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NPP DAMPIERRE 3&4 LTO ENVIRONMENTAL IMPACT ASSESSMENT

Expert Statement

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and Environmental Protection,
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SUMMARY

The Dampierre-en-Burly (here: Dampierre) nuclear power plant (NPP) consists of four operating pressurized water reactors with a capacity of about 900 MWe each. These reactors were commissioned 1980 and 1981 respectively.

France notified the 4th Periodic Safety Review ("Public consultation procedure on the 4th safety review report") of the Dampierre NPP (reactor 3 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 Loiret. 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 NPPs 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 a 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 NPPs, 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 French Authority for Nuclear Safety and Radiation Protection (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 has been completed, 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 lifetime 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. It should be noted that the EIA documents referred to the outdated IAEA Safety Guide No. NS-G-2.12 (IAEA 2009).

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 NPP 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.

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. Without justification, it is stated that no checks are to be carried out for Dampierre 3&4 as part of the supplementary investigation program.

In the framework of the generic phase of the 5th PSR of the 900 MWe reactors, the ASNR required 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 Dampierre 3&4.

A recently reported safety-related incident involving faulty anchors for earthquake protection raises serious questions about the conformity tests carried out to date. These safety-related defects have existed since the plant was commissioned without being detected during previous inspections.

Overall, a number of safety-related incidents classified as level 1 on the International Nuclear and Radiological Event Scale (INES) have occurred over the past five years, many of which were related to non-compliance with the general operating regulations (RGE). These incidents were preceded by component failures, maintenance errors, or operational errors (in some cases, multiple errors). The cause could be a lack of safety culture combined with a large number of age-related incidents. It is particularly noteworthy that the investigations revealed that many of the shortcomings have been present for some time.

External hazards

The EIA documents provide information on hazard types considered in the safety demonstration for the Dampierre 3 and 4 and measures already implemented or decided to be implemented in order to strengthen the robustness of the reactor with respect to external hazards. For most external hazards, the methods, data and assumptions used in the hazard assessment are not specified in detail. Conformity with requirements and guidance of the Western European Nuclear Regulators Association (WENRA) therefore cannot be assessed. Non-conformity with WENRA Reference Levels is observed for earthquake and seismic ground shaking. The Design Basis Earthquake (DBE) for Dampierre, termed SMS in French regulation, is based on deterministic analysis which is no longer state of the art. Available documents indicate an SMS ground motion value (PGA) for Dampierre of 0.2 g (nuclear island) and 0.1 g (site). It remains to be demonstrated that the current SMS fulfills the WENRA requirements of a DBE with an average recurrence interval of 10,000 years (WENRA 2021).

The Dampierre site is located near the late to post-Pliocene Loire fault system. The respective faults have not been properly investigated to decide about their activity or inactivity although they have known for more than 10 years at least. According to the EIA report, EDF is envisaging a geophysical and paleoseismological investigation program to evaluate the safety significance of these faults. A tangible time schedule is not communicated. An updated PSHA-based reassessment of seismic hazards that takes advantage of fault data is not announced. For potentially active faults in the near-region of an NPP site WENRA (2020b) suggests systematic fault mapping and collecting paleoseismological information. At the background of the existing literature (JOMARD et al. 2017; RITZ et al. 2021 and references therein) it is remarkable that a process for active fault characterization has not been implemented earlier in the course of the 4th PSR. It is suggested to ASNR to require (i) dedicated paleoseismological assessments of these faults and (ii) an up-to-date Probabilistic Safety Hazard Assessment (PSHA) in line with WENRA requirements considering paleoseismological data.

With respect to safety upgrades of Dampierre 3 and 4, it is evident that one of the most important measures to provide protection against external hazards is the implementation of the Hardened Safety Core (ND). Full implementation of the ND is still pending and should be completed by 2029 and 2030, respectively. The decision to implement the ND has been made in 2012. The fact that the implementation of the ND will be completed only 18 years thereafter appears remarkable at the background that WENRA requires the "timely implementation of the reasonably practicable safety improvements identified".

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 Dampierre 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 Dampierre 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 France points to shortcomings in the 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 strengthen defence-in-depth at Dampierre 3 and 4. For long-term heat removal, the Auxiliary Feedwater System (ASG) water inventory has been secured by diversifying the connection of the ASG tank from the fire-protection network (PNPP1864), supporting accident sequences where additional feedwater is required. For thermal-hydraulic control, the increase of the GCT-a regulating-valve capacity (PNPE1141) addresses the ASG consumption calculation anomaly and supports updated dimensioning accident studies. In parallel, a lower permissible primary-circuit I-131 activity has been adopted to limit radiological consequences in accidents without fuel-rod failure.

Regarding the spent-fuel pool, integrity and cooling robustness are reinforced by the addition of a diversified, mobile cooling path (PTR bis), providing a resilient means to restore cooling and aligning with Hardened Safety Core principles. PTR bis is a FARN-deployed (nuclear rapid action force), mobile system connected to fixed pool tie-ins, used to stop boiling and support the long-term return to cooling.

Accidental draining is mitigated by automatic isolation of the PTR suction at very-low pool level (PNPP1402) with duplicated suction-line isolation (PNPE1344); strengthened and motorized transfer-tube closure (PNRL1895 and PNPP1403); automation of BR pool drain/filtration valves (PNPP1780); and a resized discharge-line siphon break (PNPP1289).

Fire propagation risk between PTR trains is proposed to be mitigated by physical separation and protection measures via the fire-protection screen between the two PTR pumps (PNPP1949).

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. Nevertheless, a number of shortcomings were identified in the EU stress tests as early as 15 years ago, and not all of them have been remedied to date. According to the ASNR, the objective of the fourth PSR 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 Dampierre 3 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.

Not all necessary and planned modifications for heat removal without using the filtered pressure relief system in the event of a core melt accident have been fully implemented yet. Important components such as valves, seals and electric equipment that are required during a core melt accident, but whose resistance cannot be guaranteed during such an accident, will also only be replaced during Phase B or the Supplementary Phase. 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 Dampierre 3 or 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 for unit 3 and 2030 for unit 4 respectively). 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 spent fuel 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, the transparency and reproducibility of the radiological impact assessment are extremely 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, specifically for 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 EIA documents do not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances of more 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 unit 3 or unit 4 of Dampierre NPP 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 beyond those described 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 which equals to June 2029 for Dampierre 3 and April 2030 for Dampierre 4) 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.

Management of spent fuel and radioactive waste

Spent fuel and radioactive waste can cause negative impacts for people and the environment. Proof of safe disposal is required to prevent this. However, the EIA documents did not provide sufficient evidence.

During the EIA, sufficient information on management of spent fuel and radioactive waste should be provided. It would be appreciated if the missing information would be added to the EIA documents.

The update of the French national plan for the management of radioactive material and waste (PNGMDR) should undergo a Strategic Environmental Assessment, also transboundary.

ZUSAMMENFASSUNG

Das Kernkraftwerk (KKW) Dampierre-en-Burly (hier: Dampierre) besteht aus vier in Betrieb befindlichen Druckwasserreaktoren mit einer Leistung von jeweils 900 MWe. Diese Reaktoren wurden 1980 bzw. 1981 in Betrieb genommen.

Frankreich hat die vierte Periodische Sicherheitsüberprüfung („Öffentliches Konsultationsverfahren zum vierten Bericht der Sicherheitsüberprüfung") des KKW Dampierre (Reaktor 3 und 4) notifiziert, die als Laufzeitverlängerung gemäß der UNECE-Espoo-Konvention über die Umweltverträglichkeitsprüfung (UVP) im grenzüberschreitenden Rahmen zu betrachten ist. Zuständige Behörde ist das französische Departement Loiret. Antragsteller 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 KKW ist zeitlich nicht begrenzt. Alle zehn Jahre werden französische KKW jedoch einer Periodischen Sicherheitsüberprüfung (PSÜ) unterzogen. Die 4. 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-Reaktoren 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 französischen Behörde für nukleare Sicherheit und Strahlenschutz (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 folgen über einen Zeitraum von etwa zehn Jahren (von 2019 bis 2031) Inspektionen aller 32 Reaktoren der 900-MWe-Reaktoren.

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.

Langfristiger Betrieb 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 umgesetzt wurde. Darauf deuten auch die Ergebnisse der ersten Topical Peer Review (TPR) gemäß Artikel 8e der Richtlinie 2014/87/EURATOM hin. 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. Es sei darauf hingewiesen, dass in den UVP-Unterlagen auf den veralteten IAEO-Sicherheitsleitfaden Nr. NS-G-2.12 (IAEO 2009) Bezug genommen wurde.

Da die meisten SSCs ursprünglich für eine nominelle Betriebsdauer von 40 Jahren ausgelegt wurden, kann die 4. PSÜ als die erforderliche Genehmigung für den Betrieb des Kernkraftwerks über dessen ursprüngliche Auslegungsdauer hinaus angesehen werden. Daher erfordert die 4. PSÜ eine detailliertere Betrachtung des Alterungsmanagements. Aus den Unterlagen zur Umweltverträglichkeitsprüfung 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 erfolgen.

Die Umsetzung des Programms für ergänzende Untersuchungen (PIC) ist ein Ansatz, der darauf abzielt, das Nichtvorhandensein von Betriebsstörungen in Bereichen zu bestätigen, die nicht regelmäßig überprüft werden. Ohne Begründung wird angegeben, dass im Rahmen des ergänzenden Untersuchungsprogramms keine Überprüfungen für Dampierre 3 und 4 durchgeführt werden sollen.

Im Rahmen der generischen Phase der 5. PSÜ der 900-MWe-Reaktoren forderte die ASNR EDF auf, bis zum 31. Dezember 2025 die Strategie festzulegen, um die Erkenntnisse aus der Entdeckung von Spannungsrisskorrosion und, allgemeiner, das Risiko einer unerwarteten Degradation von Komponenten im Primär- und Hauptsekundärkreislauf durch Kontrollen im Rahmen der zusätzlichen Inspektions- und Wartungsprogramme zu berücksichtigen. Die Ursache der Risse, die interkristalline

Spannungskorrosion, ist ein bekanntes Korrosionsphänomen, das jedoch in den betreffenden Bereichen nicht erwartet wurde und daher die Rohre auch nicht darauf geprüft wurden. Dies stellt das Alterungsmanagementkonzept für Komponenten im Primär- und Hauptsekundärkreislauf in Frage.

Der Vorschlag der ASNR während der generischen Phase der 5. PSÜ, das Alterungsmanagement über die 4. PSÜ hinaus auszuweiten, 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“ im Rahmen der 4. PSÜ ist für Dampierre 3 und 4 jedoch sehr begrenzt.

Ein kürzlich gemeldeter sicherheitsrelevanter Vorfall im Zusammenhang mit fehlerhaften Verankerungen für den Erdbebenschutz wirft ernsthafte Fragen hinsichtlich der bisher durchgeführten Konformitätsprüfungen auf. Diese sicherheitsrelevanten Mängel bestehen bereits seit der Inbetriebnahme der Anlage, ohne dass sie bei früheren Inspektionen entdeckt wurden.

Insgesamt kam es in den letzten fünf Jahren zu einer Reihe von sicherheitsrelevanten Vorfällen, die auf der Internationalen Bewertungsskala für nukleare und radiologische Ereignisse (INES) als Level 1 eingestuft wurden; viele davon standen im Zusammenhang mit Verstößen gegen die Allgemeinen Betriebsvorschriften (RGE). Diesen Vorfällen gingen Komponentenausfälle, Wartungs- oder Bedienungsfehler (in einigen Fällen mehrere Fehler) voraus. Die Ursache könnte in einer mangelnden Sicherheitskultur in Verbindung mit einer hohen Zahl altersbedingter Vorfälle liegen. Besonders bemerkenswert ist, dass die Untersuchungen gezeigt haben, dass viele der Mängel bereits seit einiger Zeit bestanden haben.

Externe Gefahren

Die UVP-Unterlagen enthalten Informationen zu den in den Sicherheitsnachweisen für Dampierre 3 und 4 berücksichtigten Gefahrenarten sowie zu den bereits umgesetzten oder beschlossenen Maßnahmen zur Erhöhung der Robustheit des Reaktors gegenüber externen Gefahren. Für die meisten externen Gefahren werden die bei der Gefahrenbewertung verwendeten Methoden, Daten und Annahmen nicht im Detail angegeben. Die Übereinstimmung mit den Anforderungen und Leitlinien der Vereinigung westeuropäischer Atomaufsichtsbehörden (Western European Nuclear Regulators' Association – WENRA) kann daher nicht beurteilt werden. Bei Erdbeben und seismischen Bodenbewegungen wird eine Abweichung von den WENRA-Referenzleveln festgestellt. Das Auslegungserdbeben (DBE) für Dampierre, in den französischen Vorschriften als SMS bezeichnet, basiert auf einer deterministischen Analyse, was nicht mehr dem aktuellen Stand der Technik entspricht. Verfügbare Dokumente weisen einen SMS-Bodenbeschleunigungswert (PGA) für Dampierre von 0,2 g (nuklearer Teil) und 0,1 g (nicht-nuklearer Teil) aus. Es bleibt noch zu zeigen, dass das derzeitige SMS die WENRA-Anforderungen an ein DBE mit einem durchschnittlichen Wiederholungsintervall von 10.000 Jahren erfüllt (WENRA 2021).

Der Standort Dampierre liegt in der Nähe des Loire-Verwerfungssystems aus dem späten bis post-pliozänen Zeitalter. Die jeweiligen Verwerfungen wurden bislang nicht ausreichend untersucht, um über ihre Aktivität oder Inaktivität zu entscheiden, obwohl sie seit mindestens zehn Jahren bekannt sind. Laut dem UVP-Bericht plant EDF ein geophysikalisches und paläoseismologisches Untersuchungsprogramm, um die sicherheitstechnische Bedeutung dieser Verwerfungen zu bewerten. Ein konkreter Zeitplan wurde nicht bekannt gegeben. Eine aktualisierte, auf der PSHA basierende Neubewertung der seismischen Gefahren unter Einbeziehung von Daten zu den Verwerfungen wurde nicht angekündigt. Für potenziell aktive Verwerfungen in der unmittelbaren Umgebung eines Kernkraftwerkstandorts empfiehlt die WENRA (2020b) eine systematische Kartierung der Verwerfungen und die Erhebung paläoseismologischer Informationen. Vor dem Hintergrund der vorhandenen Literatur (Jomard et al. 2017; Ritz et al. 2021 und die darin genannten Quellen) ist es bemerkenswert, dass im Rahmen des 4. PSR bisher kein Verfahren zur Charakterisierung aktiver Verwerfungen umgesetzt wurde. Der ASNR wird empfohlen, (i) spezielle paläoseismologische Bewertungen dieser Verwerfungen sowie (ii) eine aktuelle probabilistische Sicherheitsrisikobewertung (PSHA) gemäß den WENRA-Anforderungen unter Berücksichtigung paläoseismologischer Daten zu verlangen.

Im Hinblick auf die Sicherheitsnachrüstungen der Reaktoren Dampierre 3 und 4 ist offensichtlich, dass eine der wichtigsten Maßnahmen zum Schutz vor externen Gefahren die Umsetzung des „Hardened Safety Core“ (ND) ist, die noch aussteht und erst bis 2029 bzw. 2030 abgeschlossen sein soll. Die Entscheidung zur Umsetzung des ND wurde 2012 getroffen. Die Tatsache, dass die Umsetzung des ND erst 18 Jahre später abgeschlossen sein wird, erscheint bemerkenswert vor dem Hintergrund, dass die WENRA die „rechtzeitige Umsetzung der identifizierten, vernünftigerweise durchführbaren Sicherheitsverbesserungen“ fordert.

Terroranschläge und Sabotageakte können erhebliche Auswirkungen auf kerntechnische Anlagen haben und schwere Unfälle verursachen. Dennoch werden sie in den eingereichten UVP-Unterlagen nur sehr allgemein erwähnt. Vergleichbare UVP-Berichte haben solche Ereignisse bis zu einem gewissen Grad behandelt. Auch wenn die Maßnahmen gegen Sabotage und Terroranschläge aus Gründen der Vertraulichkeit nicht im Detail erörtert 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 das Reaktorgebäude als auch das Gebäude für abgebrannte Brennelemente des KKW Dampierre 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 KKW Dampierre 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 Analyse stützt sich sowohl auf deterministische Sicherheitsanalysen (DSA) als auch auf probabilistische Sicherheitsanalysen (PSA), um Betriebstransienten, Auslegungsstörfälle (DBA) und erweiterte Auslegungsbedingungen (DEC) neu zu bewerten.

Erhebliche Sicherheitsverbesserungen wurden bereits umgesetzt oder sind geplant, um die radiologischen Folgen zu verringern und das gestaffelte Sicherheitskonzept in Dampierre 3 und 4 zu verbessern. Für die langfristige Wärmeabfuhr wurde der Wasservorrat des Hilfsspeisewassersystems (ASG) gesichert, indem die Anbindung des ASG-Tanks an das Brandschutznetz diversifiziert wurde (PNPP1864), was Unfallsequenzen unterstützt, bei denen zusätzliches Speisewasser benötigt wird. Im Hinblick auf die thermohydraulische Regelung behebt die Erhöhung der Kapazität des GCT-a-Regelventils (PNPE1141) die Diskrepanz bei der

Berechnung des ASG-Verbrauchs und unterstützt aktualisierte Unfallstudien zur Dimensionierung. Parallel dazu wurde eine niedrigere zulässige I-131-Aktivität im Primärkreis festgelegt, um die radiologischen Folgen bei Unfällen ohne Brennstabschäden zu begrenzen.

Im Hinblick auf das Lagerbecken für abgebrannte Brennelemente werden die Integrität und die Robustheit der Kühlung durch das Hinzufügen eines diversifizierten und mobilen Kühlkreislaufs (PTRbis) verstärkt, der eine robuste Möglichkeit zur Wiederherstellung der Kühlung bietet und den Prinzipien des „Hardened Safety Core“ (Noyau Dur) entspricht. PTR-bis ist ein von FARN (nukleare Schnelleinsatztruppe) eingesetztes, mobiles System, das an feste Lagerbeckenanschlüsse angeschlossen ist und dazu dient, das Sieden zu stoppen und die langfristige Rückkehr zur Kühlung zu unterstützen.

Ein unbeabsichtigtes Entleeren wird verhindert durch: automatische Absperrung der PTR-Ansaugung bei sehr niedrigem Wasserstand im Becken (PNPP1402), doppelte Absperrung der Ansaugleitung (PNPE1344), verstärkte und motorisierte Absperrung der Transferrohre (PNRL1895 und PNPP1403), Automatisierung der Beckenentleerungs-/Filterventile im Reaktorgebäude (PNPP1780) und eine neu-dimensionierte Siphonunterbrechung in der Druckleitung (PNPP1289).

