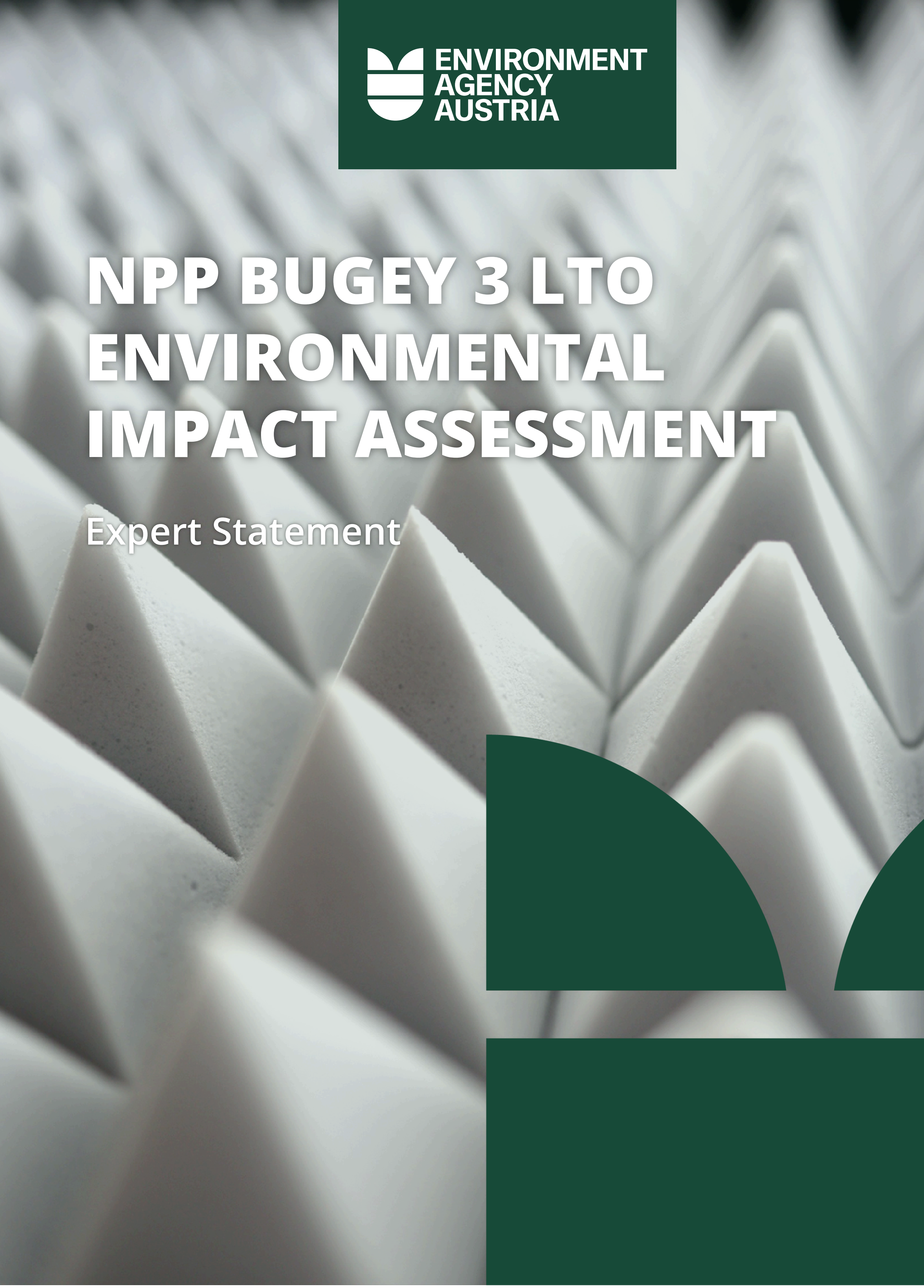

ENVIRONMENT AGENCY AUSTRIA

NPP BUGEY 3 LTO ENVIRONMENTAL IMPACT ASSESSMENT

Expert Statement

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Federal Ministry
Agriculture and Forestry, Climate
and Environmental Protection,
Regions and Water Management
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SUMMARY

The Bugey nuclear power plant (NPP) consists of four operating pressurized water reactors with a capacity of 900 MWe each. These reactors were commissioned 1978 and 1979 respectively. France notified the 4th Periodic Safety Review (“Public consultation procedure on the 4th safety review report”) of the Bugey nuclear power plant (reactor 3), 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 Ain. The project applicant is Électricité de France (EDF).

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

Procedure

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

For the 4th PSR of the 900 MWe 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) Once the generic phase is complete, inspections of all 32 reactors at the 900 MWe nuclear power plants should follow over a period of approximately ten years (from 2019 to 2031).

There is a high degree of public involvement in the process of the 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.

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 Bugey 3 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 the 4th PSR is supported. As proposed by the ASNR, the focus must be on components that are necessary for controlling accident situations. However, the scope of the program "qualification of materials under accident conditions" in the 4th PSR is very limited for Bugey 3.

A recently reported safety-related incident involving faulty anchors for earthquake protection raises serious questions about the conformity tests carried out to date. Firstly, these safety-related defects have existed since the plant was commissioned without being detected during previous inspections. Second, similar deficiencies were identified in other reactors almost five years ago, and it took that long for these deficiencies to be discovered and corrected at Bugey 3.

External hazards

The EIA documents provide information on hazard types considered in the safety demonstration for Bugey 3 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 WENRA requirements and guidance therefore cannot be assessed.

Non-conformity with WENRA Reference Levels is observed for earthquake and seismic ground shaking. The Design Basis Earthquake (DBE) for Bugey, 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 Bugey of 0.145 g. This value appears low when compared to the PGA of at least 0.25 g for the vicinity of Bugey, calculated for a recurrence interval of 1975 years by MARIN et al. (2004). Comparison suggests that the current SMS fails to meet the WENRA requirement that an exceedance frequency not higher than 10^{-4} per annum shall be used for the DBE. It remains to be demonstrated that the current SMS fulfills the WENRA requirement. It also remains to be demonstrated that 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 Bugey site is located in the Bresse Graben, a tectonic structure containing numerous potentially active faults which have not been taken into account in seismic hazard assessments. The EIA statement that “there is currently no data in the literature to suggest the presence of an active fault in the vicinity of the Bugey site” is considered to be factually wrong because various authors pointed to the existence of Quaternary and therefore active faults near the Bugey NPP. 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.

The authors of this report assume that adequate protection against an updated DBE, should it be higher than the deterministic SMS, is intended to be ensured by the Hardened Safety Core (Noyau Dur-ND). This, however, would contradict the Defence-in-Depth (DiD) concept and the separation of DiD levels because the Design Extension Conditions (DEC) equipment of the ND could become necessary to protect the plant against design basis events. The ND is a 4th DiD level system which is required as an additional and independent level compared to the 3rd DiD level. The ND therefore cannot be used to compensate for existing deficits in terms of the protection against design basis events.

With respect to safety upgrades of Bugey 3, 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) which is still pending and should be completed until 2029. The decision to implement the ND has been made in 2012. The fact that the implementation of the ND will be completed only 17 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 Bugey 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 Bugey 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 Bugey 3. 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 (PNPP0864), supporting accident sequences where additional feedwater is required. For thermal-hydraulic control, the increase of the VCD-a regulating-valve capacity (PNPE0141) 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 post-Fukushima Hardened Safety Core principles. Water make-up capabilities are strengthened by the designation of Noyau Dur (ND) as a Spent Fuel Pool (PNPP0714), with further enhancements planned on the reactor-building side (PNRL0803). Fire propagation risk between PTR trains is mitigated by the planned physical separation and protection measures (PNPP0949), complementing the broader set of pool instrumentation and actuation upgrades already deployed or in progress.

Safety aspects of core melt accidents

Severe accidents (SA) involving core meltdown were not taken into account in the design of the French 900 MWe reactors. However, as a result of previous PSRs, facilities and measures for SA management have been implemented. According to the ASNR, the objective of the fourth 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 Bugey 3. The EIA documents do not contain a systematic comparison between the safety level of the 900 MWe reactors and the safety level of the EPR in order to identify the remaining gaps.

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

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

The planned modifications for heat removal without using the filtered pressure relief system in the event of a core melt accident have not yet been fully implemented. In addition, the reinforcement of the filtered pressure relief system (U5 system) against severe earthquakes has not yet been carried out. This means that even after completion of all Phase A measures of the 4th PSR, a core melt accident with a major release of radioactive substances is still possible at Bugey 3. 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 (2030). 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-REPORT D.3b (2026) does not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances greater than 5 km after 7 days and within 1 km after one year. Additionally, the assessment omits information on ground deposition, despite its significance for evaluating long-term radiological impacts and potential contamination of the food chain.

Modelling of atmospheric dispersion and deposition conducted by the expert team demonstrate that, under certain meteorological conditions, a severe accident at Bugey 3 could lead to ground deposition of Cs-137 in Austria above the national screening threshold of 650 Bq/m^2. Although the study does not assess the probability of such conditions, the results indicate that transboundary impacts greater than those implied in the EIA documents cannot be excluded.

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

Assessment of the time frame

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

ZUSAMMENFASSUNG

Das Kernkraftwerk Bugey besteht aus vier in Betrieb befindlichen Druckwasserreaktoren mit einer Leistung von jeweils 900 MWe. Diese Reaktoren wurden 1978 bzw. 1979 in Betrieb genommen. Frankreich hat die vierte Periodische Sicherheitsüberprüfung („Öffentliches Konsultationsverfahren zum vierten Bericht der Sicherheitsüberprüfung") des Kernkraftwerks Bugey (Reaktor 3) 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 Ain. 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 Kernkraftwerke ist zeitlich nicht begrenzt. Alle zehn Jahre werden französische Kernkraftwerke 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 ASNR bestätigt. Die generische Phase endete mit der Veröffentlichung der Stellungnahme der ASNR am 23. Februar 2021, die allgemeine Vorschriften enthielt, die zuvor Gegenstand einer öffentlichen Konsultation gewesen waren. (ASN 2021) Nach Abschluss der generischen Phase sollen über einen Zeitraum von etwa zehn Jahren (von 2019 bis 2031) Inspektionen aller 32 Reaktoren der 900-MWe-Reaktoren folgen.

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.

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 eine Methode, die darauf abzielt, sicherzustellen, dass in Bereichen, die nicht regelmäßig inspiziert werden, keine Betriebsstörungen auftreten. Ohne Begründung wird angegeben, dass für Bugey 3 im Rahmen des ergänzenden Untersuchungsprogramms keine Kontrollen 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 der 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 Bugey 3 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. Erstens bestehen diese sicherheitsrelevanten Mängel bereits seit der Inbetriebnahme der Anlage, ohne dass sie bei früheren Inspektionen entdeckt wurden. Zweitens wurden ähnliche Mängel bereits vor fast fünf Jahren in anderen Reaktoren festgestellt, und es dauerte so lange, bis diese Mängel bei Bugey 3 entdeckt und behoben wurden.

Externe Gefahren

Die UVP-Unterlagen enthalten Informationen zu den in den Sicherheitsnachweisen für Bugey 3 berücksichtigten Gefahrenarten sowie zu den bereits umgesetzten oder beschlossenen Maßnahmen zur Stärkung 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 WENRA kann daher nicht beurteilt werden.

Bei Erdbeben und seismischen Bodenbewegungen wird eine Abweichung von den WENRA-Referenzwerten festgestellt. Das Auslegungserdbeben (DBE) für Bugey, 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-Bodenbewegungswert (PGA) für Bugey von 0,145 g aus. Dieser Wert erscheint niedrig im Vergleich zum PGA von mindestens 0,25 g für die Umgebung von Bugey, der von MARIN et al. (2004) für ein Wiederholungsintervall von 1975 Jahren berechnet wurde. Ein Vergleich legt nahe, dass das derzeitige SMS die WENRA-Anforderung nicht erfüllt, wonach für die DBE eine Überschreitungshäufigkeit von nicht mehr als 10^{-4} pro Jahr zugrunde gelegt werden soll. Es gilt noch nachzuweisen, dass das derzeitige SMS die WENRA-Anforderung erfüllt. Es muss außerdem noch nachgewiesen werden, dass die Erdbebensicherheit aller sicherheitsrelevanten SSCs ausreichend ist, um, wie von der WENRA (2021) gefordert, in einem DBE mit einem durchschnittlichen Wiederholungsintervall von 10.000 Jahren, die grundlegenden Sicherheitsfunktionen konservativ zu gewährleisten.

Der Standort Bugey liegt im Bresse-Graben, einer tektonischen Struktur, die zahlreiche potenziell aktive Verwerfungen aufweist, die bei der Bewertung der Erdbebengefährdung nicht berücksichtigt wurden. Die Aussage in der Umweltverträglichkeitsprüfung, dass „derzeit keine Daten in der Literatur vorliegen, die auf das Vorhandensein einer aktiven Verwerfung in der Nähe des Standorts Bugey hindeuten“, wird als sachlich falsch angesehen, da verschiedene Autoren auf das Vorhandensein von quartären und somit aktiven Verwerfungen in der Nähe des Kernkraftwerks Bugey hingewiesen haben. 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.

