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NPP TRICASTIN 3&4 LTO ENVIRONMENTAL IMPACT ASSESSMENT

Expert Statement

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CONTENTS

SUMMARY

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

France notified the 4th Periodic Safety Review ("Public consultation procedure on the 4th safety review report") of the Tricastin NPP (reactor 3 and 4), which is to be considered as a lifetime extension in accordance with the UNECE Espoo Convention on Environmental Impact Assessment (EIA) in a Transboundary Context. The competent authority is the French department of Drôme. The project applicant is Électricité de France (EDF).

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

Procedure

The operating authorization of French NPPs is not limited in time. However, every ten years, French NPPs are subject to a Periodic Safety Review (PSR). The fourth PSR plays a special role, as it marks the regulatory process for the Long-Term Operation (LTO) of 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 has been completed, inspections of all 32 reactors at the 900 MWe nuclear power plants should follow over a period of approximately ten years (from 2019 to 2031).

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

Long-Term operation and operational experience

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

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.

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

The ASNR's proposal during the generic phase of the 5th PSR to extend aging management beyond 4th PSR is supported. As proposed by the ASNR, the focus must be on components that are necessary for controlling accident situations. However, the scope of the program “qualification of materials under accident conditions” in the 4th PSR is very limited for Tricastin 3&4.

In contrast to the description in the EIA documents, the IAEA’s OSART (Operational Safety Review Team) mission, which took place at the end of 2022, identified several safety deficiencies in the area of aging management at the Tricastin NPP. Among other things, the team criticized the fact that the scope of SSCs to be included in the aging management

(AM) program is not comprehensive enough. In addition, they criticized issues that are also relevant to LTO: the backlog in maintenance and the inadequate analysis of incidents.

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

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

External hazards

The EIA documents provide information on hazard types considered in the safety demonstration for the Tricastin 3 and 4 and measures already implemented or decided to be implemented in order to strengthen the robustness of the reactor with respect to external hazards. For most external hazards, the methods, data and assumptions used in the hazard assessment are not specified in detail. Conformity with 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 Tricastin, 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 Tricastin of 0.3 g. It remains to be demonstrated that the current SMS fulfills the WENRA requirements of a DBE with an average recurrence interval of 10,000 years (WENRA 2021).

The Tricastin site is located in the Bresse Graben, a tectonic structure containing numerous active faults which have not been taken into account in seismic hazard assessments. The EIA documents name the active Mimes-, Clanseyes- and Cavenen- and Rouvière fault. The importance of active faults for the safety of Tricastin is underlined by the 2019 M=4.9 Le Teil earthquake which occurred on the Cavenen fault system only about 20 km north of Tricastin. The earthquake led to surface displacement and an epicentral ground accelerations of likely >1 g. For active faults in the near-region of an NPP site WENRA suggests systematic fault mapping and collecting paleoseismological information. It is suggested to ASNR to require (i) dedicated paleoseismological assessments of these faults and (ii) an up-to-date Probabilistic Safety Hazard Assessment (PSHA) in line with WENRA requirements considering paleoseismological data.

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

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

Information regarding the issue of terror attacks would be of great interest, considering the far-reaching consequences of potential attacks. In particular, the EIA documents should include information on the requirements for the design against the targeted crash of a commercial aircraft. This topic is particularly important, because the reactor building as well as the spent fuel building of the Tricastin 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 Tricastin 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 Tricastin 3 and 4. For long-term heat removal, the Auxiliary Feedwater System (ASG) water inventory has been secured by diversifying the connection of the ASG tank from the fire-protection network (PNPP1864), supporting accident sequences where additional feedwater is required. For thermal-hydraulic control, the increase of the GCT-a regulating-valve capacity (PNPE1141) addresses the ASG consumption calculation anomaly and supports updated dimensioning accident studies. In parallel, a lower permissible primary-circuit I-131 activity has been adopted to limit radiological consequences in accidents without fuel-rod failure.

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

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

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

Safety aspects of core melt accidents

Severe accidents (SA) involving core meltdown were not taken into account in the design of the French 900 MWe reactors. However, as a result of previous PSRs, facilities and measures for SA management have been implemented. Nevertheless, a number of shortcomings were identified in the EU stress tests as early as 15 years ago, and not all of them have been remedied to date. According to the ASNR, the objective of the fourth PSR for the 900 MWe reactors is to bring the safety level of the reactor closer to that of the EPR in Flamanville, a third-generation reactor. In third-generation reactors, features to mitigate the effects of core melt accidents are already implemented in the design; these cannot be fully transferred to second-generation reactors such as Tricastin 3 and 4. The EIA documents do not contain a systematic comparison between the safety level of the 900 MWe reactors and the safety level of the EPR in order to identify the remaining gaps.

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

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

Not all necessary and planned modifications for heat removal without using the filtered pressure relief system in the event of a core melt accident have been fully implemented yet. Important components such as valves, seals and electric equipment that are required during a core melt accident, but whose resistance cannot be guaranteed during such an accident, will also only be replaced during Phase B or the Supplementary Phase. In addition, the reinforcement of the filtered pressure relief system (U5 system) against severe earthquakes has not yet been carried out. This means that even after completion of all Phase A measures of the 4th PSR, a core melt accident with a major release of radioactive substances is still possible at Tricastin 3 or 4.

The EIA documents do not provide a complete overview of which of the planned modifications meet the ASNR requirements published at the end of the generic phase of the 4th PSR. Most of the measures are not scheduled to be implemented until the end of Phase B and the Supplementary Phase (2029 for unit 3 and 2031for unit 4 respectively). The EIA documents do not indicate whether this schedule will be adhered to.

Radiological impact of accidents / Transboundary effects

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

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

The EIA documents indicate that, for design-basis accidents, the radiological consequences are expected to remain below national reference levels and do not give rise to transboundary risks. For beyond design-basis accidents, specifically for scenarios involving core melt, the report acknowledges the potential for long-range impacts but lacks sufficient technical detail to allow independent verification of these findings. The EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) do not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances of more than 5 km after 7 days and within 1 km after one year. Additionally, the assessment omits information on ground deposition, despite its significance for evaluating long-term radiological impacts and potential contamination of the food chain.

Modelling of atmospheric dispersion and deposition conducted by the expert team demonstrate that, under certain meteorological conditions, a severe accident at unit 3 or unit 4 of Tricastin NPP could lead to ground deposition of Cs-137 in Austria above the national screening threshold of 650 Bq/m². Although the study does not assess the probability of such conditions, the results indicate that transboundary impacts beyond those described in the EIA documents cannot be excluded.

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

Assessment of the time frame

The timeframe for completing all measures under the 4th PSR (6 years after the release of the PSR report = 2029 for Tricastin 3 and 2031 for Tricastin 4) is not uncommon. However, as the period following the 4th PSR corresponds with the start of long-term operation (LTO), some of the specific measures require special attention. It is important that the agreed implementation period is not extended. A lack of financial resources or the known problems with supply chain availability, including human resources, could affect the implementation period. It is particularly noteworthy that important safety modifications listed as part of the 4th PSR were already considered necessary as part of the EU stress test (2012), and their implementation had been agreed upon.

Management of spent fuel and radioactive waste

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

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

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

ZUSAMMENFASSUNG

Das Kernkraftwerk (KKW) Tricastin besteht aus vier in Betrieb befindlichen Druckwasserreaktoren mit einer Leistung von jeweils 900 MWe. Diese Reaktoren wurden 1980 bzw. 1981 in Betrieb genommen.

Frankreich hat die vierte Periodische Sicherheitsüberprüfung („Öffentliches Konsultationsverfahren zum vierten Bericht der Sicherheitsüberprüfung“) des KKW Tricastin (Reaktor 3 und 4) notifiziert, die als Laufzeitverlängerung gemäß der UNECE-Espoo-Konvention über die Umweltverträglichkeitsprüfung (UVP) im grenzüberschreitenden Rahmen zu betrachten ist. Zuständige Behörde ist das französische Departement Drôme. Antragsteller ist Électricité de France (EDF).

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

Verfahren

Die Betriebsgenehmigung für französische KKW ist zeitlich nicht begrenzt. Alle zehn Jahre werden französische KKW jedoch einer Periodischen Sicherheitsüberprüfung (PSÜ) unterzogen. Die 4. PSÜ spielt eine besondere Rolle, da sie den Genehmigungsprozess für den Langzeitbetrieb (Long-Term Operation, LTO) eines Kernkraftwerks über 40 Jahre hinaus markiert. Der französische PSÜ-Rahmen schreibt eine umfassende Sicherheitsbewertung in zwei Phasen vor: eine generische und eine anlagenspezifische Phase.

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

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

Langfristiger Betrieb und Betriebserfahrung

Auf der Grundlage der in den UVP-Unterlagen enthaltenen Informationen kann der Schluss gezogen werden, dass ein umfassendes Alterungsmanagementprogramm zur Gewährleistung des Betriebs umgesetzt wurde. Darauf deuten auch die Ergebnisse der ersten Topical Peer Review (TPR) gemäß Artikel 8e der Richtlinie 2014/87/EURATOM hin. Das Management der mit der Alterung von Strukturen, Systemen und Komponenten (SSCs) verbundenen Probleme stellt jedoch eine große Herausforderung für das Kernkraftwerk dar, das seit mehr als 40 Jahren in Betrieb ist.

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.

Im Rahmen der generischen Phase der 5. PSÜ der 900-MWe-Reaktoren forderte die ASNR EDF auf, bis zum 31. Dezember 2025 die Strategie festzulegen, um die Erkenntnisse aus der Entdeckung von Spannungsrisskorrosion und, allgemeiner, das Risiko einer unerwarteten Degradation von Komponenten im Primär- und Hauptsekundärkreislauf durch Kontrollen im Rahmen der zusätzlichen Inspektions- und Wartungsprogramme zu berücksichtigen. Die Ursache der Risse, die interkristalline Spannungskorrosion, ist ein bekanntes Korrosionsphänomen, das jedoch in den betreffenden Bereichen nicht erwartet wurde und daher die Rohre auch nicht darauf geprüft wurden. Dies stellt das Alterungsmanagementkonzept für Komponenten im Primär- und Hauptsekundärkreislauf in Frage.

Der Vorschlag der ASNR während der generischen Phase der 5. PSÜ, das Alterungsmanagement über die 4. PSÜ hinaus auszuweiten, wird unterstützt. Wie von der ASNR vorgeschlagen, muss der Schwerpunkt auf

Komponenten liegen, die für die Beherrschung von Unfallsituationen notwendig sind. Der Umfang des Programms „Qualifizierung von Werkstoffen unter Unfallbedingungen“ im Rahmen der 4. PSÜ ist für Tricastin 3 und 4 jedoch sehr begrenzt.

Im Gegensatz zu den Angaben in den UVP-Dokumenten stellte die OSART-Mission (Operational Safety Review Team) der IAEO, die Ende 2022 stattfand, mehrere Sicherheitsmängel im Bereich des Alterungsmanagements im Kernkraftwerk Tricastin fest. Das Team kritisierte unter anderem, dass der Umfang der SSCs, die in das Alterungsmanagementprogramm (AM) einbezogen werden sollen, nicht umfassend genug sei. Darüber hinaus kritisierte es Themen, die auch für LTO relevant sind: den Wartungsrückstand und die unzureichende Analyse von Störfällen.

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

Insgesamt kam es in den letzten fünf Jahren zu einer Reihe von sicherheitsrelevanten Vorfällen, die auf der INES-Skala als Level 1 eingestuft wurden; viele davon standen im Zusammenhang mit Verstößen gegen die Allgemeinen Betriebsvorschriften (RGE). Diesen Vorfällen gingen Komponentenausfälle, Wartungs- oder Bedienungsfehler (in einigen Fällen mehrere Fehler) voraus. Die Ursache könnte in einer mangelnden Sicherheitskultur in Verbindung mit einer hohen Zahl altersbedingter Vorfälle liegen.

Externe Gefahren

Die UVP-Unterlagen enthalten Informationen zu den in den Sicherheitsnachweisen für Tricastin 3 und 4 berücksichtigten Gefahrenarten sowie zu den bereits umgesetzten oder beschlossenen Maßnahmen zur Erhöhung der Robustheit des Reaktors gegenüber externen Gefahren. Für die meisten externen Gefahren werden die bei der Gefahrenbewertung verwendeten Methoden, Daten und Annahmen nicht im Detail angegeben. Die Übereinstimmung mit den Anforderungen und Leitlinien der WENRA kann daher nicht beurteilt werden.

Bei Erdbeben und seismischen Bodenbewegungen wird eine Abweichung von den WENRA-Referenzleveln festgestellt. Das Auslegungserdbeben (DBE) für Tricastin, in den französischen Vorschriften als SMS bezeichnet, basiert auf einer deterministischen Analyse, was nicht mehr dem aktuellen Stand der Technik entspricht. Verfügbare Dokumente weisen einen SMS-Bodenbeschleunigungswert (PGA) für Tricastin von 0,3 g aus. Es bleibt noch zu zeigen, dass das derzeitige SMS die WENRA-Anforderungen an ein DBE mit einem durchschnittlichen Wiederholungsintervall von 10.000 Jahren erfüllt (WENRA 2021).

Der Standort Tricastin liegt im Bresse-Graben, einer tektonischen Struktur, die zahlreiche aktive Verwerfungen enthält, die bei der Bewertung der Erdbebengefährdung nicht berücksichtigt wurden. In den UVP-Unterlagen werden die aktiven Verwerfungen Mimes, Clanseyes, Cavenen und Rouvière genannt. Die Bedeutung aktiver Verwerfungen für die Sicherheit von Tricastin wird durch das Erdbeben von Le Teil im Jahr 2019 mit einer Stärke von M=4,9 unterstrichen, das sich am Cavenen-Verwerfungssystem nur etwa 20 km nördlich von Tricastin ereignete. Das Erdbeben führte zu Bodenverschiebungen und Bodenbeschleunigungen am Epizentrum von vermutlich über 1 g. Für aktive Verwerfungen in der unmittelbaren Umgebung eines Kernkraftwerksstandorts empfiehlt die WENRA eine systematische Kartierung der Verwerfungen und die Erfassung paläoseismologischer Daten. Der ASNR wird empfohlen, (i) spezielle paläoseismologische Bewertungen dieser Verwerfungen sowie (ii) eine aktuelle probabilistische Sicherheitsrisikobewertung (PSHA) gemäß den WENRA-Anforderungen unter Berücksichtigung paläoseismologischer Daten zu verlangen.

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

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

Angesichts der weitreichenden Folgen potenzieller Anschläge wären Informationen zum Thema Terroranschläge von großem Interesse. Insbesondere sollten die UVP-Unterlagen Angaben zu den Anforderungen an die Auslegung gegen den gezielten Absturz eines Verkehrsflugzeugs enthalten. Dieses Thema ist besonders wichtig, da sowohl das Reaktorgebäude als auch das Gebäude für abgebrannte Brennelemente des Kernkraftwerks Tricastin 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 Tricastin betragen jedoch weniger als 1,0 m. Darüber hinaus erhöhen die zunehmende Verfügbarkeit und Leistungsfähigkeit von Drohnen die potenzielle Bedrohung für kerntechnische Anlagen. Eine kürzlich durchgeführte Bewertung der nuklearen Sicherung in Frankreich weist auf Mängel im Vergleich zu den notwendigen Anforderungen an die nukleare Sicherung in Bezug auf die „Sicherungskultur“, die „Cybersicherheit“ und den „Schutz vor Insider-Bedrohungen“ hin.

Sicherheitsaspekte von Unfällen ohne Kernschmelze und im Brennelementlagerbecken

Die Analyse stützt sich sowohl auf deterministische Sicherheitsanalysen (DSA) als auch auf probabilistische Sicherheitsanalyse (PSA), um Betriebstransienten, Auslegungsstörfälle (DBA) und erweiterte Auslegungsbedingungen (DEC) neu zu bewerten.

Erhebliche Sicherheitsverbesserungen wurden bereits umgesetzt oder sind geplant, um die radiologischen Folgen zu verringern und das gestaffelte Sicherheitskonzept in Tricastin 3 und 4 zu verbessern. Für die langfristige Wärmeabfuhr wurde der Wasservorrat des Hilfsspeisewassersystems (ASG) gesichert, indem die Anbindung des ASG-Tanks an das Brandschutznetz diversifiziert wurde (PNPP1864), was Unfallsequenzen unterstützt, bei denen zusätzliches Speisewasser benötigt wird. Im Hinblick auf die thermohydraulische Regelung behebt die Erhöhung der Kapazität des GCT-a-Regelventils (PNPE1141) die Diskrepanz bei der Berechnung des ASG-Verbrauchs und unterstützt aktualisierte Unfallstudien zur Dimensionierung. Parallel dazu wurde eine niedrigere zulässige I-131-Aktivität im Primärkreis festgelegt, um die radiologischen Folgen bei Unfällen ohne Brennstabschäden zu begrenzen.

