station are robust against the new spectral accelerations.

The Gravelines site is geologically located near the junction of the London-Brabant Massif and the Paris basin immediately north of the Variscan deformation front with the Midi fault, the Nord-Artois-Shear Zone (ZCNA) and the Brabant Fault (ZFB; EIA-REPORT G2 P.2 2025 p. 222). The EIA documents note that the ZCNA consists of several faults including the Marqueffles Fault, for which neotectonics activity is indicated. Active faults are also indicated in the Dover Strait. The ZFB could be associated with the Renaix Oudenaarde (1938) earthquake[9]. EIA-REPORT G2 P.2 2025 (p. 223) states that EDF conducted some new investigations to analyse the named faults including literature review and analyses of reflection seismic. EDF concluded that currently no proven evidence existed for the existence of active faults in region around the Gravelines site. It is, however, announced that *“based on the results obtained and any additional analysis from morphotectonic studies, the acquisition of new data (high-resolution subsurface geophysics, and possibly paleoseismological trenching and dating) could be undertaken in the coming years”.*

The soil profile at the Gravelines site comprises of about 24 m thick sand deposits and underlying Hercynian (Silurian) rock. The s-wave velocity of the top soil (Vs30) is 290 m/s. According to RFS 2001-01 sites with s-wave velocities <300 m/s require considering site effects in the calculation of ground motion spectra. The EIA documents note that *“numerical modeling led to the conclusion that, in Gravelines, according to RFS 2001-01, a special site effect exists in a very limited frequency range”.* No further information is given.

As part of LTO process, EDF supplemented seismic hazard analyses by a Probabilistic Safety Assessment (PSA) Level 1. with the following main steps (EIA-REPORT P.2 2025 p. 164-166):

- A probabilistic seismic hazard study determining the occurrence frequencies of seismic events as a function of their maximum ground acceleration (PGA)
- System analysis and functional analysis to identify model failures that could initiate accident sequences initiated by earthquake, and identify SSCs involved in mitigating these sequences
- Establishing fragility curves to determine the conditional probability of failure of SSCs as a function of seismic ground motion
- Risk quantification by combining seismic hazard, system analysis, functional analysis and the seismic fragility of SSCs, to estimate the Core Damage Frequency (CDF) and the probability to uncover spent fuel in the Spent Fuel Pool.

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[8] Technical descriptions in the EIA documents are not unambiguous.

[9] EIA documents refer to the 1938-06-11 Renaix Oudenaarde M=5.0 earthquake.

The Level 1 PSA estimated that the average contribution of seismic ground motion hazards to the CDF is $8*10^{-7}$ per reactor-year for *"a monitoring window corresponding to a return period of 150,000 years".* 70% of the CDF is contributed by seismic accelerations exceeding the Gravelines Hardened Safety Core (SND) earthquake (0.41 g). The values stated for Unit 2 and Unit 4 are identical (EIA-REPORT G2 P.2 2025 p. 304; EIA-REPORT G2 P.2 2025 p. 301).

The ground motion corresponding to the occurrence probability of 10^{-4} per year is not quantified in the EIA documents.

High temperatures: The maximum long-time air temperature at which all safety-relevant materials are subject to acceptable environmental conditions, projected over the next 30 years (TLD; Température Longue Durée) was set at 30 °C, the exceptional air temperature (TE; Température Exceptionnelle) defining functional limits is 43.1 °C (EIA-REPORT G.2/G4 2025 p. 35). Values apply to both reactors, Gravelines 2 and 4. The re-assessment of high temperatures was initiated after the heatwaves of 2003 and 2006 to account for the temperature changes up to 2023. Methods and assumptions used to derive the TLD and TE values are not specified.

Analyses also accounted for high water temperature by reviewing all requirements related to the ultimate heat sink and identifying and reviewing the cases that adversely affect the cooling function.

Extremely low temperatures: Protection requirements for extremely low temperatures were developed based on lessons learned from the coldest winters (notably 1984-1985 and 1986-1987) and implemented during the second PSR. Protection is said to be ensured for all Emergency Intervention Systems (EIPS) under cold conditions corresponding to the design cold level of the reactor platform, and beyond the design cold level for the EIPS. Assessments include IPCC forecasts indicating a reduction of the number of cold days per year. Methods and assumptions used to derive temperature values are not specified in detail.

External flooding: As part of the 4th PSR, EDF was reviewing the robustness of the NPP with regard to hazards described in ASNR Guidance No. 13 on the protection against external flooding. Analyses for Gravelines, located on at the North Sea (Strait of Dover), included the (re-) assessments of extreme precipitation (rain and peak rainfall intensity), high ground water and flooding by the North Sea. The latter led to the introduction of a new maximum flood level for flooding by the North Sea. The new flood level accounts for inundation by waves and sea level rise in connection with global warming (EIA-REPORT G2 P.2 2025, p. 208-217). Detailed information on methods, data and assumptions used to derive the maximum flood level are not communicated.

EDF also analyzed the volumetric flood protection devices and its resistance against seismic impact up to the SMS.

High wind and tornado: The EIA documents state that the reassessment of hazards by storm do not require any update (EIA-REPORT G2/G4 P.2 2025). Details of the hazard assessment are not provided. The design basis tornado corresponds to intensity EF0 on the Enhanced Fujita tornado scale with velocities of

29 m/s. Probabilistic assessments revealed occurrence probabilities for this tornado intensity of $3,1*10^{-5}$ per year for oceanic domains and $1,1*10^{-5}$ per year for the inland French territory (EIA-REPORT G2 P.2 2025, p. 259). The occurrence probability of the design basis tornado consequently is $<10^{-4}$ per year and in line with international requirements. Assessments consider the dynamic wind pressure; the sudden pressure drop in the center of the vortex and wind-blown projectiles. The EIA documents conclude that protection against high wind and wind-blown projectiles is sufficient to also protect the NPP against effects of the reference tornado (EF0 on the Enhanced Fujita tornado scale).

Availability of the ultimate heat sink: Within the framework of the 4th PSR EDF targeted to verify the robustness of the installations with respect to hazards threatening the ultimate heat sink and review the implementation of the safety requirements for pumping stations (EIA-REPORT G2 P.2 2025, p. 240). The activities were initiated after the clogging of water intakes of the NPPs Choos, Cruas and Blayais by frazil ice and flotsam in 2009. Analyses for the Gravelines site include the re-assessment of the minimum safety water level, phenomena threatening the cooling water intake by clogging, sedimentation in the feeder channels (silting) and pollution of the cooling water with hydrocarbons. According to EIA REPORT G2/G4 P.2 (2025), following implementation of modification PNPP1874 “Replacement of pre-filter grids with robust grids in case of external ‘massive clogging’,” the site is robust against massive accumulation of deposits.[10]

Lightning: The determination of potential lightning strike points and the probability that a target will be struck by lightning follows the standard NF-EN-62305-1. Assessments analyze the vulnerability of connections between buildings by performing calculations to determine overvoltage and create a list of protective devices to be installed on the connections requiring protection (EIA-REPORT G2 P.2 2025, p. 263).

Human-made hazards (industrial facilities, pipelines and transport of dangerous materials): Hazard assessment is based on ASNR Regulation RFS I-2.d. (ASN 1982). Analyses include external explosion and hazards resulting from transportation of hazardous materials outside of the site and on the site. ASNR (ASN 1982) requires a maximum probability of 10^{-6} per year for unacceptable radioactive releases caused by human-made hazards. (EIA-REPORT G2 P.2 2025, p. 271) states that with respect to external explosion all hazards are excluded by deterministic analyses except for the explosion of a methane cloud due to damage to the LNG terminal Dunkerque, located less than 5 km from the NPPs, or a methane tanker at sea. Probabilistic analysis revealed a neglectable probability of 10^{-7} per year for external explosion (EIA-REPORT G2 P.2 2025, p. 272).

Accidental aircraft crash: Analyses of the hazard of accidental airplane crash is based on Règle Fondamentale de Sûreté (RFS) I-2.a. The probabilistic assessment of air traffic hazards used updates of the following data: accident analysis

[10] It should be noted that Gravelines NPP experienced significant disruption in August 2025 when massive, unpredicted swarms of jellyfish clogged its water intake filters, forcing automatic shutdowns of four reactors for safety to prevent overheating. (see chapter 3)

parameter values, environmental data specific to each site (airport/airfield locations, air traffic data) and virtual surface area values (surface areas of structures exposed to aircraft impact risk). Results show that the probability of unacceptable release of radioactive substances at the Gravelines site limit due to air traffic is less than 10^{-6}/reactor year for the reactor and spent fuel storage. The EIA documents do not specify the airplane type for which the value was calculated.

Upgrades of protection measures:

Decisions regarding upgrading measures were made on the basis of PSAs considering fire, earthquake, internal flooding, high water, sea level rise, and internal explosion. PSAs show that fires in the electrical building and earthquakes contribute significantly to the CDF of both units (EIA-REPORT G2/G4 P.1 2025, p. 38).

As a general measure to strengthen the protection of the Gravelines 2 and 4 reactor units, EDF plans to achieve safety improvements by installing a Hardened Safety Core (Noyau Dur) to increase the robustness of the NPPs against hazards such as earthquakes, tornadoes and floods. In addition to this general measure, the EIA documents[11] list a number of specific improvements including the following measures to protect the NPP from external hazards:

Safety upgrades that have already been completed and those that are planned are comprehensively listed in the Annexes of EIA-REPORT G2 P.2 2025 and EIA-REPORT G4 P.2 2025. Table 1 of the current report lists the measures relevant for external hazards.

Regulatory requirements for the 4th PSR are summarized under [AGR-F] by ASNR (ASN 2021). This report cannot determine whether the relevant requirements have been fully implemented.

Table 1: Upgrading measures for SSCs important to safety with respect to external hazards, reactors Gravelines 2 and 4 (adapted from EIA-REPORT G2/G4 P.2 2025)

PNPE1039	External Flooding	Peripheral protection against external flooding	Completed
PNPE1069	High Temperature	Installation of a warm air generator in the DEG cooling unit room	Completed
PNPE1070	High Temperature	Improvement of the air conditioning in the DVL-MT/BT rooms	Completed
PNPE1118	Earthquake	Seismic reinforcement of the local ventilation system battery rooms	Completed
PNPE1118	Earthquake	Earthquake protection of the battery room (DVE) ventilation	Completed
PNPE1138	External Flooding	Protection of the safety block (BDS) against external flooding	Completed
PNPE1165	High wind	Protection against wind-blown projectiles	Completed

[11] EIA-REPORT G2 P.1 2025, p. 32-38; EIA-REPORT G4 P.1 2025, p. 32-38; REPORT G2 P.2 2025; EIA-REPORT G4 P.2 2025.

PNPE1174	High Temperature	Installation of ventilation in the CRF/CFI rooms, replacement of the ventilation in the SEC rooms and in the pumping station hall	Completed
PNPE1191	Earthquake	Earthquake protection of cable shafts	Completed
PNPE1335	Earthquake	Robustness of the automatic shutdown of the circulating water against earthquake-induced flooding	Completed
PNPP1123	Heat sink	Implementation of a water level gauge downstream of the filter unit, triggering production pumps at low water level	Completed
PNPP1675	External Flooding	Protection against flooding caused by direct leakage onto the platform	Completed
PNPP1688 Tome C	HSC	Implementation of a Hardened Safety Core control system for new equipment	Completed
PNPP1723	Heat sink	Implementation of a winter recirculation system for non-robust sites in situations with ice formation	Completed
PNPP1874	Heat sink	Replacement of pre-filter grids to be resistant to "massive clogging"	Completed
PNPP1898	Earthquake	Reinforcement of the pole bridge in earthquake hard core	Completed
PNPP1951	Electromagnetic interference	Installation of surge protection devices	Completed
PNRL1823	High temperature	Replacement of the 6.6 kV AC backup power supply (generator sets) diesel air cooler motors	Completed
PNRL1835	High temperature	Update of the parameters for automatic fouling monitoring of heat exchangers	Completed
PNRL1922	External flooding	Treatment of volumetric protection bypasses	Completed
PNRL1927	External flooding	Elimination of volumetric protection bypass risks	Completed
PNRL1955	Low temperatures	Increase of thresholds of the ventilation and air conditioning system (general ventilation of the nuclear auxiliary building)	Completed
PNRL1990	Earthquake	Increased seismic resistance of the JP* fire protection network	Completed
ILGB1188	Heat sink	Resistance of filtration of circulating water measuring devices against wind-blown projectiles	Completed
PNPE1115	Earthquake	Automatic shutdown commands for reactors during earthquakes and information on significant earthquakes, earthquake-resistant reactor core	Open(1)
PNPE1119	Tornado	Passive protection of the reactor building against tornadoes	Open(1)
PNPE1238	Earthquake	Increased seismic resistance of fuel tanks for earthquakes exceeding the SMS	Open(1)
PNPE1285	Earthquake	Earthquake resistant cable ducts (Hardened Safety Core)	Open(1)
PNPE1305	Earthquake	Implementation of a robust detection system for total loss of heat sink (SND)	Open(1)
PNPE1323	Earthquake, tornado	Reinforcement of the chimney of the BAN (SMS, storm and tornado EF2)	Open(2)
PNPE1332	Earthquake	Earthquake resistant piping (SND) (Hardened Safety Core)	Open(1)
PNPE1333 tome B	Earthquake	Seismic protection of the core area of the main primary circuit, the main secondary circuit, and the SND support	Open(1)
PNPE1358	Earthquake	SND and tornado robustness of Noyau Dur	Open(1)

PNPP1722	Low temperatures	Trace heating and thermal insulation of the ASG supply by SER	Open(1)
PNPE1675	External flooding	Volumetric protection – bypass bubble rooms	Open(1)
PNPE1477	Lightning	Adding a lightning rod to the secondary side of the auxiliary transformers	Open(1)

(1) Modifications that will be deployed on Units 2 and 4 of the Gravelines Nuclear Power Plant as part of Phase B of the modifications to the 4th PSR.
(2) Modifications that will be deployed on Unit 2 and 4 of the Gravelines NPP as part of a specific program

Earthquake: With respect to the protection of Gravelines 2 and 4 against seismic ground motion the EIA documents mention the reinforcement of cable ducts and reinforcement of the BAN[12] chimney to prevent it from damaging safety important SSCs in the event of a collapse. (EIA-REPORT G2 P.2 2025, p. 220-221; EIA-REPORT G4 P.2 2025, p. 219-220). Both measures are said to be not related to a re-assessment of seismic hazards. The main task to increase the robustness of both units is the implementation of the Hardened Safety Core to withstand the „Noyau Dur" earthquake (SND). This remains to be completed in Phase B of the PSR (EIA-REPORT G2 P.1 2025; EIA-REPORT G4 P.1 2025).

External flooding: Flood protection at the plant perimeter is re-enforced by a sheet-plank wall on the seafront and an embankment dike that connects to an existing firewall along the land-side of the reactor blocks, installations of watertight gates, the laying of cover plates on the channels of the filtration system of circulating water and the emergency raw water system, and the removal of bypasses of the volumetric protection. (EIA-REPORT G2 P.2 2025, EIA-REPORT G4 P.2 2025, p. 207-216). Implementation of the measures is completed for both units.

High temperatures: EIA-REPORT G2/G4 P.1 (2025, p. 35) and EIA-REPORT G4 P.2 (2025, p. 235-243) lists the following measures implemented between 2013 and 2017: modification of the monitoring of intermediate cooling system and emergency raw water system heat exchangers to improve cooling by the ultimate heat sink (North Sea), replacement or protection of temperature-sensitive equipment with heat shields (diesel valves, current transformers, cables, sensors, fire alarm control panels, etc.), installation or replacement of cooling units, improvements of air conditioning of buildings containing SSCs important to safety by increasing ventilation performance and/or cooling capacity and installation of air conditioning systems. Implementation of the listed measures for mitigating effects of extreme temperatures at Gravelines 2 and 4 are implemented and meeting the deadlines defined by ANSR requirement [AGR-A] of (ASN 2021).