Die Autoren dieses Berichts gehen davon aus, dass ein angemessener Schutz vor einem aktualisierten DBE – sollte dieser über dem deterministischen SMS liegen – durch den Hardened Safety Core (Noyau Dur – ND) gewährleistet werden soll. Dies würde jedoch dem gestaffelten Sicherheitskonzept (Defence-in-Depth – DiD) und der Trennung der DiD-Ebenen widersprechen, da die Einrichtungen für die erweiterten Auslegungsbedingungen (Design Extension Conditions – DEC) des ND erforderlich werden könnten, um die Anlage vor Auslegungsereignissen zu schützen. Der ND ist ein System der 4. DiD-Ebene, die als zusätzliche und unabhängige Ebene gegenüber der 3. DiD-Ebene erforderlich ist. Der ND kann daher nicht dazu verwendet werden, bestehende Defizite beim Schutz vor Auslegungsereignissen auszugleichen.

Im Hinblick auf die Sicherheitsnachrüstungen am Reaktor Bugey 3 ist offensichtlich, dass eine der wichtigsten Maßnahmen zum Schutz vor externen Gefahren die Umsetzung des „Hardened Safety Core“ (ND) ist, die noch aussteht und bis 2029 abgeschlossen sein soll. Die Entscheidung zur Umsetzung des ND wurde 2012 getroffen. Die Tatsache, dass die Umsetzung des ND erst 17 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 Kernkraftwerks Bugey durch Flugzeugabstürze gefährdet sind. Es ist wichtig zu erwähnen, dass die 1,8 m dicke äußere Stahlbetonhülle des EPR so ausgelegt ist, dass sie dem Aufprall eines großen Passagierflugzeugs standhält. Die Wandstärken im Kernkraftwerk Bugey betragen jedoch weniger als 1,0 m. Darüber hinaus erhöht die zunehmende Verfügbarkeit und Leistungsfähigkeit von Drohnen die potenzielle Bedrohung für kerntechnische Anlagen. Eine kürzlich durchgeführte Bewertung der nuklearen Sicherung in Frankreich weist auf Mängel im Vergleich zu den notwendigen Anforderungen an die nukleare Sicherung in Bezug auf die „Sicherungskultur“, die „Cybersicherheit“ und den „Schutz vor Insider-Bedrohungen“ hin.

Sicherheitsaspekte von Unfällen ohne Kernschmelze und im Brennelementlagerbecken

Die Analysen nutzen sowohl deterministische Sicherheitsanalysen als auch probabilistische Sicherheitsanalysen (PSA), um Betriebstransienten, Auslegungsstörfälle (DBA) und erweiterte Auslegungsbedingungen (DEC) neu zu bewerten. Es wurden erhebliche Sicherheitsverbesserungen umgesetzt oder sind geplant, um die radiologischen Auswirkungen zu verringern und das gestaffelte Sicherheitskonzept im Kernkraftwerk Bugey 3 zu verbessern. Zur langfristigen Wärmeabfuhr wurde die Wasserversorgung des Hilfsspeisewassersystems (ASG) durch eine diversifizierte Anbindung des ASG-Behälters an das Brandschutznetz (PNPP0864) sichergestellt, was Unfallabläufe unterstützt, bei denen zusätzliches Speisewasser benötigt wird.

In Bezug auf die thermohydraulische Regelung behebt die Erhöhung der Kapazität des Regelventils VCD-a (PNPE0141) die Anomalie bei der

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

Im Hinblick auf das Lagerbecken für abgebrannte Brennelemente werden die Integrität und die Robustheit der Kühlung durch die Hinzufügung eines diversen, mobilen Kühlpfads (PTR bis) verstärkt, der eine robuste Methode zur Wiederherstellung der Kühlung bietet und den Prinzipien des „Hardened Safety Core“ (Noyau Dur) nach Fukushima entspricht. Die Möglichkeiten zur Wassernachspeisung werden durch die Einrichtung des Noyau Dur (ND) für das SFP-Gebäude (PNPP0714) verbessert; weitere Verbesserungen sind im Reaktorgebäude geplant (PNRL0803). Das Risiko einer Brandausbreitung zwischen den PTR-Strängen wird durch die geplanten physischen Trennungs- und Schutzmaßnahmen (PNPP0949) gemindert, die die bereits umgesetzten oder in Arbeit befindlichen umfassenderen Modernisierungen der Beckeninstrumentierung und -steuerung ergänzen.

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. 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 Bugey 3 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.

Die geplanten Umbauten zur Wärmeabfuhr ohne Einsatz des gefilterten Druckentlastungssystems im Falle eines Kernschmelzunfalls sind noch nicht vollständig umgesetzt. 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 Bugey 3 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 (2030) 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 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 bereitgestellt. 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. Der UVP-Bericht D.3b (2026) enthält 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 im Kernkraftwerk Bugey 3 zu einer Bodenkontamination von Cs-137 in Österreich führen könnte, die über dem nationalen Schwellenwert von 650 Bq/m^2 liegt. Obwohl die Studie die Wahrscheinlichkeit solcher Bedingungen nicht bewertet, deuten die Ergebnisse darauf hin, dass grenzüberschreitende Auswirkungen, die über die in den UVP-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 = 2029/2030) 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 Sicherheitsänderungen, die im Rahmen der 4. PSÜ aufgeführt sind, bereits im Rahmen des EU-Stresstests (2012) als notwendig erachtet und deren Umsetzung vereinbart worden waren.

RESUME

La centrale nucléaire du Bugey comprend quatre réacteurs à eau pressurisée en service d'une capacité de 900 MWe chacun. Ces réacteurs ont été mis en service respectivement en 1978 et 1979. La France a notifié le quatrième réexamen périodique (« Procédure de consultation publique sur le quatrième rapport du réexamen ») de la centrale nucléaire de Bugey (réacteur 3), qui doit être considéré comme une prolongation de durée de vie conformément à la Convention d'Espoo de la CEE-ONU sur l'évaluation de l'impact sur l'environnement (EIE) dans un contexte transfrontalier. L'autorité compétente est le département français de l'Ain. Le demandeur du projet est Électricité de France (EDF).

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

Procédure

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

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

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

Exploitation à long terme et expérience opérationnelle

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

La mise en œuvre du programme d'investigations complémentaires (PIC) est une approche qui vise à confirmer l'absence de défaillances opérationnelles dans les domaines qui ne font pas l'objet d'inspections régulières. Sans justification, il est indiqué qu'aucun contrôle ne doit être effectué pour Bugey 3 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 intercristalline, est un phénomène de corrosion bien connu, mais il n'était pas susceptible de se produire dans les zones concernées et les tuyaux n'ont donc pas été inspectés à cet effet. Cela signifie que le concept de gestion du vieillissement des composants des circuits primaire et secondaire principal est remis en question.

La proposition de l'ASNR, visant à étendre la gestion du vieillissement au-delà du 4e RP pendant la phase générale du 5e RP est soutenue.

Comme le propose l'ASNR, l'accent doit être mis sur les composants nécessaires au contrôle des situations accidentelles. Cependant, la portée du programme « qualification des matériaux en conditions accidentelles » du 4e RP est très limitée pour Bugey 3.

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. D'une part, ces défauts liés à la sûreté existaient depuis la mise en service de la centrale sans avoir été détectés lors des inspections précédentes. D'autre part, des défaillances similaires avaient été identifiées dans d'autres réacteurs il y a près de cinq ans, et il a fallu tout ce temps pour que ces défaillances soient découvertes et corrigées à Bugey 3.

Risques externes

Les documents de l'EIE fournissent des informations sur les types de risques pris en compte dans la démonstration de sûreté de Bugey 3, 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 la 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 Bugey, appelé SMS dans la réglementation française, repose sur une analyse déterministe qui n'est plus à la pointe de la technologie. Les documents disponibles indiquent une valeur de mouvement du sol (PGA) de 0,145 g pour le Bugey dans le cadre du SMS. Cette valeur semble faible par rapport à la PGA d'au moins 0,25 g pour les environs du Bugey, calculée pour un intervalle de récurrence de 1 975 ans par MARIN et al. (2004). La comparaison suggère que le SMS actuel ne satisfait pas à l'exigence de la WENRA selon laquelle une fréquence de dépassement ne dépassant pas 10^{-4} par an doit être utilisée pour le DBE. Il reste à démontrer que le SMS actuel satisfait à l'exigence de la WENRA. Il reste également à démontrer que la résistance sismique de tous les SSC importants pour la sûreté est suffisante pour garantir de manière prudente les fonctions de sûreté fondamentales pour un DBE avec un intervalle de récurrence moyen de 10 000 ans, comme l'exige la WENRA (2021).

Le site du Bugey est situé dans le graben de la Bresse, une structure tectonique comportant de nombreuses failles potentiellement actives qui n'ont pas été prises en compte dans les évaluations des risques sismiques. L'affirmation contenue dans l'EIE selon laquelle « il n'existe actuellement aucune donnée dans la littérature suggérant la présence d'une faille active à proximité du site du Bugey » est considérée comme factuellement erronée, car divers auteurs ont mis en évidence l'existence de failles quaternaires, et donc actives, à proximité de la centrale nucléaire du Bugey. 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.

Les auteurs du présent rapport partent du principe qu'une protection adéquate contre un événement de référence actualisé, s'il s'avérait plus grave que le scénario de référence déterministe, est censée être assurée par le noyau de sûreté renforcé (Noyau Dur-ND). Cela serait toutefois en contradiction avec le concept de défense en profondeur (DiD) et la séparation des niveaux de DiD, car les équipements relevant des conditions d'extension de conception (DEC) du ND pourraient s'avérer nécessaires pour protéger la centrale contre les événements de référence. Le ND est un système de 4e niveau de DiD qui est requis en tant que niveau supplémentaire et indépendant par rapport au 3e niveau de DiD. Le ND ne peut donc pas être utilisé pour compenser les déficits existants en matière de protection contre les événements de référence.

En ce qui concerne les renforcements de la sûreté de Bugey 3, 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. 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 17 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 Bugey 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 Bugey 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 Bugey 3. 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 (PNPP0864), 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 VCD-a (PNPE0141) 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 de « Hardened Safety Core » adoptés après l'accident de Fukushima. Les capacités de réalimentation en eau sont renforcées par la désignation du Noyau Dur (ND) comme piscine de stockage du combustible usé (PNPP0714), avec des améliorations supplémentaires prévues du côté du bâtiment réacteur (PNRL0803). Le risque de propagation d'incendie entre les trains PTR est atténué par les mesures de séparation physique et de protection prévues (PNPP0949), qui complètent l'ensemble plus large de mises à niveau de l'instrumentation et des systèmes d'actionnement de la piscine déjà déployées ou en cours.

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 adoptés après l'accident de Fukushima. Les capacités de réalimentation en eau sont renforcées par la désignation du Noyau Dur (ND) comme piscine de stockage du combustible usé (PNPP0714), avec des améliorations supplémentaires prévues du côté du bâtiment réacteur (PNRL0803). Le risque de propagation d'incendie entre les trains PTR est atténué par les mesures de séparation physique et de protection prévues (PNPP0949), qui complètent l'ensemble plus large de mises à niveau de l'instrumentation et des systèmes d'actionnement de la piscine déjà déployées ou en cours.

Aspects de sûreté des accidents de fusion du cœur

Les accidents graves (SA) impliquant une fusion du cœur n'ont pas été pris en compte dans la conception des réacteurs français de 900 MWe. Cependant, à la suite des examens périodiques de sûreté (RP) précédents, des installations et des mesures de gestion des SA ont été mises en place. Selon l'ASNR, l'objectif de la quatrième RP pour les réacteurs de 900 MWe est de rapprocher le niveau de sûreté du réacteur de celui de l'EPR de Flamanville, un réacteur de troisième génération. Dans les réacteurs de troisième génération, des dispositifs visant à atténuer les effets des accidents de fusion du cœur sont déjà intégrés dans la conception ; ceux-ci ne peuvent pas être entièrement transposés aux réacteurs de deuxième génération tels que Bugey 3. Les documents d'EIE ne contiennent pas de comparaison systématique entre le niveau de sûreté des réacteurs de 900 MWe et celui de l'EPR afin d'identifier les écarts restants.

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

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

Les modifications prévues pour sur l'évacuation de la puissance résiduelle du cœur sans ouverture du dispositif de décompression et filtration de l'enceinte en cas d'accident de fusion du cœur n'ont pas encore été entièrement mises en œuvre. En outre, le renforcement du système de décompression filtré (système U5) contre les séismes violents n'a pas encore été réalisé. Cela signifie que même après l'achèvement de toutes les mesures de la phase A du 4e RP, un accident de fusion du cœur avec un rejet important de substances radioactives est toujours possible à Bugey. 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 (2030). 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. Le rapport (EIA-REPORT D.3b 2026) ne présente pas d'analyses quantitatives pour appuyer les affirmations selon lesquelles la contamination alimentaire resterait inférieure aux limites fixées par l'UE à des distances supérieures à 5 km après 7 jours et à moins de 1 km après un an. En outre, l'évaluation omet les informations sur les dépôts au sol, malgré leur importance pour l'évaluation des impacts radiologiques à long terme et de la contamination potentielle de la chaîne alimentaire.

La modélisation de la dispersion atmosphérique et déposition réalisée par l'équipe d'experts démontre que, dans certaines conditions météorologiques, un accident grave à réacteur 3 de Bugey pourrait entraîner des contaminations du sol de Cs-137 en Autriche supérieurs au seuil national de 650 Bq/m². Bien que l'étude n'évalue pas la probabilité de telles conditions, les résultats indiquent que des impacts transfrontaliers supérieurs à ceux impliqués dans les documents d'EIE ne peuvent être exclus.

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

Évaluation du calendrier

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

1 INTRODUCTION

The Bugey NPP consists of four operating pressurized water reactors with a capacity of 900 MWe each. These reactors were commissioned 1978 and 1979 respectively.