Es ist geplant, das Risiko einer Brandausbreitung zwischen den PTR-Strängen durch eine physische Trennung und Schutzmaßnahmen mittels einer Brandschutzwand zwischen den beiden PTR-Pumpen (PNPP1949) zu mindern.

Sicherheitsaspekte von Unfällen mit Kernschmelze

Schwere Unfälle (SA) mit Kernschmelze wurden bei der Auslegung der französischen 900-MWe-Reaktoren nicht berücksichtigt. Infolge früherer PSÜ wurden jedoch Einrichtungen und Maßnahmen für das Management schwerer Unfälle implementiert. Dennoch wurden bereits vor 15 Jahren eine Reihe von Mängeln bei den EU-Stresstests festgestellt, von denen nicht alle bis heute behoben wurden. Laut ASNR besteht das Ziel der 4. PSÜ für die 900-MWe-Reaktoren darin, das Sicherheitsniveau des Reaktors dem des EPR in Flamanville, einem Reaktor der dritten Generation, anzunähern. In Reaktoren der dritten Generation werden bereits bei der Auslegung Funktionen zur Minderung der Auswirkungen von Kernschmelzunfällen berücksichtigt; diese lassen sich nicht vollständig auf Reaktoren der zweiten Generation wie Dampierre 3 und 4 übertragen. Die UVP-Unterlagen enthalten keinen systematischen Vergleich zwischen dem Sicherheitsniveau der 900-MWe-Reaktoren und dem des EPR, um die verbleibenden Lücken zu identifizieren.

Die im Rahmen der 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 worden, und zum anderen ist es anhand der verfügbaren Informationen nicht möglich zu beurteilen, ob die Modifikationen (insbesondere die Verstärkung des Fundaments) ausreichend sind.

Noch sind nicht alle notwendigen und geplanten Umbauten für die Wärmeabfuhr ohne Einsatz des gefilterten Druckentlastungssystems im Falle eines Kernschmelzunfalls vollständig umgesetzt worden. Wichtige Komponenten wie Ventile, Dichtungen und elektrische Ausrüstung, die bei einem Kernschmelzunfall benötigt werden, deren Funktionsfähigkeit während eines solchen Unfalls jedoch nicht gewährleistet werden kann, werden ebenfalls erst in Phase B oder der Ergänzungsphase ausgetauscht. Zudem wurde die Verstärkung des gefilterten Druckentlastungssystems (U5-System) gegen schwere Erdbeben noch nicht durchgeführt. Dies bedeutet, dass auch nach Abschluss aller Phase-A-Maßnahmen der 4. PSÜ ein Kernschmelzunfall mit einer massiven Freisetzung radioaktiver Stoffe in Dampierre 3 oder 4 weiterhin möglich ist.

Die UVP-Unterlagen geben keinen vollständigen Überblick darüber, welche der geplanten Modifikationen den am Ende der generischen Phase des 4. PSÜ veröffentlichten ASNR-Anforderungen entsprechen. Die meisten Maßnahmen sollen erst bis zum Ende der Phase B und der Ergänzungsphase (2029 für Block 3 bzw. 2030 für Block 4) umgesetzt werden. Aus den UVP-Unterlagen geht nicht hervor, ob dieser Zeitplan eingehalten wird.

Strahlungsauswirkungen von Unfällen / Grenzüberschreitende Auswirkungen

Die UVP-Unterlagen behandeln Ereignisse und Unfallabläufe, die drei Kategorien von Auslegungsunfällen entsprechen, sowie eine zusätzliche Kategorie, die auslegungsüberschreitende Unfälle umfasst, einschließlich Szenarien mit Kernschmelze und Szenarien im Lagerbecken für abgebrannte Brennelemente.

Der im Bericht vorgelegten Analyse der radiologischen Auswirkungen mangelt es an ausreichenden technischen Details. Wesentliche Informationen, die für eine unabhängige Überprüfung erforderlich sind, wie Radionuklidinventare, Annahmen zum Quellterm, Freisetzungsanteile und die Methodik für die Ausbreitungsmodellierung, werden nicht zur Verfügung gestellt. Folglich sind die Transparenz und die Reproduzierbarkeit der Bewertung der radiologischen Auswirkungen äußerst begrenzt.

Aus den UVP-Unterlagen geht hervor, dass bei Auslegungsstörfällen die radiologischen Auswirkungen voraussichtlich unter den nationalen Referenzwerten bleiben und keine grenzüberschreitenden Risiken verursachen. Bei auslegungsüberschreitenden Unfällen, insbesondere bei Szenarien mit Kernschmelze, räumt der Bericht zwar die Möglichkeit für weitreichende Auswirkungen ein, es fehlen jedoch ausreichende technische Details, um eine unabhängige Überprüfung dieser Ergebnisse zu ermöglichen. Die UVP-Berichte enthalten keine quantitativen Analysen, die die Aussagen untermauern, dass die Lebensmittelkontamination in Entfernungen von mehr als 5 km nach 7 Tagen und in einem Umkreis von 1 km nach einem Jahr unter den EU-Grenzwerten bleiben würde. Darüber hinaus fehlen in der Bewertung Informationen zur Bodenkontamination trotz ihrer Bedeutung für die Bewertung langfristiger radiologischer Auswirkungen und der potenziellen Kontamination der Nahrungskette.

Die vom Expert:innenteam durchgeführten Modellierungen der atmosphärischen Ausbreitung und der Bodenkontamination zeigen, dass unter bestimmten meteorologischen Bedingungen ein schwerer Unfall in Block 3 oder 4 des Kernkraftwerks Dampierre zu einer Bodenkontamination von Cs-137 in Österreich führen könnte, die über dem nationalen Schwellenwert von 650 Bq/m² liegt. Obwohl die Studie die Wahrscheinlichkeit solcher Bedingungen nicht bewertet, deuten die Ergebnisse darauf hin, dass grenzüberschreitende Auswirkungen, die über die in den UVP-Dokumenten angegebenen hinausgehen, nicht ausgeschlossen werden können.

Insgesamt liefern die UVP-Unterlagen zwar eine Bewertung der radiologischen Folgen, enthalten jedoch keine vollständigen Informationen zur Bewertungsmethodik und zu den zugrunde liegenden Daten, die die Aussagen untermauern, insbesondere im Hinblick auf schwere Unfälle mit potenziellen grenzüberschreitenden Auswirkungen. Um die potenziellen Auswirkungen auf Österreich umfassend zu bewerten und die in den UVP-Unterlagen gemachten Aussagen zu untermauern, wären detailliertere Angaben zum Freisetzungsszenario, zu den Eingangsgrößen der Ausbreitungsmodellierung sowie zu den Bewertungen der Kontamination der Nahrungskette erforderlich.

Bewertung des Zeitrahmens

Der Zeitrahmen für die Umsetzung aller Maßnahmen im Rahmen der 4. PSÜ (5 Jahre nach Veröffentlichung des PSÜ-Berichts, entsprechend Juni 2029 für Dampierre 3 und April 2030 für Dampierre 4) ist nicht ungewöhnlich. Da der Zeitraum nach der 4. PSÜ jedoch mit dem Beginn des Langzeitbetriebs (LTO) zusammenfällt, 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 in der Lieferkette, einschließlich der personellen Ressourcen, könnten sich auf den Umsetzungszeitraum auswirken. Besonders hervorzuheben ist, dass wichtige Sicherheitsmaßnahmen, die im Rahmen der 4. PSÜ aufgeführt sind, bereits im Rahmen des EU-Stresstests (2012) als notwendig erachtet und deren Umsetzung vereinbart worden waren.

Entsorgung von abgebrannten Brennelementen und radioaktiven Abfällen

Abgebrannte Brennelemente und radioaktive Abfälle können negative Auswirkungen auf Menschen und Umwelt haben. Um dies zu verhindern, ist der Nachweis einer sicheren Entsorgung erforderlich. Die UVP-Unterlagen liefern jedoch keine ausreichenden Belege.

Im Rahmen der UVP sollten ausreichende Informationen zur Entsorgung von abgebrannten Brennelementen und radioaktiven Abfällen vorgelegt werden. Es wäre wünschenswert, wenn die fehlenden Informationen in die UVP-Unterlagen aufgenommen würden.

Die Aktualisierung des französischen nationalen Plans für die Entsorgung radioaktiver Stoffe und Abfälle (PNGMDR) sollte einer strategischen Umweltprüfung unterzogen werden, auch im grenzüberschreitenden Kontext.

RESUME

La centrale nucléaire de Dampierre-en-Burly (ci-après Dampierre) comprend quatre réacteurs à eau pressurisée en service d'une capacité de 900 MWe chacun. Ces réacteurs ont été mis en service respectivement en 1980 et 1981.

La France a notifié le quatrième réexamen périodique (« Procédure de consultation publique sur le quatrième rapport du réexamen ») de la centrale nucléaire de Dampierre (réacteur 3 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 du Loiret. 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'Autorité de sûreté nucléaire et de radioprotection (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. Il convient de noter que les documents relatifs à l'EIE faisaient référence au Guide de sûreté n° NS-G-2.12 de l'AIEA (AIEA 2009), qui est désormais obsolète.

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.

La mise en œuvre du Programme d’investigations complémentaires (PIC) est une démarche visant à s’assurer de l’absence de défaillances opérationnelles dans des zones qui ne font pas l’objet d’inspections régulières. Sans justification, il est indiqué qu’aucun contrôle ne doit être effectué sur les réacteurs Dampierre 3 et 4 dans le cadre du Programme d’investigations complémentaires.

Dans le cadre de la phase générique du 5e RP des réacteurs de 900 MWe, l'ASNR a demandé à EDF de définir, au plus tard 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 inter-cristalline, 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 matériaux en conditions accidentelles » du 4e RP est très limitée pour Dampierre 3 et 4.

Un incident lié à la sûreté récemment signalé, impliquant des ancrages défectueux destinés à la protection sismique, soulève de sérieuses questions quant aux essais de conformité réalisés jusqu'à présent. Ces défauts liés à la sûreté existaient depuis la mise en service de la centrale sans avoir été détectés lors des inspections précédentes.

Dans l'ensemble, plusieurs incidents liés à la sûreté, classés au niveau 1 de l'échelle internationale des événements nucléaires et radiologiques (INES), se sont produits au cours des cinq dernières années, dont beaucoup étaient liés au non-respect des règles générales d'exploitation (RGE). Ces incidents ont été précédés par des défaillances de composants, des erreurs de maintenance ou des erreurs d'exploitation (dans certains cas, plusieurs erreurs). La cause pourrait être un manque de culture de sûreté, associé à un nombre important d'incidents liés au vieillissement. Il convient tout particulièrement de noter que les enquêtes ont révélé que bon nombre de ces lacunes existaient depuis un certain temps déjà.

Risques externes

Les documents de l'EIE fournissent des informations sur les types de risques pris en compte dans la démonstration de sûreté de Dampierre 3 et 4, ainsi que sur les mesures déjà mises en œuvre ou dont la mise en œuvre a été décidée afin de renforcer la résistance du réacteur face aux risques externes. Pour la plupart des risques externes, les méthodes, les données et les hypothèses utilisées dans l'évaluation des risques ne sont pas précisées en détail. Il n'est donc pas possible d'évaluer la conformité aux exigences et aux recommandations de l'Association des autorités de sûreté nucléaire d'Europe occidentale (Western European Nuclear Regulators' Association - WENRA). On constate un non-respect des niveaux de référence de la WENRA en matière de séismes et d'accélérations sismiques au sol. Le séisme de référence (DBE) pour Dampierre, appelé séisme majoré de sécurité (SMS) dans la réglementation française, repose sur une analyse déterministe qui n’est plus à la pointe de la technologie. Il ressort des documents disponibles que la valeur SMS de l'accélération au sol (PGA) à Dampierre est de 0,2 g (zone de la centrale nucléaire) et de 0,1 g (zone hors centrale nucléaire). Il reste à démontrer que le système SMS actuel satisfait aux exigences de la WENRA concernant un événement de conception de base (DBE) dont l'intervalle de récurrence moyen est de 10 000 ans (WENRA 2021).

Le site de Dampierre est situé à proximité du système de failles de la Loire, datant de la fin du Pliocène et de la période post-pliocène. Les failles en question n’ont pas fait l’objet d’études approfondies permettant de déterminer si elles sont actives ou inactives, bien qu’elles soient connues depuis au moins dix ans. Selon le rapport d’EIE, EDF envisage de mettre en place un programme d’études géophysiques et paléosismologiques afin d’évaluer l’importance de ces failles en matière de sûreté. Aucun calendrier concret n’a été communiqué. Aucune réévaluation actualisée des risques sismiques, fondée sur l’analyse de la sensibilité aux risques sismiques (PSHA) et s’appuyant sur les données relatives aux failles, n’a été annoncée. Pour les failles potentiellement actives situées à proximité immédiate d’un site de centrale nucléaire, la WENRA (2020b) recommande une cartographie systématique des failles et la collecte d’informations paléosismologiques. Au vu de la littérature existante (Jomard et al. 2017 ; Ritz et al. 2021 et les références qui y sont citées), il est surprenant qu’un processus de caractérisation des failles actives n’ait pas été mis en œuvre plus tôt au cours du 4e PSR. Il est suggéré à l'ASNR d'exiger (i) des évaluations paléosismologiques spécifiques de ces failles et (ii) une évaluation probabiliste des risques de sûreté (PSHA) actualisée, conforme aux exigences de la WENRA et tenant compte des données paléosismologiques.

En ce qui concerne les renforcements de la sûreté de Dampierre 3 et 4, il apparaît clairement que l’une des mesures les plus importantes pour assurer la protection contre les aléas externes est la mise en œuvre du « noyau de sûreté renforcé » (ND), qui est toujours en cours et devrait être achevée d’ici 2029 et 2030. La décision de mettre en œuvre le ND a été prise en 2012. Le fait que la mise en œuvre du ND ne soit achevée que 18 ans plus tard semble remarquable, sachant que la WENRA exige la « mise en œuvre en temps opportun des améliorations de sûreté identifiées et raisonnablement réalisables ».

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 Dampierre 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 Dampierre 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 récente évaluation de la sécurité nucléaire en France met en évidence des lacunes dans les exigences requises en matière de sécurité nucléaire, notamment en ce qui concerne la « culture de la sécurité », la « cybersécurité » et la « protection contre les menaces internes ».

Aspects liés à la sûreté en cas d'accident sans fusion du cœur et 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).

D'importantes améliorations en matière de sûreté ont été mises en œuvre ou sont prévues afin de réduire les conséquences radiologiques et de renforcer la défense en profondeur à Dampierre 3 et 4. Pour l'évacuation de la chaleur à long terme, la disponibilité en eau du système auxiliaire d'alimentation en eau (ASG) a été garantie en diversifiant le raccordement du réservoir ASG au réseau de protection incendie (PNPP1864), ce qui permet de faire face aux séquences d'accident nécessitant un apport supplémentaire d'eau d'alimentation. En matière de contrôle thermohydraulique, l'augmentation de la capacité de la vanne de régulation GCT-a (PNPE1141) corrige l'anomalie de calcul de la consommation de l'ASG et prend en compte les études d'accidents dimensionnées mises à jour. Parallèlement, une activité admissible plus faible de l'I-131 dans le circuit primaire a été adoptée afin de limiter les conséquences radiologiques en cas d'accidents sans défaillance des barres de combustible.

En ce qui concerne la piscine de stockage du combustible usé, l'intégrité et la fiabilité du système de refroidissement sont renforcées par l'ajout d'un circuit de refroidissement mobile et diversifié (PTR bis), qui offre un moyen résilient de rétablir le refroidissement et s'inscrit dans le respect des principes du « noyau dur ». Le PTR bis est un système mobile déployé par la FARN (Force d'action rapide nucléaire) et relié à des raccordements fixes au bassin, utilisé pour mettre fin à l'ébullition et faciliter le retour progressif au refroidissement.

Le risque de vidange accidentelle est atténué par : Fermeture automatique de la vanne PTR d’aspiration dans la piscine d’entreposage du combustible sur niveau très bas (PNPP1402), doublement de l’automatisme d’isolement de la ligne d’aspiration du circuit PTR (PNPE1344) ; la fermeture renforcée et motorisée du tube de transfert (PNRL1895 et PNPP1403) ; l'automatisation des vannes de vidange/filtration de la piscine du bâtiment réacteur (PNPP1780) ; et un casse-siphon redimensionné sur la conduite de refoulement (PNPP1289).

Il est prévu de limiter le risque de propagation d'un incendie entre les trains PTR grâce à une séparation physique et à des mesures de protection mises en œuvre au moyen d'un écran coupe-feu situé entre les deux pompes PTR (PNPP1949).

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. Pourtant, plusieurs lacunes avaient déjà été identifiées dans les tests de résistance de l'UE il y a quinze ans, et toutes n'ont pas encore été comblées à ce jour. 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 Dampierre 3 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.

Toutes les modifications nécessaires et prévues pour assurer l'évacuation de la chaleur sans recourir au système de décompression filtré en cas d'accident entraînant la fusion du cœur n'ont pas encore été entièrement mises en œuvre. Des composants essentiels, tels que les vannes, les joints et les équipements électriques, qui sont indispensables en cas d'accident de fusion du cœur, mais dont la résistance ne peut être garantie dans une telle situation, ne seront eux aussi remplacés qu'au cours de la phase B ou de la phase supplémentaire. 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 Dampierre 3 ou 4.

Les documents d'EIE ne fournissent pas un aperçu complet des modifications prévues desquelles / 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 pour l'unité 3 et 2030 pour l'unité 4 respectivement). Les documents d'EIE n'indiquent pas si ce calendrier sera respecté.

Impact radiologique des accidents / Effets transfrontaliers

Les documents de l'EIE traitent des événements et des séquences d'accidents correspondant à trois catégories d'accidents de référence, ainsi que d'une catégorie supplémentaire représentant les événements dépassant le cadre des accidents de référence, notamment les scénarios de fusion du cœur et de la piscine de stockage 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 extrêmement 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, et plus particulièrement les scénarios impliquant une fusion du cœur, le rapport reconnaît l'existence d'impacts potentiels à longue distance, mais ne fournit pas suffisamment de détails techniques pour permettre une vérification indépendante de ces conclusions. Les rapports de l'EIE ne présentent 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 au réacteur n° 3 ou réacteur n° 4 de la centrale nucléaire de Dampierre 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, soit juin 2029 pour Dampierre 3 et avril 2030 pour Dampierre 4) 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.