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

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

Das Risiko einer Brandausbreitung zwischen den PTR-Strängen wird durch eine physische Trennung und Schutzmaßnahmen mittels einer Brandschutzwand zwischen den beiden PTR-Pumpen gemindert. (PNPP1949)

Sicherheitsaspekte von Unfällen mit Kernschmelze

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

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

Nach dem aktuellen Kenntnisstand kann ein Versagen des Sicherheitsbehälters nach Umsetzung der Modifikation zur Stabilisierung und

Kühlung des geschmolzenen Kerns nicht ausgeschlossen werden. Zum einen sind noch nicht alle wichtigen Modifikationen umgesetzt worden, und zum anderen ist es anhand der verfügbaren Informationen nicht möglich zu beurteilen, ob die Modifikationen (insbesondere die Verstärkung des Fundaments) ausreichend sind.

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

Die UVP-Unterlagen geben keinen vollständigen Überblick darüber, welche der geplanten Modifikationen den am Ende der generischen Phase des 4. PSÜ veröffentlichten ASNR-Anforderungen entsprechen. Die meisten Maßnahmen sollen erst bis zum Ende der Phase B und der Ergänzungsphase (2029 für Block 3 bzw. 2031 für Block 4) umgesetzt werden. Aus den UVP-Unterlagen geht nicht hervor, ob dieser Zeitplan eingehalten wird.

Strahlungsauswirkungen von Unfällen / Grenzüberschreitende Auswirkungen

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

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

Aus den UVP-Unterlagen geht hervor, dass bei Auslegungsstörfällen die radiologischen Auswirkungen voraussichtlich unter den nationalen Referenzwerten bleiben und keine grenzüberschreitenden Risiken verursachen. Bei auslegungsüberschreitenden Unfällen, insbesondere bei Szenarien mit Kernschmelze, räumt der Bericht zwar die Möglichkeit für weitreichende Auswirkungen ein, es fehlen jedoch ausreichende technische Details, um eine unabhängige Überprüfung dieser Ergebnisse zu ermöglichen. Die UVP-Berichte (EIA-REPORT T3/4 D.3b 2026) enthalten keine quantitativen Analysen, die die Aussagen untermauern, dass die Lebensmittelkontamination in Entfernungen von mehr als 5 km nach 7 Tagen und in einem Umkreis von 1 km nach einem Jahr unter den EU-Grenzwerten bleiben würde. Darüber hinaus fehlen in der Bewertung Informationen zur Bodenkontamination trotz ihrer Bedeutung für die Bewertung langfristiger radiologischer Auswirkungen und der potenziellen Kontamination der Nahrungskette.

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

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

Bewertung des Zeitrahmens

Der Zeitrahmen für die Umsetzung aller Maßnahmen im Rahmen der 4. PSÜ (6 Jahre nach Veröffentlichung des PSÜ-Berichts = 2029 für Tricastin 3 und 2031 für Tricastin 4) ist nicht ungewöhnlich. Da der Zeitraum nach der 4. PSÜ jedoch mit dem Beginn des Langzeitbetriebs (LTO) zusammenfällt, erfordern einige der spezifischen Maßnahmen besondere Aufmerksamkeit. Es ist wichtig, dass der vereinbarte Umsetzungszeitraum nicht verlängert wird. Ein Mangel an finanziellen Ressourcen oder die bekannten Probleme mit der Verfügbarkeit in der Lieferkette, einschließlich der personellen Ressourcen, könnten sich auf den Umsetzungszeitraum auswirken. Besonders hervorzuheben ist, dass wichtige Sicherheitsmaßnahmen, die im Rahmen der 4. PSÜ aufgeführt sind, bereits im Rahmen des EU-Stresstests (2012) als notwendig erachtet und deren Umsetzung vereinbart worden waren.

Entsorgung von abgebrannten Brennelementen und radioaktiven Abfällen

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

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

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

RESUME

La centrale nucléaire du Tricastin comprend quatre réacteurs à eau pressurisée en service d'une capacité de 900 MWe chacun. Ces réacteurs ont été mis en service respectivement en 1980 et 1981.

La France a notifié le quatrième réexamen périodique (« Procédure de consultation publique sur le quatrième rapport du réexamen ») de la centrale nucléaire du Tricastin (réacteur 3 et 4), qui doit être considéré comme une prolongation de durée de vie conformément à la Convention d'Espoo de la CEE-ONU sur l'évaluation de l'impact sur l'environnement (EIE) dans un contexte transfrontalier. L'autorité compétente est le département français de la Drôme. Le demandeur du projet est Électricité de France (EDF).

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

Procédure

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

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

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

Exploitation à long terme et expérience opérationnelle

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

Dans le cadre de la phase générique du 5e RP des réacteurs de 900 MWe, l'ASNR 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 Tricastin 3 et 4.

Contrairement à ce qui est indiqué dans les documents de l'EIE, la mission OSART (Operational Safety Review Team) de l'AIEA, qui s'est déroulée fin 2022, a mis en évidence plusieurs lacunes en matière de sûreté dans le domaine de la gestion du vieillissement à la centrale nucléaire du Tricastin. L'équipe a notamment critiqué le fait que le champ d'application des SSCs devant être inclus dans le programme de gestion du vieillissement (AM) n'était pas suffisamment exhaustif. En outre, elle a relevé des problèmes qui concernent également LTO : le retard accumulé en matière de maintenance et l'analyse insuffisante des incidents.

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

Dans l'ensemble, plusieurs incidents liés à la sûreté, classés au niveau 1 de l'échelle INES, se sont produits au cours des cinq dernières années, dont beaucoup étaient liés au non-respect du règlement général d'exploitation (RGE). Ces incidents ont été précédés par des défaillances de composants, des erreurs de maintenance ou des erreurs d'exploitation (dans certains cas, plusieurs erreurs). La cause pourrait être un manque de culture de sûreté, associé à un nombre important d'incidents liés au vieillissement.

Risques externes

Les documents de l'EIE fournissent des informations sur les types de risques pris en compte dans la démonstration de sûreté de Tricastin 3 et 4, ainsi que sur les mesures déjà mises en œuvre ou dont la mise en œuvre a été décidée afin de renforcer la résistance du réacteur face aux risques externes. Pour la plupart des risques externes, les méthodes, les données et les hypothèses utilisées dans l'évaluation des risques ne sont pas précisées en détail. Il n'est donc pas possible d'évaluer la conformité aux exigences et aux recommandations de 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 Tricastin, 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 0,3 g pour le Tricastin dans le cadre du SMS.

Il reste à démontrer que le système SMS actuel satisfait aux exigences de la WENRA concernant un événement de conception de base (DBE) dont l'intervalle de récurrence moyen est de 10 000 ans (WENRA 2021)

Le site de Tricastin est situé dans le fossé de la Bresse, une structure tectonique comportant de nombreuses failles actives qui n’ont pas été prises en compte dans les évaluations des risques sismiques. Les documents d'EIE mentionnent les failles actives de Mimes, Clanseyes, Cavenen et Rouvière. L'importance des failles actives pour la sûreté de Tricastin est soulignée par le séisme de Le Teil (M = 4,9) survenu en 2019 sur le système de failles de Cavenen, à seulement une vingtaine de kilomètres au nord de Tricastin. Il est suggéré à l'ASNR d'exiger (i) des évaluations paléosismologiques spécifiques de ces failles et (ii) une évaluation probabiliste des risques de sûreté (PSHA) actualisée, conforme aux exigences de la WENRA et tenant compte des données paléosismologiques.

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

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

Les informations relatives aux attentats terroristes seraient d'un grand intérêt, compte tenu des conséquences considérables que pourraient avoir de telles attaques. Les documents d'EIE devraient notamment inclure des informations sur les exigences en matière de conception visant à prévenir le crash ciblé d'un avion commercial. Ce sujet est particulièrement important, car le bâtiment du réacteur ainsi que le bâtiment de stockage du combustible usé de la centrale nucléaire de Tricastin 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 Tricastin 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 Tricastin 3 et 4. Pour l'évacuation de la chaleur à long terme, la disponibilité en eau du système auxiliaire d'alimentation en eau (ASG) a été garantie en diversifiant le raccordement du réservoir ASG au réseau de protection incendie (PNPP1864), ce qui permet de faire face aux séquences d'accident nécessitant un apport supplémentaire d'eau d'alimentation. En matière de contrôle thermohydraulique, l'augmentation de la capacité de la vanne de régulation GCT-a (PNPE1141) corrige l'anomalie de calcul de la consommation de l'ASG et prend en compte les études d'accidents dimensionnées mises à jour. Parallèlement, une activité admissible plus faible de l'I-131 dans le circuit primaire a été adoptée afin de limiter les conséquences radiologiques en cas d'accidents sans défaillance des barres de combustible.

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

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

Le risque de propagation du feu entre les trains PTR est atténué par des mesures de séparation physique et de protection via l'écran de protection contre l'incendie situé entre les deux pompes PTR (PNPP1949).

Aspects de sûreté des accidents de fusion du cœur

Les accidents graves (SA) impliquant une fusion du cœur n'ont pas été pris en compte dans la conception des réacteurs français de 900 MWe. Cependant, à la suite des examens périodiques de sûreté (RP) précédents, des installations et des mesures de gestion des SA ont été mises en place. Pourtant, plusieurs lacunes avaient déjà été identifiées dans les tests de résistance de l'UE il y a quinze ans, et toutes n'ont pas encore été comblées à ce jour.

Selon l'ASNR, l'objectif de la quatrième RP pour les réacteurs de 900 MWe est de rapprocher le niveau de sûreté du réacteur de celui de l'EPR de Flamanville, un réacteur de troisième génération. Dans les réacteurs de troisième génération, des dispositifs visant à atténuer les effets des accidents de fusion du cœur sont déjà intégrés dans la conception ; ceux-ci ne peuvent pas être entièrement transposés aux réacteurs de deuxième génération tels que Tricastin 3 and 4. Les documents d'EIE ne contiennent pas de comparaison systématique entre le niveau de sûreté des réacteurs de 900 MWe et celui de l'EPR afin d'identifier les écarts restants.

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

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

Toutes les modifications nécessaires et prévues pour assurer l'évacuation de la chaleur sans recourir au système de décompression filtré en cas d'accident entraînant la fusion du cœur n'ont pas encore été entièrement mises en œuvre. Des composants essentiels, tels que les vannes, les joints et les équipements électriques, qui sont indispensables en cas d'accident de fusion du cœur, mais dont la résistance ne peut être garantie dans une telle situation, ne seront eux aussi remplacés qu'au cours de la phase B ou de la phase supplémentaire. En outre, le renforcement du système de décompression filtré (système U5) contre les séismes violents n'a pas encore été réalisé. Cela signifie que même après l'achèvement de toutes les mesures de la phase A du 4e RP, un accident de fusion du cœur avec un rejet important de substances radioactives est toujours possible à Tricastin 3 ou 4.

Les documents d'EIE ne fournissent pas un aperçu complet des modifications prévues desquelles / qui répondent aux exigences de l'ASNR publiées à la fin de la phase générique de la 4e RP. La plupart des mesures ne sont pas prévues avant la fin de la phase B et de la phase supplémentaire (2029 pour l'unité 3 et 2031 pour l'unité 4 respectivement). Les documents d'EIE n'indiquent pas si ce calendrier sera respecté.

Impact radiologique des accidents / Effets transfrontaliers

Les documents de l'EIE traitent des événements et des séquences d'accidents correspondant à trois catégories d'accidents de référence, ainsi que d'une catégorie supplémentaire représentant les événements dépassant le cadre des accidents de référence, notamment les scénarios de fusion du cœur et de la piscine de stockage du combustible usé.

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

Les documents EIE indiquent que, pour les accidents de référence, les conséquences radiologiques devraient rester inférieures aux niveaux de référence nationaux et ne pas entraîner de risques transfrontaliers.

Pour les accidents dépassant les limites de conception, et plus particulièrement les scénarios impliquant une fusion du cœur, le rapport reconnaît l'existence d'impacts potentiels à longue distance, mais ne fournit pas suffisamment de détails techniques pour permettre une vérification indépendante de ces conclusions. Les rapports (EIA-REPORT T3/4 D.3b 2026) ne présentent pas d'analyses quantitatives pour appuyer les affirmations selon lesquelles la contamination alimentaire resterait inférieure aux limites fixées par l'UE à des distances supérieures à 5 km après 7 jours et à moins de 1 km après un an. En outre, l'évaluation omet les informations sur les dépôts au sol, malgré leur importance pour l'évaluation des impacts radiologiques à long terme et de la contamination potentielle de la chaîne alimentaire.

La modélisation de la dispersion atmosphérique et déposition réalisée par l'équipe d'experts démontre que, dans certaines conditions météorologiques, un accident grave au réacteur n° 3 ou réacteur n° 4 de la centrale nucléaire du Tricastin 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 (6 ans après la publication du rapport RP =2029 pour Tricastin 3 et 2031 pour Tricastin 4) n'est pas inhabituel en principe. Cependant, comme la période suivant le 4e RP correspond au début de l'exploitation à long terme (LTO), certaines mesures spécifiques nécessitent une attention particulière. Il est important que la période de mise en œuvre convenue ne soit pas prolongée. Le manque de ressources financières ou les problèmes connus liés à la disponibilité de la chaîne d'approvisionnement, y compris les ressources humaines, pourraient avoir un impact sur la période de mise en œuvre. Il convient de noter en particulier que d'importantes modifications de sécurité figurant dans la liste des modifications du 4e RP avaient déjà été jugées nécessaires dans le cadre du test de résistance de l'UE (2012) et que leur mise en œuvre avait été convenue.

Gestion du combustible usé et des déchets radioactifs

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

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

1 INTRODUCTION

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

France notified the 4th Periodic Safety Review ("Public consultation procedure on the 4th safety review report") of the Tricastin NPP (reactor 3&4), which is to be considered as a lifetime extension in accordance with the UNECE Espoo Convention on Environmental Impact Assessment (EIA) in a Transboundary Context. The competent authority is the French department of Drôme. The project applicant is Électricité de France (EDF).

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

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

2 PROCEDURE

2.1 Treatment in the EIA documents

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

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

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

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

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

900 MWe reactors (32 units):

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

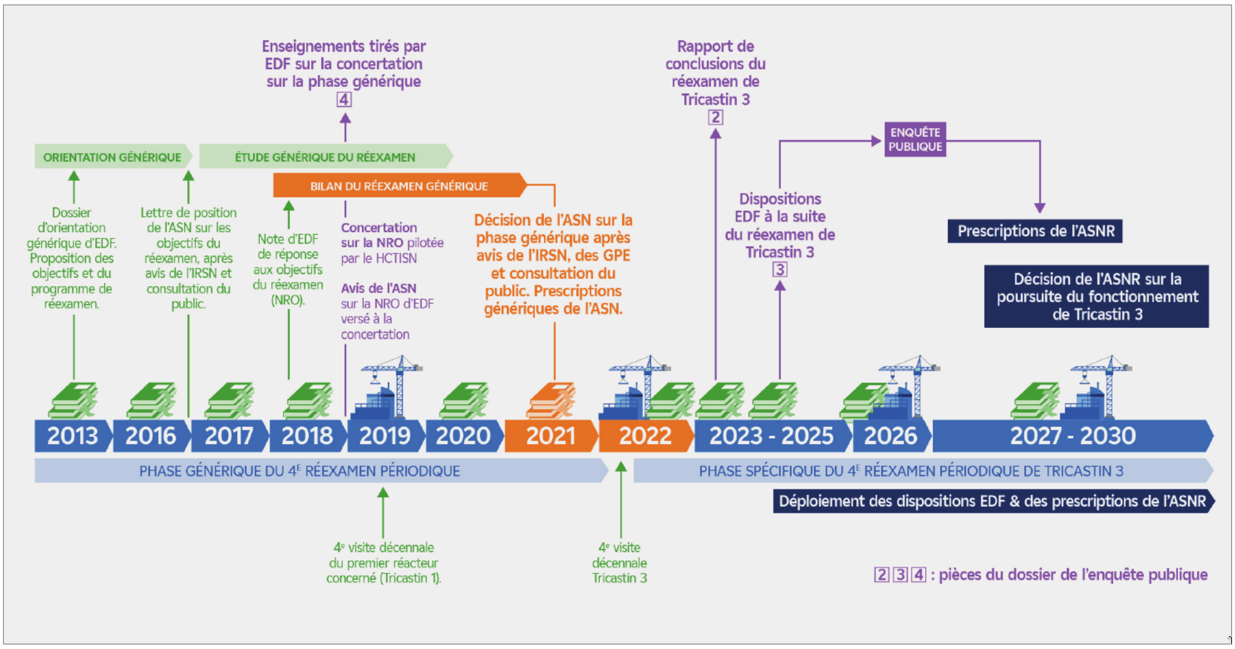
The subject of this report is the 900 MWe fleet. The 900 MWe fleet consists of 32 reactors of the CP type, which are 3-loop pressurized water reactors. This fleet includes three sub-types: CP0, CP1, and CP2 (with CP1 and CP2 often jointly referred to as CPY). While Fessenheim units 1 and 2 (CP0) were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

France is conducting the 4th PSR in two phases a general and a specific phase. 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 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 NPPs 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 timelines show the slightly different main stages of the 4th PSR for Tricastin 3 and 4.