France notified the 4th Periodic Safety Review ("Public consultation procedure on the 4th safety review report") of the Bugey nuclear power plant (reactor 3), 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 Ain. 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 submit-ted EIA documents in the framework of an expert statement.

2 PROCEDURE

2.1 Treatment in the EIA documents

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

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

The 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 reactor, which is considered the current state-of-the-art reference.

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

900 MWe reactors (32 units):

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

1300 MWe reactors (20 units):

- Timeline: Construction periods generally started in the late 1970s and continued into the late 1990s.
- Sub-types: Divided into type P4 and type P'4. Plants include Paluel, Cattenom, and Belleville.

1450 Mwe Reactors (4 units):

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

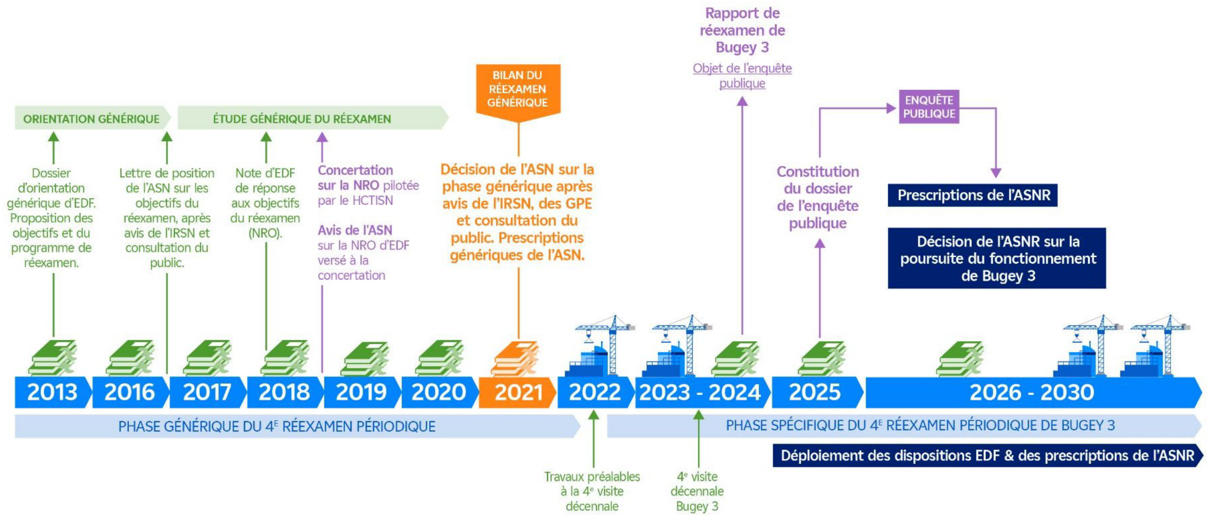
The subject of this report is the 900 MWe fleet. The 900 MWe fleet consists of 32 reactors of the CP type, which are 3-loop pressurized water reactors. This fleet includes three sub-types: CP0, CP1, and CP2 (with CP1 and CP2 often jointly referred to as CPY). While Fessenheim units 1 and 2 (CP0) were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

France is conducting the 4th PSR in two phases a general and a specific phase. For the 4th PSR of the 900 MWe nuclear power plants, EDF has set as a general guideline the objective of achieving the nuclear safety targets of the latest generation of reactors, whose reference reactor for EDF is the EPR-Flamanville 3. This guideline has been confirmed by the ASNR. The general 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 general phase is complete, inspections of all 32 reactors at the 900 MWe nuclear power plants will follow over a period of approximately ten years (from 2019 to 2031). EDF submits a review report to the government and the ASNR. This is prepared after the ten-year reactor inspection, during which modifications and inspection and maintenance work are carried out. The following timeline shows the main stages of the 4th PSR for Bugey 3.

Figure 1: Main stages of the 4th PSR for Bugey 3 (EIA-REPORT D.1 2026)

Main stages for Bugey 3



Source: Environment Agency Austria

Public Involvement in the PSR

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

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

While the public involvement process had similarities to an 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.

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

It is stated that the method used is in line with international best practices and consistent with the approach recommended by the IAEA in its Safety Guide No. NS-G-2.12 "Ageing Management for Nuclear Power Plants." (EIA-REPORT D.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 Bugey 3 from May 5 to 7, 2024, with the results meeting expectations.

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

Components whose performance may deteriorate due to aging and whose failure may have an impact on safety are documented and regularly inspected. In this context, inspections, checks, and maintenance work were carried out on the following SSCs during the 4th PSR:

structures, control and monitoring systems, electrical cables, mechanical and electromechanical equipment, electrical equipment, and instrumentation.

Following completion of the aging control analysis of the SSCs of Bugey 3, maintenance and control measures were carried out, along with modifications to ensure the continued suitability of this unit 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 D.2 2026)

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

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

- Containment,
- Structures.

The studies conducted for the 900 MWe CP0 reactors were adopted by the Bugey nuclear power plant in order to decide on the management of the aging of SSCs. The Bugey nuclear power plant teams thus adopted the studies conducted at the generic level and identified any specific features related to the SSCs of unit 3. During the ten-year inspection, the SSCs underwent a series of maintenance operations, inspections, tests, non-destructive tests, or modifications. (EIA-REPORT D.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 reactor 3 at the Bugey nuclear power plant for continued operation under satisfactory safety conditions for the ten-year period VD4-VD5. (EIA-REPORT D.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,
- containment.

Without justification, it is stated that no checks are to be carried out for Bugey 3 as part of the supplementary investigation program. (EIA-REPORT D.1 2026)

Stress corrosion of the auxiliary lines

As part of the proceedings initiated at the end of 2021 concerning “stress corrosion” on the auxiliary lines of the main primary circuit, investigations on the various reactors have shown that 900 MWe reactors such as those at Bugey are hardly susceptible, if at all, to this phenomenon. In consultation with ASNR, a strategy for dealing with the nuclear power plants and a corresponding inspection program were established. During inspections of 11 welds on production block reactor 3 of the Bugey nuclear power plant as part of its 4th PSR, no deviations were found that would have required repair. (EIA-REPORT D.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 D.2 2026)

Of the 256 materials inspected that are qualified for accident conditions (MQCA), deviations were found in fan motors, a pump motor, a valve switch and a switchboard battery. All deviations found have been corrected.

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 com-prehensive procedure involves a considerable amount of work and makes it possible to guarantee the extension of the service life up to the 5th PSR.

The following two projects are mentioned (EIA-REPORT D.3 2026):

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

Safety relevant events

According to the EIA-REPORT D.2 (2026), between January 2013 and December 2022, the Bugey nuclear power plant reported 16 significant events. None of these had any noticeable impact on the environment. Each time, corrective and preventive measures were implemented and their effectiveness was verified. This analysis of ten years of operating experience confirms that the management of significant events is correctly integrated into the Bugey nuclear power plant's management system.

It is further explained that, at the time of publication of the EIA-REPORT, reactor 3 of the Bugey nuclear power plant has reported a safety-related incident classified as level 1 on the INES scale, for which corrective measures have been planned but not yet completed in accordance with the applicable regulations.[1] This incident should has been corrected by 2025 at the latest. (EIA-REPORT D.2 2026)

3.2 Discussion

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

[1] This concerns the SEB system components in the event of an earthquake.

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 is therefore closed. (ASN 2021b)

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

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

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

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

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

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

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

Evaluation of significant effects

As part of this expert statement, an evaluation of safety-related events in reactor 3 of the Bugey NPP from January 2021 to February 2026 was carried out based on reports of the ASNR.[2]

[2] https://annual-report.asn.fr/controle/l-asnr-en-region/auvergne-rhone-alpes/centrale-nucleaire-du-bugey/avis-d-incidents#active-tab

Non-compliance with the general operating rules (RGE)

An evaluation of safety-related incidents over the past five years published by the ASNR revealed a number of incidents that were related to non-compliance with the general operating rules (RGE). The RGE are a collection of regulations approved by the ASNR that define the permissible operating range of the plant and the associated regulations for reactor operation. In particular, they specify the maximum repair periods in the event that the systems required for reactor safety are unavailable. The following non-compliance events at Bugey 3 were preceded, for example, by a component failure or a maintenance/operating error. The reason could be a lack of safety culture combined with a large number of ageing related events.

The following paragraph lists these events for Bugey 3:

- On 18 November 2025, the operator of the Bugey nuclear power plant reported a safety-related incident involving the delayed detection of the **unavailability of the emergency power supply** for reactors 3 and 4. On 14 November 2025, the operator discovered in reactor 4 a failure in the emergency power circuit. After an inspection, the operator discovered an identical fault in reactor 3. Both faults were caused by a maintenance error. The emergency power supply was unavailable for longer than permitted by the general operating rules (RGE), so the event was classified as Level 1 on the INES scale.
- On 3 April 2025, the operator of the Bugey nuclear power plant reported a safety-related event for reactor 3 due to the **reactor exceeding its permissible thermal power**. As the general operating rules (RGE) were not complied with, this event was classified as level 1 on the INES scale.
- On 4 October 2024, the operator of the Bugey nuclear power plant reported a safety-related incident involving non-compliance with material requirements for reactor 3. The pressurizer, which regulates the pressure in the primary circuit, is equipped with three pairs of valves. These must be functional under all circumstances in accident situations. In 2022, the operator noticed damage that could impair the control of these valves. However, the other two valve pairs were only examined during the fourth ten-year inspection in 2023/24, and similar damage was found. In view of the danger of the **unavailability of all SEBIM valves** in the event of an accident, this incident was classified as level 1 on the INES scale.
- On 16 January 2024, the operator of the Bugey nuclear power plant reported a safety-related incident involving the **heat exchangers in**

the intercooling system. Reactor operation leads to natural fouling of the heat exchangers, reducing their cooling capacity. The heat exchangers are cleaned when a degree of fouling calculated using software is reached. On 10 January 2024, EDF detected anomalies in the software's calculation parameters. A review of the test results carried out since 2018 revealed that in 22 cases the degree of fouling had been incorrectly considered to be sufficiently low. As several operating conditions were not permitted by the technical operating specifications, this event was classified as level 1 on the INES scale.

- On 14 October 2021, EDF reported a safety-related event involving the simultaneous **failure of both lines of the raw water supply** of reactor 3. Investigations revealed that a valve had not been fully reopened. As a result, this pump was unable to perform its function. Due to non-compliance with a general operating rule (RGE), the ASNR classified this event as level 1 on the INES scale.

Deficiencies of the seismic resistance of various components

On January 27, 2026, EDF reported a significant safety event to the ASNR concerning anchoring defects in the civil engineering of certain equipment critical to safety. EDF's report is the fourth update to its report of September 21, 2021. This update affects 30 reactors, including reactor 3 at the Bugey nuclear power plant. The significant event now concerns all reactors in France, except the EPR reactor in Flamanville.

As part of the conformity monitoring program, EDF checks the compliance of civil engineering anchors for safety-critical equipment (piping, electrical equipment, motors, pumps, etc.). These checks revealed discrepancies in certain anchors, particularly in terms of the number, diameter, and placement of dowels. These discrepancies date back to the construction of the reactors and could have compromised the stability of the supported equipment in the event of an earthquake. Given its potential consequences for these reactors, this event is classified as level 1 on the INES scale

This incident raises serious questions about the conformity checks carried out during the PSR. Firstly, these safety-related deficiencies have existed since the plant began operating without being detected in previous inspections. Second, similar discrepancies had already been found in other plants of the same design almost five years ago, and it took so long for these defects to be found and fixed in Bugey 3.

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 structures, systems and components (SSCs) poses a major challenge for the plant, which has been in operation for more than 40 years.

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

A recently reported safety-related incident involving faulty anchors for earthquake protection raises serious questions about the conformity tests carried out to date. Firstly, these safety-related defects have existed since the plant was commissioned without being detected during previous inspections. Second, similar deficiencies were identified in other reactors almost five years ago, and it took that long for these deficiencies to be discovered and rectified in Bugey 3.

- The justification that no checks are to be carried out for Bugey 3 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 resistance with regard to earthquakes should be considered.

4 EXTERNAL HAZARDS

4.1 Treatment in the EIA documents

EIA-REPORT D.1 (2026, p. 32-37) provides a general overview of the external hazard types considered in the LTO process. The list accounts for the requirements stipulated by ASNR (ASN 2021) for the 4th PSR of the French 900 MWe reactor fleet. The following external hazards (natural or human-made) are addressed: earthquakes, extreme weather or climatic conditions (flooding, snow, heat wave, extreme cold, high wind, tornado), influences from rivers (ice drift, icing, siltation, oil spills, silting, low water), lightning and electromagnetic interference, fire, industrial hazards (explosion, release of hazardous substances), aircraft crash, and malicious acts. The EIA documents note that studies on external hazards take into account the international standards set by WENRA. It is also stated that *"the use of the "Noyau Dur" [hardened safety core] to handle extreme events (earthquakes, floods, etc.) exceeding previously assumed values helps to meet these requirements"* (EIA-REPORT D.1 2026, p. 34).

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 (EIA-REPORT D.1 2026, p. 38). The SND is required to (i) envelope the ground motion of an earthquake with a recurrence interval of 20,000 years, based on probabilistic seismic hazard assessment, (ii) envelope the SMS increased by 50%, and (iii) take site effects into account.

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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

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

EIA-REPORT D.2 (2026, p. 132-135) states that the seismic hazard was re-assessed 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 re-assessment led to new seismic ground motion spectral accelerations applicable to the 4th PSR. The new SMS ground motion spectra for Bugey (4th PSR) exceeds the SMS of the 3rd PSR by 9% across the entire frequency range (EIA-REPORT D.2 (2026, p. 133). EDF claims that *"This exceedance is not significant (less than 10%)"*, without providing information on the actual ground motion values (e.g., PGA_H). EDF further claims that *"the studies conducted have shown that the development of SMS between the 3rd and 4th PSR had no influence on the earthquake safety review of the EIPS [elements important to safety]"*. EIA documents, however, provide no information on the content, methodology and results of studies performed.