Gestion du combustible usé et des déchets radioactifs

Le combustible usé et les déchets radioactifs peuvent avoir des effets néfastes sur la population et l'environnement. Pour éviter cela, il est nécessaire de prouver que leur élimination est sûre. Or, les documents d'EIE ne fournissaient pas de preuves suffisantes. Au cours de l'EIE, des informations suffisantes sur la gestion du combustible usé et des déchets radioactifs devraient être fournies. Il serait apprécié que les informations manquantes soient ajoutées aux documents d'EIE.

La mise à jour du Plan national français de gestion des matières et déchets radioactifs (PNGMDR) devrait faire l'objet d'une évaluation environnementale stratégique, y compris au niveau transfrontalier.

1 INTRODUCTION

The Dampierre-en-Burly (here: Dampierre) nuclear power plant (NPP) consists of four operating pressurized water reactors with a capacity of about 900 MWe each. These reactors were commissioned 1980 and 1981 respectively.

France notified the 4th Periodic Safety Review ("Public consultation procedure on the 4th safety review report") of the NPP (reactor 3&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 Loiret. 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 Forestry, Climate and Environmental Protection, Regions and Water Management Action 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 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 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 4th 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 upgrades) the NPP to minimize these deltas as far as reasonably achievable. EDF and the safety authority (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, 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., Dampierre, 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 (CP0) were permanently shut down, EDF is planning to extend the operational life of all the other units beyond forty years. (ASN 2022a)

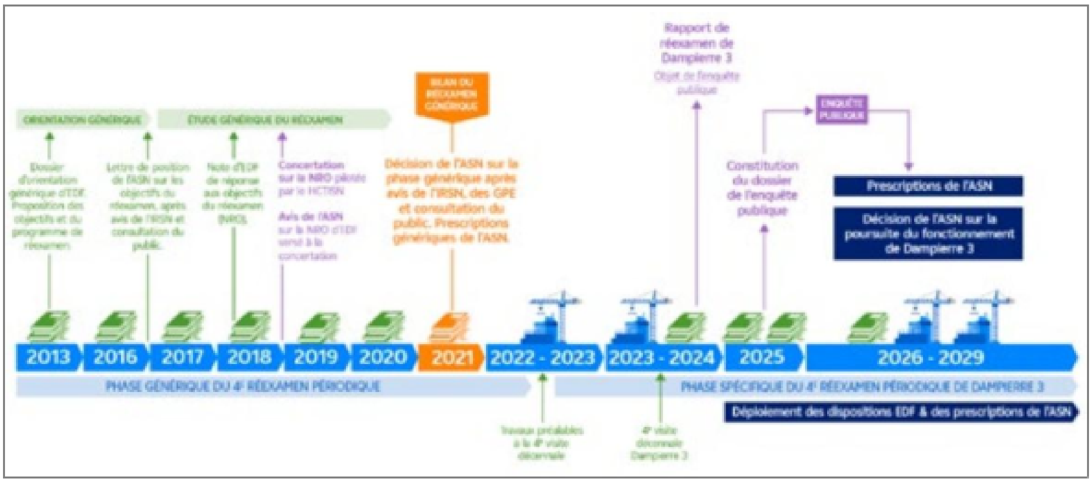
In France the 4th PSR is carried out in two complementary phases: a generic phase common to all 900 MWe reactors and a specific phase for each individual reactor.

For the 4th PSR of the 900 MWe NPPs, 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)

After the generic phase, the reviews of the 32 reactors of the 900 MWe nuclear plants are carried out over approximately ten years, from 2019 to 2031. EDF then transmits a review report to the government and the ASNR; this report is prepared after the reactor’s ten-year outage, during which modifications are implemented, and inspections, controls and maintenance operations are carried out. For Dampierre 3, the ten-year outage ran from 23 September 2023 to 2 March 2024, and EDF transmitted the conclusions report on 27 June 2024. (EIA-REPORT D3, P.1 2026) For Dampierre 4, the outage ran from 13 July 2024 to 21 December 2024, and EDF transmitted the conclusions report on 4 April 2025. The following timelines show the slightly different main stages of the 4th PSR for Dampierre 3 and 4. (EIA-REPORT D 4, P.1 2026)

Figure 1: Main stages of the 4th PSR for Dampierre 3 (EIA-REPORT D3, P.1 2026)

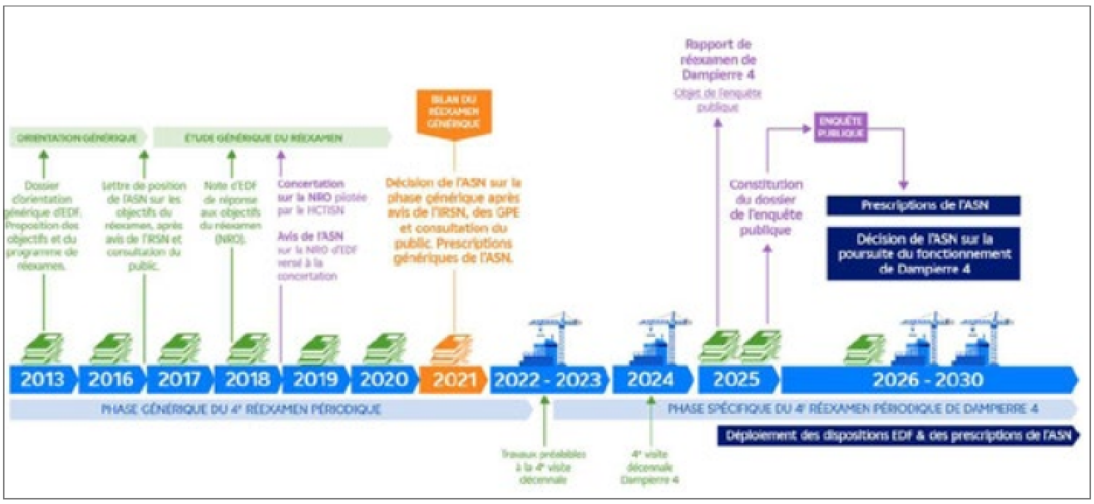
Main stages for Dampierre 3



Source: EIA-REPORT D3, P.1 2026

Figure 2: Main stages of the 4th PSR for Dampierre 4 (EIA-REPORT D4, P.1 2026)

Main stages for Dampierre 4



Source: EIA-REPORT D4, P.1 2026

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 (from 6 September 2018 to 31 March 2019). 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 environmental impact assessment (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.

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.

In this context, it should be noted that an assessment is currently underway under the Espoo Convention to determine whether an EIA procedure would have been required for the lifetime extension of another 900 Mwe rector (Tricastin unit 1). (UNECE 2026).

2.3 Conclusions

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 OPERATION EXPERIENCE

3.1 Treatment in the EIA documents

Ageing and obsolescence control

The EIA-REPORT D3/4, P.2 (2026) deals 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 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.

For Dampierre 3&4, it is stated that a systematic methodology is applied to ensure that aging phenomena do not lead to difficulties in fulfilling a safety function during the period under consideration. This method is consistent with international best practices and is in line with the approach recommended by the IAEA in its Safety Guide No. NS-G-2.12, "*Ageing Management for Nuclear Power Plants*." (EIA-REPORT D3/4, P.2 2026)

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 on the containment of Dampierre 3 from January 11 to 14, 2024, and on the containment of Dampierre 4 from November 8 to 10, 2024 with the results meeting expectations. (EIA-REPORT D3/4, P.1 2026

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 Dampierre 3&4, maintenance and control measures were carried out, along with modifications to ensure the continued suitability of these 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, Research & Development programs, laboratory tests, particularly in the field of materials, quality assurance procedures, etc. The DAPEs are updated every five years. (EIA-REPORT D3/4, P.2 2026)

There are currently 12 DAPEs for the following components of 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 at the CPY level were adapted by the Dampierre NPP to determine the management of the aging of the SSCs of units 3&4. During the ten-year inspection, the SSCs underwent a series of maintenance operations, inspections, tests, non-destructive tests, or modifications. (EIA-REPORT D3/4, P.2 2026)

It is stated that, in summary, the tests, inspections and maintenance work carried out during the VD4 shutdown will help to demonstrate the suitability of reactors 3 and 4 at the Dampierre NPP for continued operation under satisfactory safety conditions for the ten-year period VD4-VD5. (EIA-REPORT D3/4, P.2 2026)

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,
- civil engineering and containment.

No tests are to be carried out for Dampierre 3&4 as part of Program for Complementary Investigations (PIC). (EIA-REPORT D3/4, P.1 2026)

Stress corrosion of the auxiliary lines

As part of the ‘stress corrosion’ incident involving the auxiliary pipes of the main primary circuit, which began at the end of 2021, a strategy for addressing the issue across the entire nuclear power station and an associated monitoring programme were established; these are regularly reviewed in consultation with the ANSR. The expert assessments carried out on the various reactors within the nuclear power plant fleet have shown that 900-MW reactors, such as those at Dampierre, are hardly susceptible, if at all, to the phenomenon of stress corrosion. (EIA-REPORT D3/4, P.1 2026)

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. All of the inspections required under the programme on the qualification of materials under accident conditions were carried out. (EIA-REPORT D3/4, P.2 2026)

A total of 258 components qualified for use in accident conditions (MQCA) were inspected, and all identified deviations were corrected. (EIA-REPORT D3/4, P.2 2026)

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.

Safety relevant events

The analysis shows that safety-related events reported as Level 1 events in unit 4 of the Dampierre NPP, which were addressed by material or design changes following its 4th PSR shutdown, have been corrected, with the exception of an issue related to MOX fuel assemblies. Mitigation measures are being implemented or are planned for this generic event until the identified anomaly is permanently resolved through long-term measures. This issue is the subject of regular discussions with the ASNR. (EIA-REPORT D4, P.2 2026)

The analysis shows that not all of the safety-related events reported as INES Level 1 events in unit 3 of the Dampierre NPP were corrected following its 4th PSR shutdown. In addition to the above-mentioned event relating to MOX fuel assemblies, two further events could not be resolved before restart of the plant:

- A safety-related incident involving the failure of a relay, which was detected during a routine inspection and resulted in the unavailability of a signal from the safety injection system.
- A safety-related incident concerning compliance with the operating guidelines following the failure of a diesel generator set.

3.2 Discussion

As in any industrial plant, the quality of the materials used in a NPP deteriorates during operation, particularly as a result of physical aging. 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.

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), 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 Programmes (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 was therefore closed. (ASN 2021b)

However, the EIA-REPORT (D3/4, P.2 2026) for Dampierre 3&4 referred to the outdated IAEA Safety Guide No. NS-G-2.12, "*Aging Management for Nuclear Power Plants*", (IAEA 2009). This IAEA Safety Guide was published in 2009 and replaced by the specific IAEA Safety Guide SSG-48 "*Aging Management and Development of a Program for the Long-Term Operation of Nuclear Power Plants*," (IAEA 2018).

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 a successful accident management.
(UMWELTBUNDESAMT 2024b)

OSART Mission

The last mission of the IAEA Operational Safety Review Team (OSART)[1] visited the Dampierre NPP in 2015 with a follow-up visit in 2017. (IAEA 2026a)

The OSART missions provide a good opportunity to receive recommendations for operational improvements from an international team of experts. This opportunity should be utilised once again for the Dampierre NPP.

Evaluation of significant effects

As part of this expert statement, an evaluation of safety-related events of Dampierre 3&4 from January 2021 to June 2026 was carried out based on reports of the ASNR.[2]

Non-compliance with General Operating Procedures (RGE)

An analysis of safety-related incidents published by the ASNR over the past five years that were classified as Level 1 on the INES scale revealed a number of incidents that were frequently linked to non-compliance with General Operating Procedures (RGE). The RGE are a set of regulations approved by the ASNR which define the plant's permissible operating range and the associated operating procedures. In particular, they stipulate the maximum repair times in the event of a failure of systems essential to reactor safety. The incidents at Dampierre 3&4 were preceded by component failures, maintenance or operational errors, and in some cases multiple operational errors. The reason could be a lack of safety culture combined with a large number of age-related incidents. It is particularly noteworthy that the investigations revealed that many of the shortcomings had been present for some time.

The following paragraphs list these events for Dampierre 3 (11 events) and Dampierre 4 (9 events):

- On 1 December 2025, the operator reported a safety-related event for r**eactor 4 concerning a malfunction of the containment spray system (EAS)** and the device for removing residual power from the containment (EAS-ND). Whilst conducting a routine

[1] Since 1982, the IAEA has offered internationally staffed review teams (OSART missions) to assess the status of the safety-oriented interaction of personnel, technical and organizational factors in nuclear power plants.

[2] https://reglementation-controle.asnr.fr/recherche?page=20&search_text=Dampierre%20INES&sort_type=date

inspection, the operator discovered that air was present in a pipe within the EAS system. In such a situation, the procedure set out in the RGE stipulates that the reactor must be shut down within three days. However, the analysis revealed that the EAS and EAS-ND systems had already been unavailable for several weeks. The event compromised the safety function and, due to its delayed detection and failure to comply with the RGE, this event was classified as level 1 on the INES scale.

- On 30 May 2025, the operator reported a safety-related event for **reactor 3 in connection with the failure of the post-accident monitoring system for the secondary circuit**. On 26 May 2025, the operator identified inconsistencies in the display of a recording device within the system, which is used to monitor the water level in the steam generator. Investigations led to the conclusion that the recording device had been out of service for more than ten hours. Under these circumstances, the RGE were not complied with. Due to the delayed detection and failure to comply with the RGE, this event was classified as level 1 on the INES scale.
- On 4 July 2025, the operator reported a safety-related incident involving **reactor 4** in connection with a **partial failure of the accident monitoring system**. The system provides information in the control room that is intended to enable the operator to assess the state of the reactor in the event of an accident. Investigations led to the conclusion that the information displayed by the recording device had been incorrect for more than 10 days. The RGE that stipulate that the reactor must be shut down within 7 days in such cases, were not complied with. Due to the delayed detection and failure to comply with the RGE, this incident was classified as level 1 on the INES scale.
- On 7 April 2025, the operator reported a significant **radiation protection incident at reactor 3.** During a check at the exit from the controlled area, contamination was detected on the employee's cheek. As a quarter of the annual exposure limit for a worker, as laid down by law, had been exceeded, this incident was classified as level 1 on the INES scale.
- On 25 March 2025, the operator reported a safety-related event for **reactor 4** in connection with the **partial unavailability of the post-accident monitoring system**. Investigations led to the conclusion that the information displayed had been incorrect for more than 20 days. The RGE that stipulate that the reactor must be shut down within 7 days in such cases, were not complied with. Due to

the delayed detection and failure to comply with the RGE, this incident was classified as level 1 on the INES scale.

- On 21 August 2024, EDF reported a safety-related incident involving the **partial failure of the emergency raw water circuit at reactor 3**. Investigations revealed that the failure was due to an error that occurred during related maintenance work. Due to the delay in detection, this incident, which compromised the safety function, was classified as level 1 on the INES scale.
- On 22 September 2023, the operator reported a safety-related incident for **reactor 3** relating to the **failure of the automatic start-up of a diesel-powered emergency generator.** During its investigations, the operator noticed that an electrical relay required for the automatic start-up of this emergency generator had not been connected correctly. The faulty setting had been in place since the last inspection of this emergency generator two months earlier. The RGE that stipulate that the reactor must be shut down within 8 hours, were not complied with. Due to the delayed detection and non-compliance with the RGE, this incident was classified as level 1 on the INES scale.
- On 18 August 2023, the operator reported a safety-related incident relating to the **power supply for reactor 3**. Due to non-compliance with the RGE and the delayed detection, this incident, which affected the power supply function of the control and monitoring system, was classified as level 1 on the INES scale.
- On 7 February 2023, EDF reported a safety-related incident concerning the **delayed detection of a malfunction in one of the two loops of the safety injection system for reactor 4**. The safety injection system (RIS) enables boron-containing water to be injected under pressure into the reactor’s primary circuit in the event of an accident leading to a significant leak, in order to suppress the nuclear reaction and ensure the cooling of the reactor core. During a routine inspection of one of the low-pressure pumps in the safety injection system, mechanical anomalies were detected in the pipework connections. As the operator did not identify the non-compliance with RGE, this incident was classified as level 1 on the INES scale.
- On 29 November 2022, the operator reported a safety-related event for **reactor 3** in connection with the l**imitation of thermal and mechanical stress on the steam generator**s. Due to non-compliance with RGE, this event was classified as level 1 on the INES scale.

- On 22 July 2022, the operator reported a safety-related incident relating to the delayed detection of a **malfunction in the high-pressure safety injection system of reactor 3**. On 14 July 2022, an on-site employee noticed a significant amount of boron on a valve in the system, indicating a leak. Investigations revealed that this valve had been leaking for several days, which had gone unnoticed by on-site staff despite daily rounds in the relevant room. Due to the delayed detection of the system's unavailability, this incident, which affected safety functions, was classified as level 1 on the INES scale.
- On 6 July 2022, the operator reported a safety-related incident relating to the u**navailability of several switchboards for the emergency power supply to reactor 4**. On 4 May 2022, during scheduled maintenance work on reactor 4, an employee made incorrect settings to an electrical relay. On 11 May 2022, the same employee made further, equally incorrect settings to a relay in a second switchgear assembly. Due to the incorrect settings, the operator classified the emergency diesel generator and the switchgear as unavailable from 4 May to 28 June 2022, meaning that the maximum repair time limits specified in the RGE were exceeded. As the operator did not identify the non-compliance with the RGE until a late stage, this event, which impaired the functioning of the emergency power supply, was classified as level 1 on the INES scale.
- On 29 June 2022, the operator reported a safety-related event for **reactor 4, as the alarm for the neutron flux measurement chains** did not trigger when an increased neutron flux was detected. Due to the operator's delayed detection and non-compliance with the RGE, this event, which compromised the safety function, was classified as level 1 on the INES scale.
- On 9 May 2022, the operator reported a safety-related event for r**eactor 3** in connection with the exceeding of the time limit specified in the RGE for the shutdown of this reactor in the event of a **partial failure of the safety injection system (RIS)**. During operation, the operators noticed that a valve in this system had failed to close, resulting in a partial failure of the RIS. Under these circumstances, the RGE required the reactor to be shut down within one hour. As the operator did not detect the non-compliance with the RGE later, this event, which compromised safety functions, was classified as level 1 on the INES scale.
- On 15 April 2022, the operator reported a safety-related event for **reactor 4**, relating to the failure of the **safety injection system (RIS).** As the operator did not recognise the non-compliance with

the RGE until a late stage, this event, which impaired the safety functions, was classified as level 1 on the INES scale.