With respect to high cooling water temperature, criteria were set for the maximum permissible fouling of heat exchangers. It is shown that compliance with the maximum permissible fouling enables the normal operation of the units at high water temperatures of the heat sink.

[12] Buildings for Nuclear Auxiliary Facilities

Low temperatures: Results of the 4th PSR gave rise to a number of safety upgrades including the update of the list of Emergency Intervention Systems that need to be protected against low temperature, performance of regular tests to ensure ventilation capacity and the use of new software to model temperature effects (EIA-REPORT G2 P.1 2025, p. 245; EIA-REPORT G4 P.1 2025, p. 243). EDF plans to install thermal insulations for a number of SSCs.

High wind and tornado: The installation of protective devices on the filter systems of the cooling source for wind-blown projectiles against strong winds and tornadoes has been completed (EIA-REPORT G2/G4 P.1 2025, p. 36). Reinforcement of the BAN chimney is open (PNPE1323, see Table 1).

Availability of the ultimate heat sink: Measures to protect the availability of the ultimate heat sink include the installation of filtration devices (pre-filter screens, screens, chain filters) in the water intakes and managing the risk of sedimentation/siltation by implementing regular bathymetric monitoring and carrying out dredging operations. Threats to the cooling water by oil spill are mitigated by an agreement with authorities to receive early warning and administrative measures up to a precautionary shutdown of the reactors.

Lightning: The safety requirements applicable to the 4th PSR of the 900 MWe reactor fleet include new requirements for lightning protection. Accordingly, new lightning arresters are installed close to auxiliary transformers. The measures are not completed for either unit (EIA-REPORT G2/G4 P.1 2025, p. 37).

Human-made hazards (industrial facilities, pipelines and transport of dangerous materials): The EIA documents state that resistance to detonation-type explosions of buildings and civil structures housing or containing SSCs important to safety is provided by design. Analyses revealed no necessities for retrofitting. (EIA-REPORT G2 P.2 2025 p. 272) concludes that sufficient protection is in place at both NPP units.

Accidental aircraft crash: The risk assessment carried out within the 4th PSR justified the adequacy of the protective measures in place. No safety upgrades are made for Units 2 and 4 of the Gravelines NPP.

Malicious acts

The EIA REPORT P.1 (2025) mentions that the events considered, which are specified in the regulations, also include external impacts due to malicious acts. No further information is provided.

The EIA-REPORT P.4 (2025) provides some general information: The security of nuclear power plants is subject to coordination between EDF and the state (including the Ministry of the Interior and the Ministry of Defence). In particular, the authorities ensure continuous monitoring of nuclear power plants and their airspace.

Nuclear power plants are divided into different areas in terms of their design and organization and are protected by a multi-level security system. The protective measures for nuclear power plants are diverse and must remain confidential in order to ensure their effectiveness. The security measures, which are subject to various nuclear safety regulations, are not part of the fourth periodic review. EDF is implementing a €750 million investment program for all nuclear power plants to further strengthen security measures against intruders and meet the requirements for robustness in the event of an attack.

4.2 Discussion

Generic aspects

The contents and procedures of a PSR are only loosely defined in the French legal framework, leaving it to the nuclear regulator to specify conditions and contents of the review. The objectives of the 5th PSR of the 900 MWe fleet were defined by ASNR in a process that involved a proposal by EDF, a review and conclusive guidelines issued by ASNR. With respect to external hazards, ASNR stipulates that definitions of design basis events and design extension considerations must follow the requirements set by WENRA. The main implications of this requirement are:

- The mandatory contents of PSR including plant design, deterministic safety analyses, probabilistic safety analyses and hazard analyses are described in detail in Issue P, Reference Level P2.2 of WENRA (2021).
- Issue E, Reference Level E11.1 requires regular reviews of the actual design basis to determine whether the design basis is still appropriate.
- Issue F, Reference Level F5.1 requires the same regular review for Design Extension Conditions (DEC)
- Issue TU summarizes requirements for external hazard assessment, most importanty the definition of design basis events with exceedance frequencies not higher than 10^{-4} per annum, and the requirement to provide protection against design basis events. Protection shall be of sufficient reliability that the fundamental safety functions are conservatively ensured.
- Issue TU, Reference Levels TU6.1 to TU6.3 list requirements for considering DEC.
- In addition to the requirements stipulated in the WENRA Safety Reference Levels, WENRA provides ample guidance on how to consider external hazards in safety demonstration (WENRA 2020a-d).

In sum, WENRA requires that external hazards be addressed as part of the PSR. The design basis of existing plants is not considered fixed by the initial plant design but rather as a “floating” value that can change over the life of a reactor. The same applies to DEC.

The EIA documents provide no clear evidence if these WENRA requirements were followed by EDF. For most external hazards, the methods, data and assumptions used in the hazard assessment are not specified. Conformity with WENRA requirements and guidance can therefore not be assessed. It remains particularly unclear if design basis events with exceedance frequencies not higher than 10^{-4} per annum have been determined for all external hazards that apply to the site, if the assessment of design basis events is in line with WENRA regulations and guidance, and how DEC are addressed for the identified hazards.

Non-conformity with WENRA Reference Levels is observed for earthquake and seismic ground shaking. The Design Basis Earthquakes (DBE) for Gravelines and the other reactors of the French 900 MWe fleet are still based on deterministic analyses. Demonstration that the deterministically determined DBE is equivalent to a PSHA-derived design basis earthquake with an average recurrence interval of 10,000 years is missing (see discussion below). It therefore remains to be demonstrated that the seismic resistance of all SSCs important to safety is sufficient to conservatively ensure the fundamental safety functions for a DBE with an average recurrence interval of 10,000 years as required by WENRA (2021). The authors of this report assume that adequate protection against a probabilistically derived DBE, should it be higher than the deterministic value for which the plant was designed, is intended to be ensured by the Hardened Safety Core (Noyau Dur). This, however, would contradict the Defence-in-Depth (DiD) concept and the separation of DiD levels because the DEC equipment of the Noyau Dur could become necessary to protect the plant against design basis hazards, i.e., the probabilistically derived DBE. The Hardened Safety Core is classified as a 4th DiD level system which is required as an additional and independent level compared to the 3rd DiD level. The Hardened Safety Core can therefore not be used to compensate for existing deficits in terms of the protection against design basis events.

Site-specific aspects

Seismic hazard and definition of the design basis earthquake: Design basis ground motion values for the French 900 MWe reactors were established by a deterministic approach. The fact that the deterministic approach was originally stipulated in RFS 1.2.C (1981)[13] suggests that design basis values were only established after the start of construction of the Gravelines units 2 and 4 (Table 2: Design basis ground motions (peak ground acceleration) of the Gravelines reactors according to
(ASN 2011a)). At the background of the standardized reactor series operated in France, EDF introduced the notion to define the DBE as the envelope spectrum of the various SMS spectra associated with the different sites of the same plant series (ASN 2011a). This approach allowed pooling the design studies for the re-

[13] Règle fondamentale de sûreté - RFS 1.2.c of 1st October 1981 concerning the determination of the seismic motion to be taken into account for the safety of the facilities.

actors on the respective nuclear islands. All plants of a specific series consequently share the same seismic design. Other structures, referred to as "site structures", were specifically designed for each site (Table 2: Design basis ground motions (peak ground acceleration) of the Gravelines reactors according to
(ASN 2011a)).

Table 2: Design basis ground motions (peak ground acceleration) of the Gravelines reactors according to (ASN 2011a)

NPP	Start of construction	Start of commercial operation	DBE Nuclear island	DBE Site structure	SND
Gravelines 1	1975	1980	EDF normalized to 0.2 g zero period	EDF normalized to 0.2 g zero period	PGA= 0,41 g
Gravelines 2	1975	1980			
Gravelines 4	1975	1980			
Gravelines 4	1976	1981			
Gravelines 5	1979	1985			
Gravelines 6	1979	1985			

In 2001 the RFS 1.2.C (1981) was replaced by RFS 2001-01[14]. The replacement retained the general deterministic approach. The main changes concerned new definitions of seismotectonic zones, intensity-magnitude correlations, the replacement of a fixed response spectrum by a site spectrum, the consideration of site effects, and the account for paleo-earthquakes in addition to historical/instrumental earthquakes of the SISFRANCE earthquake catalogue. In addition, it was required that the DBE is higher than a minimum level that encompasses a M=4 earthquake at a distance of 10 km from the site, and a M=6.6 event at 40 km distance (ASN 2011a).

Defining the Design Basis Earthquake exclusively deterministically is not state of the art and does not conform with the WENRA Reference Levels (WENRA 2014; 2021). The Stress Tests ENSREG (2012b) therefore recommended introducing probabilistic methods (PSHA) to determine design basis earthquakes. The French National Action Plan (NAcP) consequently announced that probabilistic methods are to be used to determine the site-specific seismic hazard.

For Gravelines it is evident that a Probabilistic Seismic Hazard Assessment (PSHA) has been completed to define the ground motion parameters of the SND. The ground shaking level of the SND is relevant to the design of the Hardened Safety Core (Noyau Dur). The PSHA revealed a ground acceleration of 0,41 g for the SND[15] (EIA-REPORT G2 P.2 2025 p. 301; EIA-REPORT G4 P.2 2025 p. 301). By definition of the SND, this value corresponds to an average earthquake

[14] Règle fondamentale de sûreté - RFS 2001-01 of 31st May 2001 concerning the determination of the seismic risk for the safety of surface basic nuclear installation.

[15] The EIA documents leave open whether the value refers to Peak Ground Acceleration or Peak Horizontal Ground Acceleration.

return period of 20.000 years. No PSHA results other the single value characterizing the SND are communicated. Documents, in particular, do not show hazard curves and do not state a ground motion value characterizing the 10,000 years earthquake (occurrence probability of 10^{-4} per year) which, according to WENRA (2021), shall be used to define the seismic design basis of existing NPPs[16]. It is therefore unclear if the deterministically derived seismic design basis value for the Gravelines units, the SMS with 0,2 g[17], envelopes the ground motion value of a PSHA-derived design basis earthquake with an average recurrence interval of 10,000 years. The high value for the SND (0,41 g) suggests that this may be not the case.

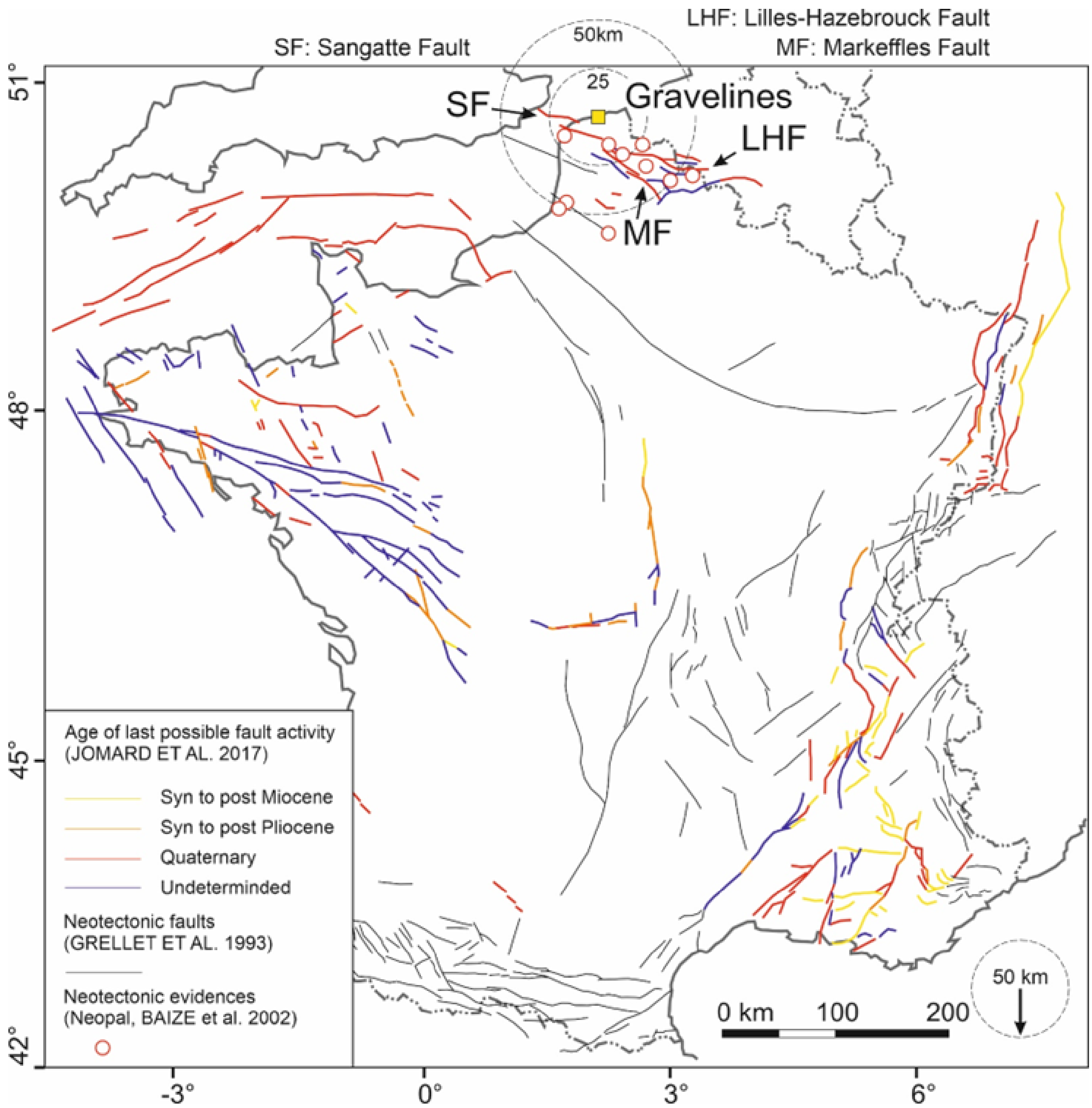
The EIA documents do not provide information on the methods, data and assumptions of the PSHA other than claiming that *"seismic studies [are] compliant with international best practices (Type 1 study)"*. The notion of type 1 study remains unexplained. With respect to methods and data it is worth noting that state-of-the art PSHA is based on both, earthquake and active fault data. Both, EIA-REPORT G2 P.2 (2025) and EIA-REPORT G4 P.2 (2025) provide evidence that EDF is aware that geological and seismological data indicate the existence of several active faults in the near-region and region around the Gravelines site, but it seems that these faults are not taken into account as fault sources in the PSHA performed to establish the SND and/or the seismic PSA. The active fault map by JOMARD et al. (2017) shows several Quaternary faults within a distance of 25 km from the site (near-region of the site to IAEA 2022), and numerous locations in the same area for which data exists in the Neopal neotectonic database (BAIZE et al. 2002; NEOPAL 2009). Evidence for Quaternary faulting particularly exists for offshore faults in the Dover Strait (Sangatte fault), the Marqueffles fault and the Lille-Hazebrouck fault (RITZ et al. 2021; GARCIA-MORENO et al. 2015). All of the named faults belong to the fault systems between the London-Brabant Massif and the Paris Basin which are known to EDF and referred to in the EIA documents (Nord-Artois-shear zone and the Brabant fault). For such proved or potentially active faults WENRA (2020b p.11ff) suggests systematic fault mapping and collecting paleoseismologic information. Efforts should at least be made in the near-region of the site (not less than 25 km) to collect geological, geophysical, geomorphologic, geodetic and paleoseismological data for identifying and characterizing active faults. Noteworthy, EDF considers that *"the acquisition of new data (high-resolution subsurface geophysics, and possibly paleoseismological trenching and dating) could be undertaken in the coming years"*. At the background of the existing literature (RITZ et al. 2021 and references therein) it is remarkable that the procedure has not yet been definitively established.