The soil profile at the Bugey site comprises of about 100 m thick clastic deposits (Quaternary terrace deposits and „Molasse“), and underlying Jurassic carbonate rock. It is claimed that this soil profile makes it possible to rule out seismic site effects for Bugey as defined in RFS 2001-01[7]. Further details, e.g., the s-wave velocity of the top soil (Vs30), a crucial parameter for assessing site effects, are not provided in the EIA documents.

According to EIA-REPORT D.2 (2026, p. 133), EDF is currently involved in a pilot study to evaluate the safety significance of faults in the surrounding of Bugey. The study is announced to include literature review, geophysical, geological, morphotectonic and, finally, paleoseismological analyses of faults. EIA documents do not include a tangible time schedule for the named investigations. Notwithstanding these investigations, EDF states that *"there is currently no data in the literature to suggest the presence of an active fault in the vicinity of the Bugey site".*

High temperatures: EDF updated the assessment of the maximum long-time air temperature at which all safety-relevant materials are subject to acceptable environmental conditions, projected over the next 30 years (TLD; Température Longue Durée) and the exceptional air temperature (TE; Température Exceptionnelle) defining functional limits (EIA-REPORT D.2 2026, p. 143). The re-assessment of high temperatures

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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.

accounts for the heatwave of 2019. Temperatures are determined for exceedance probabilities of 10^{-2} per year and 70% confidence. The assessment accounts for the climate trend at the site. 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 all requirements related to the ultimate heat sink and identifying and reviewing the cases that adversely affect the cooling function.

Extremely low temperatures: Protection requirements for extremely low temperatures were developed based on lessons learned from the coldest winters of the last decades (notably 1984-1985 and 1986-1987) and implemented during the second PSR. Minimum temperatures for which protection must be provided have not changed between the 3rd and 4th PSR. This is justified by IPCC climate forecasts which predict a decrease of 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 the Bugey NPP, located on the Rhône, included the (re-) assessments of extreme precipitation (rain and peak rainfall intensity), high ground water, river flooding and the effects of flood waves caused by dam break (i.e., the Vouglans dam upstream of Bugey). Re-assessments concluded higher rainfall intensities and *"minor"* modifications of the river flood hazards (EIA-REPORT D.2 2026, p. 127-132). Detailed information on methods, data and assumptions used to derive the maximum flood level are not communicated. EDF also analyzed the volumetric flood protection devices and its resistance against seismic impact up to the SMS to account for the hazard combination earthquake and flooding.

High wind and tornado: The EIA documents state that the reassessment of hazards by storm do not require any update (EIA-REPORT D.2 2026, p. 151). 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 D.2 2026, p. 153). The occurrence probability of the design basis tornado consequently is $<10^{-4}$ per year and in line with international requirements. Assessments consider the dynamic wind pressure; the sudden pressure drop in the center of the vortex and wind-blown projectiles. The EIA documents conclude that protection against high wind and wind-blown projectiles is sufficient to also protect the NPP against effects of the reference tornado (EF0 on the Enhanced Fujita tornado scale).

Availability of the ultimate heat sink: 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 D.2 2026, p. 148). The activities were initiated after the clogging of water intakes of the NPPs Choos, Cruas and Blayais by frazil ice and flotsam in 2009. Analyses for Bugey include the re-assessment of low water level, the combination low water level/icing, frazil ice and other phenomena threatening the cooling water intake by clogging, sedimentation in the feeder channels (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 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 D.2 2026, p. 154).

Snow: SSCs are designed according to the "Snow and Wind Regulations" of 1965, which were in effect at the time of the construction of Bugey NPP. These regulations have been updated several times since their initial publication, most recently in 2009. Under 4th PSR 900, 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 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 900 EDF revised the data on industrial facilities in the surrounding 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^{-10} per year. No necessities for retrofitting were revealed (EIA-REPORT D.2 2026, p. 159-160).

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 Bugey site limit due to air traffic is less than 10^{-6}/reactor year. 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 D2 2026, p. 160-161).

Upgrades of protection measures:

Safety upgrades that have already been completed and planned safety upgrades are described in EIA-REPORT D.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 Bugey 3, 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. In addition to this general measure, the EIA documents[8] list a number of specific improvements including the following measures to protect the NPP from external hazards:

Table 1: Upgrading measures for SSCs important to safety with respect to external hazards, reactor Bugey 3 (adapted from EIA-REPORT D.2 2026)

LLBU2583	External flooding	Correction of bypass of volumetric protection	Completed
PNPE0048	Earthquake	Earthquake resistance of fire dampers	Completed
PNPE0055	Earthquake	Earthquake safety and increased safety of fire protection devices	Completed
PNPE0118	Earthquake	Seismic resistance of ventilation of battery rooms	Completed
PNPE0165	High wind	Protection against wind-blown projectiles	Completed
PNPE0186	Low temperature	Insolation and electrical heating of demineralisation facility	Completed

[8] EIA-REPORT D.1 2026; EIA-REPORT D.2 2026

PNPE0238	Earthquake	Increase of the seismic resistance of heating oil tanks to withstand ground motion exceeding the SMS	Completed
PNPE0294	Earthquake	Correction of non-conformity piping of sys-tems	Completed
PNPP0419	Earthquake	Implementation of automatic shutdown for earthquake	Completed
PNPP0543	High temperature	Air conditioning for pump station Bugey	Completed
PNPP0675	External flooding	Protection against external flooding by direct discharge onto the platform	Completed
PNPP0679 tome A	Earthquake	Seismic reinforcement of the water level gouge of the spent fuel pool	Completed
PNPP0764	External flooding	Treatment of collection basins for rainwater drains from diesel engines	Completed
PNPP0883	External flooding	Upgrade of protection of hardened safety core against external flooding	Completed
PNRL0841	High wind	Modification of protection against wind-blown projectiles	Completed
PNRL0846	External flooding	Elimination of risks of bypassing volumetric protection in the pumping station	Completed
PNRL0853	High temperature	Update of parameters for the automatic foul-ing monitoring of heat exchangers (cooling system/emergency domestic hot water)	Completed
PNRL0916	Earthquake	Earthquake resistance of cabinets	Completed
PNRL0922	Earthquake	Bypasses of volumetric flood protection	Completed
PNPE0056	Earthquake	Earthquake resistance hardened safety core buildings	Open(1)
PNPE0048 tome B	Earthquake / fire	Earthquake safety plus safety of fire protection	Open(1)
PNPE0115	Earthquake	Automatic reactor shutdown in case of earth-quake and information about a significant earthquake, hardened safety core	Open(1)
PNPE0191	Earthquake	Seismic resistance of cable trays	Open(1)
PNPE0285	Earthquake	Seismic resistance cable trays hardened safety core	Open(1)
PNPE0305	Earthquake	Establishment of a robust H1 situation detec-tion system for earthquakes, hardened safety core	Open(1)

PNPRE0332	Earthquake	Earthquake resistance piping, hardened safety core	Open[1]
PNPE0335	Earthquake	Robustness of automatic shutdown of circulating water at earthquake-induced flooding	Open[1]
PNPE0357	Earthquake	Seismic resistance electrical equipment and control systems, hardened safety core	Open[1]
PNPE0358	Earthquake	Seismic resistance ventilation, hardened safety core	Open[1]
PNPE0428	Earthquake	Seismic resistance isolation demineralisation, hardened safety core	Open[1]
PNPE0478	Earthquake	Seismic resistance instrumentation and control, hardened safety core	Open[1]
PNPP0913	Earthquake	Earthquake resistance building bridges, hardened safety core	Open[1]
PNPP04198	Earthquake	Implementation of automatic reactor shutdown for earthquake	Open[1]
TCDI0020	Earthquake	Seismic robustness of instrumentation against SND, hardened safety core	Open[1]
PNPE0119	Tornado	Passive protection of reactor building against tornado	Until end of 2027
PNPE0333 line A	Earthquake	Earthquake resistance of the main primary circuit, the main secondary circuit and the supporting structure	Shutdown VD4 latest
PNPE0333 line B	Earthquake	Earthquake resistance of the main primary circuit, the main secondary circuit and the supporting structure – reinforcement for SND	Phase B (30. April 2029)
PNPE0377	Earthquake	Strengthening the resilience of the compression and filtration system, building U5 to SMS.	April 2030 (Phase “supplements”)
PNPE0419	Earthquake	Implementation of automatic scram for earthquake - corrections	Shutdown VD4 latest

[1] *Changes whose implementation will be completed as part of Phase A of 4th PSR 900 at Unit 3 of the Bugey Nuclear Power Plant.*

Earthquake: The attachment of EIA-REPORT D.2 (2026) lists numerous actions related to the earthquake resistance of SSCs of Bugey 3 (Table 1). 20 out of 28 upgrading / retrofitting measures have not been completed by now. Among the open issues are measures which are part of the implementation of the Hardened Safety Core (Noyau Dur). Most of the measures necessary to ensure that the Hardened Safety Core withstands the „Noyau Dur" earthquake (SND) remain to be completed in

Phase A of the PSR. Some measures are envisaged to be completed until April 2029 (EIA-REPORT D.2 2026).

External flooding: Flood protection of the site platform is re-enforced by means of sills, stoplogs and concrete walls eliminating the risks of circumventing volumetric protection, reinforcing the protection of the reactor core, and provisions to protect against flood levels exceeding the Hardened Safety Core (Noyau Dur) (EIA-REPORT D1 2026, p. 36). All measures except the last mentioned have been implemented.

High temperatures: EIA-REPORT D.1 (2026, p. 37) and EIA-REPORT D.2 (2026, p. 144-145) list the following measures implemented between 2013 and 2017: modification of the pollution monitoring of heat exchangers to improve cooling by the ultimate heat sink (water of the river Rhône), replacement or protection of temperature-sensitive equipment with heat shields (diesel valves, current transformers, cables, sensors, fire alarm control panels, etc.), installation or replacement of cooling units, improvements of air conditioning of buildings containing SSCs important to safety by increasing ventilation performance and/or cooling capacity and installation of air conditioning systems. With respect to high cooling water temperature, criteria were set for the maximum permissible fouling of heat exchangers. It is said that compliance with the maximum permissible fouling enables the normal operation of the units at high water temperatures of the heat sink.

Low temperatures: According to EIA-REPORT D.1 (2026, p. 39), protection against frazil ice is provided by recirculating warm cooling water to the heat exchangers of the emergency raw water System. The measures prevent clogging of the water intake by frazil ice. Results of the 4th PSR gave rise to a number of additional safety measures concerning the emergency diesel generator building, the pumping station, and the turbine hall (EIA-REPORT D.2 2026, p. 147).

High wind and tornado: The installation of protective devices for the filter systems of the cooling source against wind-blown projectiles (high winds and tornadoes) has been completed (EIA-REPORT D.1 2026, p. 37-38).

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

Lightning and electromagnetic interference: The safety requirements applicable to the 4th PSR of the 900 MWe reactor fleet include new requirements for lightning protection. Studies have shown that only limited changes are required at the Bugey site to achieve the lightning protection objectives implemented under the 4th PSR (EIA-REPORT D.2 2026, p. 154).

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 Bugey NPP.

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. 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, 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[9]. The cited editions are outdated by the SRLs published in 2021.

Non-conformity with WENRA Reference Levels is observed for earthquake and seismic ground shaking. The Design Basis Earthquakes (DBE) for Bugey and the other reactors of the French 900 MWe fleet are still based on deterministic analyses. Demonstration that the deterministically determined DBE is equivalent to a PSHA-derived design basis earthquake with an average recurrence interval of 10,000 years is missing (see discussion below). It therefore remains to be demonstrated that the seismic resistance of all SSCs important to safety is sufficient to conservatively ensure the fundamental safety functions for a DBE with an average recurrence interval of 10,000 years as required by WENRA (2021). The authors of this report assume that adequate protection against a probabilistically derived DBE, should it be higher than the

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

deterministic value for which the plant was designed, is intended to be ensured by the Hardened Safety Core (Noyau Dur). This, however, would contradict the Defence-in-Depth (DiD) concept and the separation of DiD levels because the DEC equipment of the Noyau Dur could become necessary to protect the plant against design basis hazards, i.e., the probabilistically derived DBE. The Hardened Safety Core is classified as a fourth DiD level system which is required as an additional and independent level compared to the third DiD level. The Hardened Safety Core therefore cannot 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)[10] suggests that design basis values were only established after the start of construction of Bugey 3 (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 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[11]. 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).

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[10] 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.

[11] 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.

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 re-assessment of the deterministically design basis earthquake SMS. A re-assessment in the course of the 4th PSR led to higher ground motion values (spectral accelerations). EDF, however, claims that these higher hazard values have no influence on seismic safety (EIA-REPORT D.2 2026, p. 133). The EIA documents provide no information on the content, methodology and results of the seismic hazard re-assessment. It remains open how EDF ensured that the seismic resistance of SSCs important to safety is in line with the requirements of the newly defined higher design basis ground accelerations. Such demonstration requires the assessments of seismic fragilities of SSCs important to safety, the determination of seismic margins etc. It is also unclear if the deterministically determined SMS ground motion values have been compared to the results of probabilistic hazard analyses as required by WENRA (2021), Issue TU.