- On 2 December 2021, the operator reported a safety-related event for **reactor 3 concerning the partial unavailability of the reactor cooling circuit** during the shutdown. Subsequent investigations carried out by the operator revealed that this reduction in flow within this circuit had already been observed on 24 November 2021, without EDF having taken any action at the time. The general operating procedures, which stipulate that repairs must be carried out within three days, were therefore not complied with in retrospect. In view of the delayed detection of the partial unavailability of the system, this incident was classified as level 1 on the INES scale.
- On 5 August 2021, the operator reported a safety-related event for **reactors 3 and 4** in connection with non-compliance with the RGE **regarding the calibration of the steam flow measurement chains** at the outlet of a steam generator. On 20 July 2021, during a routine test, the operator discovered that the equipment used to carry out this test was faulty. Due to the non-compliance with the RGE and the delayed detection, this incident was classified as level 1 on the INES scale.
- On 6 August 2021, the operator reported a safety-related event for **reactor 3 concerning the emergency power supply to the steam generators**. Whilst reactor 3 was in the restart phase following an unplanned shutdown, the operators present in the control room failed to recognize that the volume of the SG tank had fallen below the minimum volume required by the RGE. Due to the failure to comply with the RGE and the delayed detection of the insufficient volume in the SG tank, this incident was classified as level 1 on the INES scale.

Deficiencies of the seismic resistance of various components

- On 27 January 2026, EDF reported a safety-related incident concerning deficiencies in the anchoring of certain safety-critical systems within the building structure. EDF's report marks the fourth update to the report dated 21 September 2021. The safety-related incident now affects 56 reactors, i.e. all of EDF's reactors with the exception of the EPR reactor at Flamanville. This latest update adds 30 reactors, including reactor 4 at the Dampierre NPP. As part of its conformity assessment programme for its facilities, EDF is checking the compliance of the anchoring systems in the structural elements of safety-critical installations. During these inspections, discrepancies were identified in certain fixings, relating in particular to the number, diameter and arrangement of the anchors. These discrepancies date back to the reactors' construction phase and could have jeopardised the stability of the fixed equipment in the event of an earthquake. Given its potential consequences for these reactors, this incident is classified as level 1 on the INES scale.

- On 13 May 2024, EDF reported a safety-related incident relating to anchoring defects in the structural works of certain safety-critical installations. These defects also affect reactors 3 at the Dampierre NPP. Given its potential consequences for these reactors, this incident is classified as level 1 on the INES scale.

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. However, the EIA-REPORT (D3/4, P.2 2026) referred to the outdated IAEA Safety Guide No. NS-G-2.12 (IAEA 2009).

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 NPP 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.

The implementation of the PIC is an approach that aims to confirm the absence of operational failures in areas that are not regularly inspected. Without justification, it is stated that no checks are to be carried out for Dampierre 3&4 as part of the supplementary investigation program.

In the framework of the generic phase of the 5th PSR of the 900 MWe reactors, the ASNR required 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 Dampierre 3&4.

A recently reported safety-related incident involving faulty anchors for earthquake protection raises serious questions about the conformity tests carried out to date. These safety-related defects have existed since the plant was commissioned without being detected during previous inspections.

Overall, a number of safety-related incidents classified as level 1 on the INES scale have occurred over the past five years, many of which were related to non-compliance with the general operating regulations (RGE). These incidents were preceded by component failures, maintenance errors, or operational errors (in some cases, multiple errors). The cause could be a lack of safety culture combined with a large number of age-related incidents. It is particularly noteworthy that the investigations revealed that many of the shortcomings have been present for some time.

- The justification that no checks are to be carried out for Dampierre 3&4 as part of the Program for Complementary Investigations (PIC) should be provided
- 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 seismic safety should be considered.

4 EXTERNAL HAZARDS

4.1 Treatment in the EIA documents

EIA-REPORT D3/4, P.2 (2026) provides information of the external hazard types considered in the LTO process. The following external hazards (natural and human-made) are addressed: external flooding, earthquake, extreme weather or climatic conditions (extreme heat, extreme cold), impairment of the ultimate heat sink by frazil, icing, low water level, oil spill or biological flotsam, high wind and wind-generated missiles, tornado, lightning, snow, industrial hazards (explosion, release of hazardous substances), transportation of hazardous goods and aircraft crash. EIA-REPORT D3/4, P.2 (2026, p. 113-116) notes that studies on external hazards take into account the international standards set by WENRA. EIA-REPORT D3, P.2 (2026, p. 116) particularly states that requirements with respect to seismic safety, external flooding, impairment of the ultimate heat sink, lightning and tornado accord with WENRA (2014) expectations, i.e., the consideration of events with occurrence probabilities of 10^{-4} per year for design basis considerations. Design basis parameters for high winds, extreme heat and extreme cold are derived by adding margins to events with recurrence intervals of 100 years.

Hazard assessment

Earthquake: The seismic design base for the reactors of the 900 MWe fleet is deterministically derived from the maximum observed historical earthquake (SMHV[3]) increased by one degree of intensity giving the so-called maximum safety earthquake (SMS[4]) 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[5]) was defined. 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,

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[3] SMHV : Séisme Majoré Historiquement Vraisemblable – Maximal plausible historical earthquake

[4] SMS: Séisme Majoré de Sécurité – Maximum safety earthquake, equivalent to design basis earthquake (DBE)

[5] SND: Séisme Noyau Dur – Seismic level for the hardened safety core

(ii) envelope the SMS increased by 50%, and (iii) take site effects into account.

EIA-REPORT D3/4, P.2 (2026, p. 152) states that the seismic hazard was reassessed during the 4th PSR according to RFS 2001-01[6] and based on updated seismological findings (seismic-tectonic zoning, characterization of faults, etc.) and the historical seismicity data of the SisFrance 2012 database. The 4th PSR is said to also account for the "further development of calculation and modeling methods" and international lessons learned (e.g., the Kashiwazaki-Kariwa 2007 earthquake).

The reassessment of seismic hazard for the Dampierre site confirmed that the SMS spectrum applicable to the 4th PSR is covered by the SMS spectrum that was already considered in the 3rd PSR. The 4th PSR therefore did not lead to modifications of the Design Basis Earthquake (DBE) ground accelerations. EIA-REPORT D4, P.2 (2026, p. 147) adds that EDF concluded that the DBE for the nuclear island is in line with WENRA requirements. The requirements for the seismic design of SSCs important to safety remain the same as for the 3rd PSR (EIA-REPORT D3, P.2 2026, p. 154). However, EIA-REPORT D4, P.2 (2026, p. 147) identifies the need of seismic retrofit of the vent stack of Dampierre 4, which, in the event of failure and collapse, would damage relevant installations.

EIA-REPORT D4, P.2 (2026, p. 147-148) outlines the geological setting of the reactors. The site is located at the southern margin of the Paris Basin on top of thick Mesozoic to Neogene sediments overlain by fluvial sediments which are characterized by s-wave velocities that increase from 240 m/s in the fluvial sediments to more than 1600 m/s in 90 m depth. The average s-wave velocity for the uppermost 30 m (Vs30) is given with 600 m/s. Data derive from cross-borehole. EDF concludes that seismic site effects as defined in RFS 2001-01[7] are ruled out for the Dampierre site (EIA-REPORT D3/4, P.2 2026, p. 155/148). EDF further concludes that hazard assessment does not need to account for site effects.

The Dampierre NPPs are located near a number of faults belonging to the Loire fault system. EDF reports name the Sennely-, Etampes-, Sully-sur-Loire- and Sancerre-fault noting that these faults have been investigated by interpreting industrial reflection seismic. This data, however, does not allow final conclusions on the (in-) activity of the faults.

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[6] 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

[7] According to RFS 2001-01, sites with s-wave velocities <300 m/s require considering site effects in the calculation of ground motion spectra.

Although the NEOPAL database (BAIZE et al., 2002) does not record evidence for neotectonics activity within a distance of 25 km of the site EDF announces that additional investigations such as near-surface geophysical profiling and paleoseismological trenching “could be performed” to characterize the faults.

High temperatures: According to EIA-REPORT D3/4 (P.2 2026, p. 165/157) climate monitoring and the re-evaluations of air and water temperatures conducted by EDF in 2013 - taking into account methodological advancements and the results of climate monitoring for the 2009–2012 period - do not call into question previously established values. Temperatures are determined by exceedance probabilities of 10^{-2} per year and 70% confidence. Safety assessments include analyzing high temperature effects on SSCs, comparing the resulting SSC temperatures to the maximum permissible temperatures, and setting organizational measures to ensure safety functions in cases when temperature limits are exceeded. Analyses also accounted for high water temperature by reviewing 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 of the last decades (notably 1984-1985 and 1986-1987) and implemented during the 2nd PSR. Minimum temperatures for which protection must be provided have not changed during the 3rd and 4th PSR. This is justified by IPCC climate forecasts which predict a decrease in the number of cold days/nights. 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 Dampierre 3 and 4, located on the Loire, include the (re-) assessments of extreme precipitation (rain and peak rainfall intensity), river flooding and the effects of flood waves caused by dam break. Furthermore, as part of the implementation of the ASNR's 2012 technical regulations, EDF also analyzed the volumetric protection. Reassessments concluded higher rainfall intensities which led to the increase of heights of sills and sheet pile walls for flood protection. Detailed information on methods, data and assumptions used to derive the maximum Loire flood level, height of the platform, heights of protective dams etc. are not communicated. EDF also analyzed the volumetric flood protection devices and its resistance against seismic impact up to the SMS to account for the hazard combination earthquake and flooding (EIA-REPORT D3/4, P.2 2026, p. 151/144).

High wind, tornado and wind-blown projectiles: The EIA documents state that the reassessment of hazards by storm do not require any update (EIA-REPORT D3/4, P.2 2026, p. 176/169). 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 $1,1*10^{-5}$ per year for the inland French territory (EIA-REPORT D3/4, P.2 2026, p. 179/171). 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).

Impairment of the ultimate heat sink: During the 4th PSR, EDF targeted to verify the robustness of the installations with respect to hazards threatening the ultimate heat sink and reviewed the implementation of the safety requirements for pumping stations (EIA-REPORT D3/4, P.2 2026, p. 173/166). The activities were initiated after the clogging of water intakes of the NPPs Choos, Cruas and Blayais by frazil ice and flotsam in 2009. At Dampierre, these events already led to the installation of devices for pre-filtration of cooling water in addition to existing filtration. Hazard analyses for Dampierre include the re-assessment of low water level, icing, frazil ice and other phenomena threatening the cooling water intake by clogging, sedimentation (silting) and pollution of the cooling water with hydrocarbons.

Lightning and electromagnetic interference: 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 effects of lightning strikes on outdoor areas, roofs, and facades of buildings housing SSCs important to safety, and 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 D3/4, P.2 2026, p. 180/173).

Snow: SSCs are designed according to the "Snow and Wind Regulations" of 1965, which were in effect at the time of the construction of the Dampierre NPP. These regulations have been updated several times since their initial publication, most recently in 2009. Under 4th PSR, protection required in the event of snow loads is determined based on the updated 2009 regulations. EDF concludes that all SSCs comply with the 2009 requirements.

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 fixed industrial facilities such as storage sites and off-site production facilities, on- and off-site pipelines, and 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. In the framework of the 4th PSR EDF revised the data on industrial facilities in the surrounding area and re-assessed the associated hazards. Analyses revealed probabilities for hazards caused by explosion, intoxication by industrial releases and external fire between 10^{-8} and 10^{-9} per year revealing no necessities for retrofitting (EIA-REPORT D3/4, P.2 2026, p. 187/179).

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 parameter values, environmental data specific to the 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 Dampierre site limit due to air traffic is less than 10^{-6}/reactor year for each of the three safety functions. Probabilities for each of the three aircraft families (general aviation, commercial aviation and military aviation) are on the order of 10^{-7} per year at most (EIA-REPORT D3/4, P.2 2026, p. 189/181).

Upgrades of protection measures: Safety upgrades that have already been completed and planned safety upgrades are described in EIA-REPORT D3/4, P.2 (2026) comprehensively and listed in the Annexes of the cited document. Table 1 of the current report lists the measures relevant for external hazards.

As a general measure to strengthen the protection of 900 MW reactors, EDF plans to achieve safety improvements by installing a Hardened Safety Core (Noyau Dur) to increase the robustness of the NPP against hazards such as earthquakes, tornadoes and external flooding. Accoring the the EIA documents, all but two measures for implementing the Noyau Dur have been completed so far. In addition to this general measure, the EIA documents list a number of specific improvements including the following measures to protect the NPP from external hazards:

Project number Reactor Topic Title Status

Table 1: Upgrading measures for SSCs important to safety with respect to external hazards, Dampierre 3 and 4 (adapted from EIA-REPORT D3/4, P.2 2026)

Project Number	**Reactor**	**Topic**	**Title**	**Status**
PNPE1068	3 4	Noyau Dur	Installation of an electrical distribution system with a "Noyau Dur" core	Completed
PNPE1069	3 4	High temperature	Improvement of the air conditioning for the rooms of the cooling water makeup unit	Completed
PNPE1070	3 4	High temperature	Improvement of the air conditioning for the rooms of the electrical buildings	Completed
PNPE1073	3 4	Noyau Dur	Implementation of a "Noyau Dur" control system for the existing facilities	Completed
PNPE1118	3 4	Earthquake	Seismic resistance of ventilations for rooms with heat exchangers	Completed
PNPE1191	3 4	Earthquake	Seismic resistance of cable trays	Completed
PNPE1289	3 4	Noyau Dur	Establishment of a water source for the make-up supply of the "Noyau Dur"	Completed
PNPE1333	3 4	Noyau Dur	Seismic resistance of the hardened core of the primary cooling circuit including fixing	Completed
PNPP1419	3	Earthquake	Implementation of automatic reactor shutdown	Completed
PNPP1666	3 4	Noyau Dur	Emergency diesel generators	Completed
PNPP1675	3 4	External flooding	Protective measures against extreme flooding caused by direct overtopping onto the reactor platform	Completed
PNPP1679 (tome B)	3 4	Earthquake	Seismic reinforcement of certain facilities of the fuel pool	Completed
PNPP1688 (tome C)	3 4	Noyau Dur	Installation of a control and monitoring system for the core area of new facilities	Completed
PNPP1719	3 4	Heat sink	Installation of a level sensor downstream of the filtration system to trigger the feed pumps, ensuring preparedness for the impact of a massive influx of clogging material	Completed
PNPP1870	3 4	Earthquake	Seismic reinforcement of the U5 container's venting system for SMHV-level earthquakes	Completed

Project Number	Reactor	Topic	Title	Status
PNPP1883	3 4	External flooding	Close-range protection of the core against external flooding	Completed
PNPP1917	3 4	Heat sink	Replacement of the pre-filter screens to increase resistance against clogging of the water intake	Completed
PNPP1951	3 4	Lightning	Installation of surge protection devices	Completed
PNRL1823	3 4	High temperature	Replacement of the motors for diesel air coolers	Completed
PNRL1827	3	External flooding	Elimination of risks of bypassing volumetric protection (installation of flaps, reinforcement of hoppers, various control measures)	Completed
PNRL1835	3 4	High temperature	Updating the parameters for automatic fouling monitoring of heat exchangers	Completed
PNRL1846	4	External flooding	Elimination of the risk of bypassing volumetric protection in the pumping station	Completed
PNRL1933	3 4	Earthquake	Seismic resistance of boron makeup	Completed
PNRL1955	3 4	Low temperature	Change to the setpoint setting of the air condition air heaters	Completed
PNPE1115	3 4	Earthquake	Automatic reactor shutdown command upon earthquake and notification of significant earthquakes: robustness against SND loads	Completed
PNPE1119	3 4	Tornado	Passive protection of the reactor building against tornado and replacement of exhaust air ducts	Completed
PNPE1305	3 4	Earthquake	Implementation of an earthquake-resistant detection system for failure of the emergency cooling system	Completed
PNPE1336	3 4	Earthquake / Low temperature	Modifications to ensure the availability of the borated water piping after earthquake and during the cold season	Completed
PNPE1347	3 4	Earthquake	Replacement of certain electric actuators and control cabinets (resistance against SND)	Completed
PNPE1358	3 4	Earthquake / Tornado	Resilience of "Noyau Dur" ventilation systems against SND and tornadoes	Completed
PNPP1824	3 4	Noyau Dur	Addition of an analog level measurement chain for the fuel pool	Completed
PNRL1803	3 4	Noyau Dur	Installation of a "Noyau Dur" water makeup system at the reactor building pool and its steam outlet	Completed

Project Number	Reactor	Topic	Title	Status
PNRL1984	3 4	External flooding	Fastening device for anchoring a flexible connector to the fuel building inlet for refilling during high water, extending beyond the "Noyau Dur"	Completed
PNRL1990	3 4	Earthquake	Enhanced resistance of the fire protection network	Completed
PNPE1427	3 4	Noyau Dur	Implementation of an injection pump on the seals of the primary pump units "Noyau Dur"	Phase B (1)
PNPE1459	3 4	Noyau Dur	Improvement of long-term cooling for specific areas within the electrical and "survival island" in the event of a cooling source failure	Phase B (1)
PNPE1477	3 4	Lightning	Installation of surge protection in the secondary circuit of the auxiliary transformers	Phase B (1)
PNPE1323	3 4	Earthquake / Tornado	Reinforcement of the vent stack against SMS, strong winds, and tornado	Phase "Supplements" (2)
PNPE1377	3 4	Earthquake	Upgrading the seismic safety of the U5 containment venting system to SMS level	Phase "Supplements" (2)
PNPP1722	3 4	Low temperature	Installation and thermal insulation of the supply line to the emergency power supply in the machine room	Phase "Supplements" (2)

(1) Changes whose implementation will be completed as part of Phase B of 4th PSR are due on June 2029 for the reactor Dampierre 3 and April 2030 for Dampierre 4 (EIA-REPORT D3/4, P.2 2026, p. 14).

(2) Deadlines for implementation in the Supplementary Phase are March 2028 for Dampierre 3 and April 2030 for Dampierre 4.

Earthquake: The attachments of EIA-REPORT D3/4, P.2 (2026) list numerous actions related to the earthquake resistance of SSCs of Dampierre 3 and 4 (Table 1). 12 out of 13 upgrading / retrofitting measures have been completed by now, completion of the pending upgrade is scheduled for Phase "supplements" of the 4th PSR which is due 5 years after the baseline PSR. This is 27 June 2029 for Dampierre 3 (EIA-REPORT D3, P.2 2026, p.14) and April 2030 Dampierre 4 (EIA-REPORT D4, P.2 2026, p.14).

External flooding: Flood protection is increased by re-enforced enhancing volumetric protection and increase of heights of sills and sheet pile walls for flood protection (EIA-REPORT D3/4, P.2 2026). All measures have been implemented.

High temperatures: EIA-REPORT D3/4, P.2 (2026) list the following measures as implemented: replacement of motors for air cooling systems, modification of cooling systems for the electrical buildings and the buildings for cooling water makeup, and modifications of the software for the automatic monitoring of pollution of heat exchangers.