[16] WENRA 2021, Issue TU, Reference Level TU4.2

[17] The PGA value is taken from IRSN (2012). EIA documents do not provide the actual PGA value for the SMS.

Figure 2: French active fault database and location of the Gravelines reactors (redrawn from: JOMARD et al., 2017; RITZ et al. 2021).

French active fault database and location of the Gravelines reactors



Source: Environment Agency Austria

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Note: Circles around Gravelines indicate the site near-region and site region according to IAEA (2022) (radius 25 and 50 km from the site, respectively). Locations of faults near Gravelines from RITZ et al. (2021).

The Gravelines site is located on top of about 24 m thick sand deposits with s-wave velocities <300 m/s. According to RFS 2001-01 such sites require considering site effects in the calculation of ground motion spectra. The EIA documents note that *"numerical modeling led to the conclusion that, in Gravelines, according to RFS 2001-01, a special site effect exists in a very limited frequency range"*. The unclear formulation may be interpreted to indicate that the design basis spectral accelerations are exceeded for a range of frequencies.

The seismic safety of the Gravelines reactors was assessed by Level 1 seismic PSA which estimated that the average contribution of seismic ground motion hazards to the CDF is $8*10^{-7}$ per reactor-year. EIA-REPORT G2 P.2 2025 (p. 304) and EIA-REPORT G2 P.2 2025 (p. 301) adds the restriction that the value is valid for *"a monitoring window corresponding to a return period of 150,000 years".* To the authors of the current report, this restriction suggests that strong earthquakes with recurrence periods longer than 150, 000 years were not considered in the PSA. If this is the case, it cannot be excluded that the contribution of earthquakes with recurrence intervals >150,000 years to the total risk is significant or even higher than the contribution of the considered earthquakes. The observation suggests that because of a truncation at 150,000 years, the assessed seismic risk may be incomplete, i.e., the CDF value may be underestimated.

Upgrades of protective measures: Safety upgrades that have already been completed and those that are planned are comprehensively listed in the Annexes of EIA-REPORT G2 P.2 2025 and EIA-REPORT G4 P.2 2025. The Annexes do not contain a specific timetable for the implementation of the planned measures. Mandatory time schedules for individual upgrades, however, have been fixed by ASNR (SN 2021).

One of the most important measures to provide protection against external hazards is the implementation of the Hardened Safety Core (Noyau Dur). However, the implementation of the Noyau Dur is still pending as for example shown by measure no. PNPE1358 referring the earthquake and tornado robustness of the Noyau Dur (note that Table 1 contains several additional open actions that relate to the Noyau Dur). Implementation is announced for Phase B of the 4th PSR no later than March 2029 for Block 2 and December 2029 for Block 4 (EIA-REPORT G2/G4 P.3 2025, p. 6). The fact that the implementation of the Noyau Dur is still pending appears remarkable at the background that the regulatory decision for its implementation dates back to 2012 and the European Stress Tests (ASN 2012).

It is concluded that the implementation of the Hardened Safety Core (Noyau Dur) as required by [ND-A], [ND-B] and [ND-C] of ASNR (ASN 2021, p. 14) is pending. ASNR (ASN 2021) requires the following implementation timeline for the Hardened Safety Core: Gravelines 2: 21.03.2029; Gravelines 4: 19.12.2029.

External flooding: EIA-REPORT G2 P.1 and EIA-REPORT G4 P.1 (2025, p. 34) state that a new site-specific maximum flood level for flooding by the North Sea was established also mentioning that these studies take into account level rise by global warming. The EIA documents, however, lack information on whether the analyses include effects such as waves, storm surge, tsunami etc. as suggested by IAEA (2011) and WENRA (2020c) for marine flood hazards. The EIA documents further leave it open if the newly established maximum flood level meets the WENRA requirement for the design basis flood with an exceedance frequency not higher than 10^{-4} per year.

Terrorist attacks and acts of sabotage

Terrorist attacks and acts of sabotage can have a significant impact on nuclear facilities and cause serious accidents. Nevertheless, they are only mentioned in very general terms in the EIA documents submitted. Similar EIA reports have covered such events to a certain extent. Even if precautions against sabotage and terrorist attacks cannot be discussed in detail for reasons of confidentiality, the necessary legal requirements should be set out in the EIA documents.

The nuclear power plants currently in operation have a certain degree of protection against possible terrorist attacks due to their design, e.g., through relatively thick outer walls and diverse and redundant safety systems. Accidental aircraft crashes have been taken into account in the design of nuclear power plants for several decades. However, only accidents involving smaller sports aircraft and/or military aircraft were considered. It was only after the attacks of September 11, 2001, that the consequences of a deliberate crash of a commercial aircraft were considered. Older nuclear power plants, such as the Gravelines NPP, are therefore not adequately protected against such massive attacks. A targeted aircraft crash could cause a serious accident with significant consequences for the population.

According to WENRA (2013), it is expected that a deliberate crash of a commercial aircraft will not lead to a core meltdown accident in new nuclear power plants and therefore, in accordance with WENRA safety objective (O2), should only have minor radiological consequences. To prove this, the effects of direct and secondary impacts of the aircraft accident must be considered (vibrations/shocks, burning and/or explosion of the aircraft fuel). In addition, buildings or parts of buildings containing nuclear fuel and safety-relevant safety equipment should be designed in such a way that no kerosene can penetrate.

The increasing risk due to aging effects must also be taken into account for Gravelines 2 and 4: A study uses numerical simulations to investigate the influence of aging on the effects of a military aircraft impact on a nuclear power plant. The results show that the aging of a plant increases its susceptibility to large-scale or localized penetrations. The greater the degradation of the materials, the lower the residual resistance and the greater the risk of wall perforation. With the same impact force, the strength of the aged containment is reduced by approximately 30%. (FRANO 2021)

In addition to an attack with a commercial aircraft, a number of other attack scenarios are conceivable for a terrorist attack from the air. The drone flights over France in the fall of 2014 highlighted weaknesses in the air surveillance of French nuclear power plants and, above all, in the defense against such potential airborne attacks. In the fall of 2014, a total of 31 drone flights over 19 French nuclear facilities were recorded. (GP 2014)

Nuclear Threat Initiative (NTI)

In its Nuclear Security Index 2023, the US-based Nuclear Threat Initiative (NTI) assessed the measures taken by various countries to protect their nuclear facilities from terrorist attacks and sabotage. The index does not evaluate the specific measures taken by each facility, but rather the measures taken by the government and the legal requirements. In the NTI Index, 100 is the highest possible score and thus indicates compliance with current security requirements.

In the Nuclear Security Index 2023, France ranks only 20th out of 47 countries with a total score of 77 points. Low scores are shown for "security culture" (25), "cybersecurity" (63), and "protection against insider threats" (36). These low scores indicate weaknesses in protection against acts of sabotage and terrorist acts. (NTI 2025)

International Physical Protection Advisory Service (IPPAS)

The IAEA plays a key role in assisting States in protecting their civil nuclear materials and facilities. It supports States by conducting and organizing advisory security assessments and peer review missions through its International Physical Protection Advisory Service (IPPAS). An IPPAS mission is an assessment of existing practices in a State with the aim of strengthening a State's nuclear security organization, procedures, and practices. (IAEA 2021a)

The last IPPAS mission was completed in France with the follow-up mission in 2018. Due to the changed security situation in Europe and the low NTI Index score, another IPPAS mission should be considered to improve the security measures. (IAEA 2025a)

4.3 Conclusions

The EIA documents provide ample information on hazard types considered in the safety demonstration for the units 2 and 4 of Gravelines NPP and measures already implemented or decided to be implemented in order to strengthen the robustness of the reactors with respect to external hazards. The documents, however, do not provide clear evidence if the processes of the PSR and LTE follow WENRA requirements as stipulated by ASNR. For most external hazards, the methods, data and assumptions used in the hazard assessment are not specified in detail. Conformity with WENRA requirements and guidance can therefore not be assessed. It remains particularly unclear if design basis events with exceedance frequencies not higher than 10^{-4} per annum have been determined for all external hazards that apply to the site, and how DEC are addressed for the identified hazards.

Non-conformity with WENRA Reference Levels is observed for earthquake and seismic ground shaking. The Design Basis Earthquakes (DBE) for Gravelines and the other reactors of the French 900 MWe fleet are still based on deterministic analyses. Defining the Design Basis Earthquake (DBE) on deterministic methods is no longer state of the art. Demonstration that the deterministically determined DBE can be defended against a PSHA-derived DBE with an average recurrence interval of 10,000 years is missing in the EIA documents.

The EIA documents clarify that a PSHA for the Gravelines reactors was conducted to derive the SND which is relevant to the design of the Hardened Safety Core (Noyau Dur). The PSHA revealed a ground acceleration of 0.41 g for the SND which corresponds to an average earthquake return period of 20,000 years. No PSHA result other than this single number is communicated. Documents, in particular, do not state a ground motion value characterizing the 10,000 years earthquake (occurrence probability 10^{-4} per year) which, according to WENRA (2021), shall be used to define the seismic design basis of an existing NPP[18]. It is therefore unclear if the deterministically derived seismic design basis value for the Gravelines reactors, the SMS=0,2 g, envelopes the ground motion value of a PSHA-derived design basis earthquake with an average recurrence interval of 10,000 years. The relatively high value for the SND (0.41 g) suggests that this may be not the case. It therefore remains to be demonstrated that the seismic resistance of all SSCs important to safety is sufficient to conservatively ensure the fundamental safety functions for a DBE with an average recurrence interval of 10,000 years as required by WENRA (2021).

With respect to safety upgrades of the Gravelines reactors, it is evident that one of the most important measures to provide protection against external hazards is the implementation of the Hardened Safety Core (Noyau Dur). However, the implementation of the Noyau Dur is still pending. Implementation is announced for Phase B of the 4th PSR without announcing concrete time schedules in the EIA documents. The timeline prescribed by ASNR envisages implementation of the Noyau Dur in 2029. The fundamental decision to implement the Hardened Safety Core has been made in 2012 in the aftermath of the and the European Stress Tests (ASN 2012). The fact that the implementation of the Noyau Dur will be still pending 17 years thereafter appears remarkable at the background that WENRA requires the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). This suggests that the announced implementation schedules violate the WENRA requirement.

Terrorist attacks and acts of sabotage can have a significant impact on nuclear facilities and cause serious accidents. Nevertheless, they are only mentioned in very general terms in the EIA documents submitted. Similar EIA reports have covered such events to a certain extent. Even if precautions against sabotage and terrorist attacks cannot be discussed in detail for reasons of confidentiality, the necessary legal requirements should be set out in the EIA documents.

Information regarding the issue of terror attacks would be of great interest, considering the far reaching consequences of potential attacks. In particular, the EIA documents should include information on the requirements for the design against the targeted crash of a commercial aircraft. This topic is particularly

[18] WENRA 2021, Issue TU, Reference Level TU4.2

important, because reactor building as well as the spent fuel building of the Gravelines NPP is vulnerable against airplane crashes. It is important to mention that the EPR's 1.8-meter-thick outer reinforced concrete shell is designed to withstand the impact of a large passenger aircraft. However, the wall thickness at the Gravelines NPP is less than 1.0 m. Furthermore, the increasing availability and performance of drones is raising the potential threat to nuclear facilities. A recent assessment of the nuclear security in the France points to shortcomings compared to necessary requirements for nuclear security in regard to "security culture", "cybersecurity" and "protection against insider threats".

- Information on the methods, data and assumptions used for the PSHA performed to determine the SND for the Gravelines reactors should be provided, in particular, the types of seismic sources considered (source zones and/or fault sources), time coverage of the earthquake catalogue, minimum and maximum magnitudes, ground motion prediction equations, and site conditions.
- Information on the ground motion value corresponding to the occurrence probability of 10^{-4} per year derived from the PSHA which was performed to determine the SND for Gravelines reactors should be provided.
- A comparison of the ground motion values (PGAH, spectral accelerations) of the current deterministically derived design basis earthquake and the corresponding values derived by PSHA should be provided.
- Information on protection requirements of the Gravelines NPP with regard to the intentional crash of a commercial aircraft should be provided.
- The PSHA performed for determining the SND by assessing the validity of methods, data and assumptions used in the PSHA and to benchmark the PSHA with regard to WENRA requirements (WENRA 2021) and recommendations (WENRA 2020 a,b).
- Dedicated assessments of near-regional faults for which it cannot be excluded that they are active should be required, in line with WENRA (2020b). The approach may be similar to the one currently applied by EDF to the site of Cruas NPP including field geology, morphostructural and dating studies, and paleoseismology.
- The deterministically derived SMA and the current seismic design basis of Gravelines reactors with the ground motion values derived from probabilistic seismic hazard assessment (PSHA) for a DBE with the occurrence probability of 10^{-4} per year should be compared.
- Additional safety demonstrations to ensure that all SSCs relevant to safety can cope with a probabilistically derived new Design Basis Earthquake (DBE) in case the probabilistically derived DBE exceeds the ground motion parameters of the current seismic design basis of the plant should be required.
- The methods, data and assumptions used to derive hazard values for all external hazards considered in the EIA documents should be reviewed, in line with WENRA requirements and guidance (WENRA 2020a-d; 2021).

- Design basis events and design basis parameters should be defined for external hazards conform with WENRA (2021) requirements.
- It should be ensured that the use of the Noyau Dur's DEC equipment is not required to protect the facility against design events, i.e., events with recurrence intervals of 10,000 years or less (e.g., earthquakes). This is to ensure the independence of Defence-in-Depth (DiD) levels 3 and 4
- It should be evaluated if the long timeframe for implementing the Noyau Dur at the Gravelines reactors is in line with the requirement of the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). Background: the timeframe for implementing the Noyau Dur at the Gravelines reactors extends up to 2029, i.e., 17 years after ASNR's initial decision to implement Hardened Safety Cores at the French NPP fleet.
- In this context the following questions should be addressed:
  - Is it correct that strong earthquakes with recurrence periods longer than 150,000 years were not considered in the seismic PSA for the Gravelines NPP which, according to the EIA documents, revealed a contribution to the CDF of $8*10^{-7}$ per year? If yes: What would be the CDF if earthquakes with longer recurrence intervals were taken into account as well?
  - Have design basis events with exceedance frequencies not higher than 10^{-4} per annum and corresponding design basis loads been defined for all natural hazards considered in the EIA documents (extreme temperatures, river floods, high wind, tornado etc.)?
  - What are the main reasons for the excessively long timeframe (up to 2029) for implementing the Noyau Dur at the Gravelines reactors?
  - Have any studies been or will be carried out on the threat posed by newer technologies, in particular potential attacks using civilian or military drones?
  - How is the result of the Nuclear Security Index 2023 for France assessed? Are improvements planned with regard to "security culture", "cybersecurity" and "protection against insider threats"?

5 SAFETY ASPECT OF ACCIDENT WITHOUT CORE MELT AND SPENT FUEL POOL

5.1 Treatment in the EIA documents

As established in the Chapter on Procedure, the Periodic Safety Review (PSR) framework in France is structured into two distinct phases: a generic assessment and a plant-specific assessment.