RFS 2001-01[12] requires that site effects be taken into account in seismic hazard assessment when the s-wave velocity of the top soil (Vs30) is less than 300 m/s. Bugey 3 is located on Quaternary terrace sediments of the Rhone and on top of about 100 m thick „Molasse“ deposits. Such sediments are typically characterized by low Vs30 values. The EIA documents nevertheless claim that *“possible amplifications of seismic signals due to the geological conditions of the subsurface at a site cannot be considered site-specific”* concluding that [this] *“makes it possible to exclude site effects for Bugey as defined in RFS 2001-01”* (EIA-REPORT D.2 2026, p. 134). The latter conclusion appears doubtful in view of the lack of evidence from geophysical surveys to support EDF's claims.

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 Bugey nuclear island and site structure of 0.145 g. PGA values reported for the SND are

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

0.3 and 0.4 g, respectively (Table 2). These values appear low compared to the ground motion derived from the published probabilistic seismic hazard assessment (PSHA) by MARIN et al. (2004) who shows a PGA value of at least 0.25 g for the site and a recurrence interval of 1975 years (Figure 1)[13]. This value (0.25 g) is significantly higher than the ground motion reported for the SMS (0.145 g) indicating that the deterministically derived SMS fails to meet WENRA (2021) requirements stipulating that *"an exceedance frequency not higher than 10^{-4} per annum shall be used for the design basis events"*[14].

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

NPP	Bugey 3
Start of construction	1973
Start of commercial operation	1978
DBE Nuclear island	0.1 g 2011: 0.145 g
DBE Site structure	0.1 g 2011: 0.145 g
SND	PGA=0.3 g (1) PGA=0.4 g (2)

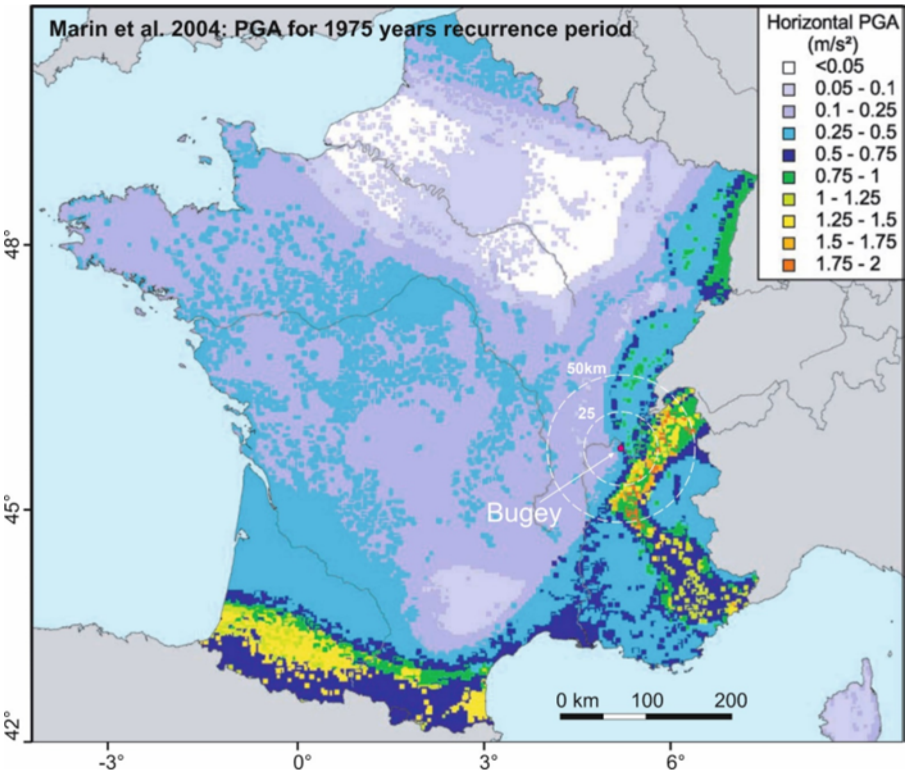
(1) IRSN 2012 (2) IRSN 2013

[13] PGA colour codes of the hazard map by MARIN et al. (2004) place the site of Bugey in the class 0.25 – 0.5 g or 0.5 – 0.75 g

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

Figure 2: Peak ground acceleration for 1975 years recurrence period in mainland France. (Redrawn from: MARIN et al. (2004))
The site of Bugey NPP (red circle) is located area color-coded in light blue (0.25 – 0.5 g) or dark blue (0.5 – 0.75 g). Circles around the Bugey NPP indicate the site near-region and site region according to IAEA (2022) (radius 25 and 50 km from the site, respectively).

PGA for 1975 years recurrence period



Source: Environment Agency Austria

In tectonic terms, the Bugey site is located in the Bresse Graben, which together with the lower and upper Rhine Graben forms the Cenozoic European rift system. EIA-REPORT D.2 (2026, p. 134-134) provides evidence that EDF is aware that geological and seismological data indicate the existence of several potentially active faults in the near-region and region around the Bugey site, but it seems that these faults have not taken into account as potential seismic sources in the seismic hazard assessments performed so far. EDF is currently performing a “pilot study” to evaluate the safety significance of active faults in the surrounding of Bugey. Notwithstanding these investigations, EDF states that *“there is currently no data in the literature to suggest the presence of an active fault in the vicinity of the Bugey site”* (EIA-REPORT D.2 2026, p. 133). We consider the quoted statement to be factually wrong. The active fault map by JOMARD et al. (2017) shows four Quaternary faults within a distance of 25 km from the site (near-region of the site to IAEA 2022), including one fault within a few kilometers distance from the NPP. Another four faults are shown within a distance of 50 km. Furthermore, the Neopal database (BAIZE et al. 2002; NEOPAL 2009) includes numerous locations in the same area for which data on neotectonic fault activity are available. For proved or potentially active faults as those shown by JOMARD et al. (2017), WENRA (2020b p.11ff) suggests systematic fault mapping and collecting paleoseismologic information. Efforts should at least be made in the near-region of the site (not less than 25 km) to collect geological, geophysical, geomorphologic, geodetic and paleoseismological data for identifying and characterizing active faults. At the background of the existing literature (RITZ et al. 2021 and references therein) it is remarkable that a process for active fault characterization and paleoseismological investigations has not been implemented much earlier.

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 for exceedance probabilities of 10^{-2} per year and 70% confidence accounting for the climate trend at the site. The basis for considering snow loads seem to be common building codes (EIA documents 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 nuclear power plants currently in operation have a certain degree of protection against possible terrorist attacks due to their design, e.g., through relatively thick outer walls and diverse and redundant safety systems. Accidental aircraft crashes have been taken into account in the design of nuclear power plants for several decades. However, only accidents involving smaller sports aircraft and/or military aircraft were considered. It was only after the attacks of September 11, 2001, that the consequences of a deliberate crash of a commercial aircraft were considered. Older nuclear power plants, such as the Bugey 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 nuclear power plants and therefore, in accordance with WENRA safety objective (O2), should only have minor radiological consequences. To prove this, the effects of direct and secondary impacts of the aircraft accident must be considered (vibrations/shocks, burning and/or explosion of the aircraft fuel). In addition, buildings or parts of buildings containing nuclear fuel and safety-relevant safety equipment should be designed in such a way that no kerosene can penetrate.

The increasing risk due to aging effects must also be taken into account for Bugey 3: A study uses numerical simulations to investigate the influence of aging on the effects of a military aircraft impact on a nuclear power plant. The results show that the aging of a plant increases its susceptibility to large-scale or localized penetrations. The greater the degradation of the materials, the lower the residual resistance and the greater the risk of wall perforation. With the same impact force, the strength of the aged containment is reduced by approximately 30%. (FRANO 2021)

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

Nuclear Threat Initiative (NTI)

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

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

International Physical Protection Advisory Service (IPPAS)

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

The last IPPAS mission was completed in France with the follow-up mission in 2018. Due to the changed security situation in Europe and the low NTI Index score, another IPPAS mission should be considered to improve the security measures. (IAEA 2026a)

4.3 Conclusions

The EIA documents provide ample information on hazard types considered in the safety demonstration for Bugey 3 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 Bugey and the other reactors of the French 900 MWe fleet are still based on deterministic analyses. Defining the Design Basis Earthquake (DBE) on deterministic methods is no longer state of the art. 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 Bugey nuclear island and site structure of 0.145 g. PGA values reported for the SND are 0.3 and 0.4 g, respectively. These values appear low when compared to the PSHA derived ground motion of at least 0.25 g for the Bugey site, calculated for a recurrence interval of 1975 years (MARIN et al. 2004). This value (0.25 g) is significantly higher than the ground motion reported for the SMS (0.145 g) indicating that the deterministically derived SMS fails to meet the WENRA (2021) requirement stipulating that *"an exceedance frequency not higher than 10^{-4} per annum shall be used for the design basis events"*[15]. We consequently 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 Bugey site is located in the Bresse Graben, a tectonic structure containing numerous potentially active faults which have not been taken into account in seismic hazard assessments so far. EDF states that *"there is currently no data in the literature to suggest the presence of an active fault in the vicinity of the Bugey site"* (EIA-REPORT D.2 2026, p. 133). We consider this statement to be factually wrong because several authors pointed to the existence of Quaternary and therefore active faults in the vicinity of the site (BAIZE et al. 2002; NEOPAL 2009; JOMARD et al. 2017; RITZ et al. 2021).

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 cannot be used to compensate for existing deficits in terms of the protection against design basis events.

With respect to safety upgrades of Bugey 3, it is evident that one of the most important measures to provide protection against external

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

hazards is the implementation of the Hardened Safety Core (Noyau Dur). However, the implementation of the Noyau Dur is still pending. Implementation of most of the measures is announced for Phase A of the 4th PSR, some for Phase B. For the latter 30. April 2029 is stated as a deadline. The fundamental decision to implement the Hardened Safety Core has been made in 2012 in the aftermath of the and the European Stress Tests (ASN 2012). The fact that the implementation of the Noyau Dur will be completed only 17 years thereafter appears remarkable at the background that WENRA requires the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). This suggests that the announced implementation schedules 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 Bugey 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 Bugey 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 Bugey 3, 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 ground motion values for the SMS and the SND.
- The methodology and database used for the deterministic definition of the SMS should be reviewed, in particular, if it is justified to disregard site effects. Background: Bugey 3 is located on "soft" fluvial and Molasse sediments which may be characterized by low Vs30 values.
- 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). Background: published PSHA results show PGA values of at least 0.25 g for a 1975 years recurrence period for the Bugey area indicating that the recurrence period of the SMS ground motion is much shorter than 10,000 years.
- Additional safety demonstrations should be required to ensure that all SSCs relevant to safety can cope with a probabilistically derived new Design Basis Earthquake (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 Bugey 3.
- The methods, data and assumptions used to determine the SND for Bugey 3, 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 2020 a,b).
- Dedicated assessments of near-regional faults for which it cannot be excluded that they are active should be performed in line with WENRA (2020b). The approach should include field geology, geophysical mapping, morphostructural and dating studies, and paleoseismology. Background: existing literature highlights numerous Quaternary faults in the 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 Bugey 3 against design events, i.e.,

events with recurrence intervals of 10,000 years or less (e.g., earthquakes). This is to ensure the independence of Defence-in-Depth (DiD) levels 3 and 4.

- It should be evaluated if the long timeframe for implementing the Noyau Dur at the Bugey 3 reactor is in line with the requirement of the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). Background: the timeframe for implementing the Noyau Dur extends up to 2029, i.e., 17 years after ASNR's initial decision to implement Hardened Safety Cores at the French NPP fleet.
- 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 for France assessed? Are improvements planned with regard to "security culture", "cybersecurity" and "protection against insider threats"?

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

5.1 Treatment in the EIA documents

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

Each phase addresses two core objectives:

- Safety Requirements Compliance: A thorough assessment of the plant's adherence to the defined and evolving Design Basis safety requirements.
- State-of-the-Art Upgrades: Identification and specification of measures required to align the plant with the Current State of the Art (SOTA) in nuclear technology.

The Flamanville 3 EPR (European Pressurized Reactor) serves as the reference standard for the Current State of the Art.

Scope of Measures and Review Focus: This chapter details the modifications and upgrades specified in EIA-REPORT D.1 (2026), EIA-REPORT D.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 D.1 (2026) and EIA-REPORT D.2 (2026) provide executive summaries and outline the highest-priority measures identified for implementation regarding Accidents Without Core Melt.

Measures Implemented:

Accidents -1: Re-supply of the ASG tank from the fire protection network (JP)

Modification: re-supply of the Auxiliary Feedwater System (ASG) tank from the fire protection systems JP (PNPP0864). Scheduled within Phase A of the 4th PSR modifications.

Rationale: Increases available water resources to the Steam Generators for heat removal during accidents without core melt.

Accidents -2: Increased discharge capacity of the VCD-a regulating valves

Modification: increase the flow capacity of VCD-a regulating valve. Scheduled within Phase A of the 4th PSR modifications (PNPE0141).

Rationale: Enhances cooldown/depressurization capability during accidents without core melt.

Accidents -3: Lowered allowable iodine activity in the primary coolant during transients

The provided documents do not specify any reduction of the allowable iodine activity or related operational changes to purification/makeup/letdown systems for Bugey 3.

Key measures for the spent fuel pool

For the spent fuel pool EIA-REPORT D.1 (2026) and EIA-REPORT D.2 (2026) list the following items:

Measures proposed:

Pool-1: Fire

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

Pool-2: Accident scenarios

Following the transpose of EPR Flamanville 3 scenarios to 900 MWe plants, to further secure spent fuel pool cooling, EDF plans to duplicate the automatic isolation device on the suction line of the pool's normal cooling circuit, ensuring reliable isolation under accident conditions even if one device fails.