Low temperatures: EIA-REPORT D3/4, P.2 (2026, p. 172/164) identify several SSCs as vulnerable against extremely low temperatures. The results of the 4th PSR gave rise to a number of additional safety measures such as organizational measures and measures to improve the insulation of vulnerable pumps, piping and tanks.

High wind and tornado: Measures include re-enforcement of the vent stacks of Dampierre unit 3 and 4 (EIA-REPORT D3/4, P.2 2026, p. 179/172).

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) in the water intakes and managing the risk of sedimentation/siltation by sensors to monitor water levels behind the filtration devices. The hazard of silting of the channel connecting the NPPs with the Loire River is managed by regular bathymetric controls and dredging of the channel

Threats to the cooling water by oil spill are not considered due to the very low probability of such events.

Lightning and electromagnetic interference: Assessments for the Dampierre reactors led to the implementation of surge arrestors and surge protection (EIA-REPORT D3/4, P.2 2026, p. 181/174).

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.

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 the Dampierre reactors.

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 include 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 importantly 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 so 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 demonstrations (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 in detail. Conformity with WENRA requirements and guidance therefore cannot be assessed. It remains particularly unclear if design basis events with exceedance frequencies not higher than 10^{-4} per annum or design basis events that provide an equivalent level of protection 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. In this context it appears remarkable that the EIA documents refer to WENRA

(2008) and WENRA (2014)[8]. The cited editions are outdated by the SRLs published in 2021.

Conformity with WENRA Reference Levels still needs to be proved for earthquake and seismic ground shaking. The Design Basis Earthquakes (DBE) for Dampierre 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. 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, may be 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 fourth DiD level system which is required as an additional and independent level compared to the third DiD level. The Hardened Safety Core therefore could 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 methodology was originally stipulated in RFS 1.2.C (1981)[9] suggests that design basis values were only established after the start of construction of Dampierre 3 and 4 (Table 2:). 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 reactors on the respective nuclear islands. All plants of a specific series consequently

[8] WENRA Safety Reference Levels (SRLs) for Existing Reactors

[9] 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.

share the same seismic design. Other structures, referred to as "site structures", were specifically designed for each site (Table 2:).

In 2001 the RFS 1.2.C (1981) was replaced by RFS 2001-01[10]. 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 therefore recommended introducing probabilistic methods (PSHA) to determine design basis earthquakes (ENSREG 2012). The French National Action Plan (NAcP) consequently announced that probabilistic methods are to be used to determine the site-specific seismic hazard.

The EIA documents provide some information on the derivation and reassessment of the deterministically design basis earthquake SMS. A reassessment in the course of the 4th PSR confirmed ground motion values. The EIA documents provide no information on the content, methodology and results of the seismic hazard reassessment.

The EIA documents include neither numerical information on the deterministically derived SMS (design basis earthquake) ground motion nor on the ground motion values for a 20,000-year recurrence interval relevant for the SND. Available documents by ASN (2011a) and IRSN (2012; 2013) indicate an SMS ground motion value (PGA) for the Dampierre nuclear island and site structure of 0.2 g and 0.1 g respectively. PGA values reported for the SND are 0.25 g. (Table 2).

Table 2: Design basis ground motions (peak ground acceleration) of the reactors Dampierre 3 and 4 according to (ASN 2011a) and ground motion for the SND (Séisme Noyau Dur; IRSN 2012; 2013).

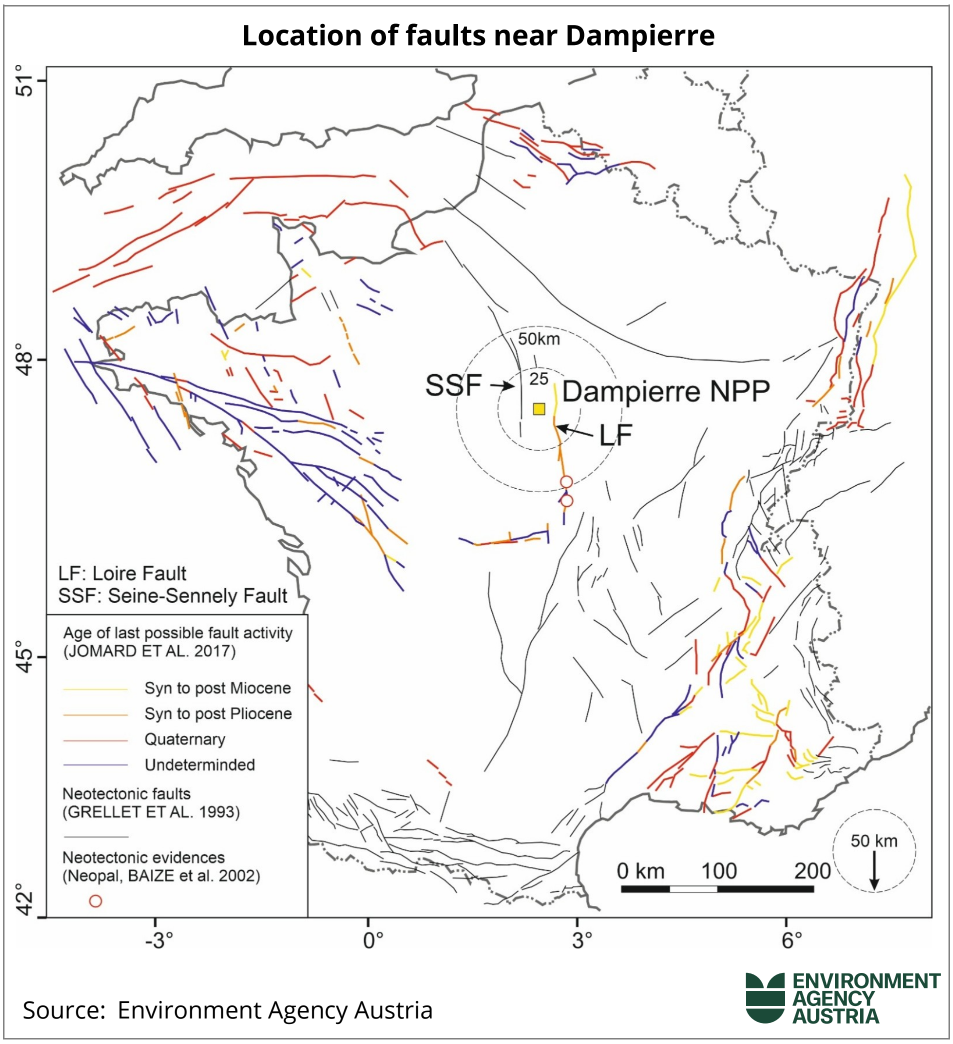
NPP	Dampierre 3	Dampierre 4
Start of construction	01/09/1975	01/12/1975

[10] 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.

Start of commercial operation	27/05/1981	20/11/1981
DBE Nuclear island	0.2 g(1)	0.2 g(1)
DBE Site structure	0.1 g(1)	0.1 g(1)
SND	PGA=0.25 g(2)(3)	PGA=0.25 g(2)(3)

(1) ASN 2011a (2) IRSN (2012) (3) IRSN (2013)

Figure 3: French active fault database and location of the Dampierre reactors (redrawn from: JOMARD et al. 2017; RITZ et al. 2021). Circles around Dampierre indicate the site near-region and site region according to IAEA (2022) (radius 25 and 50 km from the site, respectively).



In tectonic terms, the Dampierre site is located in the Paris Basin, a sedimentary basin filled with more than 3 km thick sediments of Permian to Neogene age (e.g., GUILLOCHEAU et al. 2000). The basin is crossed by a number of faults including the Loire and Seine-Sennely fault systems which pass the NPP site at distances of less than about 25 km (Figure 3).

For the Loire fault, JOMARD et al. (2017) infer a syn- to post-Pliocene age for the latest fault movement (ca. 5 to less than 2.6 million years). The NEOPAL database (BAIZE et al. 2002)[11] lists two locations at or near the Loire fault for which geological data indicate late to post-Pliocene faulting (Figure 3; CLOZIER & GROS 1985). According to the IAEA (2022), seismic hazard analysis in areas of low seismicity, such as the Paris Basin, should take into account faults of this age.

EIA-REPORT D3, P.2 (2026, p. 134) and EIA-REPORT D4, P.2 (2026, p. 134) provide evidence that EDF is aware that geological and tectonic data prove the existence of several faults in the near-region[12] and region around the Dampierre site, naming the Sennely, Etampes, Sully-sur-Loire and Sancerre fault. EDF already started investigating these faults by literature review and mapping of industrial reflection seismic and reports plans to further investigate the faults with geophysical and paleoseismological methods (EIA-REPORT D3/4, P.2 2026, p. 155/148). These plans are appreciated. EIA documents, however, do not include a tangible time schedule for the investigations announced in the EIA documents.

For active faults in the near-region of the NPP site WENRA (2020b p.11ff) suggests systematic fault mapping and collecting paleoseismological 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. At the background of the existing literature (JOMARD et al. 2017; RITZ et al. 2021 and references therein) it is remarkable that a process for fault characterization and paleoseismological investigations has not been implemented earlier, i.e., in the course of 4th PSR. WENRA (2021) requires *"the actual design basis shall regularly, and when relevant as a result of operating experience and significant new safety information, be reviewed"* (Reference Level E11.11).

Meteorological hazards: The EIA documents provide evidence that hazard assessments for at least some of the meteorological hazards considered in the 4th PSR 900 do not meet the WENRA (2021) requirements. Temperatures are determined by exceedance probabilities of 10^{-2} per year and 70% confidence. The EIA documentation contained no information as to whether climate trends were taken into account in the calculations and the values are extrapolated accordingly. The basis for considering snow loads seem to be common building codes (EIA documents

[11] https://data.oreme.org/fact/fact_map

[12] Near-region: <25 km from the site; region: <50 km from the site according to IAEA (2022)

are not particularly clear on this). The assessment of meteorological hazards is admittedly cumbersome due to short data rows and the non-stationary processes introduced by climate change. However, defining design bases for temperature extremes on design basis events with an occurrence probability of 10^{-2} (!) and 70% (!) confidence only is not appropriate and contradicts the requirements of WENRA (2021) Issue TU, Reference Level TU4.2.

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 NPPs 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 NPPs 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 NPPs, such as the Dampierre 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 melt accident in new NPPs 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 Dampierre 3 and 4: A study uses numerical simulations to investigate the influence of aging on the effects of a military aircraft impact on a NPP. 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 NPPs 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 2026a)

4.3 Conclusions

The EIA documents provide ample information on hazard types considered in the safety demonstration for Dampierre 3 and 4 and measures already implemented or decided to be implemented in order to strengthen the robustness of the reactor with respect to external hazards. The documents, however, do not provide clear evidence if the processes of the PSR and LTO 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 therefore cannot 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 Dampierre and the other reactors of the French 900 MWe fleet are still based on deterministic analyses. Defining the DBE on deterministic methods is no longer state of the art. EIA documents do not demonstrate that the deterministically determined DBE is in line with a DBE derived from a PSHA with an average recurrence interval of 10,000 years. Neither information on ground motion values of the currently valid SMS (design basis earthquake) nor on the ground motion of the SND is provided. ASN (2011a) and IRSN (2012; 2013) indicate an SMS ground motion value (PGA) for the Dampierre nuclear island and site structure of 0.2 and 0.1 g, respectively. PGA values reported for the SND are 0.25 g. We conclude that it 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 Dampierre site is located near the late to post-Pliocene Loire fault system (EIA-REPORT D3/4, P.2 2026; JOMARD et al. 2017). The respective faults have not been properly investigated to decide about their activity or inactivity although they are known for about 10 years at least (JOMARD et al., 2017; CLOZIER & GROS, 1985). According to EIA-REPORT D3/4, P.2 (2026), EDF is envisaging a geophysical and paleoseismological investigation program to evaluate the safety significance of faults near Dampierre. Tangible time schedule for these investigations is not communicated. An updated PSHA-based reassessment of seismic hazards that takes advantage of fault data is not announced.

For active faults in the near-region of an NPP site WENRA (2020b p.11ff) suggests systematic fault mapping and collecting paleoseismological information. At the background of the existing literature (JOMARD et al., 2017; RITZ et al. 2021 and references therein) it is remarkable that a process for active fault characterization has not been implemented earlier in the course of the 4th PSR.

The authors of this report assume that adequate protection against a probabilistically derived DBE, should it be higher than the deterministic SMS, 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 events. The Hardened Safety Core is a 4th DiD level system which is required as an additional and independent level compared to the 3rd DiD level. The Hardened Safety Core therefore could not be used to compensate for possible deficits in terms of the protection against design basis events.

With respect to safety upgrades of Dampierre 3 and 4, 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 not fully completed yet. The attachments of EIA-REPORT D3/4, P.2 (2026) list two measures related to the implementation of the Noyau Dur which are pending and should be completed in Phase B of the 4th PSR. The related deadlines are 27 June 2029 for the reactor Dampierre 3 and April 2030 for Dampierre 4, respectively (EIA-REPORT D3/4, P.2 2026, p.14).

The fundamental decision to implement the Hardened Safety Core has been made in 2012 in the aftermath of the European Stress Tests (ASN 2012). The fact that the implementation of the Noyau Dur will be completed only 18 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 contradict 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 documents 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 reactor building as well as the spent fuel building of the Dampierre 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 Dampierre 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”.

- With respect to seismic safety, the following information should be provided:
  - Methods, data and assumptions used for the PSHA performed to determine the SND for Dampierre 3 and 4, 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.
  - The actual PSHA-derived ground motion values for the SMS and the SND.
- The deterministically derived SMS ground motion should be benchmarked against a PSHA-derived ground motion for a recurrence interval of 10,000 years as required by WENRA (2021).
- Additional safety demonstrations should be required to ensure that all SSCs relevant to safety can cope with a probabilistically derived DBE for an occurrence probability of 10^{-4}/year in case the probabilistically derived DBE exceeds the ground motion parameters of the current SMS of Dampierre 3 and 4.
- The methods, data and assumptions used to determine the SND for Dampierre 3 and 4, 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 should be reviewed. The PSHA should be benchmarked against WENRA requirements (WENRA 2021) and recommendations (WENRA 2020a, b).
- Dedicated assessments of near-regional potentially active faults of the Loire fault system should be performed in line with WENRA

(2020b). The approach should include field geology, geophysical mapping, morphostructural and dating studies, and paleoseismology. Background: EIA documents and existing literature indicate a late to post-Pliocene age for the faults in the near-region (>25 km) and region (<50 km distance) from the site.

- It should be ensured that design basis events and design basis parameters defined for meteorological hazards conform with WENRA (2021) requirements. Background: this seems not to be the case for some meteorological hazards, in particular, extremely high temperatures which are determined for exceedance probabilities of 10^{-2} per year and 70% confidence.
- It should be ensured that the use of the Noyau Dur's DEC equipment is not required to protect the Dampierre 3 and 4 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.
- With respect to possible terror attacks, the following questions should be addressed:
  - 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 (NTI 2025) 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 2 on Procedure, the 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 D3/4, P.1 (2026), EIA-REPORT D3/4, P.2 (2026), 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.

Key measures for accidents without core melt

EIA-REPORT D3/4, P.1 (2026) and EIA-REPORT D3/4, P.2 (2026) provide executive summaries and outline the highest-priority measures identified for implementation regarding Accidents Without Core Melt.

Measures Implemented:

Accidents-1: <u>Reinforcement of water resources for reactor heat removal — refilling the ASG tank via the fire-fighting network</u>

Modification: Provision for refilling the emergency feedwater tank for the steam generators — the ASG tank — using the fire-fighting network.

Rationale: Updated thermohydraulic studies using more conservative assumptions showed that, in certain accident situations — steam-line break accidents or steam-generator tube rupture accidents — the water available in the ASG tank is no longer sufficient to bring the reactor back to a safe state; EDF therefore aims to increase available water resources.

Accidents-2: <u>Increase in the flow capacity of atmospheric steam-dump valves — GCTa modification</u>

Modification: Uprating of the mass flow capacity through the Main Steam Line Safety and Relief Valves. The atmospheric steam discharge capacity of the GCTa valves is increased by modifying the internal structure of the GCTa valve.

Rationale: In accident situations, this increased discharge capacity is intended to cool the reactor more quickly, reduce the duration of the accident, and limit any associated radioactive releases. 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>Lowering of the iodine-131 radioactivity limit in the primary-circuit water</u>

Modification: By operational measures, the maximum iodine-131 concentration in the primary-circuit water during power transients is lowered from 150 to 80 GBq/t.

Rationale: This measure is intended to reduce the activity of possible radioactive releases and their radiological consequences, particularly thyroid doses, for all accidents without fuel-cladding rupture, including the most penalizing steam-generator tube rupture accident.

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.

Key measures for the spent fuel pool

For the spent fuel pool EIA-REPORT D3/4, P.1 (2026) and EIA-REPORT D3/4, P.2 (2026) list the following items:

Measures proposed:

Pool-1: Fire protection — flame-arresting/fire-barrier device

In the event of a fire, and in order to avoid the loss of the two cooling trains, EDF has planned the addition of a flame-arresting/fire-barrier device to prevent the propagation of a fire from one cooling-circuit pump to the other.

This measure is marked as proposed for both Dampierre 3 and Dampierre 4 reactor.