Figure 1: Main stages of the 4th PSR for Tricastin 3 (EIA-REPORT T3, D.1 2026)

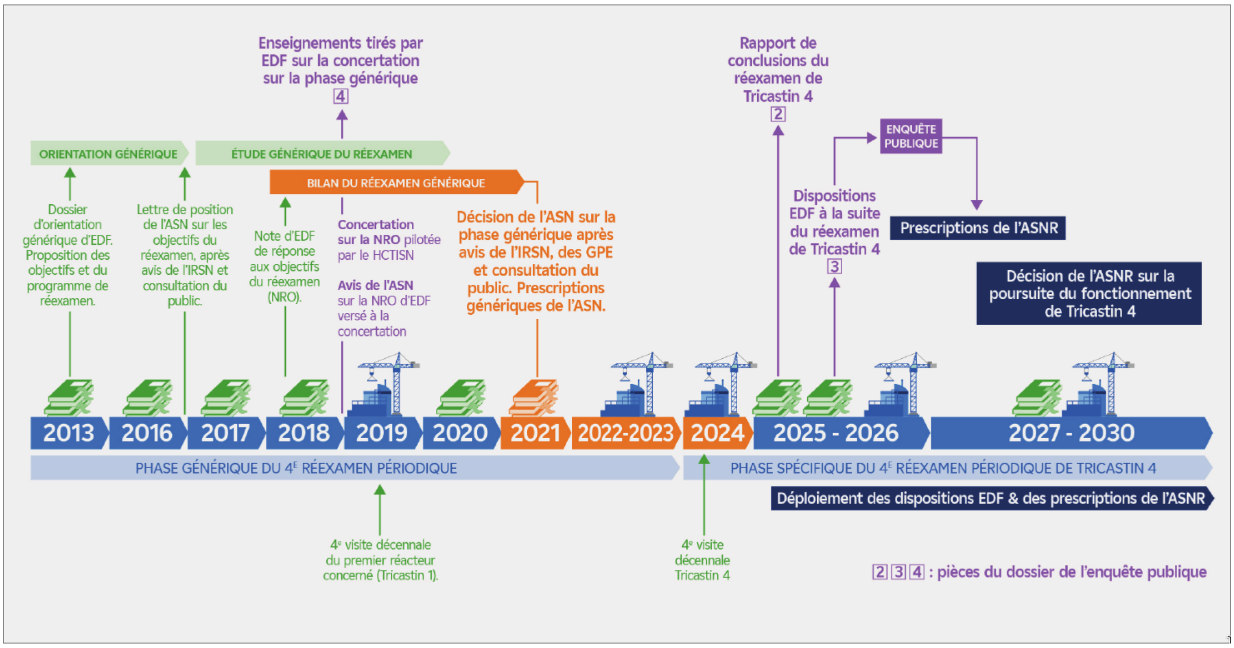
Main stages for Tricastin 3



Source: EIA-REPORT T3, D.1 2026

Figure 2: Main stages of the 4th PSR for Tricastin 4 (EIA-REPORT T4, D.1 2026)

Main stages for Tricastin 4



Source: EIA-REPORT T4, D.1 2026

Public Involvement in the PSR

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

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

While the public involvement process had similarities to an environmental impact assessment (EIA), France always emphasized that the process is not to be seen as an EIA following the EU EIA directive. Instead, France requested the High Committee for Transparency and Information on Nuclear Safety (HCTINS) to organize the process.

2.2 Discussion

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

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

2.3 Conclusions

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

3 LONG-TERM OOPERATION AND OPERATION EXPERIENCE

3.1 Treatment in the EIA documents

Ageing and obsolescence control

The EIA-REPORT T3/4, 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 SSCs, which is being continued in the 4th PSR,
- the process of inspection during operation and maintenance,
- the process for addressing the obsolescence of materials and spare parts.

For Tricastin 3, 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. SG-48 "*Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants*." (EIA-REPORT T3, D.2 2026)

For Tricastin 4, it is stated that a systematic methodology is applied to ensure that aging phenomena do not lead to difficulties in fulfilling a safety function during the period under consideration. This method is consistent with international best practices and is in line with the approach recommended by the IAEA in its Safety Guide No. NS-G-2.12, "*Ageing Management for Nuclear Power Plants*." (EIA-REPORT T4, 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 Tricastin 3 from June 20 to 23, 2022, and on the containment of

Tricastin 4 from May 16 to 19, 2024 with the results meeting expectations. (EIA-REPORT T3/4, D.2 2026)

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

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

Following completion of the aging control analysis of the SSCs of Tricastin 3 and 4, maintenance and control measures were carried out, along with modifications to ensure the continued suitability of these units for operation for a period of ten years after the 4th PSR shutdown.

Risk of obsolescence

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

“Dossier of Suitability for Continued Operation” (DAPE)

The “Dossier of Suitability for Continued Operation” (DAPE) examines in detail the control of aging risks for a component or a structure. It describes the associated aging management program, including aspects such as in-service monitoring, regular and extraordinary maintenance, operating conditions, possible changes, supplementary studies, Research & Development programs, laboratory tests, particularly in the field of materials, quality assurance procedures, etc. The DAPEs are updated every five years. (EIA-REPORT T3/4, D.2 2026)

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

- Reactor pressure vessel,
- Internal core components,

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

The studies conducted at the CPY level were adapted by the Tricastin NPP to determine the management of the aging of the SSCs of units 3&4. During the ten-year inspection, the SSCs underwent a series of maintenance operations, inspections, tests, non-destructive tests, or modifications. (EIA-REPORT T3/4, 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 reactors 3 and 4 at the Tricastin NPP for continued operation under satisfactory safety conditions for the ten-year period VD4-VD5. (EIA-REPORT T3/4, 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.

For Tricastin 3&4, the measures included performing non-destructive tests (penetrant testing of welds and thickness measurements on pipes and branches, as well as visual and TV inspections of rainwater drains). No deviations were found during these tests. (EIA-REPORT T3/4, 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 Tricastin are hardly susceptible, if at all, to this phenomenon. In consultation with ASNR, a strategy for dealing with the NPPs and a corresponding inspection program were established. During the inspection, no defects were found in either unit 3 or unit 4 of the Tricastin NPP that would have required repairs. (EIA-REPORT T3/4, 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 T3/4, D.2 2026)

A total of 257 components qualified for use in accident conditions (MQCA) were inspected, and all identified deviations were corrected. (EIA-REPORT T3/4, D.2 2026)

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

The following two projects are mentioned (EIA-REPORT T3/4, 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 T3/4, D.2 (2026), between January 2013 and December 2022, the Tricastin NPP reported five safety relevant 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 Tricastin nuclear power plant's management system.

It is further explained that upon restart following the completion of the fourth 10-year review, unit 3 and unit 4 of the Tricastin NPP did not experience any safety-related events classified as Level 1 on the INES scale for which corrective actions were required but have not yet been completed. All in all, all safety-related events of Level 1 or higher on the INES scale were addressed, with the exception of an issue related to MOX fuel assemblies. Mitigation measures are being implemented or are planned for this generic event until the identified anomaly is permanently resolved through long-term measures. This issue is the subject of regular discussions with the ASNR. (EIA-REPORT T3/4, D.2 2026)

3.2 Discussion

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

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

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

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

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

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

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

However, the EIA-REPORT (T4, D.2 2026) for Tricastin 4 referred to the outdated IAEA Safety Guide No. NS-G-2.12, “*Aging Management for Nuclear Power Plants*”. This IAEA Safety Guide was published in 2009 and replaced by the specific IAEA Safety Guide SSG-48 “*Aging Management and Development of a Program for the Long-Term Operation of Nuclear Power Plants*,” (IAEA 2018). Note: The EIA-REPORT (T3, D.2 2026) for Tricastin 3 refers to this IAEA publication.

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

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

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

OSART Mission

At the request of the France government, an IAEA Operational Safety Review Team (OSART)[1] visited the Tricastin NPP from 28 November to 15 December 2022. The purpose of the mission was to review operating practices in several areas. In the report on the OSART missions, reference was made, among other things, to deficiencies identified in the area of ageing management and LTO (IAEA 2023)

- Concerning **Ageing Management**, the IAEA team highlighted the following issue: The plant practices for the identification of SSCs to be included in the scope of ageing management (AM are not comprehensive enough and the results of work done are not always adequately documented. Without a complete identification and documentation of the scope of SSCs for AM, the plant cannot demonstrate that all ageing effects of SSCs important to safety are properly managed and equipment reliability maintained. Thus, the IAEA team suggested: The plant should consider ensuring that all relevant SSCs are identified and included in the scope of the AM and the results of work done are adequately documented.

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

- **Plant modification system:** The IAEA team pointed out: The plant modifications are not always effectively controlled to ensure that plant operations are consistent with the intended design change and that the number and duration of temporary modifications are minimized. The IAEA team recommended: The plant should effectively control modifications to ensure that plant operation is consistent with the intended design change and that the number and duration of temporary modifications are minimized.
- **Maintenance backlog:** The IAEA team highlighted that since December 2021, the overdue preventive maintenance (PM) backlog items were around 500 for safety related equipment and around 1200 for non-safety equipment. Two significant safety events were caused by overdue PM related to technical specification requirements. Thus, the IAEA team suggested: The plant should enhance its work management system to ensure that work is completed to schedule, and maintenance backlogs are minimized.
- **Detailed operating feedback findings**: Furthermore, this issue identified by the IAEA team is relevant in regard of the management of LTO: The plant event and trend analyses are not always conducted to a sufficient depth in order to identify the root causes and enable effective corrective actions to be set to prevent event reoccurrence. Thus, the IAEA team suggested: The plant should consider improving the depth of event and trend analyses to identify the root causes and enable effective corrective actions to be set to prevent event reoccurrence.

The status of the recommendations/suggestions is unknown, as no information is available regarding the OSART follow-up mission. It is not known whether the mission in question has already taken place.

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

In the framework of the generic phase of the 5th PSR for the 900 MWe reactors, 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 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 degradation in 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 of Tricastin 3&4 from January 2021 to April 2026 was carried out based on reports of the ASNR.[2]

Non-compliance with General Operating Procedures (RGE)

An analysis of safety-related incidents published by the ASNR over the past five years that were classified as Level 1 on the INES scale revealed a number of incidents that were frequently linked to non-compliance with General Operating Procedures (RGE). The RGE is 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 downtime limits for systems essential to reactor safety. The incidents at Tricastin 3&4 were preceded by component failures, maintenance or operational errors, and in some cases multiple operational errors. The reason could be a lack of safety culture combined with a large number of age-related incidents.

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

The following paragraphs list these events for Tricastin 3 (9 events) and Tricastin 4 (8 events):

- On December 2, 2025, EDF reported a safety-related incident involving a **positioning deviation in the control rods in reactor 3**. On November 27, 2025, while reactor 3 was being brought up to full power, a procedure was initiated to switch the steam generator water supply to the high-flow configuration. Errors made during this procedure caused the control rods to move outside the permissible limits for a period of seven minutes. Furthermore, the actions taken did not correspond to the instructions triggered by the alarm that sounded in the control room. Due to the accumulation of errors and the associated alarms, this event was classified as level 1 on the INES scale.
- On August 21, 2025, EDF reported a safety-related event related to the **unavailability of the boron injection system in the primary circuit of reactor 4**. Since the boron injection system was unavailable for longer than specified in the RGE, this event was classified as level 1 on the INES scale.
- On May 20, 2025, EDF reported a safety-related incident involving the **failure of a cooling system pump during a plant shutdown of reactor 4.** Due to several organizational and human errors, this incident was classified as level 1 on the INES scale.
- On May 26, 2025, EDF reported a safety-related event related to the **unavailability of a diesel-powered emergency power generator for reactor 4** for a period exceeding the duration specified in the reactor's RGE. The diagnosis performed by EDF teams revealed defects in two cylinder temperature sensors, but the operator concluded that the engine's operation was not impaired. However, later an engineer from the independent safety authority questioned the availability of this emergency power generator. Due to the prolonged unavailability of an emergency power generator, this event was classified as level 1 on the INES scale.
- On January 23, 2025, EDF reported a safety-related event related to the **unavailability of two monitoring chains in the radioactivity monitoring system** used to detect a leak between the primary and secondary circuits of **reactor 3.** Due to the prolonged unavailability of the measurement systems over a period exceeding the duration specified in the reactor's RGE, this event was classified as level 1 on the INES scale.
- On November 20, 2024, EDF reported a safety-related incident involving a failure to comply with the **general operating procedures (RGE) for reactor 3**. During a test to verify the functionality of the

emergency control console, a misunderstanding between the operator in the control room and the operator at the emergency control console resulted in a valve connected to the primary circuit remaining open longer than intended, leading to a drop in water level and pressure in the primary circuit. Since the pressure in the primary circuit remained below the limit specified in the RGE for approximately 35 minutes, this event was classified as level 1 on the INES scale.

- On June 27, 2024, EDF reported a safety-related event related to the **delayed detection of the unavailability of an emergency power generator for reactor 3**. Following maintenance work on an emergency power generator, an alarm was triggered indicating a fault in the air conditioning system of the electrical room associated with that generator. Since the required components were unavailable for longer than specified in the RGE, the event was classified as level 1 on the INES scale.
- On June 14, 2024, EDF reported a safety-related event related to the d**elayed detection of a failure in the water level gauge of a safety circuit in reactor 3**. Because the deadline specified in the RGE for restoring the proper configuration of this system was not met—due to the delayed detection—this event was classified as level 1 on the INES scale.
- On January 31, 2024, EDF reported a safety-related event involving **non-compliance with RGE** during the configuration of **the water spray system in the reactor building (EAS) of reactor 4**. A detailed analysis of this first significant event revealed that the situation that occurred on January 26, 2024, had already occurred on multiple occasions, resulting in the deadline for restoring compliance specified in the RGE being exceeded. Due to several configuration errors in the EAS circuit and the exceeding of the deadline for restoring compliance specified in the RGE, this event was classified as level 1 on the INES scale.
- On July 21, 2023, EDF reported a safety-related event related to **non-compliance with a specific provision of the RGE for reactor 3**. The operator must continuously monitor the neutron flux emitted by the reactor core in order to control any unintended power increase. Given the non-compliance with the RGE, this event was classified as level 1 on the INES scale.
- On February 13, 2023, EDF reported a safety-related incident involving the **unavailability of both lines of the high-pressure safety injection system (RIS) at reactor 4**. The safety injection system (RIS) allows boron-containing water to be injected under pressure

into the reactor's primary circuit in the event of an accident leading to a rupture in that circuit, in order to stop the nuclear reaction and ensure cooling of the reactor core. Given that both lines of the high-pressure safety injection circuit were unavailable for a period longer than specified in the RGE, the event was classified as level 1 on the INES scale.

- On October 27, 2021, a safety-related event occurred involving **a positioning deviation of the control rods in reactor 3**. Since this error persisted longer than permitted by the technical operating regulations, this event was classified as level 1 on the INES Scale.
- On April 13, 2021, EDF reported a safety-related event involving the **exceeding of the time limit specified for the shutdown of reactor 4**. On April 7, 2021, the malfunction triggered the initiation of the shutdown of reactor 4. This led to a reduction in pressure and temperature, as well as the use of borated water from the storage pool water treatment and cooling system (PTR) tank. Since the time limit specified in the reactor's RGE was exceeded, the ASN classified this event as level 1 on the INES scale.
- On March 8, 2021, EDF reported a safety-related event involving the d**elayed detection of a failure in the ventilation and filtration system of the control room of reactor 3**. Given the delayed detection, this event was classified as level 1 on the INES scale.
- On December 31, 2020, EDF reported a safety-related event related to the **delayed detection of the failure of a flow sensor in the water supply circuit of a steam generator of reactor 4**. Given the delayed detection of the unavailability of a protective system required by RGE, this event was classified as level 1 on the INES scale.