Each phase addresses two core objectives:

- Safety Requirements Compliance: A thorough assessment of the plant's adherence to the defined and evolving Design Basis safety requirements.
- State-of-the-Art Upgrades: Identification and specification of measures required to align the plant with the Current State of the Art (SOTA) in nuclear technology. The Flamanville 3 EPR (European Pressurized Reactor) serves as the reference standard for the Current State of the Art.

Scope of Measures and Review Focus: This chapter details the modifications and upgrades specified in EIA-REPORT G2 P.1 (2025), EIA-REPORT G4 P.1 (2025), focusing on two critical safety topics:

- Accidents Without Core Melt: This category encompasses operational transients, Design Basis Accidents (DBA) of varying likelihood, and Design Extension Conditions (DEC) involving multiple system failures that are prevented from progressing to core melt or significant fuel damage and
- Spent Fuel Pool (SFP) Integrity and Cooling.

The documents provided for Gravelines 2 and 4 use the same generic methodology and measures for both reactors in the context of accidents without core melt and for spent fuel pools. The only notable differences are implementation details—these are scheduling/status nuances rather than major programmatic differences.

Key measures for accidents without core melt (EIA-REPORT G2/G4 P.1,2025)

EIA-REPORT G2 P.1 (2025) and EIA-REPORT G4 P.1 (2025) provide executive summaries and outline the highest-priority measures identified for implementation regarding Accidents Without Core Melt.

Measures Implemented:

Accidents-1, Augmented Ultimate Heat Sink Connection for Steam Generators (SGs):

Modification: Establishment of diversified interconnection points linking the Steam Generator Auxiliary Feedwater System (ASG) to the Fire Fighting Water Reservoir.

Rationale: To mitigate certain accident sequences involving the complete loss of both main and emergency feedwater systems. This connection provides a crucial alternate, unconventional heat removal source by ensuring a robust water supply to the Steam Generators, thereby maintaining the primary system's heat sink capability.

Accidents - 2. <u>Increased Relief Capacity of Steam Line Valves (GCT-a Modification):</u>

Modification: Uprating of the mass flow capacity through the Main Steam Line Safety and Relief Valves.

Rationale: The enhanced steam relief rate permits a significantly faster depressurization and cooldown of the Reactor Coolant System (RCS) during specific design basis or design extension transients. This capability accelerates the transition to a safe shutdown state and reduces thermal-hydraulic stress on the system components.

Accidents -3: <u>The allowable amount of Iodine in the primary system coolant was decreased</u>

While this measure is undoubtedly beneficial, the report does not indicate which operational measures were taken to achieve it. The iodine concentration in the primary coolant is the result of a balance between the release of iodine from the fuel due to micro-failures in the fuel rods and the operation of the makeup and letdown systems, which remove fission products from the primary system coolant. Was this system modified? Will it be operated for longer time periods? If so, how will this affect its reliability? How is the phenomenon of iodine spiking considered?

Key measures for the spent fuel pool (EIA-REPORT G2/G4 P.1 2025)

For the spent fuel pool EIA-REPORT G2 P.1 (2025) and EIA-REPORT G4 P.1 list the following items:

Measures proposed:

Pool-1: Fire: In the event of a fire, to prevent the loss of both cooling paths, EDF has planned the addition of a flame arrestor device to eliminate the risk of a fire spreading from one pump in the cooling circuit to the other.

Pool-2: Accident scenarios: Following the transpose of EPR FLA3 scenarios to 900 MWe plants, to further secure spent fuel pool cooling, EDF plans to duplicate the automatic isolation device on the suction line of the pool's normal cooling circuit, ensuring reliable isolation under accident conditions even if one device fails.

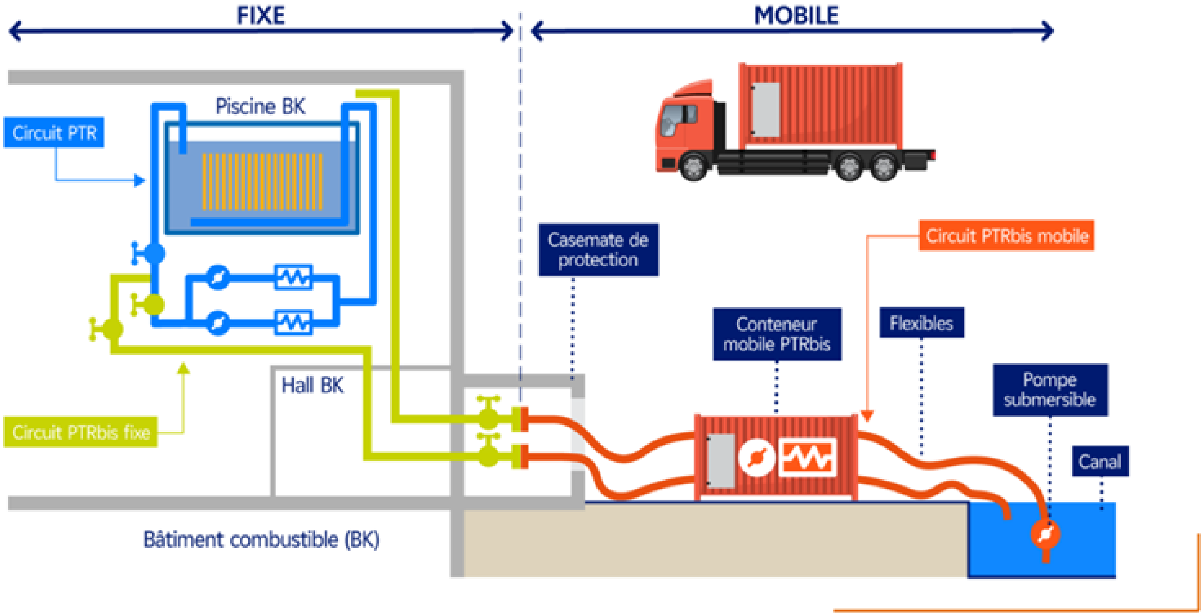
Measures implemented:

Pool-3: Additional pool cooling “PTR bis”: As part of the post-Fukushima measures, the diversified water source (SEG) allows for the replenishment of water in the fuel building pool. During 4th PSR, a new mobile cooling system (PTR bis) for the pool allows for diversification of the cold source and, in the event of a loss of the cooling circuit during normal operation, ensures a return to a cooling state for the fuel pool without boiling. This type of arrangement brings the design of 900 MWe reactors closer to that of EPR FLA3 type reactors.

While the mobile cooling system is already implemented, the fire prevention system is still in the planning phase.

Figure 3: Schematic diagram of the cooling circuit of the spent fuel pool (EIA-REPORT G2/G4 P.1 2025)

Schematic diagram of the cooling circuit of the spent fuel pool



Source: Environment Agency Austria

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EIA-REPORT G2 P.2 (2025) and EIA-REPORT G4 P.2 (2025) represent the most extensive of the five reports submitted for the Plant Lifetime Extension (PLEX) review. Their section on risks is logically segmented into two main components:

Conformity Evaluation to Applied Safety Standards: An assessment against the existing licensing basis.

- Re-evaluation (SOTA Comparison): Derivation of necessary measures by comparing the safety profile of Gravelines 2 reactor against the Current State of the Art (SOTA), as defined by the Flamanville 3 EPR design.
- The "Conformity" section is deemed outside the scope of this discussion as it does not relate to Accidents Without Core Melt or the Spent Fuel Pool (SFP). The following focuses on the considerations within the Re-evaluation chapter.

Re-evaluation of Accidents Without Core Melt

EDF's approach to the "Accidents Without Core Melt" scenario involved a comprehensive safety re-evaluation of operational transients, Design Basis Accidents (DBA), and Design Extension Conditions (DEC) Category A.

This re-evaluation utilized both deterministic safety analysis (DSA) and probabilistic safety analysis (PSA) methodologies. A primary goal of this exercise was the reduction of potential radiological consequences associated with these events, aligning the older units with the risk profile of the EPR.

The generic Periodic Safety Review (PSR) specifically mandated the investigation of the following categories of initiating events and accidents:

Reactivity Initiating Accidents.

- Uncontrolled withdrawal of control rod banks during startup.
- Uncontrolled withdrawal of control rod banks at power.
- Control rod cluster misalignment, drop of a control rod cluster, or drop of a control rod bank (group of clusters).
- Uncontrolled boron dilution.
- Withdrawal of a single Power Control Rod Cluster.
- Control rod ejection accident.

Thermal-Hydraulic and Heat Removal Transients

- Partial loss of primary coolant flow or Forced reduction of primary coolant flow.
- Total loss of load and/or turbine trip.
- Loss of normal feedwater to the Steam Generators.
- Malfunction of normal feedwater.
- Excessive load increase.
- Inadvertent opening of a secondary relief valve.
- Small break on secondary piping.
- Major Steam Line Break, Category 4.
- Major feedwater line break.
- Momentary depressurization of the primary circuit.
- Loss-of-Coolant Accidents and System Integrity Events
- Loss-of-Coolant Accident due to a small break with a diameter ≤2.5 cm.
- Intermediate Loss-of-Coolant Accident, Category 4.
- Inadvertent actuation (startup) of the safety injection system.
- Inadvertent opening of a pressurizer safety valve.
- Steam Generator Tube Rupture, Category 3.
- Category 4 Steam Generator Tube Rupture (combined with a stuck-open secondary relief valve).

Equipment and Operational Failures

- Total loss of off-site power (or Loss of external electrical power supplies).

- Seizure/Locked rotor of a Reactor Coolant Pump (RCP / motopompe primaire).
- Fuel and Core Design Events
- Class 2 Power Capability (a capacity limit check for verifying the sizing of the Reactor Protection System).
- Fuel assembly misalignment in the core.
- Fuel handling accident (in-reactor).
- Irradiated fuel container handling accident.

Gravelines 2 and 4: Reactor-Specific PSR Modification Status

During the plant-specific phase of the Periodic Safety Review directed at the Gravelines Nuclear Power Plant, EDF categorized safety enhancements into three groups based on their implementation status: *fully completed, currently deploying (Phase A), and scheduled for later deployment (Phase B).*

Accidents Without Core Melt

Fully Implemented Modifications

The following modifications have been fully completed on Gravelines units 2&4, and all associated documentation impacts have been integrated:

- PNPE1141 Increased flow rate of GCT-a- control valves
- PNPP1595 Volume B "Valve Head Replacement" SEBIM.
- PNPP1838 "Renovation of the RPN CPY in VD4".
- PNPP1864 "Refilling the ASG reservoir via the water circuit" JP* fire".
- PNPP1873 "SIP-Protection System Evolution".
- PNRL1817 "Tmoy filter – SIP C".
- PNRL1829 "Increased REA boron volume required increase in PET free volume.
- PNRL1903 "ASG repowering by JP* (STE alarms: creation on NTB JPP thresholds)".
- Generalization of hafnium-absorbing clusters in the VD4 900 reference.
- Power dilution alarm sheet.

Modifications Currently Being Deployed - Phase A

The following modifications are currently being deployed at Gravelines units 2&4, with remaining integration activities scheduled for completion within Phase A of the 4th PSR 900 modifications:

- Operating strategy for H3/DCC-LH transients in the Domaine Complémentaire - temperature threshold set to 240°C to balance preservation of primary pump seals -> plan to refine this to 190°C in Phase B

Modifications Scheduled for Phase B

The following modifications are planned for deployment at Gravelines units 2&4 during the subsequent Phase B of the 4th PSR modifications:

- PNPE1359 "Increase in accumulator pressure" RIS.
- PNRL1957 "Modification of the blocking line of group R".

Spent Fuel Pool – BK (Bâtiment combustible)

Fully Implemented Modifications

The following modifications have been fully completed on Gravelines units 2&4, and all associated documentation impacts have been integrated:

- PTR bis mobile diversified cooling system (hors tome N): A mobile, diversified cooling path has been fully implemented on Unit 2 and 4, enabling rapid restoration of pool cooling via pre-installed connections. Documentation impacts have been integrated.
- Provisional ultimate water makeup source (PNPE1348): A temporary, alternative source of ultimate makeup water has been fully implemented to meet the prescription pending completion of the permanent source (PNPE1289 is planned by end of 2024).

Modifications Scheduled for Phase B

The following modifications are planned for deployment at Gravelines units 2&4 during the subsequent Phase B of the 4th PSR 900 modifications:

- Doubling the automatic isolation of the spent fuel pool suction line (PTR) and adding Noyau Dur water makeup to the spent fuel pool: EDF plans to implement the doubled isolation logic (PNPE1344) and the Noyau Dur makeup path to the spent fuel pool (PNPP1714/PNPE1258), strengthening drain-down prevention and ensuring robust emergency top-up capability.
- ASG-ND fixed line for spent fuel pool re-supply via SEG (PNPE1258) and the reactor building arrangement (PNRL1803): These Noyau Dur dispositions will be deployed to provide fixed, seismically robust makeup routing and ensure steam exhaust doors remain open in APR states (Arrêt Pour Rechargement), enabling gravity makeup from the reactor building to the spent fuel pool when pools are connected.
- Level surveillance upgrades:
  1. Reactor building pool "Tout ou Rien" level measurement (PNPE1128) to enhance detection and response.
  2. Spent fuel pool analogue level measurement chain (PNPP1824) to improve continuous monitoring and control.
- Fire separation between PTR pumps (PNPP1949): Installation of a physical fire screen between PTR pumps to prevent simultaneous loss of both cooling paths.

- Permanent ultimate water makeup source (PNPE1289): Planned for Unit 2 by end of 2024; until then, the provisional source (PNPE1348) remains in use.

Document EIA-REPORT G2 P.3 (2025) and EIA-REPORT G4 P.3 (2025) provide easy-to-use lists of measures but no new information in respect to EIA-REPORT G2/4 P.1 (2025) and EIA-REPORT G2/G4 P.2 (2025). The documents EIA-REPORT G2 P.4 (2025) and EIA-REPORT G4 P.4 (2025) give an overview of the “Lessons learned by EDF from the consultation on the generic phase of the 4th periodic safety review of 900 MWe reactors”. Although they dedicate a section to the robustness of the spent fuel pool no additional information is given. EIA-REPORT G2 P.5 (2025) and EIA-REPORT G4 P.5 (2025) provide relevant snippets from the French Environmental Code in the context of a periodic safety review.

5.2 Discussion

Generic aspects

Accidents-1: Augmented Ultimate Heat Sink Connection (SG Feedwater)

The installation of a diversified connection to the Fire Fighting Water Reservoir for the Steam Generator (SG) Auxiliary Feedwater System (ASG) is a recognized and valuable enhancement. This measure aligns with post-Fukushima accident safety upgrades implemented across numerous Nuclear Power Plants (NPPs) globally to secure the Ultimate Heat Sink (UHS) function.

The historical operation of the Narora NPP Unit 1 (India), which utilized the fire brigade system to sustain cooling during a prolonged Station Blackout (SBO) exceeding 18 hours following a catastrophic cable fire, provides a practical precedent for the long-term effectiveness of this approach. Providing a dedicated connection ensures that mobile fire pump assets can effectively facilitate long-duration residual heat removal from the primary system.

Accident-2. Uprated Steam Line Safety and Relief Valve Capacity (GCT-a)

While the increased mass flow capacity of the Main Steam Line Safety and Relief Valves is clearly beneficial for accelerating reactor cooldown during various transients, the assessment report is deficient in providing key quantitative data.

Information Gaps: the report omits the initial and final mass flow rates (e.g., in kg/s or lbm/s) achieved by the upgrade. Crucially, a comparison is missing between the new maximum discharge capacity and the steam flow per steam line during normal operation to contextualize the magnitude of the capacity increase.