Measures implemented:

Pool-2: Duplication of the automatic isolation device on the suction line of the normal pool cooling circuit

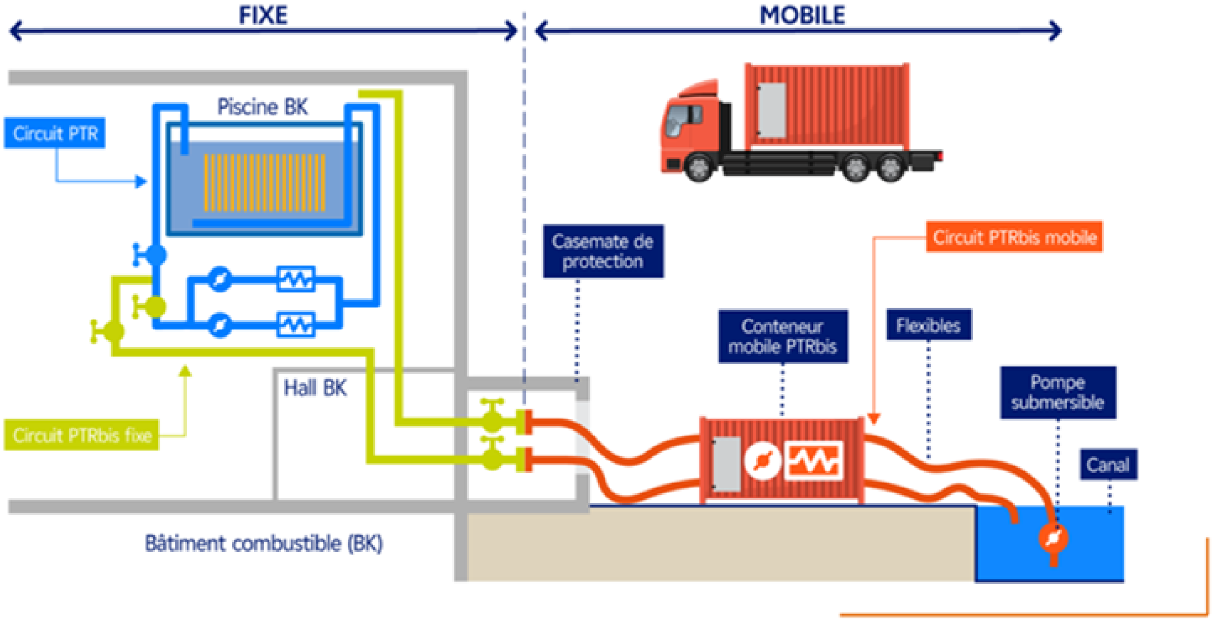
Following behaviour of fuel pools of the 900 Mwe reactors against accident scenarios considered for EPR FLA3, to further secure spent fuel pool cooling, carries out the duplication of the automatic isolation device on the suction line of the pool's normal cooling circuit. This should ensure reliable isolation under accident conditions even if one device fails.

Pool-3: Additional diversified mobile pool cooling system — PTR-bis

As part of the post-Fukushima measures, the diversified water source SEG allows water make-up to the pool in the fuel building. During 4th PSR, a new diversified mobile pool cooling means, PTR-bis, diversifies the cold source and, in the event of loss of the normal operating cooling circuit, allows a return to spent-fuel-pool cooling without boiling. This type of provision should bring the design of 900 MWe reactors closer to that of EPR FLA3-type reactors.

Figure 4: Schematic diagram of the cooling circuit of the spent fuel pool (EIA-REPORT D3/4, P.1 2026)

Schematic of the cooling circuit of the spent fuel pool



Source: EIA-REPORT D3/4, P.1 2026

EIA-REPORT D3, P.2 (2026) and EIA-REPORT D4, P.2 (2026) represent the most extensive of the five reports submitted for the LTO 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 Dampierre 3 / Dampierre 4 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 only partly relates to the Spent Fuel Pool.

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 covers operational transients, Design Basis Accidents (DBA), and Design Extension Conditions (DEC) with deterministic and probabilistic analyses, with the objective of reducing radiological consequences and aligning older units toward the EPR safety profile.

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 Accident-type events / reactivity-related events

- Withdrawal of a regulating rod cluster at power
- Control-rod ejection accident
- Uncontrolled withdrawal of control-rod groups during start-up
- Uncontrolled withdrawal of control-rod groups at power
- Mispositioning, drop of a rod cluster, or drop of a group of rod clusters
- Uncontrolled boric-acid dilution

Thermal-hydraulic and heat-removal transients

- Partial loss / 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 (OISS)
- 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 primary break with diameter ≤ 2.5 cm
- Intermediate-break LOCA, category 4
- Inadvertent actuation / start-up of the safety-injection system
- Inadvertent opening of a pressurizer safety valve
- Steam Generator Tube Rupture (SGTR) category 3

- Category 4 SGTR, i.e. steam-generator tube rupture combined with a stuck-open secondary valve

Equipment, electrical-supply and operational failures

- Total loss of external electrical power supplies
- Locked rotor of a primary motor-pump / reactor coolant pump

Fuel and core-configuration events

- Class 2 power capability check, used to verify the proper sizing of reactor protections
- Fuel assembly mispositioning in the core
- Fuel-handling accident
- Irradiated-fuel container handling accident

The reports distinguish between modifications already deployed and modifications to be deployed in Phase B or in the Supplementary Phase of the 4th PSR as well modification of the specific program.

Accidents Without Core Melt

The following lists show the adaptions implemented or planned for both units Dampierre 3 and 4.

Dampierre 3 and 4

Fully implemented:

- PNPE1141 — Increase in the flow rate of GCT-a regulating valves
- PNPP1864 — Creation of a fixed means of refilling the ASG tank by the fire-protection systems
- PNPP1595 — Modification / replacement of SEBIM valve heads, including Vol. A and B for Dampierre 4
- PNRL1817 — Improvement of the measurement system by adding a filter to the average temperature signal (SIP-C)
- PNPP1838 — RPN renovation: new architecture and functionalities for nuclear-power measurement
- PNPP1873 — Evolution of SIP-P process instrumentation and reparameterization of RPR thresholds
- PNRL1829 — Increase in required REA boron volume and increase in free TEP volume
- PNPE1152 — Substitution of the emergency turbo-alternator power supply by the Ultimate Emergency Diesel

- PNPE1166 — Electrical architecture allowing substitution by / connection to the Ultimate Emergency Diesel of the neighbouring unit
- PNPP1371 — Reliability improvement of the isolation of GMPP thermal barriers
- PNPP1811 — EAS-ND system for primary-water injection and residual-power removal from the reactor containment
- PNRL1879 — Replacement of electrical contacts for fans of the electrical-building ventilation system by latching contactors
- PNRL1954 — Safety strapping on insulation of pipes connecting the safety-injection accumulators to the main circuit and on the pressurizer expansion line
- PNRL1946 — Replacement of Microtherm® microporous insulation
- PNRL1947 — Replacement of Protect 1000S fibrous insulation on auxiliary lines of diameter 50 mm or more that could release fibers in the event of a primary-system break,
- PNPP1797 — Installation of a boron-meter on the discharge of the RCV chemical and volume control system.

Planned for Phase B /Supplementary Phase / Specific Program:

- PNPE1359 — Increase in RIS accumulator pressure
- PNRL1957 — Lowering of the group R blocking line following integration of water gaps into the safety studies
- PNPE1189 — Addition of a primary-fluid sampling device in shutdown state downstream of the CEPP heat exchanger / primary pump sealing circuit, addressing heterogeneous dilution risks due to CEPP leakage
- PNPP1932 — Installation of a tapping point on the double envelope of the RIS safety-injection and EAS spray systems for endoscopic inspection
- PNPE1442 — Operational modification concerning accessibility and use of self-checking cells to operate RCV-RIS from the BL

Spent Fuel Pool – BK (Bâtiment combustible)

The following lists show the adaptions implemented or planned for both units Dampierre 3 and 4 according to implementation status or implementation phase.

Dampierre 3 and 4

Fully Implemented:

- PNXX1752 — Analog level measurement of the spent-fuel storage pool
- PNPP1289 — Resizing of the siphon breaker on the discharge line of the spent-fuel-pool cooling system
- PNPP1401 — Installation of a second static seal on the reactor-building pool gate
- PNPP1402 — Automatic closure of the PTR suction valve in the spent-fuel storage pool on very low level
- PNPP1403 — Motorization of the PTR transfer-tube valve
- PNPP1474 — Wide-range pressure measurement for RIS accumulators
- PNPP1549 — Safe positioning of a fuel assembly
- PNPP1666 — Ultimate Emergency Diesel / Diesel d'Ultime Secours (DUS)
- PNPP1679 — Seismic reinforcement of on/off level measurements for the spent-fuel pool
- PNPP1780 — Automation of reactor-building pool drain valves
- PNPP1907 — Creation of a diversified mobile cooling system, PTR-bis (excluding Vol. N for unit 3)
- PNPE1289 — Source of water for the Noyau Dur makeup
- PNPE1344 — Duplication of the automatic isolation of the spent-fuel-pool suction line by PTR valves
- PNPE1443 — Reinforcement of a hopper / opening in the spent fuel pool
- PNRL1895 — Reliability improvement of the transfer-tube valve control for closure under flow

Planned for Phase B /Supplementary Phase / Specific Program:

- PNPE1128 — On/off level measurements for the reactor-building pool
- PNPP1824 — Addition of an analogue level-measurement chain for the spent-fuel pool
- PNPP1949 — Installation of a fire-protection screen between the PTR pumps for physical separation of the two PTR trains
- PNRL1984 — Anchoring arrangement for securing a flexible plug to provide makeup to the spent-fuel storage pool in the event of river flooding beyond the Noyau Dur level

- PNPE1258 — Installation of the ASG-ND device and fixed line for replenishing the spent-fuel storage pool by SEG; to be deployed under specific programming no later than Phase B
- for unit 3 only: PNPP1907 vol. N —PTRbis discharge and Noyau Dur makeup from the top of the spent-fuel storage pool; to be deployed under specific programming no later than Phase B

Document EIA-REPORT D3/4, P.3 (2026) provides easy-to-use lists of measures but no new information in respect to EIA-REPORT D3/4, P.1 (2026) and EIA-REPORT D3/4, P.2 (2026). EIA-REPORT D3/4, P.4 (2026) presents EDF's lessons learned from the public consultation on the generic phase of the 4th PSR of the 900 MWe reactors, which took place from September 2018 to March 2019. EIA-REPORT D3/4, D.5 (2026) provides relevant snippets from the French Environmental Code in the context of a periodic safety review.

5.2 Discussion

Generic aspects

Accidents-1: Reinforcement of water resources for reactor heat removal — refilling the ASG tank via the fire-fighting network

The installation of refilling the ASG tank via the fire-fighting network is a recognized and valuable enhancement. This measure aligns with post-Fukushima accident safety upgrades implemented across numerous 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 supplied excerpts do not provide the initial and final GCT-a mass-flow rates, nor a comparison between the upgraded

GCT-a discharge capacity and normal steam-flow values. Therefore, a quantitative assessment of the magnitude of the capacity increase cannot be made from the provided excerpts alone.

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. A 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: <u>Reduced Primary System I-131 Limit</u>

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 provided excerpts do not detail the operational monitoring or enforcement methodology for the new iodine-131 limit. They also do not state whether iodine-spiking effects were explicitly considered in the associated design-basis studies or operating procedures. 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: <u>Installation of Flame Shields in the SFP Building</u>

The proposed installation of flame shields 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.

Pool-2: <u>Duplication of the automatic isolation device on the suction line</u>

The improvement appears safety-positive in principle because it adds redundancy to a pool-cooling suction-line isolation function, but the provided excerpts leave unclear how the duplication is implemented, how independent it is, what accident signals actuate it, whether it can create spurious-isolation or cooling-availability issues, and how those potential trade-offs are managed.

Pool-3: <u>Mobile Cooling Capabilities</u>

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 increases the capability of the long-term cooling and inventory control of the SFP under Design Extension Conditions (DEC) and has been implemented.

Further measures

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])

Under prescription [Étude-B], EDF is required to evaluate, by an experimental approach, the validity of the critical heat flux correlation used at the periphery of deformed fuel assemblies and to define the work programme and associated schedule for incorporating the lessons learned.

For Dampierre 3 the evaluation and programme definition were to be completed no later than 31 December 2024, and that EDF submitted a detailed programme of test configurations to ASN in June 2021. Information on whether the experimental evaluation was completed after that submission is not provided.

For Dampierre unit 4 EDF has evaluated the validity of the correlation experimentally and has defined the work programme and associated schedule.

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

Under prescription [Étude-D], EDF performed tests to characterize the buckling limit of fuel assembly grids in a configuration more realistic than the historical test bench. The results were used to assess fuel-assembly mechanical behaviour during a category 4 primary-coolant loss accident combined with a concurrent earthquake. This evaluation confirmed that neither core cooling capability nor the control of reactivity via control rod drop were compromised.

This assessment does not call into question the capability to cool the core or to control reactivity by control-rod drop. The findings must be taken into account in the relevant safety reports by the deadline for the TTS of the 5th PSR of the 900 MWe series.

5.3 Conclusions

While the 4th PSR for Dampierre 3 and 4 has delivered substantive, EPR-aligned safety enhancements and is being deployed through the established generic and plant-specific phases, confidence in the outcome would be further strengthened by concise clarifications ahead of the detailed points that follow. In particular, it is helpful to - sharpen transparency on quantitative inputs and margins used in the updated accident studies that underpin recent and planned modifications; provide tranche-specific delivery dates within the phased program and clearly report closure of remaining VD4 actions; and concisely justify deferred items and interim protections while standardizing status reporting to distinguish installation from commissioning for robust closure tracking.

1. Enhance Transparency and Provide Clarity on Key Quantitative Data

- Quantitative Data: The documents identify PNPE1141 as an increase in the flow rate of the GCT-a regulating valves and state that it contributes to resolving an ASG water-consumption study anomaly. However, the excerpts provided do not give the initial and final GCT-a mass-flow rates or compare the upgraded capacity with nominal operating steam flows. Providing these values would help quantify the magnitude of 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 (SGTR). 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. A concise clarification would therefore be useful.

2. Establish Firm and Accountable Timelines

- Study Status and Next Steps: The documents identify the critical heat-flux correlation and the treatment of fuel-assembly deformation as important assumptions in 4th PSR accident studies. They also state that further elements concerning fuel-assembly deformation impacts are to be provided in Phase B. The provided excerpts do not give a clear status of any associated experimental programme, results, or integration schedule. A status update identifying completion, results, and planned integration would improve traceability.

3. Clarify Status Reporting and Implementation Rationale

- Resolve Discrepancies and improve transparency by status tracking for implemented and pending measures: The reports list a mixture of already deployed modifications, Phase B modifications, and modifications under specific programming. EDF also states that it will present annually the dispositions implemented during the previous year and those remaining to be carried out, with their programming. To strengthen transparency and closure tracking, it would be useful to systematically provide unit-specific completion dates and status fields for each outstanding measure, distinguishing installation, commissioning, operational availability, documentary update, and final closure.

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 main 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

In the event of a core melt accident, the corium spreads after breaking through the reactor pressure vessel (RPV) in the reactor building vessel well and in the room of the reactor core instrumentation (RIC room). To limit the risk of losing the containment integrity 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)[13]. 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,
- the conclusions of its test-based investigation program on the behaviour of basement in the event of core melt accidents.

[13] PNPP1976: "Installation of a device for dry distribution and stabilisation of the corium under water" has been implemented already.

EDF has identified the sites where the basements need to be reinforced. In accordance with ASNR regulation [AG-A-II] the thickening of the basement in the containment room and in the adjacent RIC room has been performed at the units 3&4 of the Dampierre NPP.

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 bottom of the reactor building in order to avoid any risk of corium penetration (PNPP1976 Volume B / PNPE1460)[14].

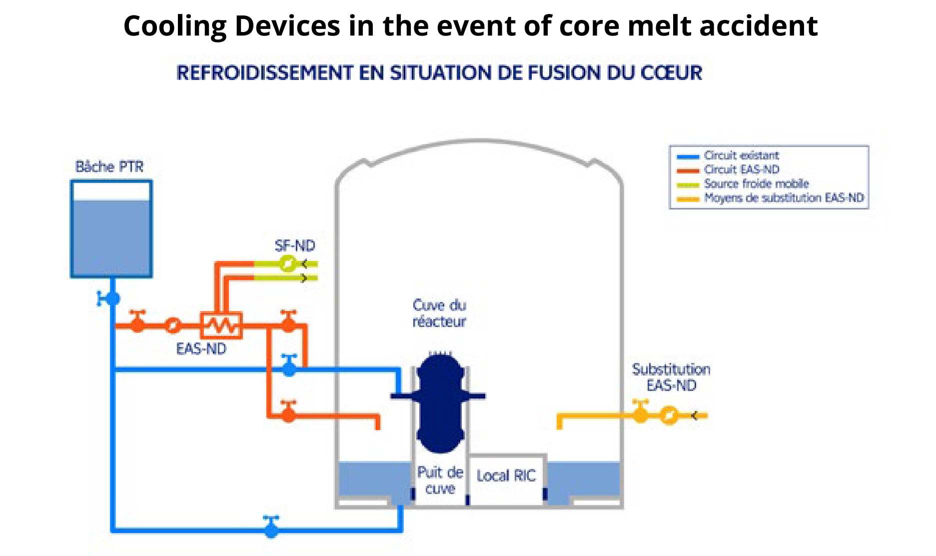
The dry distribution of the corium is ensured by the prior sealing of the containment room and the adjacent RIC room. The corium is then drowned by gravity with the water present in the sumps at the bottom of the reactor building filled by the safety injection systems (SIS), the sprinkler system (EAS) or the “Hardened Safety 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)[15] 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.

[14] PNPP1976 Volume B / PNPE1460: "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 as part of the Supplementary Phase of the 4th PSR.

[15] PNXX1746: “Detection of breaches and operation of the hydrogen recombinator at high temperatures” has been implemented.

Figure 5: Cooling devices in the event of core melt accident (EIA-REPORT D3/4, P.1 2026)



Source: Environment Agency Austria

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)[16]. 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)[17].

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)[18] will be implemented.

Probability of an accident caused by internal events

According to the EIA-REPORT D3/4, P.2 (2026), the annual frequency of breakthroughs in the basement was estimated at around 10^{-6} / year at the end of the 3rd PSR. With the measures for dry distribution and

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16 PNPE1362: "Installation of fixed injection and extraction lines in the reactor building and mobile replacement device for EAS-ND – return of water from the fuel pool to the reactor building" will be implemented as part of the Supplementary Phase of the 4th PSR.

17 PNPE1386: "Installation of a measuring point for the sump in the reactor building" will be implemented as part of the Supplementary Phase of the 4th PSR.

18 PNPE1387: "Device for detecting corium spread in the RIC room" will be implemented in Phase B of the 4th PSR.

refilling of the corium (PNPP1976)[19], along with the EAS-ND system (PNPP1811)[20] and the SF-ND, which in particular enables long-term cooling of the corium, the probability of a breakthrough of the basement is reduced to a frequency of a few 10^{-7} / year, which is in line with the objective of avoiding lasting effects on the environment.

The frequency of early releases has been reduced by a factor of 2. This improvement is largely attributable to the DUS supplying power to the reactor pressure vessel’s electrical safety shut-off valves. The risk of early releases is in the order of a few 10^{-7} per year per reactor, making the risk of early and significant releases extremely unlikely.

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)[21] 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 melt accident, 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 tank of the water treatment and cooling system of the pools (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).

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.

[19] PNPP1976: see above

[20] PNPP1811: ”Introduction of an EAS-ND system for water injection into the primary circuit and for dissipating residual power” has been already implemented.

[21] PNPP1811: see above

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 PTR tank, 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)[22] 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)[23]. These devices for collecting and recirculating comply with the regulations [AG-B-IV] and [AG-D-I].

To reduce the potential radiological consequences, sodium tetraborate baskets will be implemented in the sump basins of the reactor building (PNPE1410)[24] in accordance with regulation [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.