Measures implemented:

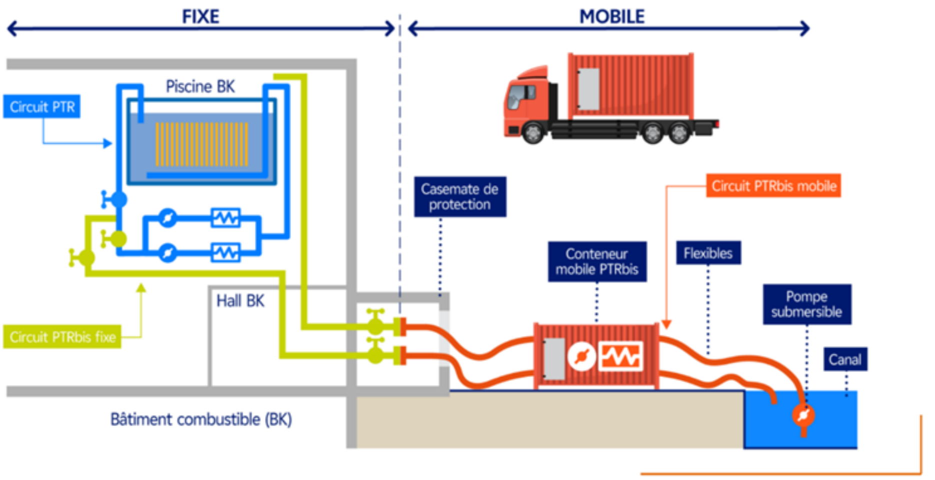
Pool-3: Additional pool cooling “PTR bis” (diversified, mobile)

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

Remaining complement: resolution of bubble injection aspects scheduled for Phase B.

Figure 3: Schematic diagram of the cooling circuit of the spent fuel pool (EIA-REPORT D.1 2026)

Schematic of the cooling circuit of the spent fuel pool



Source: Environment Agency Austria

EIA-REPORT D.2 (2026) represents the most extensive of the five reports submitted for the Plant Lifetime Extension (PLEX) review. Their section on risks is logically segmented into two main components:

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

- Re-evaluation (SOTA Comparison): Derivation of necessary measures by comparing the safety profile of Bugey 3 reactor against the Current State of the Art (SOTA), as defined by the Flamanville 3 EPR design.

- The "Conformity" section is deemed outside the scope of this discussion as it does not relate to Accidents Without Core Melt or the Spent Fuel Pool (SFP). The following focuses on the considerations within the Re-evaluation chapter.

Re-evaluation of Accidents Without Core Melt

EDF's approach to the "Accidents Without Core Melt" scenario 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 Accidents (RIA)

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

Thermal-Hydraulic and Heat Removal Transients

- Partial loss of primary coolant flow or Forced reduction of primary coolant flow
- Total loss of load and/or turbine trip
- Loss of normal feedwater to the Steam Generators (SGs)
- Malfunction of normal feedwater
- Excessive load increase
- Inadvertent opening of a secondary relief valve (OISS)
- Small break on secondary piping
- Major Steam Line Break (SLB/RTV), Category 4
- Major feedwater line break

- Momentary depressurization of the primary circuit
- Loss-of-Coolant Accidents (LOCA) and System Integrity Events
- Loss-of-Coolant Accident (LOCA) due to a small break with a diameter ≤2.5 cm
- Intermediate Break LOCA (IBLOCA/APRP BI), Category 4
- Inadvertent actuation (startup) of the safety injection system
- Inadvertent opening of a pressurizer safety valve
- Steam Generator Tube Rupture (SGTR), Category 3
- Category 4 SGTR (combined with a stuck-open secondary relief valve)

Equipment and Operational Failures

- Total loss of off-site power (or Loss of external electrical power supplies)
- Seizure/Locked rotor of a Reactor Coolant Pump (RCP / motopompe primaire)
- Fuel and Core Design Events
- Class 2 Power Capability (a capacity limit check for verifying the sizing of the Reactor Protection System)
- Fuel assembly misalignment in the core
- Fuel handling accident (in-reactor)
- Irradiated fuel container handling accident

Accidents Without Core Melt

Fully Implemented Modifications

The following modifications have been fully completed on Bugey Unit 3, and all associated documentation impacts have been integrated:

- PNPP0864 volume B: Replenishment of the ASG tank via the JP fire network
- PNPP0371: Improving the reliability of Reactor coolant pump's thermal barrier insulation
- PNPP0546: Ensuring the long-term viability of EAU auscultation (Containment instrumentation) with regard to DAO (Optimal Auscultation Device)
- PNXX0372: Installation of extensometers on the extrados of the reator building's prestressed wall
- PNPP0714: Water source for the Hard-Core backup (also classified as Fuel Pool/ND, but contribution to cases without meltdown)

Modifications Currently Being Deployed - Phase A

The following modifications are currently being deployed at Bugey Unit 3, with remaining integration activities scheduled for completion within Phase A of the 4th PSR modifications:

- PNPE0141: Increase in flow rate of VCD-a control valves
- PNPE0159: Increase in boron concentration of PTR and REA blankets and REA boron volume
- PNPP0595: Replacement of pressurizer safety valve heads (SEBIM)
- PNPP0838: Restriction of the control range on the right edge
- PNPP0864 volume A: Replenishment of the ASG tank by JP (supplement to volume B)
- PNPP0873: Evolution of the SIP-P process instrumentation system – re-parameterisation of RPR thresholds
- PNPE0152: Replacement of the power supply by the Emergency Turbo-Alternator with the Emergency Diesel Generators (DUS)
- PNPE0068: Installation of a hard-core electrical distribution system (contributes to scenarios with and without meltdown)
- PNPP0811: Installation of a System for evacuation of residual heat from the containment (EAS-ND) primary injection and residual power evacuation system (contributes to scenarios with and without meltdown)

Modifications Scheduled for Phase B

The following modifications are planned for deployment at Bugey Unit 3 during the subsequent Phase B of the 4th PSR modifications:

- PNPE0189: Primary fluid sampling device (downstream of CEPP exchanger) to prevent the risk of heterogeneous dilution (shutdown condition)
- PNPP0932: Connection to Safety Injection System/ Containment Spray System (RIS/EAS) double jacket for endoscopic sampling (interventions, sizing without core melt)

Spent Fuel Pool – BK (Bâtiment combustible)

Fully Implemented Modifications

The following modifications have been fully completed at Bugey Unit 3, and all associated documentation impacts have been integrated:

- PTR bis mobile diversified cooling system (PNPP0907): deployed on Bugey 3; documentation impacts integrated

- Automation of reactor building pool drain valves (PNPP0780): implemented to ensure automatic isolation of the reactor building's filtration lines
- Noyau Dur (hardened systems and arrangements for extreme conditions) ultimate makeup source (PNPP0714): implemented, providing a diversified water source for pool makeup
- Analog level measurement for BK pool (PNXX0752): implemented to improve continuous level monitoring
- Protection of Noyau Dur against external flooding (PNPP0883): implemented

Modifications Currently Deploying (Phase A)

The following modifications are planned for deployment at Bugey Unit 3 during the subsequent Phase B of the 4th PSR 900 modifications:

- PTR bis (PNPP0907, except line I): in deployment with Phase A completion; documentation updates in progress
- Fire protection of PTR cables, PNPP0949 line C: in deployment with Phase A completion

Modifications Scheduled for Phase B/Supplementary Phase/ Specific Phase

The following modifications are planned for deployment at Bugey 3 during the subsequent Phase B of the 4th PSR modifications:

- Doubling the automatic isolation of the spent fuel pool suction/drain line (PTR): PNPE0344 will automate isolation via PTR 017 VB, providing redundancy with existing PTR 001 VB
- Noyau Dur makeup routing to rector building via a diversified water source (SEG) (PNPE0258): fixed, seismically robust line for pool resupply.
- Reactor building arrangement for APR states (PNRL0803): Noyau Dur makeup to reactor building and steam exhaust arrangement.
- BK pool analogue level measurement chain (PNPP0824): additional chain planned for Phase B.
- Fire separation between PTR pumps (PNPP0949 line A): installation of a physical fire protection screen between PTR pumps.
- Device to amortize the drop of a spent fuel cask (PNPP0877): scheduled with deployment no later than end of 2025.

Document EIA-REPORT D.3 (2026) provides easy-to-use lists of measures but no new information in respect to EIA-REPORT D.1 (2026) and EIA-REPORT D.2 (2026). The documents EIA-REPORT D.4 (2026) gives an overview of the "Lessons learned by EDF from the consultation on the generic phase of the 4th periodic safety review of 900 MWe reactors". Although they dedicate a section to the robustness of the spent fuel pool no additional information is given. EIA-REPORT 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: ASG tank re-supply via the fire protection network (JP)

The re-supply of the Auxiliary Feedwater System (ASG) tank from the JP fire protection network is part of the 4th PSR measures for Bugey 3 (PNPP0864), with deployment split between lines A and B. This increases available water resources to the Steam Generators for heat removal in accidents without core melt. Status for Bugey 3 indicates line B is implemented and line A is planned in Phase A.

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.

Accidents-2: Increased discharge capacity of VCD-a regulating valves

The increase of VCD-a valve flow capacity (PNPE0141) addresses an analysis anomaly related to ASG water consumption and contributes to faster cooldown/depressurization and reduced radiological consequences in penalizing transients such as SGTR, as required by ASNR prescriptions under the 4th PSR.

Information Gaps: The report describes objectives and benefits but do not provide quantitative mass-flow figures before/after the upgrade. It omits the initial and final mass flow rates achieved by the upgrade. Crucially, a comparison is missing between the new maximum discharge capacity and the steam flow per steam line during normal operation to contextualize the magnitude of the capacity increase.

Potential Adverse Effects:

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

Accidents-3: Reduced allowable I-131 activity in the primary coolant during transients

The allowable equivalent I-131 activity limit during transients has been lowered (e.g., from 150 to 100 GBq/t), with a direct and proportional reduction of thyroid dose and a smaller reduction of effective dose; this specifically mitigates consequences for steam generator tube rupture scenarios (SGTR) and other accidents without core melt.

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. It does not specify operational changes to make-up/let-down or purification systems, nor do they detail handling of iodine spiking; only the limit reduction and its radiological rationale are stated.

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: Fire protection and separation of PTR trains

Fire protection and separation measures for PTR, including a physical fire protection screen between PTR pumps (PNPP0949 line A, scheduled Phase B) and fire protection of PTR cables in shared fire volumes (PNPP0949 line C, completing in Phase A) to prevent common-cause loss of both cooling paths represent a highly commendable and undoubtedly beneficial safety enhancement. The planned screen (line A) remains to be deployed; line C is in progress.

Pool-2: Mobile Cooling Capabilities

A mobile, diversified pool cooling system (“PTR bis”, PNPP0907) has been deployed on Bugey 3, providing a diversified cold source and enabling return to non-boiling cooling following loss of the normal PTR path.

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

Further studies

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

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

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

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

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

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

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

Implementation: EDF must update the relevant safety reports and integrate the associated lessons into the applicable technical specifications within the 4th RP 900 timeframe (i.e., by the tranche's Target Technical Specification milestone).

Site-specific aspects

Plant specific phase: Is there a specific date by when those measures will be fully implemented beyond that Phase B measures are due by 30 April 2029 and "Compléments" by April 2030?

PNPP0949 line A (fire screen between PTR pumps) is scheduled for Phase B—what is the justification for deferral and what interim protections apply?

What interim dispositions or compensatory measures are in place at Bugey 3 to manage safety until each deferred (Phase B/Supplementary Phase) modification is implemented?

5.3 Conclusions

While the 4th PSR for Bugey 3 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 reports should provide the initial and final mass flow rates for the VCD-a valve flow capacity upgrade (PNPE0141), along with a comparison to the nominal operational flow. This is necessary to quantify the safety benefit.
   - Adverse Effects Analysis: The analysis of the uprated VCD-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.
2. Establish Firm and Accountable Timelines
   - Phase windows and tranche-specific dates: The program commitments give windows for deferred items (Phase B by 30 April 2029; Supplementary Phase by April 2030). Requesting tranche-specific calendar dates within these windows for Bugey 3 will improve follow-up and transparency.
   - Study Status and Next Steps: For the Critical Heat Flux (CHF) experimental program (Requirement [Study-B]), EDF should provide an updated status on its completion and publicly commit to the defined work program and schedule for incorporating the findings. This is overdue, as the reporting deadline was December 31, 2024. The provided excerpts do not report the outcome/status; a status update request (completion, results, and planned integration) is therefore appropriate.
3. Clarify Status Reporting and Implementation Rationale
   - Justify Deferral: For PNPP0949 line A (fire screen between PTR pumps), scheduled in Phase B, the safety justification for deferral and detail the interim protections in place until installation should be provided. Note that line C (fire protection of PTR cables in shared fire volumes) is in deployment with completion in Phase A.
   - For the Noyau Dur makeup dispositions to the spent fuel pool, the interim arrangements that ensure drain-down prevention and makeup capability prior to installation of the fixed routes should be clarified: PNPE0258 (ASG-ND and fixed line to re-supply reactor building via SEG) is planned for Phase B, while PNPP0714 is already

implemented and PNRL0803 (reactor building makeup/exhaust arrangement) is planned for Phase B.