Deficiencies of the seismic resistance of various components

- On May 13, 2024, EDF reported a safety-related incident involving **deficiencies in the anchoring of certain safety-critical components in reactor 4**. These inspections revealed deviations in certain anchorings, specifically regarding the number, diameter, and arrangement of the anchors. These deviations date back to the reactors' construction period and could have compromised the stability of the equipment attached to them in the event of an earthquake. Given its potential consequences for these reactors, this event is classified as level 1 on the INES scale.
- On June 8, 2023, EDF already reported the same safety-related event concerning **anchoring deficiencies in certain safety-related components**. These deficiencies affect, among others,

reactor 3. Given the potential consequences for these reactors, this event is classified as level 1 on the INES scale.

3.3 Conclusions

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

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.

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

The ASNR's proposal during the generic phase of the 5th PSR to extend aging management beyond 4th PSR is supported. As proposed by the ASNR, the focus must be on components that are necessary for controlling accident situations. However, the scope of the program "qualification of materials under accident conditions" in the 4th PSR is very limited for Tricastin 3&4.

In contrast to the description at the EIA documents the IAEA's OSART (Operational Safety Review Team) mission, which took place at the end of 2022, identified several safety deficiencies in the area of aging management at the Tricastin NPP. Among other things, the team criticized the fact that the scope of SSCs to be included in the AM program is not comprehensive enough. In addition, they criticized issues that are also relevant to LTO: the backlog in maintenance and the inadequate analysis of incidents.

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

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

- 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.
- The status of the recommendations/suggestions of the OSART mission (2022) concerning the ageing management program and LTO at the Tricastin NPP should be provided.

4 EXTERNAL HAZARDS

4.1 Treatment in the EIA documents

EIA-REPORT T3, D.2 (2026) and EIA-REPORT T4, D.2 (2026) provide information of the external hazard types considered in the LTO process. The following external hazards (natural or human-made) are addressed: external flooding, earthquake, extreme weather or climatic conditions (extreme heat, extreme cold), impairment of the ultimate heat sink by frazil, icing, low water level, oil spill or biological flotsam, high wind and wind-generated missiles, tornado, lightning, snow, industrial hazards (explosion, release of hazardous substances) and aircraft crash. EIA-REPORT T3/4, D.2 (2026, p. 99) notes that studies on external hazards take into account the international standards set by WENRA.

Hazard assessment

Earthquake: The seismic design base for the reactors of the 900 MWe fleet is deterministically derived from the maximum observed historical earthquake (SMHV[3]) increased by one degree of intensity giving the so-called maximum safety earthquake (SMS[4]) which is linked to a reference spectrum (SSD). Both determine the seismic design basis of the plant and are reassessed in PSR. Following the Fukushima Daiichi accident in 2011, a new seismic level (SND[5]) was defined. The SND is required to (i) envelope the ground motion of an earthquake with a recurrence interval of 20,000 years, based on probabilistic seismic hazard assessment, (ii) envelope the SMS increased by 50%, and (iii) take site effects into account.

EIA-REPORT T3, D.2 (2026, p. 132) and EIA-REPORT T4, D.2 (2026, p. 131) state that the seismic hazard was reassessed during the 4th PSR according to RFS 2001-01[6] and based on updated seismological findings (seismic-tectonic zoning, characterization of faults, etc.) and the historical

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

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

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

[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

seismicity data of the SisFrance 2012 database. The reassessment led to new seismic ground motion spectral accelerations applicable to the 4th PSR. The SMS spectrum for the 4th PSR of Tricastin is covered by the SMS spectrum of Tricastin that was already considered in the 3rd PSR. Seismic hazard reassessment for Tricastin therefore has no impact on the design of SSCs for the 4th PSR (EIA-REPORT T3/4, D.2 2026, p. 135/134). It is further stated that the seismological parameters of the 11.11.2019 Le Tail earthquake[7] underwent a first characterization which did not lead to new SMS parameters for the Tricastin site.

EIA-REPORT T3, D.2 (2026, p. 135) and EIA-REPORT T4, D.2 (2026, p. 134) describe the tectonic setting of the Tricastin site noting its location near to the active Mimes-, Clanseyes- and Cavenen fault in the Rhone valley noting that evidence for neotectonics exist from four sites along the Nimes fault. The 11-11-2019 Le Teil earthquake occurred on the Rouvière fault which is part of the Cavenen fault system. The EIA Reports state that the closest fault of this system, the St-Montan fault, is located at a distance of 13 km from the site. The Clanseyes fault was the location of repeated earthquake swarms[8].

According to EIA-REPORT T3/4, D.2 (2026, p. 135/134), EDF is currently involved in a pilot study to evaluate the safety significance of faults near Tricastin. The study is announced to include literature review, geophysical, geological, morphotectonic and paleoseismological analyses of faults. Following the Le Teil earthquake in 2019, the Tricastin site was prioritized under the EDF program to improve knowledge of faults around NPPs. In this context, geophysical profiles across faults were conducted in 2022. EIA documents, however, include neither a time schedule for the named investigations nor results achieved so far.

The soil profile at the Tricastin site is characterized by Vs wave velocities higher than 400 m/s. Data derive from cross-borehole measurements and are supported by modelling. EDF concludes that seismic site effects as defined in RFS 2001-01[9] are ruled out for Tricastin (EIA-REPORT T3/4, D.2 2026, p. 136/135).

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, based on meteorological

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[7] The Mw=4.9 Le Teil earthquake 11-11-2019 lead to surface rupture and exceptionally high local ground motion values (CAUSSE et al. 2021)

[8] Clanseyes is located at a distance of about 8 km from the site.

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

records of the 2019 heat wave. Temperatures are projected up to the next PSR, i.e., over the next 10 years (EIA-REPORT T3/4, D.2 2026, p. 143/144). The re-assessment of high temperatures accounts for the heatwave of 2019. Temperatures are determined by exceedance probabilities of 10^{-2} per year and 70% confidence. Safety assessments include analyzing high temperature effects on SSCs, comparing the resulting SSC temperatures to the maximum permissible temperatures, and setting organizational measures to ensure safety functions in cases when temperature limits are exceeded. Analyses also accounted for high water temperature by reviewing requirements related to the ultimate heat sink and identifying and reviewing the cases that adversely affect the cooling function.

Extremely low temperatures: Protection requirements for extremely low temperatures were developed based on lessons learned from the coldest winters of the last decades (notably 1984-1985 and 1986-1987) and implemented during the 2nd PSR. Minimum temperatures for which protection must be provided have not changed between the 3rd and 4th PSR. This is justified by IPCC climate forecasts which predict a decrease in the number of cold days/nights. Methods and assumptions used to derive temperature values are not specified in detail.

External flooding: As part of the 4th PSR, EDF was reviewing the robustness of the NPP with regard to hazards described in ASNR Guidance No. 13 on the protection against external flooding. Analyses for Tricastin 3 and 4, 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 the NPPs). Furthermore, as part of the implementation of the ASNR's 2012 technical regulations, EDF also analyzed the volumetric protection. Reassessments concluded higher rainfall intensities and *“minor”* modifications of the river flood hazards (EIA-REPORT T3/4, D.2 2026, p. 128-132). 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 (EIA-REPORT T3/4, D.2 2026, p. 127).

High wind, tornado and wind-blown projectiles: The EIA documents state that the reassessment of hazards by storm do not require any update (EIA-REPORT T3/4, 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 T3/4, 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).

Impairment of the ultimate heat sink: During the 4th PSR, EDF targeted to verify the robustness of the installations with respect to hazards threatening the ultimate heat sink and reviewed the implementation of the safety requirements for pumping stations (EIA-REPORT T3/4, D.2 2026, p. 150/148). 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 Tricastin include the re-assessment of low water level, icing, frazil ice and other phenomena threatening the cooling water intake by clogging, sedimentation (silting) and pollution of the cooling water with hydrocarbons.

Lightning and electromagnetic interference: The determination of potential lightning strike points and the probability that a target will be struck by lightning follows the standard NF-EN-62305-1. Assessments analyze the 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 T3/4, D.2 2026, p. 155/156).

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

Human-made hazards (industrial facilities, pipelines and transport of dangerous materials): Hazard assessment is based on ASNR Regulation RFS I-2.d. (ASN 1982). Analyses include external explosion and hazards resulting from transportation of hazardous materials outside of the site and on the site. ASNR (ASN 1982) requires a maximum probability of 10^{-6} per year for unacceptable radioactive releases caused by human-made hazards. In the framework of the 4th PSR EDF revised the data on industrial facilities in the surrounding area and re-assessed the associated hazards. Analyses revealed probabilities for hazards caused by explosion, intoxication by industrial releases and external fire between 10^{-7} and 10^{-8} per year. No necessities for retrofitting were revealed (EIA-REPORT T3/4, D.2 2026, p. 164/163).

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 Tricastin 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 T3/4, D2 2026, p. 164/163).

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 900 MW reactors, EDF plans to achieve safety improvements by installing a Hardened Safety Core (Noyau Dur) to increase the robustness of the NPP against hazards such as earthquakes, tornadoes and external flooding. In addition to this general measure, the EIA documents list a number of specific improvements including the following measures to protect the NPP from external hazards:

Table 1: Upgrading measures for SSCs important to safety with respect to external hazards, Tricastin 3 and 4 (adapted from EIA-REPORT T3/4, D.2 2026)

PNPE1008	3 4	Explosion	Reconstruction and redesigning of the SGZ parks	Completed
PNPE1069	3 4	High temperature	Improvement of the air conditioning in rooms housing cooling units	Completed
PNPE1070	3 4	High temperature	Improvement of the air conditioning in the rooms	Completed
PNPE1109 tome D	3	Lightning	Overhead line pylon refurbishment – lightning rods	Completed
PNPE1117	3 4	External flooding	Flood protection and prevention of bypass at the pumping station	Completed

PNPE1118	3 4	Earthquake	Seismic reinforcement of batteries	Completed
PNPE1121	3 4	External flooding	Volumetric protection and prevention of bypass	Completed
PNPE1138	3 4	External flooding	Protection of the safety block system against external floods	Completed
PNPE1165	3 4	High wind	Installation of protective grilles	Completed
PNPE1191	3 4	Earthquake	Seismic resistance of cable trays	Completed
PNPE1238	4	Earthquake	Increase of earthquake resistance of heating oil tarpaulins for earthquakes exceeding the SMS	Completed
PNPE1276	3 4	Earthquake	Reinforcement of gravel dam against SND	Completed
PNPP1012 tome A	3	Earthquake	Hazards associated with gas storage facilities outside of buildings	Completed
PNPE1333 tome A	3 4	Earthquake	Earthquake protection of the core of the main primary circuit and the main secondary circuit	Completed
PNPP1419	3 4	Earthquake	Implementation of automatic shutdown	Completed
PNPP1666	3 4	Noyau Dur	Electric supply by emergency diesel	Completed
PNPP1675	3 4	External flooding	Protection against extreme floods – Noyau Dur	Completed
PNPP1688 tome A	3	Noyau Dur	Implementation of control for HSC	Completed
PNPP1688 tome C	4	Noyau Dur	Implementation of a "Noyau Dur" control system for new SSCs	Completed
PNPP1791	3 4	Heat sink	Refurbishment of the pressure drop sensors of filters and installation of a level measurement system downstream of the filter system	Completed
PNPP1898	3 4	Earthquake	Robustness against SND – reactor building polar bridges	Completed
PNPP1943	3 4	Earthquake	Prevention of earthquake-induced flooding	Completed
PNPP1951	3 4	Lightning	Installation of surge protection devices	Completed
PNPP1955	3 4	High temperature	Improvement of cooling of emergency diesels	Completed
PNRL1835	34	High temperature	Automatic contamination monitoring of heat exchangers	Completed
PNRL1844	3 4	Heat sink	Access ramps to heat sink	Completed
PNRL1924	3 4	Explosion	Grounding of channels for the ventilation of iodine rooms	Completed
PNRL1933	3 4	Earthquake	Earthquake protection of the soda mixing facility	Completed

PNRL1955	3 4	Low temperature	Changing the setpoint setting of air heater	Completed
PNRL1984	4	External flooding	Implementation of flexible hook-up connections	Completed
PNPE1115	3 4	Earthquake	Implementation of automatic reactor shutdown for earthquake	Phase B(1)
PNPE1305	3 4	Earthquake	Establishment of an earthquake-proof detector to identify loss of heat sink	Phase B(1)
PNPE1330 Tome B	3 4	Explosion	Explosion protection in battery rooms	Phase B(1)
PNPE1332	3 4	Earthquake	Seismic resistance of piping	Phase B(1)
PNPE1357	3 4	Earthquake	Earthquake resistance (Noyau Dur) of electrical equipment and control systems	Phase B(1)
PNPE1358	3 4	Earthquake tornado	SND and tornado resistance of ventilation systems	Phase B(1)
PNPE1478	4	Earthquake	Earthquake resistance of instrumentation against SND earthquake	Phase B(1)
PNPP1620	3	Earthquake	Replacement of the earthquake-resistant SND abseiling devices	Phase B(1)
PNPP1688 tome D	3 4	Noyau Dur	Implementation of Noyau Dur control systems for new SSCs	Phase B(1)
PNPP1824	3 4	Noyau Dur	Addition of an analog level measurement chain for the BK fuel pool	Phase B(1)
PNPE1427	3	Noyau Dur	Installation of an injection pump on the seals of the primary motor pump assemblies “Noyau Dur”	Phase “supplements”(2)
PNPE1459	3	Noyau Dur	Improvement of long-term cooling of certain rooms, in the event of the loss of the heat sink.	Phase “supplements” (2)
PNPE1119	4	Tornado	Tornado protection	18.06.2030
PNPE1323	3 4	Tornado	Improving the resistance of the chimney against earthquake, high wind and tornado	Phase B(1)
PNPE1333 tome B	3 4	Earthquake	Earthquake protection of the core area of the main primary circuit and main secondary circuit	Phase B(1)
PNPE1377	3 4	Earthquake	Enhancement of the resistance of the pressure relief and filter device of the tank during an SMS earthquake	18.06.2030
PNPP1722	4	Low temperature	Improvement of thermal insulation	Phase B(1)
PNPP1723	3 4	Low temperature	Setting up a winterization system for non-winterized locations in case of frost	End 2024

TCDI0021	3	External flooding	Volumetric protection	End 2023

[1] Changes whose implementation will be completed as part of Phase B of 4th PSR 900 are due on 06. March 2028 for the reactor Tricastin 3 and 18. June 2030 for Tricastin 4.

[2] Deadlines for implementation in the phase "supplements" are March 2028 for Tricastin 3 and June 2030 for Tricastin 4.

Earthquake: The attachments of EIA-REPORT T3/4, D.2 (2026) list numerous actions related to the earthquake resistance of SSCs of Tricastin 3 and 4 (Table 1). 7 out of 19 upgrading / retrofitting measures have not been completed by now, completion is scheduled for Phase B of the 4th PSR which is due 5 years after the baseline PSR. This is 06. March 2028 for Tricastin 3 (EIA-REPORT T3/4, D.2 2026, p.13) and 18. June 2030 Tricastin 4 (EIA-REPORT T4, D.2 2026, p.11). The same dates are valid for the phase "supplements". Among the Phase B issues are measures to complete the implementation of the Hardened Safety Core (Noyau Dur).

External flooding: Flood protection is increased by re-enforced enhancing volumetric protection and retrofitting the flood dams protecting from river floods to withstand the SND (EIA-REPORT T3/4, D.1 2026, p.27). All measures except some volumetric protections have been implemented.

High temperatures: EIA-REPORT T3, D.2 (2026) and EIA-REPORT T4, D.2 (2026) list the following measures as implemented: modification of the pollution monitoring of heat exchangers to improve cooling by the ultimate heat sink (water of the river Rhône), improvements of the air conditioning in vulnerable rooms and improvement of cooling of emergency diesels.

Low temperatures: EIA-REPORT T3/4, D.2 (2026, p. 150/148) identify several SSCs as vulnerable against extremely low temperatures. The results of the 4th PSR gave rise to a number of additional safety measures such as organizational measures and measures to improve the insulation of vulnerable pumps, piping and tanks.

High wind and tornado: 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 T3/4, D.2 2026, p. 154/153).