Containment System Functions and Equipment

For existing equipment identified as necessary in the event of a core melt accident, a review of its performance under the conditions of a core melt accident is conducted. Existing equipment for which no evidence of performance under these conditions can be provided is replaced with qualified equipment.

To improve the resistance of certain components of the containment structure in the event of a core melt accident, EDF has replaced the

[22] PNPP1541: "Introduction of a system for collecting wastewater in the event of a core-melt accident" have already been implemented.

[23] PNPE1362: see above

[24] PNPE1410: "Installation of sodium tetraborate baskets in the sump basins of the reactor building" will be implemented during a specific program at the latest for Dampierre 3 on 27/06/2029 and for Dampierre 4 on 07/04/2030.

viewing windows of the reactor building (BR) airlocks (PNPP1631)[25], SEBIM valves (PNPP1595)[26] and specific system cables (PNPE1264)[27].

Other changes related to the qualification of equipment for core melt accidents will not be implemented until Phase B of the 4th PSR. This involves the replacement of electric actuators and control cabinets for the EAS and PTR systems (PNPE1347)[28], the replacement of certain seals (PNRL1896)[29] and electric actuators of the SG and EAS system (PNPE1486)[30].

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 (PNPP1870)[31]. In accordance with regulation [AG-C-II], the U5-System will be further reinforced to ensure its resistance to earthquakes of magnitude SMS (PNPE1377)[32].

Management of contaminated water

As part of crisis management, compliance with the quality guidelines for water intended for human consumption must be ensured during both the short-term and long-term phases of a core meltdown accident.

In accordance with regulation [AG-D-II], EDF shall have the necessary means to reduce the contamination of water present in the reactor

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[25] PNPP1631: “Reinforcement of the BR airlock viewing windows” has been implemented.

[26] PNPP1595: Replacement of the SEBIM valves” has been done.

[27] PNPE1264: “Replacement of cables of the Reactor vessel pressure relief system“ has been performed.

[28] PNPE1347: “Replacement of certain electric actuators and control cabinets in the EAS and PTR systems“ will be performed as part of the Phase B of the 4th PSR.

[29] PNRL1896: “Replacement of the seals on a valve in the nitrogen storage and distribution system“ will be performed as part of the Phase B of the 4th PSR.

[30] PNPE1486: “Replacement of specific electric actuators of the SG and EAS system“ will be performed as part of the Phase B of the 4th PSR.

[31] PNPP1870: “Strengthening the resilience of the containment's decompression and filter device in the event of an SMHV earthquake” has been implemented.

[32] PNPE1377: “Reinforcement of the compression and filter device of the U5 container in the event of an SMS earthquake” will be implemented as part of the specific program at the latest on June 27, 2029, for unit 3 and at the latest on April 7, 2030, for unit 4.

building following a core meltdown accident, and shall ensure that these means are operational on site (PNPE1362[33] and PNPE1449[34]).

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. These investigations found no evidence to suggest that additional measures would be necessary in view of the safety-related challenges.

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 bring them closer to 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 Dampierre 3 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 LTO project. (WENRA 2013) According to the WENRA safety objectives, core melt accidents that would lead to early or large releases should be practically eliminated. The occurrence of certain severe accidents can be considered to be practically eliminated “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 elimination”. Although this concept applies only to new reactors, it should also be applied to Dampierre 3 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-REPORT does not refer to the practical elimination of early releases in the event of a core melt accident, but only to the low

[33] PNPE1362: see above

[34] PNPE1449 “Study of a mobile water treatment module for treating contaminated water” will be performed as part of the Supplementary Phase.

probability, which, with a value of a few 10^{-7} per year, is not actually that low.

The EIA documents do not include a systematic comparison between the safety level of the 900 MWe reactors and modern safety standards in order to highlight 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. High-silicate concrete, which is highly susceptible to erosion, was used for the Dampierre 3 and 4 units.

IRSN simulations regarding the eroded thickness of highly siliceous concrete differ significantly from EDF's results. According to IRSN, the corium only solidifies when the melted thickness reaches approximately three meters. The complex physical phenomena involved in the solidification of corium were the subject of extensive research. (UMWELTBUNDESAMT 2021b) Pending the results, ASNR requires EDF to prepare the necessary work to reinforce the basements made of highly siliceous concrete so that these measures can be implemented from 2025 onwards. (see [AG-A-II]). The EIA documents do not explain if the necessary thickness considered by IRSN will be achieved for Dampierre 3&4).

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 reactors.

Consequently, the coolability of the core must be considered unlikely. (ASAMPSA 2013)

Studies that have demonstrated the feasibility and effectiveness of this device, which would have important differences with the EPR core catcher are not presented. 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 the core melt.

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 (PNPP1976 Volume B / PNPE1460).

Although the “installation of a device for dry distribution and stabilization of corium under water” (PNPP1976) have been implemented already, effective medium- and long-term cooling can only be guaranteed once all measures have been implemented after Phase B and Supplementary Phase of the 4th PSR.

It was one of the important lessons learned of the Fukushima accident that is important to have instrumentation that does not lose its function under severe 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 previously 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 Supplementary Phase of the LTO 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).

Containment System Functions and Equipment

To improve the resistance of certain components of the containment structure in the event of a core melt accident, EDF has replaced some equipment. However, important components such as valves, seals and electric equipment that are required during a core melt accident, but whose resistance cannot be guaranteed during such an accident, will also only be replaced during Phase B.

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 to an extreme earthquake has not yet been carried out, although this safety deficit was already identified during the EU stress tests. Upgrade measure (PNPE1377) will be only implemented as part of the specific program at the latest on June 27, 2029 for unit 3 and at the latest on April 7, 2030 for unit 4. It is noteworthy that this upgrade is coming nearly 20 years after this vulnerability became apparent following the Fukushima accident.

Management of contaminated water

Following the accident at the Fukushima Daiichi NPP, 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 core melt accident. 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 in the next years.

It is envisaged to investigate the use of a mobile water treatment module for treating contaminated water 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 a 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. The EU stress tests, already 15 years ago, have nevertheless revealed a number of shortcomings that have still not been addressed.

According to the ASNR, the objective of the 4th PSR for the 900 MWe reactors is to bring the safety level of the reactor closer to that of the EPR Flamanville 3, a third-generation reactor. In third-generation reactors, features to mitigate the effects of core melt accidents are already implemented in the design; this approach cannot be fully transferred to second-generation reactors such as Dampierre 3 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.

Not all necessary and planned modifications for heat removal without using the filtered pressure relief system in the event of a core melt accident have yet been fully implemented. Important components such as valves, seals and electric equipment that are required during a core melt accident, but whose resistance cannot be guaranteed during such an accident, will also only be replaced during Phase B or the Supplementary Phase. In addition, the reinforcement of the filtered pressure relief system (U5 system) against severe earthquakes has not yet been carried out. This means that today, a core melt accident with a major release of radioactive substances is still possible at Dampierre 3&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 for unit 3 and 2030 for unit 4 respectively). 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.
- Information about the thickness of the basement in the containment room and in the adjacent RIC room and the studies to justify the sufficient thickness should be provided.
- Studies that prove the sufficient dimension of the spreading areas for Dampierre 3&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 Dampierre 3&4 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 the outage of the 4th PSR and after implementation of all modification of 4th PSR (including Phase B, Supplementary Phase and the specific program) should be provided.
- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Dampierre 3&4. The concept of practical elimination should be used in 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 Dampierre 3&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

The assessment of transboundary impacts of accidents at Dampierre unit 3 and unit 4 is provided in Chapter 6 of *Document 3bis – Document on Environmental Effects Associated with Reactor Operation over the Next Ten Years*, prepared separately for each reactor unit (unit 3 and unit 4), with the reactor unit number indicated in the document title (EIA-REPORT D3, D.3b (2026) and EIA-REPORT D4, D.3b (2026)).

According to the results of the assessment presented in the EIA documents, transboundary impacts are considered possible only in the event of a core melt accident at Dampierre unit 3 and unit 4. In contrast, during normal operation and in the case of a design-basis accident, cross-border radiological effects are assessed as negligible.

Chapter 6 provides an overview of the assessment of impact for 4 categories of the reactor operation: normal operation and three types of design-basis accidents historically used in plant planning, along with the corresponding impact assessment results. These categories are referred to as:

- Category 1 – Normal operation
- Category 2 – Moderately frequent accidents (1 event in 10^2 years of reactor operation) for which the effects of the releases do not exceed 1 mSv/year at the site boundary.
- Category 3 – Very rare accidents (1 event in 10^2 – 10^4 years of reactor operation) for which the acute effective dose received due to effects of the releases do not exceed 10 mSv.
- Category 4 – Hypothetical accidents (1 event in 10^4 to 10^6 years of reactor operation) for which the acute effective dose received due to effects of the releases do not exceed 50 mSv.

Further information about the parameters applied in the assessment of the effects of the design-basis accidents and the underlying assessment methodology is not provided in the EIA documents.

Although a severe accident involving core melt is an extremely unlikely scenario requiring the simultaneous failure of multiple protection and control systems, it still cannot be fully excluded and the assessment presented in the EIA documents confirms that such an unlikely scenario can have transboundary consequences.

Thus, the 4th PSR includes also three beyond design-basis accidents:

1. Loss of cooling in shutdown operating mode,
2. Loss of spent fuel pool cooling, and
3. Loss of off-site power (station blackout).

The probability of these events is given as approximately 1 in 5 000 000 years of reactor operation. For these studies, it is postulated that a core meltdown accident has occurred, meaning that a sequence of events has led to at least a partial core meltdown, and that beyond the loss of the first barrier (the fuel rods), it could lead to the loss of the second barrier (the primary circuit, including the reactor vessel) and, in the absence of appropriate measures, to the degradation of the integrity of the third barrier. No further description of accidents which would possibly affect other countries in the EU nor accidents progression analyses are provided.

The identification of plausible cumulative accident scenarios at Dampierre unit 3 and unit 4, 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.

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 further in Chapter 6 of relevant EIA report documents for unit 3 and unit 4. (EIA-REPORT D3/4, D.3b, 2026)

The EIA documents present the results of calculations demonstrating the potential impacts on public health in terms of projected doses for early (24 hours and 7 days) and late (50 years) phases of an emergency assuming no protective measures are implemented. The results are compared with the reference values for emergency exposure situations set in the French legislation. 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.

The EIA-REPORT D3, D.3b (2026) and EIA-REPORT D4, D.3b (2026) claim that the assessment of radiological consequences includes calculations of the total effective dose for early and late phase and the thyroid equivalent dose for early phase of an emergency for the population at distances of 650 m (corresponding to distances of the nearest settlements to all power plants in the 900 MWe class), 2 km, 5 km, and 10 km. For beyond design-basis accidents, transboundary impacts are also assessed for distances of up to 1000 km. This includes 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 of more 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

The EIA documents consider core melt accidents as the only type of events with potential for transboundary consequences. It assumes an accident sequence leading to at least partial melting of the reactor core. It is further assumed that, in addition to failure of the first physical barrier (fuel cladding), failure of the second barrier (reactor coolant pressure boundary, including the primary circuit) may also occur triggering complex physical processes that can further lead to failure of the third barrier and release of radioactivity into the environment. 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. Although, there is an information in the EIA-REPORT D3/4, D.3b (2026) that EDF used a tool developed by the former Institute for Radioprotection and Nuclear Safety (IRSN) to assess the dose-related effects on the population of radioactive pollutants released from nuclear power plant operation, further details on the atmospheric dispersion model used to estimate off-site consequences are not provided. The EIA-REPORT D3/4, D.3b (2026) indicate 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-REPORT D3/4, D.3b (2026) 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 (0.05 mSv).

The EIA-REPORT D3/4, D.3b (2026) 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-REPORT D3/4, D.3b (2026) 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-REPORT D3/4, D.3b (2026) 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 food measures and 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 the calculations underlying the presented results, the expert team conducted dispersion modelling for two hypothetical large-scale release scenarios for Dampierre unit 3 and unit 4 to assess whether protective action thresholds in Austria could be exceeded under specific circumstances. Since both units are of identical design and located at the same site, the modelling produced identical results; therefore, only the results for unit 3 are presented. The aim of the assessment was to assess whether a severe accident at unit 3 or unit 4 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 Dampierre 3 & 4, the reference source term is scaled to reflect the characteristics of the French 900 MWe series reactors.

The release category FK2 considers an accident at a PWR resulting in core melt with large containment release happening one hour after the reactor shutdown. The release category FK3 considers an accident at a 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 core inventory	Kr-Xe	1,0	1,0
	I	$7{,}0\cdot10^{-3}$	$7{,}0\cdot10^{-3}$
	I2-Br	$4{,}0\cdot10^{-1}$	$1{,}5\cdot10^{-2}$
	Cs-Rb	$2{,}9\cdot10^{-1}$	$4{,}4\cdot10^{-2}$
	Te-Sb	$1{,}9\cdot10^{-1}$	$4{,}0\cdot10^{-2}$
	Ba-Sr	$3{,}2\cdot10^{-2}$	$4{,}9\cdot10^{-3}$
	Ru[1)]	$1{,}7\cdot10^{-2}$	$3{,}3\cdot10^{-3}$
	La[2)]	$2{,}6\cdot10^{-3}$	$5{,}2\cdot10^{-4}$

[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 Dampierre 3 or 4 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 which confirm the possibility of ground contamination in Austria from a release in Dampierre 3. For this task, meteorological conditions for the period 3 – 7 August 2025 were chosen. The JRODOS calculation was performed using a meteorological data set with the wind direction selected to represent meteorological conditions that would result in plume dispersion over Austrian territory. With acknowledging the methodological limitation, purpose of this calculation was to assess the possibility and not the probability of the transboundary impact of a severe accident at Dampierre 3 above the Austrian reference levels. Calculations are performed for two release scenarios, both assuming the same release start time, and consequently, the same meteorological conditions.

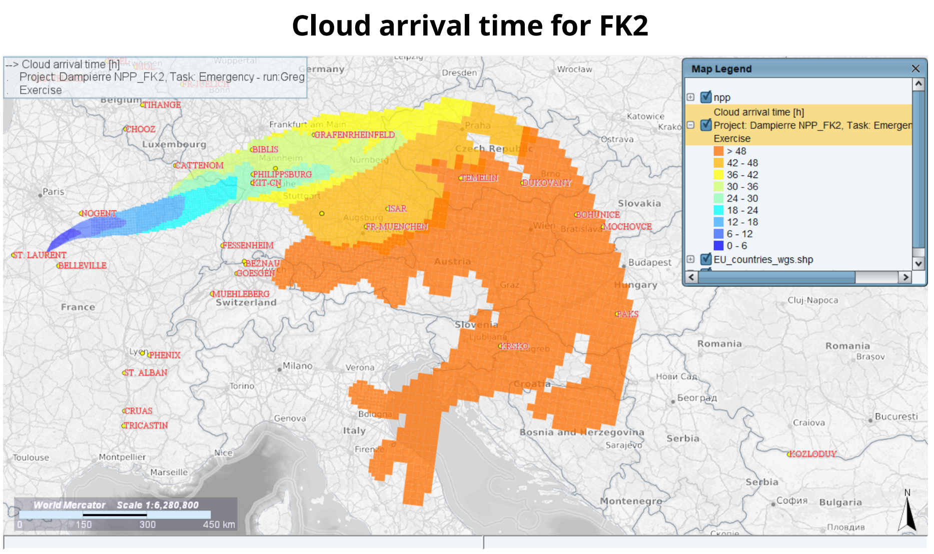
Location:....................Dampierre 3, France

Release start: ....3 August 2025, 18:00 UTC

Prognosis duration: ..96 hours

Figure 6 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 3 August 2025 at 18:00 UTC, the cloud is projected to reach Austrian territory in 48 hours. Meteorological conditions are the dominant factor influencing cloud arrival time, and this result may vary significantly under different weather conditions.

Figure 6: Cloud arrival time for the release category FK2

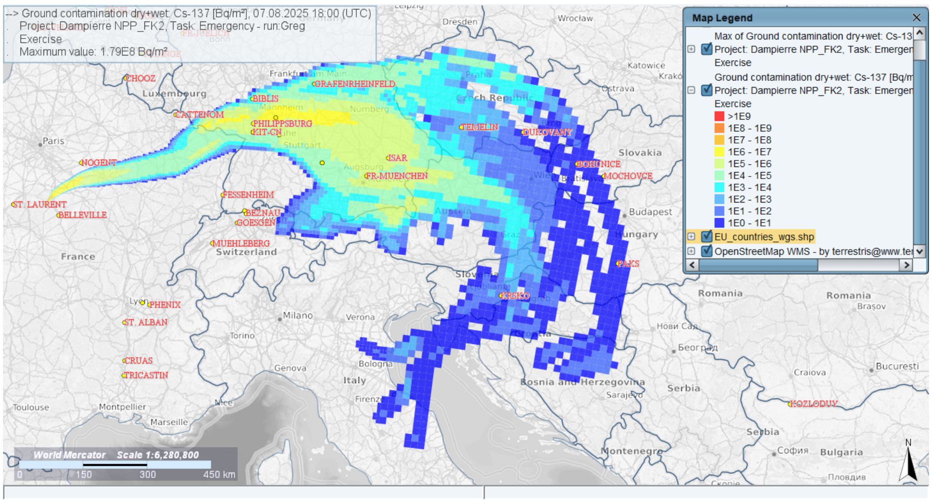


Source: Environment Agency Austria

Deposition of the radioactive material released in an accident depends on a number of factors: characteristics of a release, including particle size distribution, meteorological conditions, deposition surface and others. The results of the JRODOS calculations performed by the expert team for release categories FK2 and FK3 are presented in Figure 7 and Figure 8.