- Resolve Discrepancies and improve transparency: The Bugey 3 report provides tranche-specific status tables (fully implemented, Phase A, Phase B, “Complementation” and “Specific program”), and some items carry explicit deadlines (e.g., PNPP0877: no later than end-2025). Tranche-specific calendar dates for each item within the committed windows (Phase B by 30 April 2029; Supplementary Phase by April 2030, etc.) should be confirmed, and for each item reported as “implemented”, it should be specified, whether this denotes installation completion and commissioning (in line with requalification). In general, a simpler and reduced deadline system would increase transparency.

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

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 in the event of a core melt accident due to erosion of the basement, a device is used that is based on stabilizing the corium underwater after it has spread in the dry (PNPP0976)[16]. According to EDF, this solution is similar in principle to that used in EPR (core catcher). This arrangement complies with regulation [AG-A-I].

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

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

[16] PNPP0976: "Installation of a device for dry distribution and stabilisation of the corium under water" will be implemented during Phase A of the 4th PSR.

The reactors at the Bugey nuclear power plant are not affected by regulation [AG-A-II] on thickening the foundations, as their foundations are made of limestone-silica concrete.

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 (PNPE0460)[17].

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

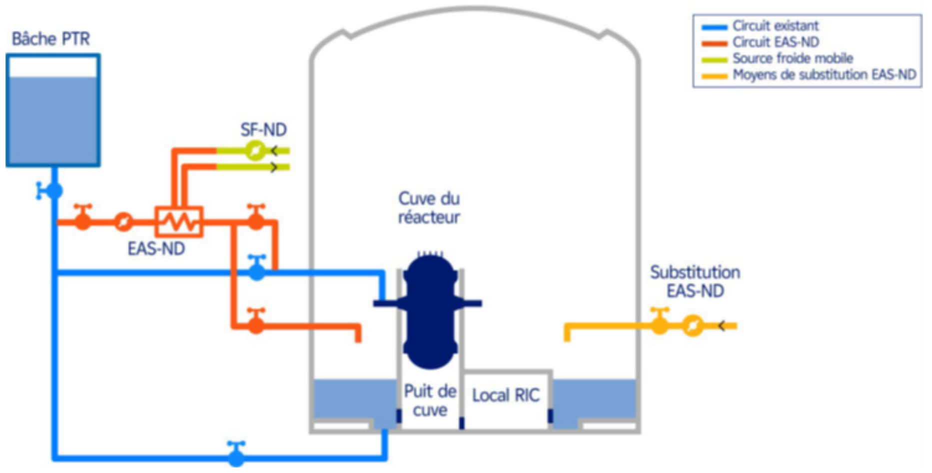
The measurement for detecting a vessel penetration (PNXX0746)[18] makes it possible to ensure water injection onto the corium at the most effective time. The cooling of the corium and the long-term removal of residual power are ensured by the EAS-ND and hard-core cooling source (SF-ND) measures.

[17] PNPE0460: "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.

[18] PNXX0746: "Detection of breaches and operation of the hydrogen recombinator at high temperatures" has been implemented.

Figure 4: Cooling devices in the event of core melt accident (EIA-REPORT D.1 2026)

Cooling Devices in the event of core melt accident



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 (PNPE0362)[19]. This measure complies with regulation [AG-B-III]. This replenishment is controlled by measuring the water level at the bottom of the reactor building (PNPE0386)[20].

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 (PNPE0387)[21] will be implemented.

According to the EIA-REPORT D.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 refilling of

[19] PNPE0362: "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.

[20] PNPE0386: "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.

[21] PNPE0387: "Device for detecting corium spread in the RIC room" will be implemented in Phase B of the 4th PSR.

the corium (PNPP0976)[22], along with the EAS-ND system (PNPP0811)[23] 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 approximately 10^{-7} / year, which is in line with the objective of avoiding lasting effects on the environment.

Removal of residual heat without filtered venting

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

The implementation of the EAS-ND provision (PNPP0811)[24] 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.

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

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[22] PNPP0976: see above

[23] PNPP0811: "Introduction of an EAS-ND system for water injection into the primary circuit and for dissipating residual power" will be completed as part of Phase A of the 4th PSR.

[24] PNPP0811: see above

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 (PNPP0541)[25] 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 (PNPE0362)[26]. These devices for collecting and recirculating comply with the regulations [AG-B-IV] and [AG-D-I].

To reduce the potential radiological consequences, the sodium tetraborate baskets will be implemented in the sump basins of the reactor building (PNPE0410)[27] at the latest for Bugey 3 in April, 2029 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.

Components whose resistance cannot be guaranteed during a core melt accident will be replaced with qualified materials (PNPE0347[28] und PNRL0986[29]).

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 (PNPP0870)[30]. In accordance with regulation [AG-C-II], the U5-System will

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[25] PNPP0541: “Introduction of a system for collecting wastewater in the event of a core-melt accident” will be completed as part of Phase A of the 4th PSR.

[26] PNPE0362: see above

[27] PNPE0410: “Installation of sodium tetraborate baskets in the sump basins of the reactor building” will be implemented during a specific program for Bugey 3.

[28] PNPE0347: “Replacement of the RCV089VP electric actuator” will be implemented in Phase B of the 4th PSR.

[29] PNRL0986: “Replacement of valves SEB 358/360 VE, JPD 990/992 VP and replacement of the closure/seal assembly of valve RPE 903 VP” will be implemented in Phase B of the 4th PSR.

[30] PNPP0870: “Strengthening the resilience of the containment's decompression and filter device in the event of an SMHV earthquake” has been performed.

be further reinforced to ensure its resistance to earthquakes of magnitude SMS (PNPE0377)[31].

Management of contaminated water

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

In accordance with regulation [AG-D-II], EDF has the necessary means to reduce water contamination in the reactor building following a core melt accident (PNPE0362)[32] and (PNPE0449)[33].

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. EDF will determine the possible measures to be taken with regard to the safety risks and the associated schedule.

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 Bugey 3.

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

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[31] PNPE0377: "Reinforcement of the compression and filter device of the U5 container in the event of an SMS earthquake" will be implemented at the latest on as part of the Supplementary Phase of the 4th PSR.

[32] PNPE0362: see above

[33] PNPE0449 "Study of a mobile water treatment module for treating contaminated water" will be performed as part of the Supplementary Phase of the 4th PSR.

objectives, core melt accidents that would lead to early or large releases should be practically excluded. The occurrence of certain severe accidents can be considered to be practically excluded "if it is physically impossible for the conditions to occur, or if it can be assumed with high confidence that the occurrence of these conditions is extremely unlikely". The concept of "extremely unlikely with high confidence" is an essential part of the IAEA's concept of "practical exclusion". Although this concept applies only to new reactors, it should also be applied to Bugey 3 in order to reduce the existing risks. Especially since the goal of the 4th PSR is to approach the safety level of the new EPR in Flamanville. The EIA documents do not include a systematic comparison between the safety level of the 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. For the Bugey 3, limestone-silica concrete has been used. Thus, the thickening of the basement is not seen as necessary.

The coolability of the corium in the ex-vessel phase was subject to large uncertainties. The geometry of the 900 MWe reactor cavity bottom consists of a circular cylinder of inner radius 2.6 m, sided by a rectangular area facing the RIC room), whose dimensions are approximately 4.0 m x 2.6 m. Thus, the total area of reactor pit and RIC room is 31.6 m². Referring to the indicative figure of 0.02 m²/MWth this translates to a necessary area of approximately 55 m² for the 900 MWe reactor. Consequently, the coolability of the 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. 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 (PNPE0460).

Although the "installation of a device for dry distribution and stabilization of corium under water" (PNPP0976) will be implemented during Phase A of the 4th PSR for Bugey 3, effective medium- and long-term cooling can only be guaranteed once all measures have been implemented after Phase B and Supplementary Phase.

It was one of the important lessons learned of the Fukushima accident that is important to have instrumentation that do not lose its function under accident conditions. EDF plans to install temperature measuring devices and instruments for measuring the water level at the bottom of the plant. (PNPE0386) 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 (PNPP0811) is being implemented during Phase A of the 4th PSR.

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 Phase B or 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 (PNPE0362).

Important components such as valves, which 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. An upgrade measure (PNPE0377) is not planned until the Supplementary Phase.

Management of contaminated water

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

IRSN assessed the consequences of a meltdown of the basement without a special device to limit contamination. At most river sites, the radionuclide concentration in the respective river could exceed the reference dose values for drinking water (0.1 mSv/year) by a factor of approximately 1,000 several months after the 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 during the Supplementary Phase.

It is envisaged to investigate the use of a mobile water treatment module for treating contaminated water during the Supplementary Phase. (PNPE0449). Thus, it is not clear if this measure will be implemented at all.

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

6.3 Conclusions

Severe accidents (SA) involving core meltdown were not taken into account in the design of the French 900 MWe reactors. However, as a result of previous PSRs, facilities and measures for SA management have been implemented. According to the ASNR, the objective of the fourth PSA for the 900 MWe reactors is to bring the safety level of the reactor closer to that of the EPR in Flamanville, a third-generation reactor. In third-generation reactors, features to mitigate the effects of core melt accidents are already implemented in the design; these cannot be fully transferred to second-generation reactors such as Bugey 3. The EIA documents do not contain a systematic comparison be-tween the safety level of the 900 MWe reactors and the safety level of the EPR in order to identify the remaining gaps.

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

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

The planned modifications for heat removal without using the filtered pressure relief system in the event of a core melt accident have not yet been fully implemented. In addition, the reinforcement of the filtered pressure relief system (U5 system) against severe earthquakes has not yet been carried out. This means that even after completion of all Phase A measures of the 4th PSR, a core melt accident with a major release of radioactive substances is still possible at Bugey 3. The EIA documents do not provide a complete over-view 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 (2030). The EIA documents do not indicate whether this schedule will be adhered to.

- The EIA documents should include an overview of which of the planned measures are to be used to meet the ASNR requirements published at the end of the generic phase of the 4th PSR and when they are to be implemented.
- Studies that prove the sufficient thickness of the containment basements and the dimension of the spreading areas for Bugey 3 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 Bugey 3 and modern safety standards of the EPR Flamanville 3 should be included in order to identify the gaps.
- Information about the core damage frequency (CDF) and the large (early) release frequency (L(E)RF) before the 4th PSR, after implementation of all modification of 4th PSR and after the end of Phase A of the 4th PSR should be provided.
- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Bugey 3. 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 Bugey 3 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 Unit 3 of Bugey NPP is provided in the Chapter 6 of the *Document 3bis – Document regarding the environmental impact associated with the operation of the reactor during the next ten years*. (EIA-REPORT 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 Bugey 3. 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 102 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 102 – 104 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 104 to 106 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 fourth periodic safety review 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. 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 Bugey 3, 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. (EIA-REPORT 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 days) 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 D.3b (2026) claims 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 in the nearest settlements (450 m) and at distances of 2 km, 5 km, and 10 km. However, from the results for the early phase of an emergency it cannot be concluded whether they correspond to a 24-hour or a 7-day integration period. 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 greater than 5 km does not exceed limits for placing the food on the market already after 7 days; after one year, this distance is reported to be less than 1 km. However, the EIA documents do not present any additional results of the food contamination assessment, nor do they provide calculated activity concentrations in specific food items to substantiate these statements.

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

7.2 Discussion

Generic aspects

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. Similarly, the EIA-REPORT D.3b (2026) provides no details on the atmospheric dispersion model used to estimate off-site consequences. The EIA-REPORT D.3b (2026) indicates that mitigation measures intended to reduce the consequences of design-basis accidents were taken into account; however, it does not describe the assessment methodology needed to substantiate this claim or allow replication of the results.

Results for design-basis accidents indicate that projected population exposures at the nearest inhabited areas remain below French regulatory reference levels. The assessment recognizes that only core melt accidents have the potential to cause cross-border radiological impacts. The EIA-REPORT D.3b (2026) evaluates 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 (0.07–0.08 mSv).

The EIA-REPORT D.3b (2026) states 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 D.3b (2026) also claims 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 D.3b (2026) does 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 calculations of which results are presented and in order to assess whether, under specific circumstances, the limit value for the protective measures in Austria could be exceeded, the expert team conducted related dispersion modelling for large-scale release following two hypothetical accidents scenarios for Bugey 3. The aim of the assessment was to assess whether a severe accident at Bugey 3 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 Bugey 3, 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 PWR resulting in core melt with large containment release happening one hour after the reactor shutdown. The release category FK3 considers an accident at PWR resulting in core melt with medium containment release happening two hours after the reactor shutdown. In both scenarios, release lasts for 3 hours. Activities expressed as fractions of the core inventory for both release categories are shown in Table 3.