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 sensors to monitor pressure drops due to clogging and implementing organizational measures to ensure timely cleaning of intake filters. 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 Tricastin NPP to achieve the lightning protection such as the installation of additional lightning rods (EIA-REPORT T3/4, D.2 2026, p. 157).

Human-made hazards (industrial facilities, pipelines and transport of dangerous materials): The EIA documents state that resistance to detonation-type explosions of buildings and civil structures housing or containing SSCs important to safety is provided by design.

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

4.2 Discussion

Generic aspects

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

- The mandatory contents of PSR include plant design, deterministic safety analyses, probabilistic safety analyses and hazard analyses are described in detail in Issue P, Reference Level P2.2 of WENRA (2021).
- Issue E, Reference Level E11.1 requires regular reviews of the actual design basis to determine whether the design basis is still appropriate.

- Issue F, Reference Level F5.1 requires the same regular review for Design Extension Conditions (DEC)
- Issue TU summarizes requirements for external hazard assessment, most importantly the definition of design basis events with exceedance frequencies not higher than 10^{-4} per annum, and the requirement to provide protection against design basis events. Protection shall be of sufficient reliability so that the fundamental safety functions are conservatively ensured.
- Issue TU, Reference Levels TU6.1 to TU6.3 list requirements for considering DEC.
- In addition to the requirements stipulated in the WENRA Safety Reference Levels, WENRA provides ample guidance on how to consider external hazards in safety demonstrations (WENRA 2020a-d).

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

The EIA documents provide no clear evidence if these WENRA requirements were followed by EDF. For most external hazards, the methods, data and assumptions used in the hazard assessment are not specified. Conformity with WENRA requirements and guidance therefore cannot be assessed. It remains particularly unclear if design basis events with exceedance frequencies not higher than 10^{-4} per annum or design basis events that provide an equivalent level of protection have been determined for all external hazards that apply to the site, if the assessment of design basis events is in line with WENRA regulations and guidance, and how DEC are addressed for the identified hazards. In this context it appears remarkable that the EIA documents refer to WENRA (2008) and WENRA (2014)[10]. The cited editions are outdated by the SRLs published in 2021.

Conformity with WENRA Reference Levels still needs to be proved for earthquake and seismic ground shaking. The Design Basis Earthquakes (DBE) for Tricastin and the other reactors of the French 900 MWe fleet are still based on deterministic analyses. Demonstration that the deterministically determined DBE is equivalent to a PSHA-derived design basis earthquake with an average recurrence interval of 10,000 years is missing. It therefore remains to be demonstrated that the seismic resistance of all SSCs important to safety is sufficient to conservatively ensure the fundamental safety functions for a DBE with an average

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

recurrence interval of 10,000 years as required by WENRA (2021). The authors of this report assume that adequate protection against a probabilistically derived DBE, should it be higher than the deterministic value for which the plant was designed, is intended to be ensured by the Hardened Safety Core (Noyau Dur). This, however, would contradict the Defence-in-Depth (DiD) concept and the separation of DiD levels because the DEC equipment of the Noyau Dur could become necessary to protect the plant against design basis hazards, i.e., the probabilistically derived DBE. The Hardened Safety Core is classified as a 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)[11] suggests that design basis values were only established after the start of construction of Tricastin 3 and 4 (Table 2:). At the background of the standardized reactor series operated in France, EDF introduced the notion to define the DBE as the envelope spectrum of the various SMS spectra associated with the different sites of the same plant series (ASN 2011a). This approach allowed pooling the design studies for the reactors on the respective nuclear islands. All plants of a specific series consequently 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[12]. 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

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[11] 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.

[12] 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.

earthquake at a distance of 10 km from the site, and a M=6.6 event at 40 km distance (ASN 2011a).

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

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

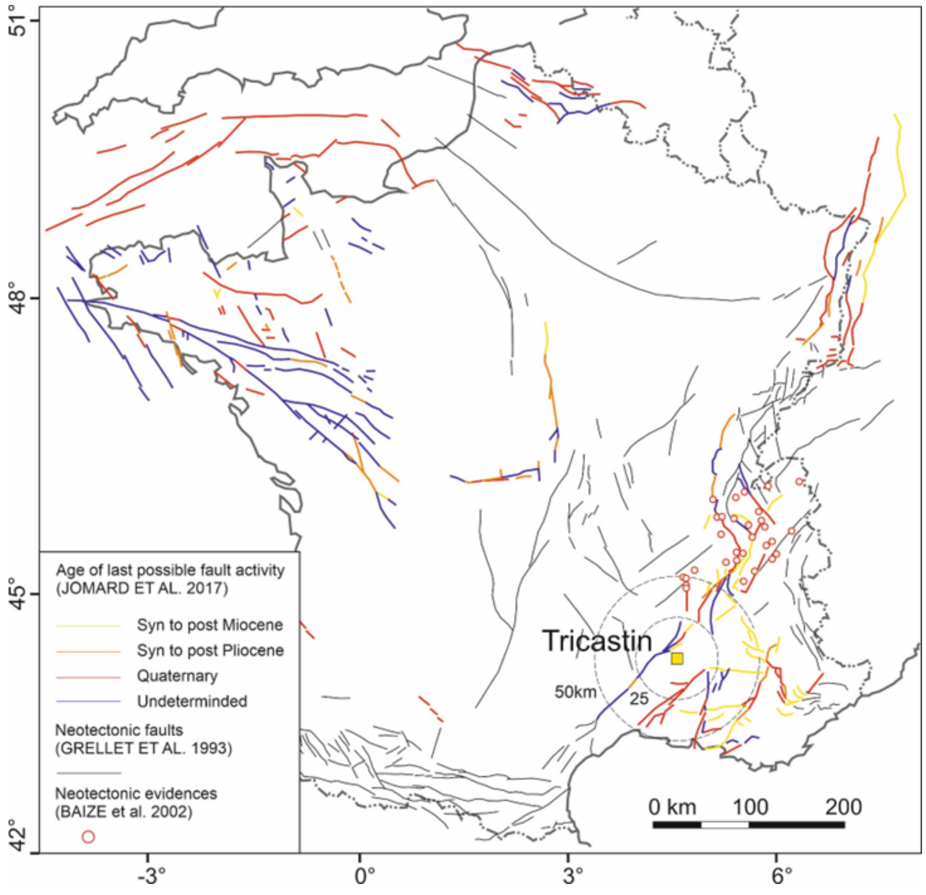
The EIA documents include neither numerical information on the deterministically derived SMS (design basis earthquake) ground motion nor on the ground motion values for a 20,000 year recurrence interval relevant for the SND. Available documents by ASN (2011a) and IRSN (2012; 2013) indicate an SMS ground motion value (PGA) for the Tricastin nuclear island and site structure of 0.2 g and 0.3 g respectively. PGA values reported for the SND are 0.4 g. (Table 2).

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

NPP	Tricastin 3	Tricastin 4
Start of construction	01.04.1975	01.05.1978
Start of commercial operation	11.05.1981	01.11.1981
DBE Nuclear island	0.2 g	0.2 g
DBE Site structure	0.2 g; (2011): 0.3 g$^{(1)}$	0.2 g; (2011): 0.3 g$^{(1)}$
SND	PGA=0.4 g$^{(2)(3)}$	PGA=0.4 g$^{(2)(3)}$

(1) Application of RFS I-2-C and 2001-01 (2) IRSN (2012) (3) IRSN (2013)

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



Location of faults near Tricastin

Source: Environment Agency Austria

In tectonic terms, the Tricastin site is located near or within the Bresse Graben System, which together with the lower and upper Rhine Graben forms the Cenozoic European rift system. EIA-REPORT T3, D.2 (2026, p. 134) and EIA-REPORT T4, D.2 (2026, p. 134) provide evidence that EDF is aware that geological and seismological data indicate the existence of several active faults in the near-region and region around the Tricastin site, naming the active Mimes-, Clanseyes- and Cavenen fault (EIA-REPORT T3/4, D.2 2026, p. 135; 134) and mentioning neotectonic evidence from four sites along the Nimes fault. This is in line with data by JOMARD et al. (2017) who identified numerous Quaternary faults in the near-region and region (IAEA 2022) of Tricastin (Figure 3).

The importance of active faults for the safety of Tricastin is underlined by the M=4.9 Le Teil earthquake of 11. November 2019 which occurred only about 20 km North of Tricastin. The earthquake caused surface displacement and led to near-fault ground accelerations of likely >1 g (RITZ et al. 2020; CAUSSE et al. 2021). The Le Teil earthquake occurred on the Rouvière fault which is part of the Cavenen fault system. This fault system extends into the near-region (<25 km distance) of the site.

It seems that these faults have not been taken into account as potential seismic sources in the seismic hazard assessments performed so far. This is remarkable in the light of the Le Teil earthquake. According to EIA-REPORT T3/4, D.2 (2026, p. 135/134), EDF is currently involved in a “pilot study” to evaluate the safety significance of faults near the Tricastin NPP. It is said that the Tricastin site was “prioritized” under the EDF program to improve knowledge of faults around NPPs. EIA documents, however, neither include a tangible time schedule for investigations announced in the EIA documents nor any results of investigations performed so far.

For active faults in the near-region of the NPP site WENRA (2020b p.11ff) suggests systematic fault mapping and collecting paleoseismological information. Efforts should at least be made in the near-region of the site (not less than 25 km) to collect geological, geophysical, geomorphologic, geodetic and paleoseismological data for identifying and characterizing active faults. At the background of the existing literature (JOMARD et al., 2017; RITZ et al. 2021 and references therein) it is remarkable that a process for active fault characterization and paleoseismological investigations has not been implemented much earlier. WENRA (2021) requires *“the actual design basis shall regularly, and when relevant as a result of operating experience and significant new safety information, be reviewed”* (Reference Level E11.11). The Le Teil earthquake proving both, the existence of capable faults and the possibility of ground motion values well in excess of the SMS and SND of the Tricastin reactors is clearly a *“significant new safety information”*.

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

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

The increasing risk due to aging effects must also be taken into account for Tricastin 3 and 4: A study uses numerical simulations to investigate the influence of aging on the effects of a military aircraft impact on a NPP. The results show that the aging of a plant increases its susceptibility to large-scale or localized penetrations. The greater the degradation of the materials, the lower the residual resistance and the greater the risk of wall perforation. With the same impact force, the strength of the aged containment is reduced by approximately 30%. (FRANO 2021)

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

Nuclear Threat Initiative (NTI)

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

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

International Physical Protection Advisory Service (IPPAS)

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

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

4.3 Conclusions

The EIA documents provide ample information on hazard types considered in the safety demonstration for Tricastin 3 and 4 and measures already implemented or decided to be implemented in order to strengthen the robustness of the reactor with respect to external hazards. The documents, however, do not provide clear evidence if the processes of the PSR and LTO follow WENRA requirements as stipulated by ASNR. For most external hazards, the methods, data and assumptions used in the hazard assessment are not specified in detail. Conformity with WENRA requirements and guidance therefore cannot be assessed. It remains particularly unclear if design basis events with exceedance frequencies not higher than 10^{-4} per annum have been determined for all external hazards that apply to the site, and how DEC are addressed for the identified hazards.

Non-conformity with WENRA Reference Levels is observed for earthquake and seismic ground shaking. The Design Basis Earthquakes (DBE) for Tricastin 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 Tricastin nuclear island and site structure of 0.3 g. PGA values reported for the SND are 0.4 g. We conclude that it remains to be demonstrated that the seismic resistance of all SSCs important to safety is sufficient to conservatively ensure the fundamental safety functions for a DBE with an average recurrence interval of 10,000 years as required by WENRA (2021).

The Tricastin site is located near the active Mimes-, Clanseyes- and Cavenen fault (EIA-REPORT T3/4, D.2 2026; JOMARD et al. 2017). The importance of active faults for the safety of Tricastin is underlined by the 2019 M=4.9 Le Teil earthquake with an epicenter located only about 20 km North of Tricastin. The earthquake led to surface displacement and an epicentral ground accelerations of likely >1 g (RITZ et al., 2020; CAUSSE et al., 2021). It occurred on the Cavenen fault system which extends into the near-region (distance <25 km; IAEA 2022) of the site. The named active faults have apparently not been taken into account as seismic sources in the seismic hazard assessments although they are known for about 10 years at least (JOMARD et al., 20217). According to EIA-REPORT T3/4, D.2 (2026), EDF is currently conducting a “pilot study” to evaluate the safety significance of faults near Tricastin, however, without providing a tangible time schedule for these investigations. An updated PSHA-based reassessment of seismic hazards that takes advantage of fault data is not announced.

For active faults in the near-region of an NPP site WENRA (2020b p.11ff) suggests systematic fault mapping and collecting paleoseismological information. At the background of the existing literature (JOMARD et al., 2017; RITZ et al. 2021 and references therein) it is remarkable that a process for active fault characterization has not been implemented much earlier. WENRA (2021) requires *"the actual design basis shall regularly, and when relevant as a result of operating experience and significant new safety information, be reviewed"* (Reference Level E11.11). The Le Teil earthquake proved both, the existence of capable faults and the possibility of ground motion values well in excess to the SMS and SND of the Tricastin reactors. It is clearly regarded a *"significant new safety information"*.

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 possible deficits in terms of the protection against design basis events.

With respect to safety upgrades of Tricastin 3 and 4, it is evident that one of the most important measures to provide protection against external hazards is the implementation of the Hardened Safety Core (Noyau Dur). However, the implementation of the Noyau Dur is not completed yet. The attachments of EIA-REPORT T3/4, D.2 (2026) list several measures related to the implementation of the Noyau Dur which are pending and should be completed in Phase B or the phase "supplements" of the 4th PSR. The related deadlines are 6. March 2028 for Tricastin 3 (EIA-REPORT T3, D.2 2026, p.13) and 18. June 2030 Tricastin 4 (EIA-REPORT T4, D.2 2026, p.11).

The fundamental decision to implement the Hardened Safety Core has been made in 2012 in the aftermath of the European Stress Tests (ASN 2012). The fact that the implementation of the Noyau Dur will be completed only 18 years thereafter appears remarkable at the background that WENRA requires the *"timely implementation of the reasonably practicable safety improvements identified"* (WENRA 2021, Issue A, Reference Level A2.3). This suggests that the announced implementation schedules contradict the WENRA requirement.

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

Information regarding the issue of terror attacks would be of great interest, considering the far reaching consequences of potential attacks. In particular, the EIA documents should include information on the requirements for the design against the targeted crash of a commercial aircraft. This topic is particularly important, because reactor building as well as the spent fuel building of the Tricastin 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 Tricastin 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 Tricastin 3 and 4, in particular, the types of seismic sources considered (source zones and/or fault sources), time coverage of the earthquake catalogue, minimum and maximum magnitudes, ground motion prediction equations, and site conditions.
    - The actual ground motion values for the SMS and the SND.
- The deterministically derived SMS ground motion should be benchmarked against a PSHA-derived ground motion for a recurrence interval of 10,000 years as required by WENRA (2021).
- Additional safety demonstrations should be required to ensure that all SSCs relevant to safety can cope with a probabilistically derived 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 Tricastin 3 and 4.
- The methods, data and assumptions used to determine the SND for Tricastin 3 and 4, in particular, the types of seismic sources considered (source zones and/or fault sources), time coverage of the

earthquake catalogue, minimum and maximum magnitudes, ground motion prediction equations, and site conditions should be reviewed. The PSHA should be benchmarked against WENRA requirements (WENRA 2021) and recommendations (WENRA 2020a, b).

- Dedicated assessments of near-regional active faults and 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: EIA documents and 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 the Tricastin 3 and 4 against design events, i.e., events with recurrence intervals of 10,000 years or less (e.g., earthquakes). This is to ensure the independence of Defence-in-Depth (DiD) levels 3 and 4.
- It should be evaluated if the long timeframe for implementing a research program to identify, assess and parametrize active faults in the near-region and region around the Tricastin site 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 a “prioritized pilot study” following the Le Teil 2019 earthquake, i.e., 7 years after the event, appears unreasonably long. The typical timeframe for completing a paleoseismological assessment of an active fault is 1 or 2 years including all preparatory investigations.
- With respect to possible terror attacks, the following questions should be addressed:
  - Have any studies been or will be carried out on the threat posed by newer technologies, in particular potential attacks using civilian or military drones?
  - How is the result of the Nuclear Security Index 2023 (NTI 2025) for France assessed? Are improvements planned with regard to “security culture”, “cybersecurity” and “protection against insider threats”?

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

5.1 Treatment in the EIA documents

As established in the Chapter 2 on Procedure, the PSR framework in France is structured into two distinct phases: a generic assessment and a plant-specific assessment.