Potential Adverse Effects: Increasing valve capacity could potentially introduce adverse effects in specific high-pressure scenarios, such as a Steam Generator

Tube Rupture (SGTR) accident. An SGTR constitutes a containment bypass scenario which typically leads to a transient increase in SG pressure. While the valve opening is intended to relieve pressure, an excessively large discharge capacity could intensify the uncontrolled release of primary coolant (contaminated with radioactive material) to the atmosphere, thus challenging the integrity of the release mitigation strategy.

Accidents-3. Reduced Primary System I-131 Limit

The measure to enforce a lower permissible concentration of Iodine-131 (I-131) in the Reactor Coolant System (RCS) water is undeniably beneficial for reducing the potential radiological source term during accidents.

Implementation Gaps: The report lacks crucial details on the methodology for implementing and enforcing this reduced limit.

The assessment does not specify whether the effects of iodine spiking—a rapid, transient increase in iodine concentration during depressurization events—have been adequately considered in the design basis or operational procedures related to this new limit.

Pool-1: Installation of Flame Traps in the SFP Building

The planned installation of flame traps within the Spent Fuel Pool (SFP) building ventilation system represents a highly commendable and undoubtedly beneficial safety enhancement, particularly against hydrogen combustion or other potential ignition sources.

Implementation Status Gap: The benefit of this measure is currently mitigated by the fact that it is not yet fully implemented, and the report fails to provide a firm, committed timeline for its completion.

Pool 2: Mobile Cooling Capabilities

The establishment of infrastructure and procedures to enable SFP cooling via mobile, diverse sources is a critical defence-in-depth measure. This measure is directly aligned with the lessons learned and subsequent industry requirements arising from the Fukushima Daiichi accident. This enhancement ensures the long-term cooling and inventory control of the SFP under Design Extension Conditions (DEC) and has been successfully implemented.

The re-evaluation during the generic phase has resulted in a large number of safety improvements, many of which are already implemented. However, the status of two crucial measures mandated by the ASNR following the conclusions of the 4th PSR remains to be clarified. EDF is currently carrying out supplementary studies on these two fuel-related topics:

1. Critical Heat Flux (CHF) Correlation Validity (Requirement [Study-B])

Requirement: By December 31, 2024, EDF must evaluate, using an experimental approach, the validity of the Critical Heat Flux (CHF) correlation applied to the periphery of deformed fuel assemblies. Concurrently, EDF must define the work program and schedule to integrate the lessons learned.

Action & Status Question: EDF submitted a detailed test configuration program to the ASNR in June 2021. The text provides no information on whether the CHF experimental program has been completed or what its current status is.

2. Fuel Assembly Grid Buckling Limit (Requirement [Study-D])

Requirement: EDF performed tests to characterize the buckling limit of fuel assembly grids under a more realistic configuration than historical test rigs.

Finding: The test results were used to evaluate fuel assembly mechanical behavior during a Category 4 Loss-of-Coolant Accident (LOCA) concurrent with a contemporary seismic event. This evaluation confirmed that neither core cooling capability nor the control of reactivity via control rod drop were compromised.

Implementation: EDF must update the relevant safety analysis reports and integrate the findings into the Target Technical Specifications (TTS) by the deadline of the 5th PSR of the 900 MWe series. This timeline is standard for integrating complex, regulator-approved technical specifications that affect operational procedures.

For the site-specific measures for Gravelines 2 and 4, the question remains open as to whether there is a specific date by which these measures will be fully implemented.

5.3 Conclusions

While the generic and plant-specific phases have resulted in numerous beneficial safety improvements, several key issues require immediate resolution. Firstly, the reports suffer from a lack of quantitative data necessary to fully assess the benefit and potential adverse effects (e.g., during an SGTR) of the GCT-a valve uprate. Secondly, the implementation status of some critical measures, such as the SFP Flame Trap Installation, is currently unconfirmed with a firm timeline, creating an unquantified safety risk. Finally, there is a lack of justification for deferring beneficial State of the Art upgrades like the RIS Accumulator Pressure Increase. Transparency in reporting, commitment to firm deadlines, and clarification of technical justifications are necessary to fully validate the safety improvements derived from the PSR.

Enhance Transparency and Provide Clarity on Key Quantitative Data

- Quantitative Data: The reports should provide the initial and final mass flow rates for the GCT-a Valve Uprate (PNPE1141), along with a comparison to the nominal operational flow. This is necessary to quantify the safety benefit.
- Adverse Effects Analysis: The analysis of the uprated GCT-a capacity should be expanded to quantify the risk of increased radioactive release

during a Containment Bypass scenario like a Steam Generator Tube Rupture. This ensures that the modification does not introduce new, unacceptable risks.

- Radiological Implementation: Detailed methodology on how the Reduced Primary System I-131 Limit will be implemented and monitored should be provided, explicitly addressing how iodine spiking will be accounted for in operational procedures and design basis analyses.

Establish Firm and Accountable Timelines

- Missing Deadlines: EDF and the ASNR should establish a firm, committed timeline for the completion of the SFP Flame Trap Installation (Pool-1). The absence of a fixed date creates an unquantified safety risk.
- Study Status and Next Steps: For the Critical Heat Flux (CHF) experimental program (Requirement [Study-B]), EDF should immediately provide an updated status on its completion and publicly commit to the defined work program and schedule for incorporating the findings, as the reporting deadline was December 31, 2024.

Clarify Status Reporting and Implementation Rationale

- Justify Deferral: A comprehensive safety justification for deferring beneficial state-of-the-art measures like the RIS Accumulator Pressure Increase to Phase B of the implementation cycle should be provided. This justification should explicitly weigh the cost/complexity against the temporary safety margin reduction.
- Resolve Discrepancies: The conflicting status between the different report parts should be clarified. Future reporting should clearly define the criteria for "implemented" (design complete vs. installation complete) should be clarified to prevent ambiguity, the same is true for planned measures. The reports do support a distinction between Phase A and Phase B, but it is often difficult to follow if the mentioned dates deal with the design or the installation.

6 SAFETY ASPECTS OF CORE MELT ACCIDENTS

6.1 Treatment in the EIA documents

As part of 4th PSR, EDF's goal is to significantly reduce the risk of early and significant releases in the event of core-melt accidents in order to avoid lasting effects on the environment. Two projects are planned to achieve this goal:

- Stabilization of the corium on the reactor building basement by distributing and cooling it. The aim is to prevent the basement from breaking through in order to retain the contaminated water resulting from the accident in the reactor building, treat it to remove the radionuclides it contains, and thus prevent the spread of liquid radioactive substances outside the site ("waterway").
- the removal of residual heat from the core without opening the containment pressure relief and filtration system (U5-System), in order to prevent the release of radioactive substances into the air ("air route").

Stabilization and Cooling of the Corium

The corium spreads after breaking through the reactor pressure vessel in the vessel well and in the room of the reactor core instrumentation (RIC room). To limit the risk of losing the containment integrity in the event of a core-melt accident due to erosion of the basement, a device is used that is based on stabilizing the corium underwater after it has spread in the dry (PNPP1976)[19]. According to EDF, this solution is similar in principle to that used in EPR (core catcher). This arrangement complies with regulation [AG-A-I].

In application of regulation [AG-A-II], EDF has submitted

- a detailed preliminary draft for the reinforcement of the containment basement, whose concrete has a high silica content,
- submitted the conclusions of its test-based investigation program on the behavior of basement in the event of core-melt accidents.

EDF has identified the sites where the basements need to be reinforced. The thickening of the basement will be carried out specifically at the sites concerned.

The units at the Gravelines site are not affected by regulation [AG-A-II] on thickening the basements of highly silicate containment buildings, as their basements are made of silicate concrete. (EIA-REPORT G2 P.2 2025)

In addition, and in accordance with regulation [AG-A-III], EDF will reinforce the walls between the RIC room and the area of the water collection basins at the

[19] PNPP1976: "Installation of a device for dry distribution and stabilization of corium under water" has been fully implemented at Gravelines 2 and 4.

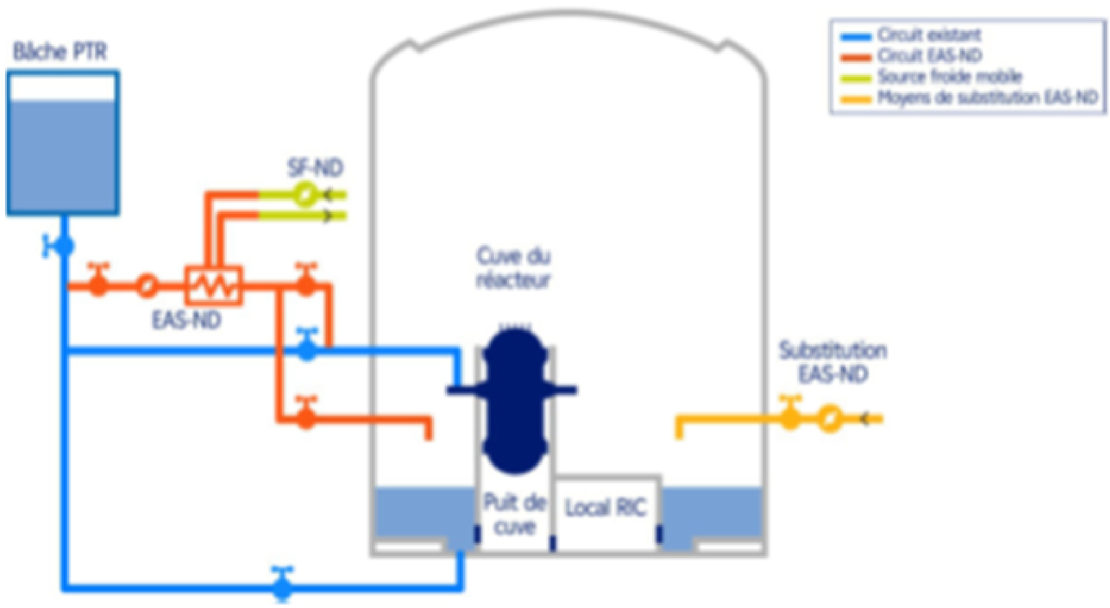
bottom of the reactor building in order to avoid any risk of corium penetration (PNPE1460) [20].

The dry distribution of the corium is ensured by the prior sealing of the containment room and the adjacent RIC room. The corium is then drowned by gravity with the water present in the sumps at the bottom of the reactor building filled by the safety injection systems (SIS), the sprinkler system (EAS) or the "Hard Core" sprinkler system (EAS-ND). Gravity refilling of the corium is ensured by redundant holes in the walls of the vessel and RIC rooms, which are closed by passive valves (or flaps) that ensure tightness between the water accumulated at the bottom of the building and the spreading area. This guarantees dry spreading of the corium. The removal of the sealing device is triggered after the corium has spread by the tearing of fusible plugs.

The measurement for detecting a vessel penetration (PNXX1746)[21] makes it possible to ensure water injection onto the corium at the most effective time. The cooling of the corium and the long-term removal of residual power are ensured by the EAS-ND and hard-core cooling source (SF-ND) measures.

Figure 4: Cooling devices in the event of core meltdown (EIA-REPORT G2 P.1 2025)

Schematic diagram of the cooling circuit of the spent fuel pool



Source: Environment Agency Austria

umweltbundesamt

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[20] PNPE 1460: "Reinforcement of the walls between the internal instrumentation room of the reactor core (RIC) and the sump area at the bottom of the containment" will be implemented in the Supplementary Phase for Gravelines 2 and 4.

[21] PNXX1746: "Detection of an RDB breakthrough and hydrogen recombinator function at high temperatures" have been fully implemented at Gravelines 2 and 4.

EDF will implement an additional measure that, in the event of a medium- to long-term failure of the EAS-ND, allows water to be replenished using mobile means for a sufficient period of time to limit erosion of the basement (PNPE1362)[22]. This measure complies with regulation [AG-B-III]. This replenishment is controlled by measuring the water level at the bottom of the reactor building (PNPE1386)[23].

In addition, following the investigation by the Permanent Group of Experts on Reactors (GPR), special instrumentation to detect the spread of corium over the entire area of the RIC room (PNPE1387)[24] will be implemented.

According to the EIA-REPORT G2/G4 P.2 (2025), the annual frequency of breakthroughs in the basement was estimated at around 10^{-6} / year at the end of the 3rd PSR. Due to the planned measures, the probability of a breakthrough of the basement is reduced to approximately 10^{-7} / year, which is in line with the goal of avoiding effects on the environment.

Removal of residual heat without filtered venting

The evaporation of water on the corium and the formation of non-condensable gases during the interaction between corium and concrete lead to a slow increase in pressure in the containment. The pressure can reach the design pressure of the containment and necessitate the opening of the pressure relief and filter device (U5-System), resulting in radioactive releases into the environment.

The implementation of the EAS-ND provision (PNPP1811)[25] as part of the 4th PSR also enables the residual heat to be dissipated from the containment. The EAS-ND arrangement is dimensioned in such a way that situations involving core meltdown, which would lead to the opening of the containment filter device, are avoided.

The “EAS-ND” arrangement comprises:

- A pump that can be operated either with direct injection from the PTR tank into the primary circuit or with recirculation from the collection tanks of the reactor building,
- A heat exchanger that transfers the heat from the primary circuit pumped by the pump (EAS-ND) to the hard-core cooling source (SF-ND),

[22] PNPE1362: “Installation of fixed injection and extraction lines in the reactor building and mobile replacement device for the EAS-ND, return of wastewater from the BK to the BR” will be implemented in the Supplementary Phase for Gravelines 2 and 4.

[23] PNPE1386: Installation of a sump level measurement system in the reactor building will be implemented in the Supplementary Phase for Gravelines 2 and 4.

[24] PNPE1387: “Installation of a detection system for the spread of corium in the RIC room (core instrumentation) and emergency power supply via the DUS (Diesel Ultime Secours) upon detection of a containment breach” will only be implemented in Phase B for Gravelines 2 and 4.

[25] PNPP1811: “Installation of an EAS-ND system for feeding water into the primary circuit and for dissipating residual power” has been implemented in Gravelines 2 and 4.

- The SF-ND consists of a mobile pumping device that is transported and deployed by the FARN. It is connected to the cooling circuit via flexible pipes connected to connections at the edge of the reactor building.

Certain valves or valve seals on auxiliary lines of the EAS-ND device will be replaced as part of measure (PNPE1471)[26] to ensure their resistance under accident conditions involving a core-melt accident.

In order to further limit the risk of a pressure increase in the containment building, EDF has defined measures in accordance with regulation [AG-B-II-1], that, in addition to the water contained in the tank of the water treatment and cooling system of the pools (PTR), will allow a further quantity of boron-containing water to be fed into the reactor building in the short term in order to remove residual heat from the containment in the event of a core-melt accident.

The long-term management of core-melt accidents is based on the circulation operation of the EAS-ND system to keep the corium submerged and remove residual power from the reactor. EDF is setting up a system to manage any leaks that may occur in the EAS-ND circuit (PNPP1541)[27] outside the containment building. In addition, EDF is installing a device to return the wastewater present in the collection tanks of the spent fuel building to the reactor building (PNPE1362)[28]. These devices for collecting and recirculating comply with the regulations with regulations [AG-B-IV] and [AG-D-I].

To reduce the potential radiological consequences, the modification "Installation of sodium tetraborate baskets in the sump basins of the reactor building" (PNPE1410) will be implemented at the latest for Gravelines 2 in March 2029 and for Gravelines 4 by December 12, 2029 in accordance with [CR-B] The proposed arrangement consists of installing fixed devices in the floor of the reactor building that contain an alkali salt that dissolves in water and retains the iodine in the water, thus limiting its transition to the gas phase. The devices are passive and consist of baskets filled with disodium tetraborate decahydrate.