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

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

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

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

Ideally, atmospheric dispersion modelling for a specific type of accident with a release would be done with daily meteorological data for at least one year to understand transport and deposition of a radioactive plume in all meteorological conditions. As the goal of modelling in this study was only to confirm whether a deposition of Cs-137 above 650 Bq/m^2 from an accident in Bugey 3 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 Bugey 3. For this task, meteorological conditions for the period 21 – 24 July 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 Bugey 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:....................Bugey 3, France

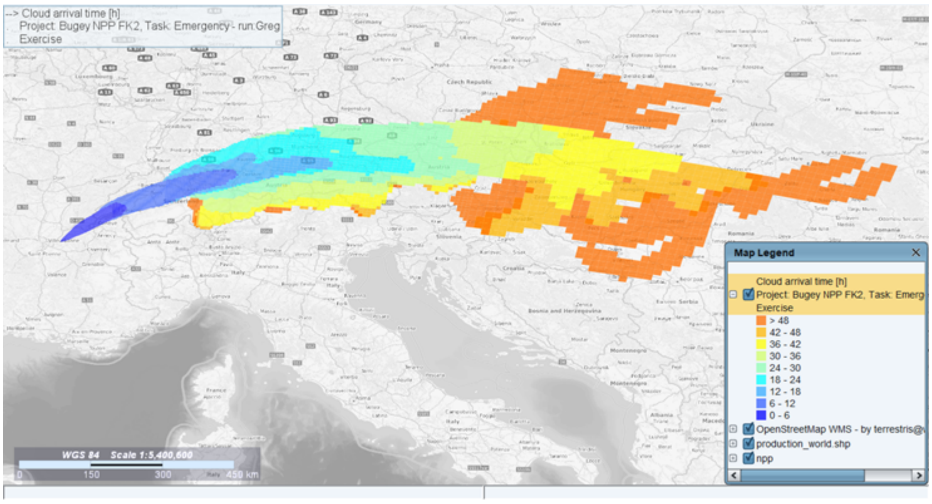
Release start: ....21 July 2025, 06:00 UTC

Prognosis duration: ..72 hours

Figure 5 presents information on cloud arrival time, indicating when the radioactive cloud is expected to reach the affected country. In both scenarios, with the release assumed to start on 21 July 2025 at 06:00 UTC, the cloud is projected to reach Austrian territory in approximately 12 hours. Meteorological conditions are the dominant factor influencing cloud arrival time, and this result may vary significantly under different weather conditions.

Figure 5: Cloud arrival time for the release category FK2

Cloud arrival time for FK2

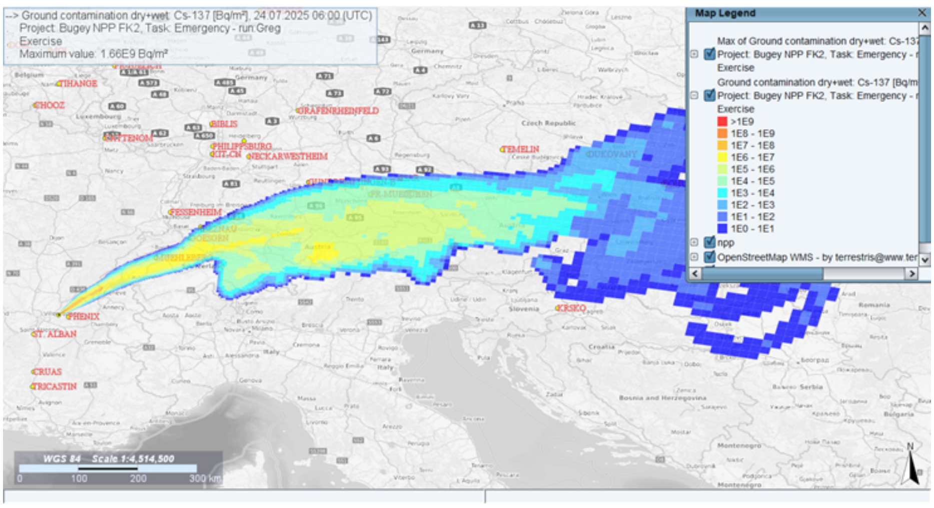


Source: Environment Agency Austria

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 reviewer for release categories FK2 and FK3 are presented in Figure 6 and Figure 7.

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

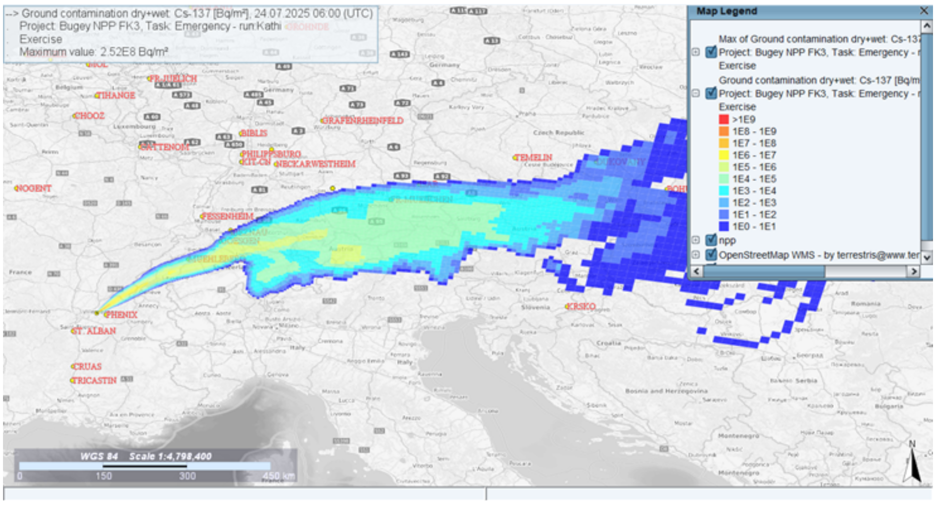
Ground contamination with CS-137 for FK2



Source: Environment Agency Austria

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

Ground contamination with Cs-137 for FK3



Source: Environment Agency Austria

The results of the JRODOS calculations demonstrate that contamination levels in Austria exceeding 650 Bq/m² are possible. The maximum calculated ground deposition exceeds 1×10^6 Bq/m² for release category FK2 and 1×10^5 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 D.3b (2026) does not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances greater than 5 km after 7 days and within 1 km after one year. Additionally, the assessment omits information on ground deposition, despite its significance for evaluating long-term radiological impacts and potential contamination of the food chain.

Modelling of atmospheric dispersion and deposition conducted by the expert team demonstrate that, under certain meteorological conditions, a severe accident at Bugey 3 could lead to ground deposition of Cs-137 in Austria above the national screening threshold of 650 Bq/m². Although the study does not assess the probability of such conditions, the results indicate that transboundary impacts greater than those implied in the EIA documents cannot be excluded.

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

- 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 Bugey 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 Bugey NPP Unit 3. (EIA-REPORT D.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, each plant has a period of five years following the release of the PSR report, to implement all safety measures identified.

For Bugey 3, 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.

Certain set of measures are scheduled for implementation within Phase B, which is planned to be completed by April 2029. Measures that are not completed within Phase B (or its extension,) are then to be completed within further phase, to be finalised by April 2030. This coincides with the "5 years after the release of the PSR report", as required by the regulator ASNR.

8.2 Discussion

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

8.3 Conclusions

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

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

9 LIST OF PRELIMINARY RECOMMENDATIONS

9.1 Long-term operation and operational experience

- The justification that no checks are to be carried out for Bugey 3 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 resistance with regard to earthquakes should be considered.

9.2 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 Bugey 3, 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 ground motion values for the SMS and the SND.

- The methodology and database used for the deterministic definition of the SMS should be reviewed, in particular, if it is justified to disregard site effects. Background: Bugey 3 is located on “soft” fluvial and Molasse sediments which may be characterized by low Vs30 values.
- 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). Background: published PSHA results show PGA values of at least 0.25 g for a 1975 years recurrence period for the Bugey area indicating that the recurrence period of the SMS ground motion is much shorter than 10,000 years.
- Additional safety demonstrations should be required to ensure that all SSCs relevant to safety can cope with a probabilistically derived new Design Basis Earthquake (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 Bugey 3.
- The methods, data and assumptions used to determine the SND for Bugey 3, 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 2020 a,b).
- Dedicated assessments of near-regional faults for which it cannot be excluded that they are active should be performed in line with WENRA (2020b). The approach should include field geology, geophysical mapping, morphostructural and dating studies, and paleoseismology. Background: existing literature highlights numerous Quaternary faults in the 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 Bugey 3 against design events, i.e., events with recurrence intervals of 10,000 years or less (e.g., earthquakes). This is to ensure the independence of Defence-in-Depth (DiD) levels 3 and 4.

- It should be evaluated if the long timeframe for implementing the Noyau Dur at the Bugey 3 reactor is in line with the requirement of the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). Background: the timeframe for implementing the Noyau Dur extends up to 2029, i.e., 17 years after ASNR's initial decision to implement Hardened Safety Cores at the French NPP fleet.
- 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 for France assessed? Are improvements planned with regard to "security culture", "cybersecurity" and "protection against insider threats"?

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

1. Enhance Transparency and Provide Clarity on Key Quantitative Data
   - Quantitative Data: The reports should provide the initial and final mass flow rates for the VCD-a valve flow capacity upgrade (PNPE0141), along with a comparison to the nominal operational flow. This is necessary to quantify the safety benefit.
   - Adverse Effects Analysis: The analysis of the uprated VCD-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.

2. Establish Firm and Accountable Timelines
   - Phase windows and tranche-specific dates: The program commitments give windows for deferred items (Phase B by 30 April 2029;

Supplementary Phase by April 2030). Requesting tranche-specific calendar dates within these windows for Bugey 3 will improve follow-up and transparency.
- Study Status and Next Steps: For the Critical Heat Flux (CHF) experimental program (Requirement [Study-B]), EDF should provide an updated status on its completion and publicly commit to the defined work program and schedule for incorporating the findings. This is overdue, as the reporting deadline was December 31, 2024. The provided excerpts do not report the outcome/status; a status update request (completion, results, and planned integration) is therefore appropriate.

3. Clarify Status Reporting and Implementation Rationale

- Justify Deferral: For PNPP0949 line A (fire screen between PTR pumps), scheduled in Phase B, the safety justification for deferral and detail the interim protections in place until installation should be provided. Note that line C (fire protection of PTR cables in shared fire volumes) is in deployment with completion in Phase A.
- For the Noyau Dur makeup dispositions to the spent fuel pool, the interim arrangements that ensure drain-down prevention and makeup capability prior to installation of the fixed routes should be clarified: PNPE0258 (ASG-ND and fixed line to re-supply reactor building via SEG) is planned for Phase B, while PNPP0714 is already implemented and PNRL0803 (reactor building makeup/exhaust arrangement) is planned for Phase B.
- Resolve Discrepancies and improve transparency: The Bugey 3 report provides tranche-specific status tables (fully implemented, Phase A, Phase B, “Complementation” and “Specific program”), and some items carry explicit deadlines (e.g., PNPP0877: no later than end-2025). Tranche-specific calendar dates for each item within the committed windows (Phase B by 30 April 2029; Supplementary Phase by April 2030, etc.) should be confirmed, and for each item reported as “implemented”, it should be specified, whether this denotes installation completion and commissioning (in line with requalification). In general, a simpler and reduced deadline system would increase transparency.

9.4 Safety aspects of core melt accidents

- The EIA documents should include an overview of which of the planned measures are to be used to meet the ASNR requirements published at the end of the generic phase of the 4th PSR and when they are to be implemented.
- Studies that prove the sufficient thickness of the containment basements and the dimension of the spreading areas for Bugey 3 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 Bugey 3 and modern safety standards of the EPR Flamanville 3 should be included in order to identify the gaps.
- Information about the core damage frequency (CDF) and the large (early) release frequency (L(E)RF) before the 4th PSR, after implementation of all modification of 4th PSR and after the end of Phase A of the 4th PSR should be provided.
- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Bugey 3. 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 Bugey 3 should be issued only after the planned measures to mitigate the release in the event of a core melt accident have been fully implemented.

9.5 Radiological impact of accidents / Transboundary Effects

- Information on the release parameters is needed for the reconstruction of the results of the assessment provided in the EIA 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.

9.6 Assessment of the time frame

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

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12 GLOSSARY

ASG .... Steam Generator Auxiliary Feedwater System

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

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

BAN .... Buildings for Nuclear Auxiliary Facilities

BK.................................. Spent Fuel Building

Bq .... Becquerel

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

CHF .... Critical Heat Flux

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

DAPE .... Dossier of Suitability for Continued Operation

DBA .... Design Basis Accidents

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

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

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

DID .... Defence-in-Depth

DVN................................ Ventilation and air conditioning system

EAS .... Sprinkler System

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

EDF .... Électricité de France

EDG .... Emergency Diesel Generator

EIA .... Environmental Impact Assessment

EIPS .... Emergency Intervention Systems

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

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

GPR .... Permanent Group of Experts on Reactors

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

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

I-131 .... Iodine-131

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

IBLOCA.......................... Intermediate Break Loss-of-Coolant Accident

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

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

LOCA .... Loss of Coolant Accident

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

mSv .... Millie-Sievert

MW................................ Mega Watt

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

PGA .... Peak Ground Acceleration

PIC .... Program for Complementary Investigations

PSR .... Periodic Safety Review

PSHA .... Probabilistic Safety Hazard Assessment

PSA .... Probabilistic Safety Assessment

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

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

PWR .... Pressurized Water Reactor

REA .... Boron and Water Storage Tank

RIA .... Reactivity Initiating Accidents

RIC .... Reactor Core Instrumentation

RCP .... Reactor Coolant Pump

RCS .... Reactor Cooling System

RFS .... Règle Fondamentale de Sûreté

RGE .... General Operating Rules

RPN .... Reactor Protection System

RPV .... Reactor Pressure Vessel

SA .... Severe Accidents

SBO .... Station Black Out

SEG .... Diversified water source

SFP .... Spent Fuel Pool

SF-ND .... Hardened Core Cooling Source

SG .... Steam Generator

SGTR .... Steam generator tube ruptures

SIS .... Safety Injection Systems

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

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

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

SOTA .... State of the Art

SSCs .... Structures, Systems and Components

TBq .... Tera-Becquerel, E12 Bq

TLD .... Température Longue Durée

TE.................................. Température Exceptionnelle

TPR .... Topical Peer Review

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

UHS .... Ultimate Heat Sink

VCD-a .... Main turbine bypass system with discharge into the atmosphere

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

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

Vs30 .... s-wave velocity of the top soil

WENRA.......................... Western European Nuclear Regulators' Association

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