Each phase addresses two core objectives:

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

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

Scope of Measures and Review Focus: This chapter details the modifications and upgrades specified in EIA-REPORT T3/4, D.1 (2026), EIA-REPORT T3/4, 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 T3/4, D.1 (2026) and EIA-REPORT T3/4, D.2 (2026) provide executive summaries and outline the highest-priority measures identified for implementation regarding Accidents Without Core Melt.

Measures Implemented:

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

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

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

Accidents-2: Increased Relief Capacity of Steam Line Valves (GCT-a Modification):

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

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

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

While this measure is undoubtedly beneficial, the report does not indicate which operational measures were taken to achieve it. The iodine concentration in the primary coolant is the result of a balance between the release of iodine from the fuel due to micro-failures in the fuel rods and the operation of the makeup and letdown systems, which remove fission products from the primary system coolant.

Key measures for the spent fuel pool

For the spent fuel pool EIA-REPORT T3/4, D.1 (2026) and EIA-REPORT T3/4, D.2 (2026) list the following items:

Measures implemented:

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 / shields to eliminate the risk of a fire spreading from one pump in the cooling circuit to the other.

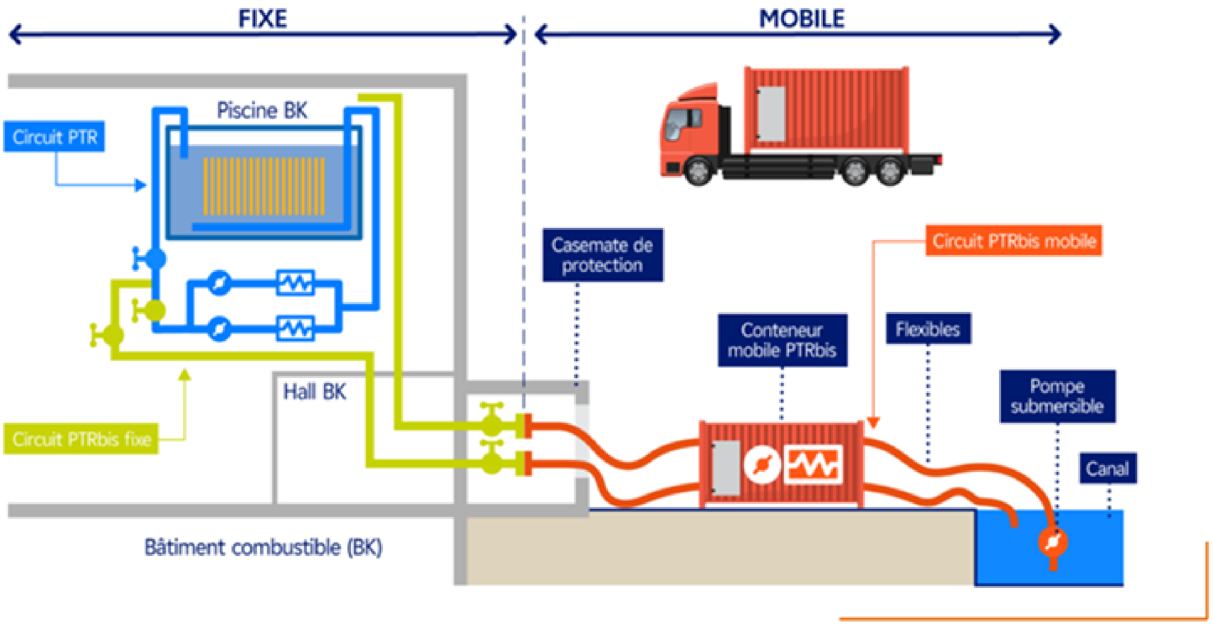
Pool-2: Accident scenarios: Following the transpose of EPR FLA3 scenarios to 900 MWe plants, to further secure spent fuel pool cooling, EDF plans to duplicate the automatic isolation device on the suction line of the pool’s normal cooling circuit, ensuring reliable isolation under accident conditions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

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

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

Figure 4: Schematic diagram of the cooling circuit of the spent fuel pool (EIA-REPORT T3/4, D.1 2026)

Schematic of the cooling circuit of the spent fuel pool



Source: EIA-REPORT T3/4, D.1 2026

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EIA-REPORT T3, D.2 (2026) and EIA-REPORT T4, D.2 (2026) represent the most extensive of the five reports submitted for the LTO review. Their section on risks is logically segmented into two main components:

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

- Re-evaluation (SOTA Comparison): Derivation of necessary measures by comparing the safety profile of Tricastin 3 / Tricastin 4 reactor against the Current State of the Art (SOTA), as defined by the Flamanville 3 EPR design.

- The "Conformity" section is deemed outside the scope of this discussion as it 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, 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), 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

The report defines three implementation phases:

Phase A: Modifications deployed before or during the ten-year outage, enabling large-scale hardware changes and associated RGE updates.

Phase B: Completion of remaining hardware and intellectual modifications. For Tricastin 3 and 4, Phase B items must be deployed no later than 5 years after the Review Conclusions Reports.

Additionals (“Supplements”): Certain post-review modifications requiring longer preparation (e.g., new equipment qualified for very harsh conditions), with about 5 years of preparation. Deadline depends on reactor category set by ASNR for VD4-900: reactors whose re-evaluation began before 31/12/2021 have up to 6 years after the Review Conclusions Report; others have 5 years.

Accidents Without Core Melt

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

Tricastin 3

Fully implemented:

- PNPE1141 Increase in flow rate of GCT-a regulating valves,
- PNPP1864 Refilling of the ASG reservoir by the JP* fire water circuit,
- PNPP1595 SEBIM valve head replacement,
- PNRL1817́ Filter Tmoy – SIP C,
- PNPP1838 Renovation of the RPN CPY,
- PNPP1873 SIP-Protection System Evolution,
- PNRL1829 Increase in REA boron volume required - Increase in PET free volume, Generalization of hafnium-absorbing clusters in the VD4 900 reference standard, Power dilution alarm sheet,
- PNPE1152 Substitution of the power supply by the Emergency Turbo Alternator by the Ultimate Emergency Diesel Fuel
- PNPP1811 Implementation of an EAS-ND system for primary water injection and residual power evacuation,
- PNRL1894 Replacement of primary circuit temperature probes in cold branch,
- PNRL1879 Contactors / latching (reliability of electrical-building ventilation, particularly in shutdown states),
- PNPP1811 Implementation of an EAS-ND system for primary water injection and residual power evacuation,
- PNPP1371 Reliability of the insulation of GMPP thermal barriers,
- PNPE1152 Substitution of the electrical power supply by the Emergency Turbo Alternator with a power supply by the Ultimate Emergency Diesel,
- PNPE1166 – Addition of an electrical architecture enabling the replacement of the Ultimate Emergency Diesel by the Ultimate Emergency Diesel from the neighbouring unit

Planned/Phase B (to be deployed):

- PNPE1359 Increase in RIS accumulator pressure,
- PNRL1957 Modification of the blocking line of group R,
- PNPE1189 Addition of a primary fluid sampling device in standstill state downstream of the heat exchanger CEPP (Primary Pump

Sealing Circuit) to meet the risks of heterogeneous dilution by CEPP leak,

- PNPE1442 Self-monitoring cells for operating RCV-RIS from the BL (simultaneous injection)

Tricastin 4

Fully implemented:

- PNPE1141 Increasing the flow rate of GCT-a control valves, Generalization of hafnium absorption clusters in the VD4 900 reference model, Power dilution alarm sheet,
- PNPP1864 Creation of a fixed means of replenishing the reservoir of the ASG Steam Generator Emergency Supply System by the fire protection systems,
- PNPP1595 SEBIM valve head replacement,
- PNRL1817 Evolution of the instrumentation system by adding a filter to the temperature signal average,
- PNPP1838 RPN Renovation: New RPN Architecture and Functionalities (Power Measurement),
- PNPP1873 Evolution of the process instrumentation system - threshold reparameterization,
- PNRL1829 Increased REA boron volume required - Increased free volume,
- PNPP1811 Implementation of an EAS-ND system for primary water injection and evacuation of residual power from the containment,
- PNRL1894 Replacement of primary circuit temperature probes in the cold branch,
- PNPE1152 Substitution of the power supply by the LLS Emergency Turbo Alternator by the Ultimate Emergency Diesel,
- PNPP1419 Implementation of an AAR for earthquakes,
- PNPP1714 Water source for the Hard-Core backup system,
- PNPE1152 Substitution of the electrical power supply by the Emergency Turbo Alternator with a power supply by the Ultimate Emergency Diesel Engine,
- PNPE1166 Addition of an electrical architecture allowing the substitution of the Ultimate Emergency Diesel by the Ultimate Emergency Diesel from the neighbouring unit,
- PNPP1371 Improving the reliability of GMPP thermal barrier insulation,

- PNPP1811 Implementation of an EAS-ND system for primary water injection and evacuation of residual power from the reactor containment,
- PNRL1879 Replacement of electrical contractors for fans in electrical building ventilation systems with latching contactors

Planned/Phase B (to be deployed):

- PNRL1957 Lowering of the blocking line of group R following the integration of the water sheets in safety studies,
- PNPE1359 Increase in RIS system accumulator pressure,
- PNPE1189 Addition of a primary fluid sampling device in standby mode downstream of the heat exchanger CEPP (Primary Pump Sealing Circuit) with regard to the risks of heterogeneous dilution due to CEPP leakage,
- PNPE1442 Operational modification: Accessibility - use of self-monitoring cells to operate the RCV-RIS from the BL (simultaneous ISHP injection)

Spent Fuel Pool – BK (Bâtiment combustible)

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

Tricastin 3

Fully Implemented:

- PNXX1752 Analog level measurement of the fuel storage pool,
- PNPP1289 Resizing of the siphon break located on the discharge line of the system cooling of the fuel storage pool,
- PNPP1401 Installation of a second static joint on the pool cofferdam of the reactor building,
- PNPP1402 Automatic closure of the suction PTR valve in the fuel storage pool at a very low level,
- PNPP1403 PTR transfer tube valve motorization,
- PNPP1474 Pressure measurement for RIS accumulators, wide range,
- PNPP1549 Safe positioning of a fuel assembly,
- PNPP1666 Ultimate Emergency Diesel,
- PNPP1679 Strengthening of the on/off level measurement chains for fuel storage pools,

- PNPP1780 Automation of reactor building pool drain valves

Planned/Phase B (to be deployed):

- PNPE1128 On/Off Level Measurements of the reactor building pool,
- PNPE1258 – Installation of the ASG-ND device and fixed line for pool power supply,
- PNRL1803 Installation of a Hardened Safety Core water booster, the reactor building pool and its steam outlet,
- PNPP1824 Installation of an analogue level measurement chain for the storage pool,
- PNPP1949 Installation of a fire protection screen between the PTR pumps for the physical separation of the two PTR lanes,
- PNRL1984 Hook (anchoring point) to secure a flexible "plug"/connection used to fill the fuel storage pool, so that emergency raw-water hoses can be held in place and deliver makeup reliably under flood conditions

Tricastin 4

Fully Implemented:

- PNXX1752 Analog level measurement of the fuel storage pool,
- PNPP1289 Resizing of the siphon break located on the discharge line of the fuel storage pool cooling system,
- PNPP1401 Installation of a second static joint on the cofferdam of the reactor building pool,
- PNPP1402 Automatic closure of the PTR suction valve in the fuel storage pool at very low levels,
- PNPP1403 PTR transfer tube valve motorization,
- PNPP1474 RIS Accumulator Pressure Measurement Wide Range,
- PNPP1549 Safe positioning of a fuel assembly,
- PNPP1666 Ultimate Emergency Diesel Engine,
- PNPP1780 Automation of reactor building pool drain valves,
- PNPP1907 Creation of a diversified mobile cooling system PTR bis,
- PNPE1344 Doubling of the fuel pool suction line isolation automation BK via the PTR valves,
- PNPE1443 Reinforcement of a hopper in the BK,
- PNRL1895 Improving the reliability of the transfer tube valve control for closing under flow,

- PNRL1984 Hook for securing a flexible plug to fill the fuel storage pool in case of river flooding beyond the Hard Core,
- PNPP1679 Seismic reinforcement levels TOR fuel pool BK,
- PNPP1949 Installation of a fire protection screen between the PTR pumps for the physical separation of the two PTR lanes

Planned/Phase B (to be deployed):

- PNPE1128 On/Off Level Measurements of the reactor building pool,
- PNPE1258 Implementation of the ASG-ND system and fixed line for replenishing the fuel storage pool by SEG,
- PNPP1824 Adding an analog level measurement chain for the BK fuel pool

Document EIA-REPORT T3/4, D.3 (2026) provides easy-to-use lists of measures but no new information in respect to EIA-REPORT T3 / T4, D.1 (2026) and EIA-REPORT T3/4, D.2 (2026). The documents EIA-REPORT T3/4, 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 T3/4, D.5 (2026) provides relevant snippets from the French Environmental Code in the context of a periodic safety review.

5.2 Discussion

Generic aspects

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

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

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

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

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

Information Gaps: the report omits the initial and final mass flow rates (e.g., in kg/s) 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. A SGTR constitutes a containment bypass scenario which typically leads to a transient increase in SG pressure. While the valve opening is intended to relieve pressure, an excessively large discharge capacity could intensify the uncontrolled release of primary coolant (contaminated with radioactive material) to the atmosphere, thus challenging the integrity of the release mitigation strategy.

Accidents-3: Reduced Primary System I-131 Limit

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

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

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

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

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

Pool 2: Mobile Cooling Capabilities

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

Further measures

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

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

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

Site-specific aspects

Pending Hard-Core top-up booster (unit 4): The creation of a Hard-Core booster water source is noted as “not deployed” on Tricastin 4 while it is listed for Tricastin 3 as implementation planned for phase B. It is not mentioned why it is not deployed.

5.3 Conclusions

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

1. Enhance Transparency and Provide Clarity on Key Quantitative Data
   - Quantitative Data: The reports should provide the initial and final mass flow rates for the GCT-a valve flow capacity upgrade (PNPE1141), along with a comparison to the nominal operational flow. This is necessary to quantify the safety benefit.
   - Adverse Effects Analysis: The analysis of the uprated GCT-a capacity should be expanded to quantify the risk of increased radioactive release during a Containment Bypass scenario like a Steam Generator Tube Rupture (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
   - 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

- Resolve Discrepancies and improve transparency: The reports present a mix of implemented and pending measures, with some items completed (e.g., PNPE1191 cable-axis reinforcement on unit 3) and others scheduled via specific programs with stated target dates (e.g., PNPE1323 BAN-chimney reinforcement to be deployed no later than end-2025), alongside ongoing monitoring/modelling enhancements to support future updates. To strengthen transparency and closure tracking, it would be useful to systematically provide unit-specific completion dates for each outstanding item marked “to be deployed” or “specific program,” and to concisely document the interim protections and verification in place until final resolution. Additionally, the report sometimes states facts without further explanation why a specific measure was not deployed (e.g. hard-core water source booster PNPE 1289 for unit 4).

6 SAFETY ASPECTS OF CORE MELT ACCIDENTS

6.1 Treatment in the EIA documents

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

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

Stabilization and Cooling of the Corium

In the event of a core melt accident, the corium spreads after breaking through the reactor pressure vessel (RPV) in the reactor building vessel well and in the room of the reactor core instrumentation (RIC room). To limit the risk of losing the containment integrity due to erosion of the basement, a device is used that is based on stabilizing the corium underwater after it has spread in the dry (PNPP1976)[13]. According to EDF, this solution is similar in principle to that used in EPR (core catcher). This arrangement complies with regulation [AG-A-I].

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

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

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

The reactors at the Tricastin NPP are not affected by regulation [AG-A-II] on thickening the foundations, as their foundations are made of 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 (PNPE1460)[14].

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

The measurement for detecting a vessel penetration (PNXX1746)[15] makes it possible to ensure water injection onto the corium at the most effective time. The cooling of the corium and the long-term removal of residual power are ensured by the EAS-ND and hard-core cooling source (SF-ND) measures.