Reinforcement of the U5-System

Based on the lessons learned from the Fukushima accident, the pressure relief and filter system of the containment (U5-System) was initially reinforced to ensure its resistance to an SMHV earthquake. In accordance with regulation [AG-C-II], the U5-System will be further reinforced to ensure its resistance to earthquakes of magnitude SMS (PNPE1377)[29].

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[26] PNPE1471: "Replacement of valves or valve seals on the EAS ND" will be carried out in Phase B for Gravelines 2. It has been already performed for Gravelines 4.

[27] PNPP1541: "Introduction of a system for collecting wastewater in the event of a core-melt accident" has been implemented for Gravelines 2 and 4.

[28] PNPE1362: see above

[29] PNPE1377: "Reinforcement of the compression and filter device of the U5 container in the event of an SMS earthquake" at the latest on (19/12/2029) for Gravelines 4 and in March 2030 for Gravelines 2.

Management of contaminated water

As part of crisis management, short- and long-term compliance with drinking water quality guidelines following a core-melt accident must be ensured as follows:

In accordance with regulation [AG-D-II], EDF has the necessary means to reduce water contamination in the reactor building following a core-melt accident and ensures that these means are operational on site (PNPE1362)[30] and (PNPE1449)[31].

In accordance with regulation [AG-D-III], EDF has investigated ways of limiting the spread of radioactive substances via the soil and groundwater outside the site in order to limit water contamination in the environment following a core-melt accident. According to EDF, these investigations have not revealed any need for additional measures with regard to safety risks.

6.2 Discussion

Severe accidents (SA) were not taken into account in the design of the French 900 MWe reactors. However, as a result of previous PSRs, equipment and measures for SA management have been implemented. The EU stress tests have nevertheless revealed a number of shortcomings.

According to ASNR, the objective of the 4th PSR for the 900 MWe reactors is to approximate the safety level of the third-generation reactor in Flamanville (EPR). In third-generation reactors, core-melt accidents are already taken into account in the design of the reactors; the measures taken for these reactors cannot be fully transferred to second-generation reactors such as Gravelines 2 and 4.

It is state of the art to use the WENRA "Safety Goals for New Power Reactors" as a reference for identifying meaningful safety improvements during an LTE project. (WENRA 2013) According to the WENRA safety objectives, core-melt accidents that would lead to early or large releases should be practically excluded. The occurrence of certain severe accidents can be considered to be practically excluded "if it is physically impossible for the conditions to occur, or if it can be assumed with high confidence that the occurrence of these conditions is extremely unlikely". The concept of "extremely unlikely with high confidence" is an essential part of the IAEA's concept of "practical exclusion". Although this concept applies only to new reactors, it should also be applied to the Gravelines 2 and 4 in order to reduce the existing risks. Especially since the goal of the 4th PSR is to approach the safety level of the new EPR in Flamanville. The EIA documents do not include a systematic comparison between the safety level of the

[30] PNPE1362: see above

[31] PNPE1449 "Investigation of a mobile water treatment module for treating contaminated water" will be implemented as part of the "Supplementary Phase" for Gravelines 2 and 4.

900 MWe reactors and modern safety standards in order to identify the remaining gaps.

EDF's modifications focused on heat removal without opening the filtered venting devices and stabilizing and cooling the corium on the basement.

Stabilization and Cooling of the Corium

The strategy envisaged by EDF in the context of the 4th PSR to limit the risk of the basement melting through consists of solidifying the corium after failure of the reactor pressure vessel (RPV) and cooling it over the long term. In order to implement this strategy, adaptation work must be carried out inside the reactor building and new circuits must be installed.

The concrete dissolves under the influence of the heat of the corium, which can cause the basement to melt through. The solidification of the corium and the thickness of the melted concrete depend on the type of concrete used in the basements. For Gravelines 2 and 4 the siliceous concrete has been used. Thus, the thickening of the basement is not seen as necessary.

The coolability of the corium in the ex-vessel phase was subject to large uncertainties. The geometry of the 900 MWe reactor cavity bottom consists of a circular cylinder of inner radius 2.6 m, sided by a rectangular area facing the RIC room), whose dimensions are approximately 4.0 m x 2.6 m. Thus, the total area of reactor pit and RIC room is 31.6 m². Referring to the indicative figure of 0.02 m²/MWth this translates to a necessary area of approximately 55 m² for the 900 MWe reactor. Consequently, the coolability of the corium is unlikely (ASAMPSA 2013)

Studies that have demonstrated the feasibility and effectiveness of this device, which would have important differences with the EPR core catcher. The limitation of the spreading area due to building constraints impedes the realization of the new device.

From the point of view of the current knowledge, a failure of the containment function cannot be excluded after implementation of the modification for the stabilization of molten.

Furthermore, there is a risk of lateral failure of the walls of the RIC room. ASNR therefore considers the strength of the walls to be insufficient and calls for reinforcement. (see [AG-A III]) The walls to the RIC room have not yet been reinforced, although this is necessary to avoid the risk of the corium breaking through. This will be only implemented as part of the Supplementary Phase (PNPE1460).

Although the "installation of a device for dry distribution and stabilization of corium under water" (PNPP1976) has been implemented for Gravelines 2 and 4, effective medium- and long-term cooling can only be guaranteed once all measures have been implemented after Phase B.

It was one of the important lessons learned of the Fukushima accident that is important to have instrumentation that do not lose its function under accident conditions. EDF plans to install temperature measuring devices and instruments for measuring the water level at the bottom of the plant. (PNPE1386) In addition, measuring devices are to be installed to monitor the spread of corium in the RIC room. However, these necessary devices will only be installed in the Supplementary Phase.

Removal of residual heat without filtered venting

The EAS system is designed to dissipate residual heat from the containment in the event of a severe accident. The EAS system is used both to prevent severe accidents and to limit the consequences of severe accidents. A malfunction in one component of the system could therefore disable two safety levels. It does not comply with current IAEA safety requirements for a safety system to be assigned to multiple safety levels.

ASNR requires that the injection of an additional volume of borated water be enabled in order to significantly reduce the risk of a pressure increase. (see [AG-B]) The EAS-ND system for feeding water into the primary circuit and for dissipating residual power (PNPP1811) has been implemented.

In ASNR's view, numerous additional components and measures beyond those previous planned by EDF are necessary to ensure that the residual heat removal system functions effectively in the long term. However, these important modifications are only to be carried out in Phase B or Supplementary Phase of the program.

In the event of leaks, contaminated water could run onto the floor of the fuel building, where the components of the EAS system are installed, and impair its availability and reliability. Early reinjection of water from the floor of the fuel building into the reactor building would limit the impact. The measure provided for this purpose will only be implemented during the Supplementary Phase (PNPE1362). The necessary “replacement of valves or valve seals on the EAS ND” will also be carried out only in Phase B for Gravelines 2. (PNPE1471) The replacement has already been performed for Gravelines 4.

Reinforcement of the U5-System

The U5-System is to be used in the event of a failure of the EAS system to enable filtered venting into the atmosphere during a severe accident in the event of excessive pressure in the containment. ASNR requires that the U5-System remain operational even after a severe earthquake. (see [AG-C])

The backfitting of the U5-System with regard to its lack of resistance against an extreme earthquake has not yet been carried out, although this safety deficit was already identified during the EU stress tests. The backfitting is scheduled to take place by 19/12/2029 for Gravelines 4 and in March 2030 for Gravelines 2.

Management of contaminated water

Following the accident at the Fukushima Daiichi nuclear power plant, ASNR instructed EDF to submit a feasibility study for the installation of a geotechnical barrier to prevent the spread of contaminated water in the event of a serious accident. According to a 2012 EDF study, the benefits of such barriers do not justify the costs.

IRSN assessed the consequences of a meltdown of the basement without a special device to limit contamination. At most river sites, the radionuclide concentration in the respective river could exceed the reference dose values for drinking water (0.1 mSv/year) by a factor of approximately 1,000 several months after the meltdown. In addition, even without penetration of the basement, contaminated water can leak from the reactor building and cause the reference values for drinking water to be exceeded. (UMWELTBUNDESAMT 2021a). EDF has therefore committed to providing measures to reduce the risk of contamination of the surrounding water. (see [AG-D]).

The development and implementation of a sufficiently effective measure to limit the spread of contaminated water into the environment is still ongoing. The measures designated as the second and third lines of defense will only be implemented or investigated during the Supplementary Phase.

A mobile water treatment module for treating contaminated water is envisaged to investigate during the Supplementary Phase. (PNPE1449). Thus, it is not clear if this measure will be implemented at all.

Overall, it cannot be ruled out that contaminated water will be released into the environment following a core-melt accident.

6.3 Conclusions

Severe accidents (SA) involving core meltdown were not taken into account in the design of the French 900 MWe reactors. However, as a result of previous PSRs, facilities and measures for SA management have been implemented. According to the ASNR, the objective of the fourth PSA for the 900 MWe reactors is to bring the safety level of the reactor closer to that of the EPR in Flamanville, a third-generation reactor. In third-generation reactors, features to mitigate the effects of core melt accidents are already implemented in the design; these cannot be fully transferred to second-generation reactors such as Gravelines 2 and 4. The EIA documents do not contain a systematic comparison between the safety level of the 900 MWe reactors and the safety level of the EPR in order to identify the remaining gaps.

The modifications planned as part of the 4th PSR in the event of a core-melt accident focus on heat removal from the containment without opening the filtered pressure relief system and on stabilizing and cooling the corium on the basement.

Based on current knowledge, a failure of the containment cannot be ruled out after the modification to stabilize and cool the molten core has been implemented. On the one hand, not all important modifications have been implemented yet, and on the other hand, it is not possible to assess whether the modifications (especially the reinforcement of the basement) are sufficient based on the available information.

The planned modifications for heat removal without using the filtered pressure relief system in the event of a core-melt accident have not yet been fully implemented. In addition, the reinforcement of the filtered pressure relief system (U5 system) against severe earthquakes has not yet been carried out. This means that even after completion of all Phase A measures of the 4th PSR, a core-melt accident with a major release of radioactive substances is still possible at Gravelines 2 and 4. The EIA documents do not provide a complete overview of which of the planned modifications meet the ASNR requirements published at the end of the generic phase of the 4th PSR. Most of the measures are not scheduled to be implemented until the end of phase B and the supplementary phase (2029). The EIA documents do not indicate whether this schedule will be adhered to.

- The EIA documents should include an overview of which of the planned measures are to be used to meet the ASNR requirements published at the end of the generic phase of the 4th PSR and when they are to be implemented.
- Studies that prove the sufficient thickness of the containment basements and the dimension of the spreading areas for Gravelines 2 and 4 should be provided.
- It should be explained which options were examined to limit the spread of radioactive substances via soil and groundwater after a core melt accident in accordance with regulation [AG-D-III] How is it justified that there is no need for additional measures with regard to safety risks?
- A systematic comparison between the safety level of the 900 MWe reactors and modern safety standards of the EPR Flamanville 3 should be included in order to identify the gaps.
- Information about the core damage frequency (CDF) and the large (early) release frequency L(E)RF before the 4th PSR, after implementation of all modification of 4th PSR and after the end of Phase A of the 4th PSR should be provided.
- Information why the necessary “replacement of valves or valve seals on the EAS ND” (PNPE1471) has been performed at Gravelines 4 but not at Gravelines 2 should be provided.
- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Gravelines 2 and 4. The concept of practical elimination should be used for this approach. Especially since the goal of the 4th PSR is to move closer to the safety level of the EPR Flamanville 3.
- The authorization for continued operation of Gravelines 2 and 4 should be issued only after the planned measures to mitigate the release in the event of a core-melt accident have been fully implemented.

7 RADIOLOGICAL IMPACT OF ACCIDENTS / TRANSBOUNDARY EFFECTS

7.1 Treatment in the EIA documents

Assessment of impact of accidents at unit 2 and unit 4 of Gravelines NPP is provided in the EIA-REPORT G2/G4 P.3b (2025) for the respective units. No differences were identified in the postulated events or in the assessed impacts on the public and the environment for the two units. This expert opinion therefore considers both units.

According to the results presented in Chapters 4 – 6 of the EIA-REPORT G2/G4 P.3b (2025), no transboundary impacts are expected during normal operation or for the design-basis accident scenarios.

Chapter 6 of the report provides an overview of the three types of design-basis accidents historically used in plant planning, along with the corresponding impact assessment results. However, the parameters applied in the assessment and the underlying assessment methodology are not specified.

The identification of plausible, albeit very rare, cumulative accident scenarios at the Gravelines reactors, which were not considered in the original plant design, led to the development of supplementary safety measures and more than 30 additional improvements in the plant operation. Although a severe accident involving core melt is an extremely unlikely scenario requiring the simultaneous failure of multiple protection and control systems, it cannot be excluded. Given its potential for transboundary consequences, this scenario is included in the EIA documents. The assessment of the potential dispersion of radioactive material within a radius of up to 1,000 km for the core-melt scenario is provided in the report.

Thus, the fourth periodic safety review includes also three beyond design-basis accidents:

1. Loss of shutdown cooling,
2. Loss of fuel element storage pool cooling, and
3. Loss of off-site power (station blackout).

Probability of these events is given as approximately 1 in 5 000 000 years of operation. No further description of accidents which would possibly affect other countries in the EU nor accidents progression analyses are provided.

Main measures to mitigate radiological consequences following accidents without core melt (design-basis accidents), and beyond design-basis accidents that were implemented during the plant construction and complemented by additional measures implemented as a result of improvements in plant's safety were described in the previous chapters.

The EIA documents present the results of calculations demonstrating the potential impacts on public health in terms of projected doses assuming no protective measures are implemented. Although the report states that the assessment of radiological consequences is based on an 'acceptably pessimistic' estimate of releases and on 'realistic scenarios' that do not incorporate protective measures, it does not define the criteria for an acceptably pessimistic assessment nor provide a description or justification of the scenarios considered realistic.

For the three categories of design-basis accidents, only the results for the nearest settlements are reported. For events classified as Category 4 – additional accidents, which in practice correspond to beyond design-basis accidents, transboundary impacts are also assessed for distances of up to 1,000 km, including the territory of Austria.

The EIA documents also refer to results of activity concentrations in food, stating that contamination of food for human consumption at distances greater than 5 km does not exceed limits for placing the food on the market already after 7 days; after one year, this distance is reported to be less than 1 km. However, the EIA documents do not present any additional results of the food contamination assessment, nor do they provide calculated activity concentrations in specific food items to substantiate these statements.

The radiological impact of accidents, whether design-basis or beyond design-basis, on the environment in terms of ground deposition is not provided in the EIA documents.

7.2 Discussion

Generic aspects

For beyond-design basis events, the EIA documents consider several scenarios, including loss of shutdown cooling, an incident involving the fuel element storage pool, and a station blackout. Although the assessment states that parameters leading to increased radioactive releases were used to ensure conservative, 'worst-case' outcomes, the underlying source term data are not provided. No radionuclide inventories, release fractions, or other essential parameters are included, and the document does not contain sufficient information to reproduce or independently verify the calculations. Similarly, the EIA documents provide no details on the atmospheric dispersion model used to estimate off-site consequences. The report indicates that mitigation measures intended to reduce the consequences of design-basis accidents were taken into account; however, it does not describe the assessment methodology needed to substantiate this claim or allow replication of the results.

Results for design-basis accidents indicate that projected population exposures at the nearest inhabited areas remain below French regulatory reference levels. The assessment recognizes that only core-melt accidents have the potential to cause cross-border radiological impacts. The EIA documents evaluate the long-range transport of radioactive material within a 1,000-km radius under "worst-case" conditions, a distance that includes Austrian territory. Reported results expressed as effective dose for different age groups suggest that the lifetime dose to the Austrian population would not exceed 1 mSv (0.03–0.04 mSv).