Figure 7: Ground contamination with Cs-137 for the release category FK2

Ground contamination with CS-137 for FK2



Source: Environment Agency Austria

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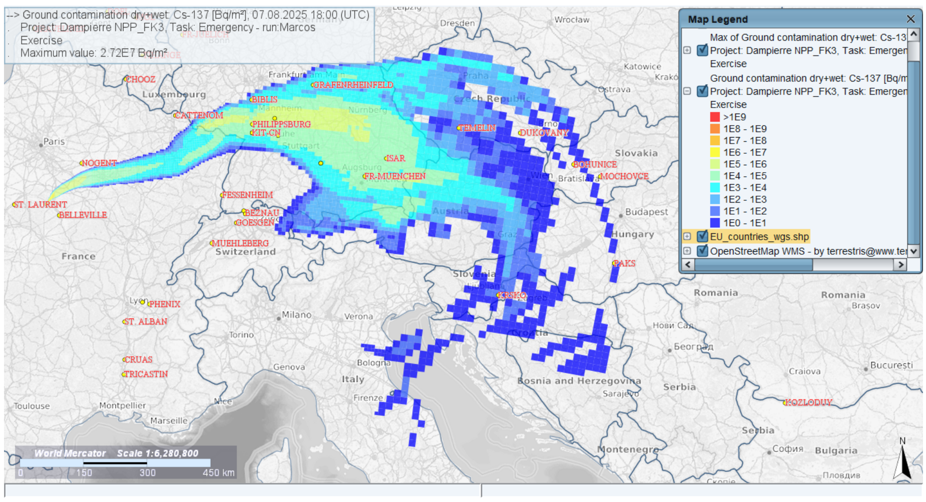
Figure 8: Ground contamination with Cs-137 for the release category FK3

Ground contamination with Cs-137 for FK3



Source: Environment Agency Austria

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The results of the JRODOS calculations demonstrate that contamination levels in Austria exceeding 650 Bq/m² are possible. The maximum calculated ground deposition exceeds 1×10^5 Bq/m² for release category FK2 and 1×10^4 Bq/m² for release category FK3.

These results indicate that transboundary radiological consequences more significant than those presented in the EIA documents cannot be excluded. The probability of occurrence of such contamination levels was not assessed within this calculation.

At the same time, the EIA documents do not provide information on the methodology applied for the radiological impact assessment. In the absence of a transparent description of input assumptions, meteorological data sets, dispersion modelling approach, and evaluation criteria, it is not possible to verify how the conclusions regarding limited transboundary impact were derived, nor whether scenarios with potentially more significant transboundary consequences were adequately considered.

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 spent fuel 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, the transparency and reproducibility of the radiological impact assessment are extremely 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, specifically for 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 EIA-REPORT D3, D.3b (2026) and EIA-REPORT D4, D.3b (2026) do not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances of more 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 unit 3 or unit 4 of Dampierre NPP 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 beyond those described 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.

- Information on the release parameters is needed for the reconstruction of the results of the assessment provided in the EIA documents. 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 unit 3 and unit 4 of Dampierre NPP. In particular the document "Description of the measures proposed by the operator in after the completion of the period review" summarises the measures that are planned to be implemented for Dampierre NPP unit 3 and unit 4. (EIA-REPORT D3/4, P.3 2026)

The introduction of measures to be implemented put emphasis on four important areas including:

- "risks", which included four different groups of scenarios, accidents with or without core damage, external hazards as well as accidents related with the spent fuel pool. An important consideration of the "risks" is that the safety improvements are designed to assure that a standard 900 MWe reactor approaches those comparable to Generation III reactors, with Flamanville 3 (EPR) as a reference reactor.
- "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 (ASN 2021), Dampierre 3 and 4 have a period of five years following the release of the PSR report, to implement all safety measures identified.

For Dampierre 3 and 4, the implementation of the safety measures 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 documents.

For Dampierre 3, certain set of measures are scheduled for implementation. They are planned to be implemented either as part of "Phase B", or as part of the "Phase B Supplements", or as part of a specific implementation package, but in any case, no later than June 2029. This coincides with the "five years after the release of the PSR report", as required by the regulator.

For Dampierre 4, certain set of measures are scheduled for implementation. They are planned to be implemented either as part of "Phase B", or as part of the "Phase B Supplements", or as part of a specific implementation package, but in any case, no later than by April 2030. This coincides with the "five years after the release of the PSR report", as required by the regulator.

8.2 Discussion

It is important that the agreed implementation period of five 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 which equals to June 2029 for Dampierre 3 and April 2030 for Dampierre 4) 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 five 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 MANAGEMENT OF SPENT FUEL AND RADIOACTIVE WASTE

9.1 Treatment in the EIA documents

Spent fuel is in France not defined as radioactive waste but as material. After a period of storage in the fuel pools at the NPP sites, the spent fuel is transported to the reprocessing plant. (EIA-REPORT D3/4, D.3b 2026) The resulting vitrified high-level waste is foreseen to be disposed of in the future deep geological repository Cigéo.

EDF is assessing the construction of additional storage capacity for spent fuel. (EIA-REPORT D3/4, P.2 2026)

In France, radioactive waste is categorized according to its radioactivity and its half-life and is or shall be disposed of in different final repositories. (EIA-REPORT D3/4, D.3b 2026)

- Very low-level waste (VLLW - in French: très faible activité TFA) has an activity below 100 Bq/g; VLLW is disposed of in the CIRES surface repository in Aube.
- Low level waste (LLW - in French: faible activité FA) has an activity up to one million Bq/g.
  Short-lived LLW (LLW-SL - in French: vie courte VC) with half-life below 31 years is disposed of in the CSA surface repository in Aube (formerly also in the CSM repository, which is already closed).
  For long-lived LLW (LLW-LL - in French: vie longue VL) with half-life over 31 years,), a final disposal solution has to be developed.
- Intermediate level waste (ILW - in French: moyenne activité MA) has an activity of up to one billion Bq/g.
  ILW-SL is also disposed of in the CSA.
  ILW-LL will finally be disposed of deep geological repository Cigéo in the future.
- High level waste (HLW - in French: haute activité HA) will be finally disposed of in the future deep geological repository Cigéo.

VLLW, LLW and ILW with a half-life below 100 days (in French: vie très courte) will be allowed to decay.

The NPP Dampierre produces the following amounts of radioactive waste (VLLW and LILW-SL) per year.

Table 4: Radioactive waste VLLW and LILW-SL from Dampierre NPP (EIA-REPORT D3/4, D.3b 2026)

Radioactive waste	**Average volume in m³ per year (in period 2010-2019)**	**Prognosis of average volume in m³per year (in period 2025-2028)**
VLLW: solid, to be disposed of in CIRES	107	440
ILW-SL: solid, to be disposed of in CSA	229	215
LLW-SL: solid, to be disposed of in CSA	170	75
LLW-SL: solid for melting*	87	320
LLW-SL: solid for incineration*	298	540
LLW-SL: liquid for incineration*	97	16

* *Treatment and conditioning are done in the CENTRACO facilities, operated by Cyclife; the French waste management organization ANDRA is responsible for the disposal.*

The increase in the volume of VLLW is explained by the upcoming refurbishment program (Grand Carénage). The increase of LLW-SL for melting and incineration will reduce the volume that has to be disposed of in the CSA facility; this corresponds with the reduction of solid LLW-SL for direct disposal. The reduction of liquid LLW-SL for incineration is due to increased treatment by vaporization.

9.2 Discussion

Spent fuel and radioactive waste can cause negative impacts for people and the environment. Proof of safe and secure disposal is required to prevent this. This proof shall include an estimate of the inventory of spent fuel and radioactive waste expected from the lifetime extension. For proof of disposal, it is decisive whether the required disposal facilities are available in the required capacity in good time.

Spent fuel and other high-level radioactive waste represent the greatest risk for transboundary impacts, especially when significant amounts are stored centrally while awaiting final disposal.

The deep geological repository Cigéo is planned to start operation in 2050. (IAEA 2026b) The original start of operation was 2035. (EC COM 2024) Proof is needed that the radioactive waste resulting from reprocessing (HLW and ILW-LL) can be stored safely for a longer time than planned, especially If the operation start of Cigéo would be postponed again. No information on this is available in the EIA documents.

The French Joint Convention national report from 2024 informs that the French national plan for the management of radioactive material and waste (PNGMDR) 2022-2026 requires EDF to refine the estimates of when the existing storage facilities in the spent fuel pools in La Hague will be filled to capacity. Spent fuel producers are required to develop dry interim storage strategies. EDF is required to guarantee the provision of a new centralised spent fuel pool as soon as possible. (RÉPUBLIQUE FRANÇAISE 2024) Asides from that one sentence that EDF is assessing the construction of additional storage capacity for spent fuel, no further information is available in the EIA documents.

ANDRA informs on its website[35] that the Cigéo inventory for which the licenses are being submitted does not include the so-called spare inventory waste. This spare inventory waste includes possible direct disposal of spent fuel in case of a change in France's reprocessing policy; furthermore, it includes spent fuel or radioactive waste resulting from lifetime extensions of the existing reactors beyond 50 years. A change in the inventory would require a new license. The period of the NPP lifetime extension could be beyond 50 years. In this case, proof of disposal needs to be provided for additional capacities for a deep geological repository. No information on this is available in the EIA documents.

Summarizing, the EIA documents do not provide sufficient information

- on the amount of spent fuel that results from 10, 20 or more years of lifetime extension,
- on the conclusion of a contract for reprocessing of the spent fuel from the lifetime extension,
- on the capacities of the spent fuel interim storage at La Hague,
- on a possible enlargement or addition to the capacity of Cigéo in case of NPP lifetime extensions over 50 years,
- on the available capacity of the CSA repository for LILW-SL, and
- on the timetable for the planned repository search for LLW-LL.

[35] https://international.andra.fr/stepwise-development-cigeo-and-timeline-associated-decisions, seen 16.7.2026

9.3 Conclusions

Spent fuel and radioactive waste can cause negative impacts for people and the environment. Proof of safe disposal is required to prevent this. However, the EIA documents did not provide sufficient evidence.

- During the EIA, sufficient information on management of spent fuel and radioactive waste should be provided. It would be appreciated if the missing information would be added to the EIA documents:
  - on the amount of spent fuel that results from 10, 20 or more years of lifetime extension,
  - on the conclusion of a contract for reprocessing of the spent fuel from the lifetime extension,
  - on the capacities of the spent fuel interim storage at La Hague,
  - on a possible enlargement or addition to the capacity of Cigéo in case of NPP lifetime extensions over 50 years,
  - on the available capacity of the CSA repository for LILW-SL, and
  - on the timetable for the planned repository search for LLW-LL.
- The update of the French national plan for the management of radioactive material and waste (PNGMDR) should undergo a Strategic Environmental Assessment, also transboundary.

10 LIST OF PRELIMINARY RECOMMENDATIONS

10.1 Long-term operation and operational experience

- The justification that no checks are to be carried out for Dampierre 3&4 as part of the Program for Complementary Investigations (PIC) should be provided
- 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 seismic safety should be considered.

10.2 External hazards

- With respect to seismic safety, the following information should be provided:
  - Methods, data and assumptions used for the PSHA performed to determine the SND for Dampierre 3 and 4, 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.

  - The actual PSHA-derived ground motion values for the SMS and the SND.
- The deterministically derived SMS ground motion should be benchmarked against a PSHA-derived ground motion for a recurrence interval of 10,000 years as required by WENRA (2021).
- Additional safety demonstrations should be required to ensure that all SSCs relevant to safety can cope with a probabilistically derived DBE for an occurrence probability of 10^{-4}/year in case the probabilistically derived DBE exceeds the ground motion parameters of the current SMS of Dampierre 3 and 4.
- The methods, data and assumptions used to determine the SND for Dampierre 3 and 4, 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 should be reviewed. The PSHA should be benchmarked against WENRA requirements (WENRA 2021) and recommendations (WENRA 2020a, b).
- Dedicated assessments of near-regional potentially active faults of the Loire fault system should be performed in line with WENRA (2020b). The approach should include field geology, geophysical mapping, morphostructural and dating studies, and paleoseismology. Background: EIA documents and existing literature indicate a late to post-Pliocene age for the faults in the near-region (>25 km) and region (<50 km distance) from the site.
- It should be ensured that design basis events and design basis parameters defined for meteorological hazards conform with WENRA (2021) requirements. Background: this seems not to be the case for some meteorological hazards, in particular, extremely high temperatures which are determined for exceedance probabilities of 10^{-2} per year and 70% confidence.
- It should be ensured that the use of the Noyau Dur's DEC equipment is not required to protect the Dampierre 3 and 4 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.
- With respect to possible terror attacks, the following questions should be addressed:
  - 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 (NTI 2025) for France assessed? Are improvements planned with regard to

"security culture", "cybersecurity" and "protection against insider threats"?

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

1. Enhance Transparency and Provide Clarity on Key Quantitative Data

- Quantitative Data: The documents identify PNPE1141 as an increase in the flow rate of the GCT-a regulating valves and state that it contributes to resolving an ASG water-consumption study anomaly. However, the excerpts provided do not give the initial and final GCT-a mass-flow rates or compare the upgraded capacity with nominal operating steam flows. Providing these values would help quantify the magnitude of 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 (SGTR). 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. A concise clarification would therefore be useful.

2. Establish Firm and Accountable Timelines

- Study Status and Next Steps: The documents identify the critical heat-flux correlation and the treatment of fuel-assembly deformation as important assumptions in 4th PSR accident studies. They also state that further elements concerning fuel-assembly deformation impacts are to be provided in Phase B. The provided excerpts do not give a clear status of any associated experimental programme, results, or integration schedule. A status update identifying completion, results, and planned integration would improve traceability.

3. Clarify Status Reporting and Implementation Rationale

- Resolve Discrepancies and improve transparency by status tracking for implemented and pending measures: The reports list a mixture of already deployed modifications, Phase B modifications, and modifications under specific programming. EDF also states that it will present annually the dispositions implemented during the previous year and those remaining to be carried out, with their programming. To strengthen transparency and closure tracking, it would be useful to systematically provide unit-specific completion dates and status fields for each outstanding measure, distinguishing installation, commissioning, operational availability, documentary update, and final closure.

10.4 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.
- Information about the thickness of the basement in the containment room and in the adjacent RIC room and the studies to justify the sufficient thickness should be provided.
- Studies that prove the sufficient dimension of the spreading areas for Dampierre 3&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 Dampierre 3&4 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 the outage of the 4th PSR and after implementation of all modification of 4th PSR (including Phase B, Supplementary Phase and the specific program) should be provided.
- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Dampierre 3&4. The concept of practical elimination should be used in 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 Dampierre 3&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.

10.5 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 documents. 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.

10.6 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 five 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.

10.7 Radioactive Waste Management

- During the EIA, sufficient information on management of spent fuel and radioactive waste should be provided. It would be appreciated if the missing information would be added to the EIA documents:
  - on the amount of spent fuel that results from 10, 20 or more years of lifetime extension,

  - on the conclusion of a contract for reprocessing of the spent fuel from the lifetime extension,
  - on the capacities of the spent fuel interim storage at La Hague,
  - on a possible enlargement or addition to the capacity of Cigéo in case of NPP lifetime extensions over 50 years,
  - on the available capacity of the CSA repository for LILW-SL, and
  - on the timetable for the planned repository search for LLW-LL.
- The update of the French national plan for the management of radioactive material and waste (PNGMDR) should undergo a Strategic Environmental Assessment, also transboundary.

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GLOSSARY

AM .... Ageing Management

ASG .... Steam Generator Auxiliary Feedwater System

ASN........... French Authority for Nuclear Safety

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

BK .... Spent Fuel Building

Bq .... Becquerel

BR .... Reactor Building

CDF........... Core Damage Frequency

CEPP.......... Primary Pump Sealing Circuit

CHF .... Critical Heat Flux

CIRES......... Repository for VLLW

Cs-137....... Caesium-137

CSA........... Repository for LILW-SL

DAPE .... Dossier of Suitability for Continued Operation

DBA .... Design Basis Accidents

DBE........... Design Basis Earthquake

DEC........... Design Extension Conditions

DID .... Defence-in-Depth

DSA........... Deterministic Safety Analysis

DUS .... Ultimate Emergency Diesel / Diesel d'Ultime Secours

EAS .... Sprinkler System

EAS-ND ..... Hardened Safety Core Sprinkler System

EDF .... Électricité de France

EDG .... Emergency Diesel Generator

EIA .... Environmental Impact Assessment

ENSREG .... European Nuclear Safety Regulators Group

EPR .... European Pressurized Reactors

EU .... European Union

FARN .... Force d'Action Rapide Nucléaire = Nuclear *Rapid* Action Force

FK............... Release Category

FLA3 .... Flamanville Unit 3

GCT-a .... Main turbine bypass system with direct venting of steam produced by the steam generators to the atmosphere

GMPP .... Reactor coolant pump (primary pump) motor-pump group

GPR .... Permanent Group of Experts on Reactors

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

HLW........... High-level radioactive waste

HSC............ Hardened Safety Core

I-131 .... Iodine-131

IAEA........... International Atomic Energy Agency

ILW .... Intermediate-level radioactive waste

INES........... International Nuclear and Radiological Event Scale

IPCC........... Intergovernmental Panel on Climate Change

IPPAS......... International Physical Protection Advisory Service

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

LILW .... Low- and Intermediate-level radioactive waste

LL............... Long-lived

LLW .... Low-level radioactive waste

LOCA .... Loss of Coolant Accident

LTO............ Long-Term Operation

MQCA........ Equipment qualified for use in accident situations

mSv .... Millie-Sievert

MWe.......... Mega Watt electric

NAcP .... National Action Plan

ND .... Noyau Dur = Hardened Safety Core

NPP .... Nuclear Power Plant

NTI............ Nuclear Threat Initiative

OAMP........ Overall Ageing Management Programme

OISS........... Spurious opening of a secondary relief valve at 0% rated power

OSART....... Operational Safety Review Team

PGA .... Peak Ground Acceleration

PIC .... Program for Complementary Investigations

PNGMDR .. French national plan for the management of radioactive material and waste

PSA............ Probabilistic Safety Assessment

PSHA .... Probabilistic Safety Hazard Assessment

PSR .... Periodic Safety Review

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

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

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

RCS............ Reactor Cooling System

RCV............ Chemical and Volume Control system

RCV-RIS..... Interface between the RCV and RIS systems

REA............ Boron and Water Storage Tank

RFS .... Règle Fondamentale de Sûreté

RGE............ General Operating Procedures

RIC .... Reactor Core Instrumentation

RIS .... Safety Injection system

RPN .... Nuclear Power Measurement System

RPR............ Reactor Protection System

RPV .... Reactor Pressure Vessel

SA .... Severe Accidents

SBO .... Station Black Out

SEG .... Diversified water source

SF-ND .... Hardened Core Cooling Source

SFP .... Spent Fuel Pool

SG .... Steam Generator

SGTR .... Steam generator tube ruptures

SIP-C .... Control part of the Process Instrumentation System

SIP-P .... Process instrumentation system / process part of SIP

SIS .... Safety Injection Systems

SL .... Short-lived

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

SRL .... Safety Refence Level

SSCs .... Structures, Systems and Components

SSD .... Reference Spektrum

TEP .... Primary effluent treatment system

TPR .... Topical Peer Review

TTS .... Target Technical Specifications

UHS .... Ultimate Heat Sink

VD4 .... Quatrième Visite Décennale (Fourth ten-year inspection/outage)

VD5............Fifth ten-year inspection/outage

VLLW ....Very Low-Level radioactive Waste

Vs..............s-wave velocity of the top soil

WENRA......Western European Nuclear Regulators’ Association

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