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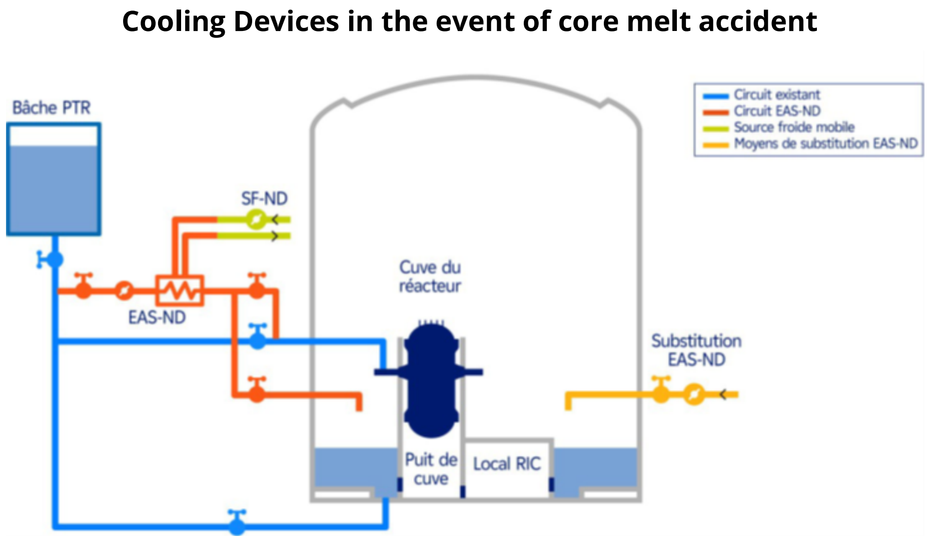
[14] PNPE1460: "Reinforcement of the walls between the internal instrumentation room of the reactor core (RIC) and the sump area at the bottom of the containment" will be implemented for unit 3 as part of the Supplementary Phase and for unit 4 of the Phase B of the 4th PSR.

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

Figure 5: Cooling devices in the event of core melt accident (EIA-REPORT D.1 2026)



Source: Environment Agency Austria

EDF will implement an additional measure that, in the event of a medium- to long-term failure of the EAS-ND, allows water to be replenished using mobile means for a sufficient period of time to limit erosion of the basement (PNPE1362)[16]. This measure complies with regulation [AG-B-III]. This replenishment is controlled by measuring the water level at the bottom of the reactor building (PNPE1386)[17].

In addition, following the investigation by the Permanent Group of Experts on Reactors (GPR), special instrumentation to detect the spread of corium over the entire area of the RIC room (PNPE1387)[18] will be implemented.

According to the EIA-REPORT T3/4, 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 the corium (PNPP1976)[19], along with the EAS-ND system

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[16] PNPE1362: “Installation of fixed injection and extraction lines in the reactor building and mobile replacement device for EAS-ND – return of water from the fuel pool to the reactor building” will be implemented for unit 3 as part of the Phase B and for unit 4 as part of the Supplementary Phase of the 4th PSR.

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

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

[19] PNPP1976: see above

(PNPP1811)[20] and the SF-ND, which in particular enables long-term cooling of the corium, the probability of a breakthrough of the basement is reduced to a frequency of 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 (PNPP1811)[21] as part of the 4th PSR also enables the residual heat to be dissipated from the containment. The EAS-ND arrangement is dimensioned in such a way that situations involving core melt accident, which would lead to the opening of the containment filter device, are avoided.

The “EAS-ND” arrangement comprises:

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

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

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

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

[21] PNPP1811: see above

The long-term management of core melt accidents is based on the circulation operation of the EAS-ND system to keep the corium submerged and remove residual power from the reactor. EDF is setting up a system to manage any leaks that may occur in the EAS-ND circuit (PNPP1541)[22] outside the containment building. In addition, EDF is installing a device to return the wastewater present in the collection tanks of the spent fuel building to the reactor building (PNPE1362)[23]. These devices for collecting and recirculating comply with the regulations [AG-B-IV] and [AG-D-I].

To reduce the potential radiological consequences, sodium tetraborate baskets will be implemented in the sump basins of the reactor building (PNPE1410)[24] at the latest for Tricastin 3 in March 2028 and for Tricastin 4 in Juni 2030 in accordance with regulation [CR-B]. The proposed arrangement consists of installing fixed devices in the floor of the reactor building that contain an alkali salt that dissolves in water and retains the iodine in the water, thus limiting its transition to the gas phase. The devices are passive and consist of baskets filled with disodium tetraborate decahydrate.

Containment System Functions and Equipment

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

To improve the resistance of certain components of the containment structure in the event of a core melt accident, EDF has replaced the viewing windows of the reactor building (BR) airlocks (PNPP1631)[25] and specific system cables (PNPE1264)[26].

Other changes related to the qualification of equipment for core melt accidents will not be implemented until Phase B of the 4th PSR. This involves the replacement of electric actuators and control cabinets for the

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

[23] PNPE1362: see above

[24] PNPE1410: “Installation of sodium tetraborate baskets in the sump basins of the reactor building” will be implemented during a specific program at the latest in March 2028 for Tricastin 3 and at the latest on June 18, 2030, for Tricastin 4.

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

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

EAS and PTR systems (PNPE1347)[27], the replacement of certain seals (PNRL1896)[28] and the capture of leaks at the shut-off valve of the EAS collection basin (PNPP1541G)[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 (PNPP1870)[30]. In accordance with regulation [AG-C-II], the U5-System will be further reinforced to ensure its resistance to earthquakes of magnitude SMS (PNPE1377)[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 (PNPE1362)[32] and (PNPE1449)[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.

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

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

[29] PNPP1541G: “Capturing leaks at the shut-off valve of the EAS collection shaft” will be performed as part of the Phase B of the 4th PSR for unit 3. This measure is not mentioned for unit 4.

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

[31] PNPE1377: “Reinforcement of the compression and filter device of the U5 container in the event of an SMS earthquake” will be implemented as part of the specific program at the latest in March 2028 for Tricastin 3 and at the latest on June 18, 2030, for Tricastin 4.

[32] PNPE1362: see above

[33] PNPE1449 “Study of a mobile water treatment module for treating contaminated water” will be performed as part of the Phase B of the 4th PSR for unit 3 and in accordance with [AG-DII] in June 2030 for unit 4.

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 Tricastin 3 and 4.

It is state of the art to use the WENRA "Safety Goals for New Power Reactors" as a reference for identifying meaningful safety improvements during an LTO project. (WENRA 2013) According to the WENRA safety objectives, core melt accidents that would lead to early or large releases should be practically eliminated. The occurrence of certain severe accidents can be considered to be practically eliminated "if it is physically impossible for the conditions to occur, or if it can be assumed with high confidence that the occurrence of these conditions is extremely unlikely". The concept of "extremely unlikely with high confidence" is an essential part of the IAEA's concept of "practical elimination". Although this concept applies only to new reactors, it should also be applied to Tricastin 3 and 4 in order to reduce the existing risks. Especially since the goal of the 4th PSR is to approach the safety level of the new EPR in Flamanville. The EIA 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 Tricastin 3 and 4, 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 are not presented. The limitation of the spreading area due to building constraints impedes the realization of the new device. From the point of view of the current knowledge, a failure of the containment function cannot be excluded after implementation of the modification for the stabilization of the core melt.

Furthermore, there is a risk of lateral failure of the walls of the RIC room. ASNR therefore considers the strength of the walls to be insufficient and calls for reinforcement. (see [AG-A III]) The walls to the RIC room have not yet been reinforced, although this is necessary to avoid the risk of the corium breaking through. This will be only implemented as part of the Supplementary Phase (unit 3) or Phase B (unit 4) respectively (PNPE1460).

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

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

Removal of residual heat without filtered venting

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

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

In ASNR's view, numerous additional components and measures beyond those previously planned by EDF are necessary to ensure that the residual heat removal system functions effectively in the long term. However, these important modifications are only to be carried out in 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 (unit 4) and Phase B (unit 3) respectively (PNPE1362).

Containment System Functions and Equipment

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

Reinforcement of the U5-System

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

The backfitting of the U5-System with regard to its lack of resistance to an extreme earthquake has not yet been carried out, although this safety deficit was already identified during the EU stress tests. Upgrade measure (PNPE1377) will be only implemented as part of the specific program at the latest in March 2028 for Tricastin 3 and at the latest on June 18, 2030, for Tricastin 4. It is noteworthy that this upgrade is coming nearly 20 years after this vulnerability became apparent following the Fukushima accident.

Management of contaminated water

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

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

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

It is envisaged to investigate the use of a mobile water treatment module for treating contaminated water during the Phase B (unit 3) or in June 2030 (unit 4) (PNPE1449). Thus, it is not clear if this measure will be implemented at all.

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

Remarks of the OSART Mission to the Severe Accident Management Program

The international experts of the OSART mission in 2022 (see chapter 3) noticed an issue concerning the plant's severe accident management program for severe accidents taking place simultaneously in several units. To be prepared for an accident in several units was one of the important lessons learned from the Fukushima accident in 2011.

The IAEA team noted the following:

- Situations where the plant faces severe accidents simultaneously in several units are not fully considered in the baseline for severe accident management (SAM).
- The plant has not planned and carried out exercises that cover on-site emergency response organization training for severe accidents that may take place simultaneously in more than one unit.
- The SAM guidelines and training do not provide information on how to cope with severe accidents taking place simultaneously in several units.
- Assessments of the habitability and accessibility of local SAM actions do not cover situations where the plant faces severe accidents simultaneously in several units.

The IAEA team explained that not considering all aspects of severe accidents taking place simultaneously in several units, mitigation actions may not be performed in an effective manner during the emergency. Thus, the IAEA team gave the following suggestion: The plant should consider enhancing its severe accident management program to include all aspects of severe accidents taking place simultaneously in several units.

As mentioned before, the status of the suggestion and the severe accident program is unknown, as no information is available.

6.3 Conclusions

Severe accidents (SA) involving a core meltdown were not taken into account in the design of the French 900 MWe reactors. However, as a result of previous PSRs, facilities and measures for SA management have been implemented. The EU stress tests, already 15 years ago, have nevertheless revealed a number of shortcomings that have still not been addressed.

According to the ASNR, the objective of the 4th PSR for the 900 MWe reactors is to bring the safety level of the reactor closer to that of the EPR Flamanville 3, a third-generation reactor. In third-generation reactors, features to mitigate the effects of core melt accidents are already implemented in the design; this approach cannot be fully transferred to second-generation reactors such as Tricastin 3 and 4. The EIA documents do not contain a systematic comparison between the safety level of the 900 MWe reactors and the safety level of the EPR in order to identify the remaining gaps.

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

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

Not all necessary and planned modifications for heat removal without using the filtered pressure relief system in the event of a core melt accident have yet been fully implemented. Important components such as valves, seals and electric equipment that are required during a core melt accident, but whose resistance cannot be guaranteed during such an accident, will also only be replaced during Phase B or the Supplementary Phase. In addition, the reinforcement of the filtered pressure relief system (U5 system) against severe earthquakes has not yet been carried out. This means that today, a core melt accident with a major release of radioactive substances is still possible at Tricastin 3&4.

The EIA documents do not provide a complete overview of which of the planned modifications meet the ASNR requirements published at the end of the generic phase of the 4th PSR. Most of the measures are not scheduled to be implemented until the end of Phase B and the Supplementary Phase (2029 for unit 3 and 2031 for unit 4 respectively). The EIA documents do not indicate whether this schedule will be adhered to.

- The EIA documents should include an overview of which of the planned measures are to be used to meet the ASNR requirements published at the end of the generic phase of the 4th PSR and when they are to be implemented.

- Studies that prove the sufficient thickness of the containment basements and the dimension of the spreading areas for Tricastin 3&4 should be provided.
- It should be explained which options were examined to limit the spread of radioactive substances via soil and groundwater after a core melt accident in accordance with Regulation [AG-D-III]. How is it justified that there is no need for additional measures with regard to safety risks?
- A systematic comparison between the safety level of the Tricastin 3&4 and modern safety standards of the EPR Flamanville 3 should be included in order to identify the gaps.
- Information about the core damage frequency (CDF) and the large (early) release frequency (L(E)RF) before the 4th PSR, after the outage of the 4th PSR and after implementation of all modification of 4th PSR (including Phase B, Supplementary Phase and the specific program) should be provided.
- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Tricastin 3&4. The concept of practical elimination should be used in this approach. Especially since the goal of the 4th PSR is to move closer to the safety level of the EPR Flamanville 3.
- The status of the suggestions of the OSART mission (2022) concerning the issue of the severe accident management program for severe accidents taking place simultaneously in several units of the Tricastin NPP should be provided.
- The authorization for continued operation of Tricastin 3&4 should be issued only after the planned measures to mitigate the release in the event of a core melt accident have been fully implemented.

7 RADIOLOGICAL IMPACT OF ACCIDENTS / TRANSBOUNDARY EFFECTS

7.1 Treatment in the EIA documents

The assessment of transboundary impacts of accidents at Tricastin unit 3 and unit 4 is provided in Chapter 6 of *Document 3bis – Document on Environmental Effects Associated with Reactor Operation over the Next Ten Years*, prepared separately for each reactor unit (unit 3 and unit 4), with the reactor unit number indicated in the document title (EIA-REPORT T3, D.3b, 2026 and EIA-REPORT T4, 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 Tricastin unit 3 and unit 4. In contrast, during normal operation and in the case of a design-basis accident, cross-border radiological effects are assessed as negligible.

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

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

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

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

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

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

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

The identification of plausible cumulative accident scenarios at Tricastin unit 3 and unit 4, which were not considered in the original plant design, led to the development of supplementary safety measures and more than 30 additional improvements in the plant operation.

Main measures to mitigate radiological consequences following accidents without core melt (design-basis accidents), and beyond design-basis accidents that were implemented during the plant construction and complemented by additional measures implemented as a result of improvements in plant's safety were described further in Chapter 6 of relevant EIA report documents for unit 3 and unit 4. (EIA-REPORT T3/4, D.3b, 2026)

The EIA documents present the results of calculations demonstrating the potential impacts on public health in terms of projected doses for early (24 hours and 7 days) and late (50 years) phases of an emergency assuming no protective measures are implemented. The results are compared with the reference values for emergency exposure situations set in the French legislation. Although the report states that the assessment of radiological consequences is based on an "acceptably pessimistic" estimate of releases and on "realistic scenarios" that do not incorporate protective measures, it does not define the criteria for an acceptably pessimistic assessment nor provide a description or justification of the scenarios considered realistic.

The EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) claim that the assessment of radiological consequences includes calculations of the total effective dose for early and late phase and the thyroid equivalent dose for early phase of an emergency for the population at distances of 650 m (corresponding to distances of the nearest settlements which is 1 km for Tricastin) 2 km, 5 km, and 10 km. For beyond design-basis accidents, transboundary impacts are also assessed for distances of up to 1000 km. This includes the territory of Austria.

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

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

7.2 Discussion

Generic aspects

The EIA documents consider core melt accidents as the only type of events with potential for transboundary consequences. It assumes an accident sequence leading to at least partial melting of the reactor core. It is further assumed that, in addition to failure of the first physical barrier (fuel cladding), failure of the second barrier (reactor coolant pressure boundary, including the primary circuit) may also occur triggering complex physical processes that can further lead to failure of the third barrier and release of radioactivity into the environment. Although the assessment states that parameters leading to increased radioactive releases were used to ensure conservative, ‘worst-case’ outcomes, the underlying source term data are not provided. No radionuclide inventories, release fractions, or other essential parameters are included, and the document does not contain sufficient information to reproduce or independently verify the calculations. Similarly, the EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) provide no details on the atmospheric dispersion model used to estimate off-site consequences. The EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) indicate that mitigation measures intended to reduce the consequences of design-basis accidents were taken into account; however, it does not describe the assessment methodology needed to substantiate this claim or allow replication of the results.

Results for design-basis accidents indicate that projected population exposures at the nearest inhabited areas remain below French regulatory reference levels. The assessment recognizes that only core melt accidents have the potential to cause cross-border radiological impacts. The EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) evaluate the long-range transport of radioactive material within a 1,000-km radius under "worst-case" conditions, a distance that includes Austrian territory. Reported results expressed as effective dose for different age groups suggest that the lifetime dose to the Austrian population would not exceed 1 mSv (0.05 mSv).

The EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) state that long-distance atmospheric dispersion calculations used transfer coefficients derived from five years of meteorological data, accounting for topography, weather conditions (mainly wind), and deposition processes. It remains unclear whether simulations were performed continuously using daily meteorological input over five years, or whether only a limited number of calculations using average transport coefficients were conducted. Further, the assessment lacks information on the actual dispersion model or calculation method used.

The EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) also claim that in case of a beyond design-basis accident with core melt EU maximum levels of radionuclides in food would not be exceeded, but it does not present the methodology for calculation of the activity concentration in food, nor the calculation results to confirm this claim.