The EIA documents state that long-distance atmospheric dispersion calculations used transfer coefficients derived from five years of meteorological data, accounting for topography, weather conditions (mainly wind), and deposition processes. It remains unclear whether simulations were performed continuously using daily meteorological input over five years, or whether only a limited number of calculations using average transport coefficients were conducted. Further, the assessment lacks information on the actual dispersion model or calculation method used.

The EIA documents also claim that in case of a beyond design-basis accident with core melt EU maximum levels of radionuclides in food would not be exceeded, but it does not present the methodology for calculation of the activity concentration in food, nor the calculation results to confirm this claim.

EIA documents do not contain information on levels of ground deposition or contamination. Austria has set level for ground deposition of Cs-137 which is 650 Bq/m^2. Values of ground deposition above this value will trigger the screening of agricultural protective measures according to the catalogue of measures (BMK 2022). While doses to population might be below reference levels, ground deposition of Cs-137 above 650 Bq/m^2 could have serious non-radiological consequences, such as psychological and economic consequences in the affected areas.

Site-specific aspects

As the EIA documents did not provide sufficient data to reproduce calculations of which results are presented and in order to assess whether, under specific circumstances, the limit value for the protective measures in Austria could be exceeded, the expert team conducted related dispersion modelling for large-scale release following two hypothetical accidents scenarios for Gravelines NPP. The aim of the assessment was to assess whether a severe accident at Gravelines could possibly cause a deposition on Austrian territory above 650 Bq/m^2, a value that triggers protective actions related to prevention of food contamination. Probability of a large-scale release was not assessed nor considered in this study on atmospheric dispersion following a severe accident.

The source terms, marked as release categories FK2 and FK3, used in the JRODOS dispersion modelling to assess the deposition on Austrian territory are referenced in publication "*Übersicht über Maßnahmen zur Verringerung der Strahlenexposition nach Ereignissen mit nicht unerheblichen radiologischen Auswirkungen (Maßnahmenkatalog)*", 2010, Table 7.2-7 (SSK 2010). The source terms for both release scenarios, expressed as cumulative release fractions, are derived from a reference core inventory representative of a 1000 MWe-class PWR. For application to Gravelines, the reference source term is scaled to reflect the characteristics of the French 900-MWe series reactors. This scaling ensures that the assumed radionuclide inventory is consistent with the actual core power and isotopic inventory of the Gravelines.

The release category FK2 considers an accident at PWR resulting in core-melt with large containment release happening one hour after the reactor shutdown. The release category FK3 considers an accident at PWR resulting in core-melt with medium containment release happening two hours after the reactor shutdown. In both scenarios, release lasts for 3 hours. Activities expressed as fractions of the core inventory for both release categories are shown in Table 3.

Table 3: Cumulative release rates, based on the core inventory according to the German Risk Study Phase A (adapted from (SSK 2010))

		Release category	
		FK2	**FK3**
Start (h)		1	2
Duration (h)		3	3
Release height (m)		10	10
Thermal energy (GJ/h)		15	1
Released fraction of the core inventory	**Kr-Xe**	1,0	1,0
	I	$7{,}0 \cdot 10^{-3}$	$7{,}0 \cdot 10^{-3}$
	I2-Br	$4{,}0 \cdot 10^{-1}$	$1{,}5 \cdot 10^{-2}$
	Cs-Rb	$2{,}9 \cdot 10^{-1}$	$4{,}4 \cdot 10^{-2}$
	Te-Sb	$1{,}9 \cdot 10^{-1}$	$4{,}0 \cdot 10^{-2}$
	Ba-Sr	$3{,}2 \cdot 10^{-2}$	$4{,}9 \cdot 10^{-3}$
	Ru1)	$1{,}7 \cdot 10^{-2}$	$3{,}3 \cdot 10^{-3}$
	La2)	$2{,}6 \cdot 10^{-3}$	$5{,}2 \cdot 10^{-4}$

1) "Ru" also applies to Rh, Co, Mo, Tc

2) "La" also applies to Y, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm

Ideally, atmospheric dispersion modelling for a specific type of accident with a release would be done with daily meteorological data for at least one year to understand transport and deposition of a radioactive plume in all meteorological conditions. As the goal of modelling in this study was only to confirm whether a deposition of Cs-137 above 650 Bq/m^2 from an accident in Gravelines would be possible, a historical weather data that could support dispersion of the radioactive plume to Austria was used for the analysis.

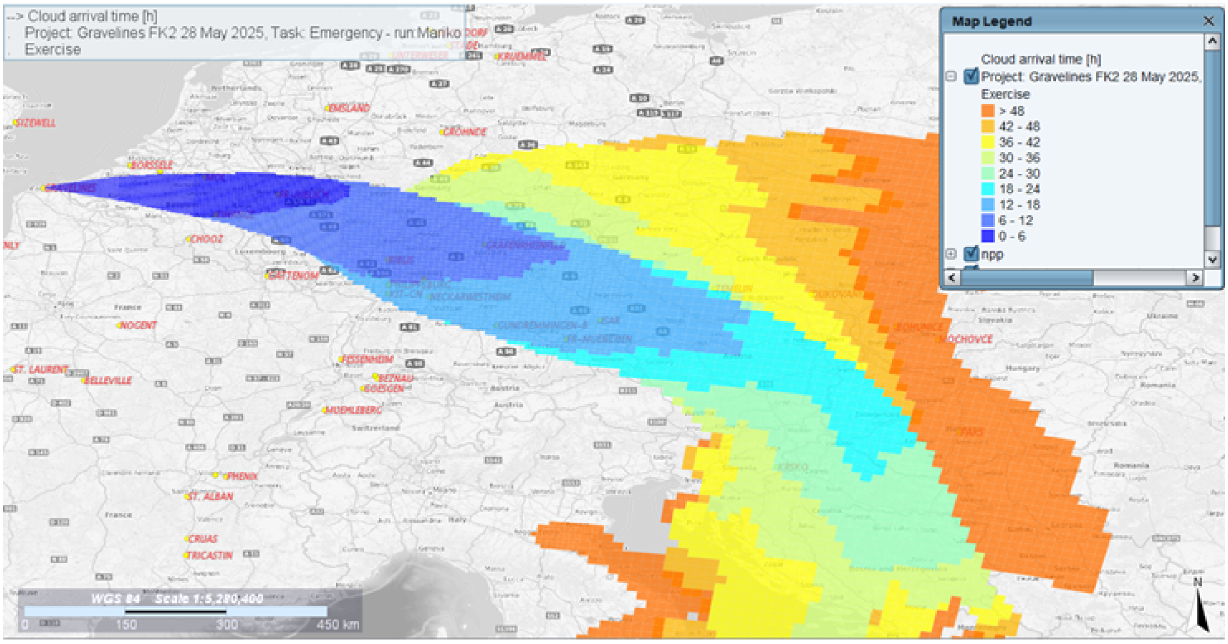
Presented here are the results of the calculations for one of the plant's units which confirm the possibility of ground contamination in Austria from a release in Gravelines. Both release scenarios were modelled assuming the same start time, and consequently, the same meteorological conditions.

Location:	Gravelines, France
Release start:	28 May 2025, 07:00 UTC
Prognosis duration:	72 hours

Figure 5 presents information on cloud arrival time, indicating when the radioactive cloud is expected to reach the affected country. In both scenarios, with the release assumed to start on 28 May 2025 at 07:00 UTC, the cloud is projected to reach Austrian territory in approximately 20 hours. Meteorological conditions are the dominant factor influencing cloud arrival time, and this result may vary significantly under different weather conditions.

Figure 5: Cloud arrival time for the release category FK2

Schematic diagram of the cooling circuit of the spent fuel pool



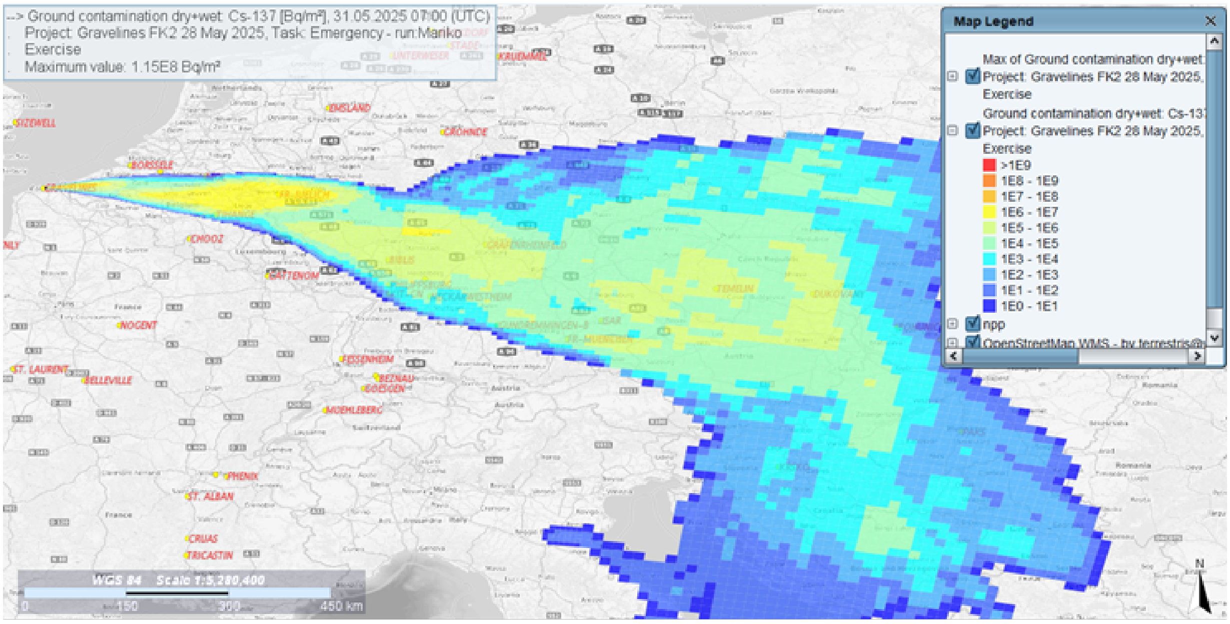
Source: Environment Agency Austria

umweltbundesamt

Deposition of the radioactive material released in an accident depends on a number of factors: characteristics of a release, meteorological conditions, deposition surface and others. For this task, meteorological conditions for the period 28 – 31 May 2025, which led to transport of a radioactive plume over Austrian territory, were chosen.

Figure 6: Ground contamination with Cs-137 for the release category FK2

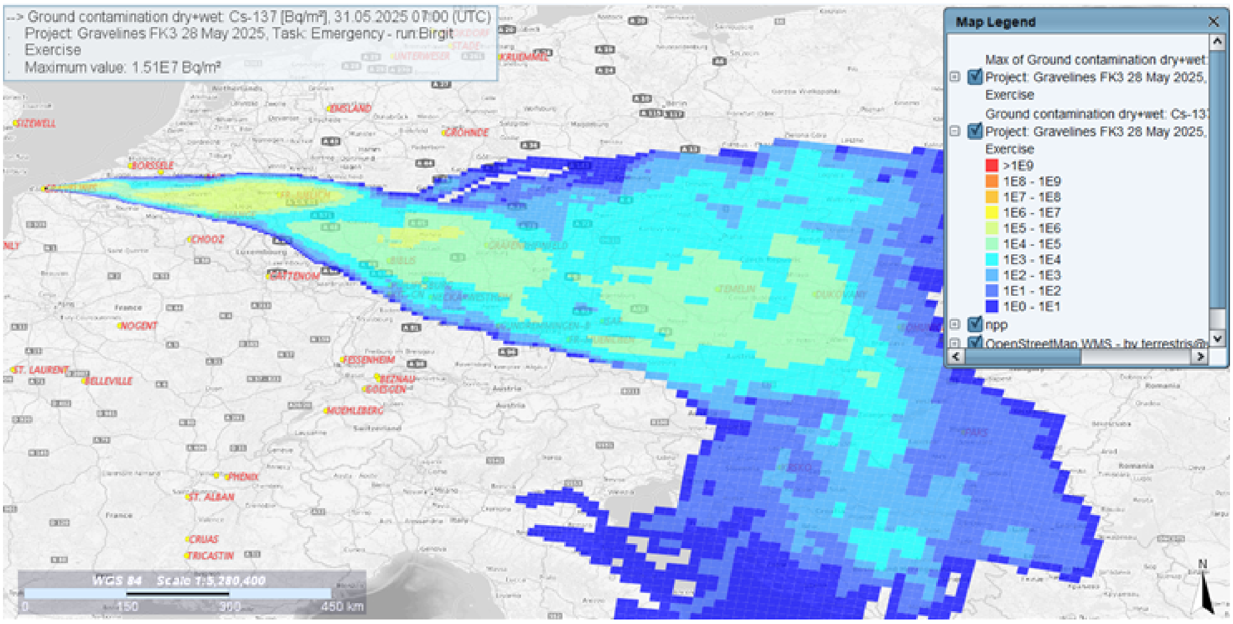
Schematic diagram of the cooling circuit of the spent fuel pool



Source: Environment Agency Austria umweltbundesamt

Figure 7: Ground contamination with Cs-137 for the release category FK3

Schematic diagram of the cooling circuit of the spent fuel pool



Source: Environment Agency Austria umweltbundesamt

Results of the JRODOS calculation for both release categories, FK2 and FK3 presented in Figure 6 and Figure 7, show that there is a possibility of contamination in Austria above 650 Bq/m2 with the maximum calculated value exceeding

1×10^5 Bq/m^2 for the release category FK2 and 1×10^4 Bq/m^2 for the release category FK3. The probability of such contamination was not assessed in this study.

7.3 Conclusions

The EIA documents address events and accident sequences corresponding to three categories of design-basis accidents, as well as an additional category representing beyond design-basis events, including core-melt and fuel element storage pool scenarios.

The analysis of radiological consequences presented in the report lacks sufficient technical detail. Essential information required for independent verification, such as radionuclide inventories, source-term assumptions, release fractions, and the methodology for dispersion modelling, is not provided. Consequently, both the transparency and reproducibility of the radiological impact assessment are limited.

The EIA documents indicate that, for design-basis accidents, the radiological consequences are expected to remain below national reference levels and do not give rise to transboundary risks. For beyond design-basis accidents, including scenarios involving core melt, the report acknowledges the potential for long-range impacts, but lacks sufficient technical detail to allow independent verification of these findings. The report does not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances greater than 5 km after 7 days and within 1 km after one year. Additionally, the assessment omits information on ground deposition, despite its significance for evaluating long-term radiological impacts and potential contamination of the food chain.

Modelling of atmospheric dispersion and deposition conducted by the expert team demonstrate that, under certain meteorological conditions, a severe accident at Gravelines 2 and 4 could lead to ground deposition of Cs-137 in Austria above the national screening threshold of 650 Bq/m^2. Although the study does not assess the probability of such conditions, the results indicate that transboundary impacts greater than those implied in the EIA cannot be excluded.

Overall, the EIA provides an assessment of radiological consequences without providing complete information on assessment methodology and underlying data to support the claims, particularly for severe accidents with potential transboundary effects. More detailed source-term information, dispersion modelling inputs, and food-chain contamination assessments would be needed to fully evaluate the potential impact on Austria and to support the claims made in the EIA documents.

- Information on the release parameters is needed for the reconstruction of the results of the assessment provided in the EIA. Where detailed information on core inventory and source terms cannot be disclosed, minimum required information to be requested is on released activities of Cs-137 and iodine for beyond design-basis accidents

- A presentation of the modelling results supporting statements of lifetime dose for transboundary impact (Austria) should be provided.
- A presentation of atmospheric dispersion and ground deposition calculations for key radionuclides, including spatial distribution maps, modelling assumptions, and uncertainty evaluation should be provided.
- Information of the calculations supporting statements on food contamination should be provided.

8 ASSESSMENT OF THE TIME FRAME

8.1 Treatment in the EIA documents

The EIA documents emphasize the goals of the investigation undertaken with the generic PSR of the 900 MWe NPPs, which included Gravelines NPP. Those are covering three areas:

- “risks”, where the plant is assessed against the requirements set by current standards and regulations, but also for opportunities to increase safety levels to those comparable to Generation III reactors, with Flamanville 3 EPR as a reference reactor. The latter includes four distinctive areas: accidents without core damage, accidents with core damage, external impacts, and spent fuel pool issues.
- “disadvantages”, where issues that lead to release that could affect people and the environment are assessed, and
- “ageing management”, where processes to prevent degradation due to aging are assessed, especially for the period beyond 40 years of operation.

The aim of the 4th PSR was to assess the status in relation to these goals, with the objective of identifying specific measures—either technical or administrative (analyses)—that would lead to enhanced safety, to comply with the goals set.

According to a decision by the French regulator ASNR, each plant has a period of five years following the release of the PSR report, to implement all safety measures identified.

For Gravelines 2 (Gravelines 4), the implementation is organised in three phases. The Phase A measures are those that could be implemented during operations or within an outage related to the 4th PSR. Those measures have already been implemented at the time of the release of the EIA document. Next, the measures that will not be implemented in Phase A are scheduled for implementation within Phase B, which is planned to be completed by March 2029 (December 2029). Measures that are not completed within Phase B (or its extension, which is also planned to be completed by April 2029 (December 2029)) are then to be completed within further phases, to be finalised by March 2030 (December 2029). This coincides with the "5 years after the release of the PSR report”, as required by the regulator ASNR.

8.2 Discussion

It is important that the agreed implementation period (5 years) is not extended. Some of the information circulating around seems to suggest uncertainties related to the financial resources needed for the implementation of the safety modifications for the 900 MWe series, including the activities related with the ageing management (LTO). Both a lack of financial resources and even more so supply chain issues including human resources could be a cause of a delay, avoiding any delays and assuring as fast as possible implementation shall remain the priority for EDF.

8.3 Conclusions

The timeframe for completing all measures under the 4th PSR (5 years after the release of the PSR report = 2029/2030) is not uncommon. However, as the period following the 4th PSR corresponds with the start of long-term operation (LTO), some of the specific measures require special attention. It is important that the agreed implementation period is not extended. A lack of financial resources or the known problems with supply chain availability, including human resources, could affect the implementation period. It is particularly noteworthy that important safety modifications listed as part of the 4th PSR were already considered necessary as part of the EU stress test (2012), and their implementation had been agreed upon.

- Maintaining agreed schedule, or when possible, accelerating the safety improvements and LTO measures to be completed, where possible, even before 5 years deadline is strongly recommended.
- EDF should put the priority on the funding for the safety upgrade measures required in the 4th PSR and those related with the LTO, rather than on construction of a series of new EPR-2.
- Additional clarity of how the post Fukushima measures are being integrated with the measures that were decided on the basis of 4th PSR would be appreciated.

9 LIST OF CONCLUSIONS

9.1 Long-term operation and operational experience

- The justification that no checks are to be carried out for Gravelines 4 as part of the Program for Complementary Investigations (PIC) should be provided.
- The evaluation of safety-related incidents over the last five years revealed a high number of safety-related incidents that were classified as INES level 1. In addition to the events regarding deficiencies in earthquake protection, a number of events that compromised safety also occurred in both reactors. The reason for the large number of safety-related events could be a lack of safety culture combined with a large number of age-related events. Also noteworthy were the incidents involving contamination of workers and the blockage of the ultimate heat sink (UHS) by jellyfish.
- In-depth investigations on components relevant for preventing external events to affect the nuclear safety of the plant should be carried out, in particular concerning those components of the original systems that connect the newly installed “hardened safety core” and systems for mitigating the effects of core-melt accidents.
- A complete analysis of the causes of the cracks in the auxiliary line due to stress corrosion cracking should be carried out and taken into account in order to take preventive protective measures against such damage and its effects already within the framework of the 4th PSR.
- The modification of the ageing management for the secondary and primary circuit components to detect unexpected degradation should be considered.
- A systematic ageing control of the components safety relevant concerning the resistance with regard to earthquakes should be considered.

9.2 External hazards

- Information on the methods, data and assumptions used for the PSHA performed to determine the SND for the Gravelines reactors should be provided, in particular, the types of seismic sources considered (source zones and/or fault sources), time coverage of the earthquake catalogue, minimum and maximum magnitudes, ground motion prediction equations, and site conditions.
- Information on the ground motion value corresponding to the occurrence probability of 10^{-4} per year derived from the PSHA which was performed to determine the SND for Gravelines reactors should be provided.

- A comparison of the ground motion values (PGAH, spectral accelerations) of the current deterministically derived design basis earthquake and the corresponding values derived by PSHA should be provided.
- Information on protection requirements of the Gravelines NPP with regard to the intentional crash of a commercial aircraft should be provided.
- The PSHA performed for determining the SND by assessing the validity of methods, data and assumptions used in the PSHA and to benchmark the PSHA with regard to WENRA requirements (WENRA 2021) and recommendations (WENRA 2020 a,b).
- Dedicated assessments of near-regional faults for which it cannot be excluded that they are active should be required, in line with WENRA (2020b). The approach may be similar to the one currently applied by EDF to the site of Cruas NPP including field geology, morphostructural and dating studies, and paleoseismology.
- The deterministically derived SMA and the current seismic design basis of Gravelines reactors with the ground motion values derived from probabilistic seismic hazard assessment (PSHA) for a DBE with the occurrence probability of 10^{-4} per year should be compared.
- Additional safety demonstrations to ensure that all SSCs relevant to safety can cope with a probabilistically derived new Design Basis Earthquake (DBE) in case the probabilistically derived DBE exceeds the ground motion parameters of the current seismic design basis of the plant should be required.
- The methods, data and assumptions used to derive hazard values for all external hazards considered in the EIA documents should be reviewed, in line with WENRA requirements and guidance (WENRA 2020a-d; 2021).
- Design basis events and design basis parameters should be defined for external hazards conform with WENRA (2021) requirements.
- It should be ensured that the use of the Noyau Dur's DEC equipment is not required to protect the facility against design events, i.e., events with recurrence intervals of 10,000 years or less (e.g., earthquakes). This is to ensure the independence of Defence-in-Depth (DiD) levels 3 and 4
- It should be evaluated if the long timeframe for implementing the Noyau Dur at the Gravelines reactors is in line with the requirement of the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). Background: the timeframe for implementing the Noyau Dur at the Gravelines reactors extends up to 2029, i.e., 17 years after ASNR's initial decision to implement Hardened Safety Cores at the French NPP fleet.
- In this context the following questions should be addressed:
  - Is it correct that strong earthquakes with recurrence periods longer than 150,000 years were not considered in the seismic PSA for the Gravelines NPP which, according to the EIA documents, revealed a contribution to the CDF of $8*10^{-7}$ per year? If yes: What would be the CDF if earthquakes with longer recurrence intervals were taken into account as well?

- Have design basis events with exceedance frequencies not higher than 10^{-4} per annum and corresponding design basis loads been defined for all natural hazards considered in the EIA documents (extreme temperatures, river floods, high wind, tornado etc.)?
- What are the main reasons for the excessively long timeframe (up to 2029) for implementing the Noyau Dur at the Gravelines reactors?
- Have any studies been or will be carried out on the threat posed by newer technologies, in particular potential attacks using civilian or military drones?
- How is the result of the Nuclear Security Index 2023 for France assessed? Are improvements planned with regard to "security culture", "cybersecurity" and "protection against insider threats"?

9.3 Safety aspect of accident without core melt and spent fuel pool

Enhance Transparency and Provide Clarity on Key Quantitative Data

- Quantitative Data: The reports should provide the initial and final mass flow rates for the GCT-a Valve Uprate (PNPE1141), along with a comparison to the nominal operational flow. This is necessary to quantify the safety benefit.
- Adverse Effects Analysis: The analysis of the uprated GCT-a capacity should be expanded to quantify the risk of increased radioactive release during a Containment Bypass scenario like a Steam Generator Tube Rupture. This ensures that the modification does not introduce new, unacceptable risks.
- Radiological Implementation: Detailed methodology on how the Reduced Primary System I-131 Limit will be implemented and monitored should be provided, explicitly addressing how iodine spiking will be accounted for in operational procedures and design basis analyses.

Establish Firm and Accountable Timelines

- Missing Deadlines: EDF and the ASNR should establish a firm, committed timeline for the completion of the SFP Flame Trap Installation (Pool-1). The absence of a fixed date creates an unquantified safety risk.
- Study Status and Next Steps: For the Critical Heat Flux (CHF) experimental program (Requirement [Study-B]), EDF should immediately provide an updated status on its completion and publicly commit to the defined work program and schedule for incorporating the findings, as the reporting deadline was December 31, 2024.

Clarify Status Reporting and Implementation Rationale

- Justify Deferral: A comprehensive safety justification for deferring beneficial state-of-the-art measures like the RIS Accumulator Pressure Increase to Phase B of the implementation cycle should be provided. This justification should explicitly weigh the cost/complexity against the temporary safety margin reduction.
- Resolve Discrepancies: The conflicting status between the different report parts should be clarified. Future reporting should clearly define the criteria for "implemented" (design complete vs. installation complete) should be clarified to prevent ambiguity, the same is true for planned measures. The reports do support a distinction between Phase A and Phase B, but it is often difficult to follow if the mentioned dates deal with the design or the installation.

9.4 Safety aspects of core melt accidents

- The EIA documents should include an overview of which of the planned measures are to be used to meet the ASNR requirements published at the end of the generic phase of the 4th PSR and when they are to be implemented.
- Studies that prove the sufficient thickness of the containment basements and the dimension of the spreading areas for Gravelines 2 and 4 should be provided.
- It should be explained which options were examined to limit the spread of radioactive substances via soil and groundwater after a core melt accident in accordance with regulation [AG-D-III] How is it justified that there is no need for additional measures with regard to safety risks?
- A systematic comparison between the safety level of the 900 MWe reactors and modern safety standards of the EPR Flamanville 3 should be included in order to identify the gaps.
- Information about the core damage frequency (CDF) and the large (early) release frequency L(E)RF before the 4th PSR, after implementation of all modification of 4th PSR and after the end of Phase A of the 4th PSR should be provided.
- Information why the necessary “replacement of valves or valve seals on the EAS ND” (PNPE1471) has been performed at Gravelines 4 but not at Gravelines 2 should be provided.
- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Gravelines 2 and 4. The concept of practical elimination should be used for this approach. Especially since the goal of the 4th PSR is to move closer to the safety level of the EPR Flamanville 3.

- The authorization for continued operation of Gravelines 2 and 4 should be issued only after the planned measures to mitigate the release in the event of a core-melt accident have been fully implemented.

9.5 Radiological impact of accidents / Transboundary Effects

- Information on the release parameters is needed for the reconstruction of the results of the assessment provided in the EIA. Where detailed information on core inventory and source terms cannot be disclosed, minimum required information to be requested is on released activities of Cs-137 and iodine for beyond design-basis accidents
- A presentation of the modelling results supporting statements of lifetime dose for transboundary impact (Austria) should be provided.
- A presentation of atmospheric dispersion and ground deposition calculations for key radionuclides, including spatial distribution maps, modelling assumptions, and uncertainty evaluation should be provided.
- Information of the calculations supporting statements on food contamination should be provided.

9.6 Assessment of the time frame

- Maintaining agreed schedule, or when possible, accelerating the safety improvements and LTO measures to be completed, where possible, even before 5 years deadline is strongly recommended.
- EDF should put the priority on the funding for the safety upgrade measures required in the 4th PSR and those related with the LTO, rather than on construction of a series of new EPR-2.
- Additional clarity of how the post Fukushima measures are being integrated with the measures that were decided on the basis of 4th PSR would be appreciated.

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12 GLOSSARY

ASG .... Steam Generator Auxiliary Feedwater System

ASN .... French Authority for Nuclear Safety

ASNR .... French Authority for Nuclear Safety and Radiation Protection

BAN .... Buildings for Nuclear Auxiliary Facilities

BK .... Spent Fuel Building

Bq .... Becquerel

CDF .... Core Damage Frequency

CHF .... Critical Heat Flux

Cs-137 .... Caesium-137

DBA .... Design Basis Accidents

DBE .... Design Basis Earthquake

DEG .... System for generating and distributing cold water

DEC .... Design Extension Conditions

DID .... Defence-in-Depth

DVN .... Ventilation and air conditioning system

EAS .... Sprinkler System

EAS-ND .... “Hard Core” Sprinkler System

EDF .... Électricité de France

EDG .... Emergency Diesel Generators

EIA .... Environmental Impact Assessment

EIPS .... Emergency Intervention Systems ()

ENSREG .... European Nuclear Safety Regulators Group

EPR .... European Pressurized Reactors

EU .... European Union

FK .... Release Category

FLA3 .... Flamanville Unit 3

GCTa .... Main turbine bypass system with venting to the atmosphere

GPR .... Permanent Group of Experts on Reactors

GW...................................... Giga Watt hour

HCTINS .... High Committee for Transparency and Information on Nuclear Safety

I-131 .... Iodine-131

IAEA .... International Atomic Energy Agency

INES .... International Nuclear and Radiological Event Scale

IPCC .... Intergovernmental Panel on Climate Change

IRSN .... Institut de Radioprotection et de Sûreté Nucléaire,

LHP/LHQ .... Emergency power supply with 6.6 kV AC

LOCA .... Loss of Coolant Accident

LTO .... Long-Term Operation

mSv.................................... Millie-Sievert

MW .... Mega Watt

NPP.................................... Nuclear Power Plant

PGA.................................... Peak Ground Acceleration

PSR .... Periodic Safety Review

PSHA .... Probabilistic Safety Hazard Assessment

PSA .... Probabilistic Safety Assessment

PTR .... Tank of Water Treatment and Cooling System of Pools

PTR bis .... Mobile auxiliary cooling system for fuel element pools

PWR................................... Pressurized Water Reactor

REA .... Boron and Water Storage Tank

RIA .... Reactivity Initiating Accidents

RIC .... Reactor Core Instrumentation

RCP .... Reactor Coolant Pump

RCS .... Reactor Cooling System

RFS .... Règle Fondamentale de Sûreté

RPN.................................... Reactor Protection System

RPV .... Reactor Pressure Vessel

SA .... Severe Accidents

SBO...................................... Station Black Out

SFP....................................... Spent Fuel Pool

SF-ND .... Hard-Core Cooling Source

SG .... Steam Generator

SGTR.................................... Steam generator tube ruptures

SIS .... Safety Injection Systems

SMHV................................... Maximal plausible historical earthquake (Séisme Majoré Historiquement Vraisemblable)

SMS .... Safe Shutdown Earthquake, Maximum safety earthquake, equivalent to design basis earthquake (Séisme Majoré de Sécurité)

SND .... Séisme Noyau Dur – Seismic level for the hardened safety core

SOTA.................................... State of the Art

SSCs..................................... Structures, Systems and Components

TBq .... Tera-Becquerel, E12 Bq

TLD .... Température Longue Durée

TE.. Température Exceptionnelle

TTS....................................... Target Technical Specifications

UHS .... Ultimate Heat Sink

WENRA................................. Western European Nuclear Regulators´ Association

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ISBN 978-3-99004-861-0