EIA-REPORT T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) do not contain information on levels of ground deposition or contamination. Austria has set level for ground deposition of Cs-137 which is 650 Bq/m^2. Values of ground deposition above this value will trigger the screening of food measures and agricultural protective measures according to the catalogue of measures (BMK 2022). While doses to population might be below reference levels, ground deposition of Cs-137 above 650 Bq/m^2 could have serious non-radiological consequences, such as psychological and economic consequences in the affected areas.

Site-specific aspects

As the EIA documents did not provide sufficient data to reproduce the calculations underlying the presented results, the expert team conducted dispersion modelling for two hypothetical large-scale release scenarios for Tricastin unit 3 and unit 4 to assess whether protective action thresholds in Austria could be exceeded under specific circumstances. Since both units are of identical design and located at the same site, the modelling produced identical results; therefore, only the results for unit 3 are presented. The aim of the assessment was to assess whether a severe accident at unit 3 or unit 4 could possibly cause a deposition on Austrian territory above 650 Bq/m^2, a value that triggers protective actions related to prevention of food contamination. Probability of a large-scale release was not assessed nor considered in this study on atmospheric dispersion following a severe accident.

The source terms, marked as release categories FK2 and FK3, used in the JRODOS dispersion modelling to assess the deposition on Austrian territory are referenced in publication "*Übersicht über Maßnahmen zur Verringerung der Strahlenexposition nach Ereignissen mit nicht unerheblichen radiologischen Auswirkungen (Maßnahmenkatalog)*", 2010, Table 7.2-7 (SSK 2010). The source terms for both release scenarios, expressed as cumulative release fractions, are derived from a reference core inventory representative of a 1000 MWe-class PWR. For application to Tricastin 3 & 4, the reference source term is scaled to reflect the characteristics of the French 900 MWe series reactors.

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

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

		Release category	
		FK2	FK3
Start (h)		1	2
Duration (h)		3	3
Release height (m)		10	10
Thermal energy (GJ/h)		15	1
Released fraction of core inventory	Kr-Xe	1,0	1,0
	I	$7{,}0\cdot10^{-3}$	$7{,}0\cdot10^{-3}$
	I2-Br	$4{,}0\cdot10^{-1}$	$1{,}5\cdot10^{-2}$
	Cs-Rb	$2{,}9\cdot10^{-1}$	$4{,}4\cdot10^{-2}$
	Te-Sb	$1{,}9\cdot10^{-1}$	$4{,}0\cdot10^{-2}$
	Ba-Sr	$3{,}2\cdot10^{-2}$	$4{,}9\cdot10^{-3}$
	Ru1)	$1{,}7\cdot10^{-2}$	$3{,}3\cdot10^{-3}$
	La2)	$2{,}6\cdot10^{-3}$	$5{,}2\cdot10^{-4}$

1) *“Ru” also applies to Rh, Co, Mo, Tc*

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

Ideally, atmospheric dispersion modelling for a specific type of accident with a release would be done with daily meteorological data for at least one year to understand transport and deposition of a radioactive plume in all meteorological conditions. As the goal of modelling in this study was only to confirm whether a deposition of Cs-137 above 650 Bq/m^2 from an accident in Tricastin 3 or 4 would be possible, a historical weather data that could support dispersion of the radioactive plume to Austria was used for the analysis.

Presented here are the results of the calculations which confirm the possibility of ground contamination in Austria from a release in Tricastin 3. For this task, meteorological conditions for the period 13 – 16 March 2026 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 Tricastin 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:....................Tricastin 3, France

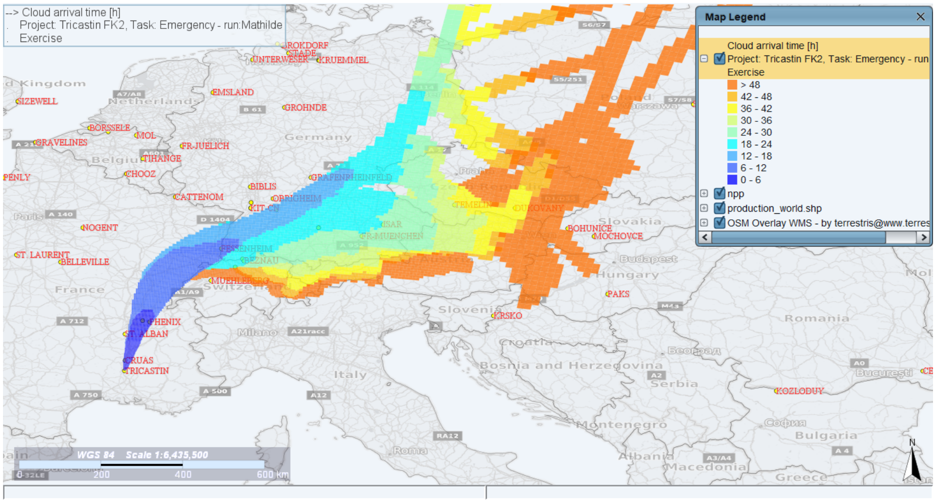
Release start:.............13 March 2026, 10:00 UTC

Prognosis duration: ..72 hours

Figure 6 presents information on cloud arrival time, indicating when the radioactive cloud is expected to reach the affected country. In both scenarios, with the release assumed to start on 13 March 2026 at 10:00 UTC, the cloud is projected to reach Austrian territory in less than 30 hours. Meteorological conditions are the dominant factor influencing cloud arrival time, and this result may vary significantly under different weather conditions.

Figure 6: Cloud arrival time for the release category FK2

Cloud arrival time for FK2

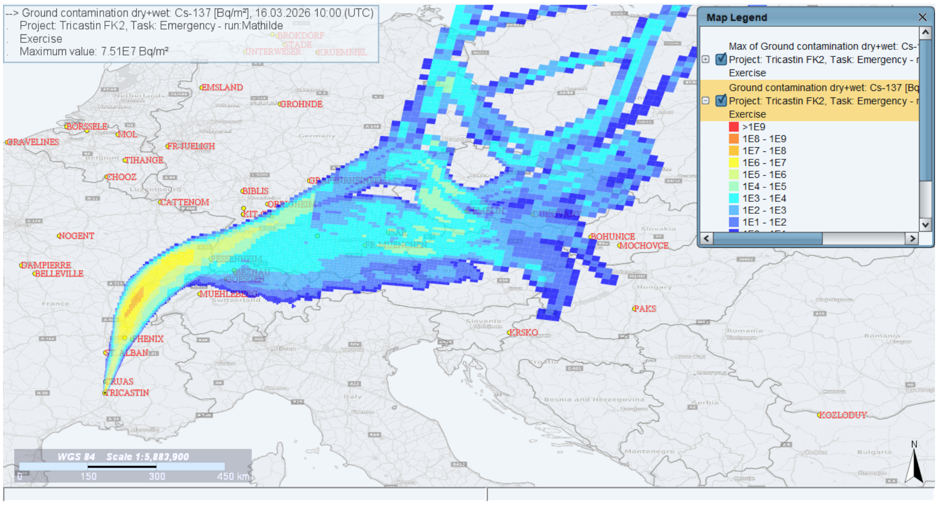


Source: Environment Agency Austria

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 expert team for release categories FK2 and FK3 are presented in Figure 7 and Figure 8.

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

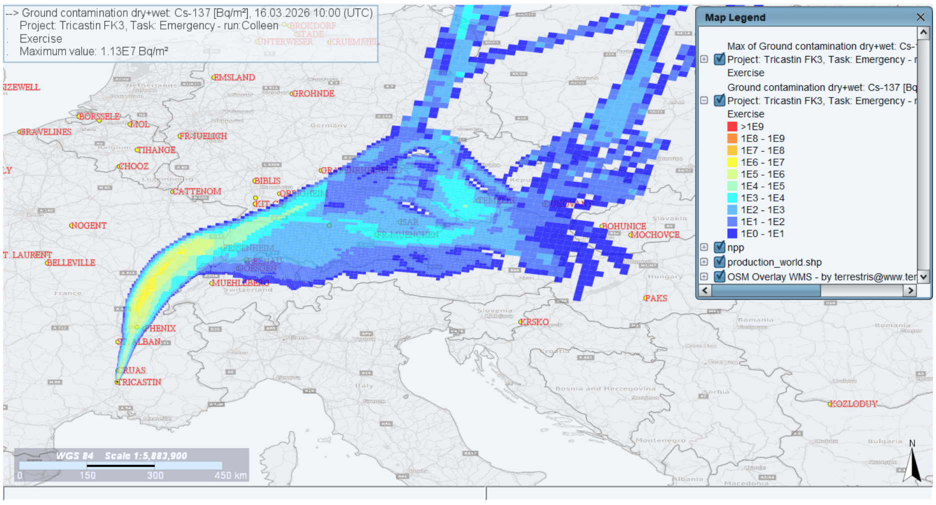
Ground contamination with CS-137 for FK2



Source: Environment Agency Austria

Figure 8: 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 1×10^4 Bq/m² for release category FK2 and 1×10^3 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 T3, D.3b (2026) and EIA-REPORT T4, D.3b (2026) do not present quantitative analyses to substantiate claims that food contamination would remain below EU limits at distances of more than 5 km after 7 days and within 1 km after one year. Additionally, the assessment omits information on ground deposition, despite its significance for evaluating long-term radiological impacts and potential contamination of the food chain.

Modelling of atmospheric dispersion and deposition conducted by the expert team demonstrate that, under certain meteorological conditions, a severe accident at unit 3 or unit 4 of Tricastin NPP 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 beyond those described in the EIA documents cannot be excluded.

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

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

8 ASSESSMENT OF THE TIME FRAME

8.1 Treatment in the EIA documents

The EIA documents emphasize the goals of the investigation undertaken with the generic PSR of the 900 MWe NPPs, which included unit 3 and unit 4 of Tricastin 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 Tricastin NPP unit 3 and unit 4. (EIA-REPORT T3/4, 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 (ASN 2021), Tricastin 3 and 4 have a period of six years following the release of the PSR report, to implement all safety measures identified.

For Tricastin 3 and 4, the implementation of the safety measures is organised in three phases. The Phase A measures are those that could be implemented during operations or within an outage related to the 4th PSR. Those measures have already been implemented at the time of the release of the EIA documents.

For Tricastin 3, certain set of measures are scheduled for implementation within Phase B, which is planned to be completed by December

2026. Measures that are not completed within Phase B (or its extension) are then to be completed within further phase, to be finalised by March 2029. This coincides with the "6 years after the release of the PSR report”, as required by the regulator.

For Tricastin 4, set of measures are scheduled for implementation within Phase B, which is planned to be completed by 2028. Measures that are not completed within Phase B (or its extension) are then to be completed within further phase, to be finalised by June 2031. This coincides with the "6 years after the release of the PSR report”, as required by the regulator

8.2 Discussion

It is important that the agreed implementation period (6 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 (6 years after the release of the PSR report = 2026/2029 for Tricastin 3 and 2028/2031 for Tricastin 4) is not uncommon. However, as the period following the 4th PSR corresponds with the start of long-term operation (LTO), some of the specific measures require special attention. It is important that the agreed implementation period is not extended. A lack of financial resources or the known problems with supply chain availability, including human resources, could affect the implementation period. It is particularly noteworthy that important safety modifications listed as part of the 4th PSR were already considered necessary as part of the EU stress test (2012), and their implementation had been agreed upon.

- Maintaining agreed schedule, or when possible, accelerating the safety improvements and LTO measures to be completed, where possible, even before 6 years deadline is strongly recommended.
- EDF should put the priority on the funding for the safety upgrade measures required in the 4th PSR and those related with the LTO, rather than on construction of a series of new EPR-2.
- Additional clarity of how the post Fukushima measures are being integrated with the measures that were decided on the basis of 4th PSR would be appreciated.

9 MANAGEMENT OF SPENT FUEL AND RADIOACTIVE WASTE

9.1 Treatment in the EIA documents

Spent fuel is in France not defined as radioactive waste but as material. After a period of storage in the fuel pools at the NPP sites, the spent fuel is transported to the reprocessing plant in La Hague operated by Orano. The resulting vitrified HLW is foreseen to be disposed of in the future deep geological repository Cigéo. (EIA-REPORT T3/4, D.3b 2026, p. 42)

In France, radioactive waste is categorized according to its radioactivity and its half-life. (EIA-REPORT T3/4, D.3b 2026, p. 42)

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

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

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

Table 4: Radioactive waste VLLW and LILW-SL from Tricastin NPP (EIA-REPORT T3/4, D.3b 2026, p. 43)

Radioactive waste	**Average volume in m³ per year (in period 2013-2022)**	**Prognosis of average volume in m³per year (in period 2026-2028)**
VLLW: solid, to be disposed of in CIRES	176	270
ILW-SL: solid, to be disposed of in CSA	187	300
LLW-SL: solid, to be disposed of in CSA	129	220
LLW-SL: solid for melting*	21	30
LLW-SL: solid for incineration*	590	612
LLW-SL: liquid for incineration*	0	300

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

The increase of the volume is explained by the upcoming refurbishment program. The types of radioactive waste and their management pathways are not expected to be changed significantly.

After conditioning, the yearly volume of LILW-SL is about 110 m^3. Between 1985 and 1995 the volume was reduced by optimizing of treatment and conditioning; before it was 360 m^3/year. In coming years, it could reach more than 110 m^3/yea^r due to the refurbishment. (EIA-REPORT T3/4, D.2 2026, chapter 2.2.4.4.1)

9.2 Discussion

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

The spent fuel represents the greatest risk for transboundary impacts, especially when significant amounts are stored centrally while awaiting final disposal.

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

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

ANDRA informs on its website[34] that the Cigéo inventory for which the licenses are being submitted does not include the so-called spare inventory waste. This spare inventory waste includes possible direct disposal of spent fuel in case of a change in France's reprocessing policy; furthermore, it includes spent fuel or radioactive waste resulting from lifetime extensions of the existing reactors beyond 50 years. A change in the inventory would require a new license. The period of the NPP lifetime extension could be beyond 50 years. In this case, proof of disposal needs to be provided for additional capacities for a deep geological repository. No information on this is available in the EIA documents.

Summarizing, the EIA documents do not provide sufficient information

- on the amount of spent fuel that results from 10, 20 or more years of lifetime extension,
- on the conclusion of a contract for reprocessing of the spent fuel from the lifetime extension,
- on the capacities of the spent fuel interim storage at La Hague,
- on a possible enlargement or addition to the capacity of Cigéo in case of NPP lifetime extensions over 50 years
- on the available capacity of the CSA repository for LILW-SL, especially when an increase of the average yearly volume of radioactive waste is expected due to refurbishment,
- on the timetable for the planned repository search for LLW-LL.

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[34] https://international.andra.fr/stepwise-development-cigeo-and-timeline-associated-decisions, seen 2.6.2026

9.3 Conclusions

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

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

10 LIST OF PRELIMINARY RECOMMENDATIONS

10.1 Long-term operation and operational experience

- 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.
- The status of the recommendations/suggestions of the OSART mission (2022) concerning the ageing management program and LTO at the Tricastin NPP should be provided.

10.2 External hazards

- With respect to seismic safety, the following information should be provided:
  - Methods, data and assumptions used for the PSHA performed to determine the SND for Tricastin 3 and 4, in particular, the types of seismic sources considered (source zones and/or fault sources), time coverage of the earthquake catalogue, minimum and maximum magnitudes, ground motion prediction equations, and site conditions.
  - The actual ground motion values for the SMS and the SND.

- The deterministically derived SMS ground motion should be benchmarked against a PSHA-derived ground motion for a recurrence interval of 10,000 years as required by WENRA (2021).
- Additional safety demonstrations should be required to ensure that all SSCs relevant to safety can cope with a probabilistically derived 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 Tricastin 3 and 4.
- The methods, data and assumptions used to determine the SND for Tricastin 3 and 4, in particular, the types of seismic sources considered (source zones and/or fault sources), time coverage of the earthquake catalogue, minimum and maximum magnitudes, ground motion prediction equations, and site conditions should be reviewed. The PSHA should be benchmarked against WENRA requirements (WENRA 2021) and recommendations (WENRA 2020a, b).
- Dedicated assessments of near-regional active faults and 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: EIA documents and 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 the Tricastin 3 and 4 against design events, i.e., events with recurrence intervals of 10,000 years or less (e.g., earthquakes). This is to ensure the independence of Defence-in-Depth (DiD) levels 3 and 4.
- It should be evaluated if the long timeframe for implementing a research program to identify, assess and parametrize active faults in the near-region and region around the Tricastin site 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 a “prioritized pilot study” following the Le Teil 2019 earthquake, i.e., 7 years after the event, appears unreasonably long. The typical

timeframe for completing a paleoseismological assessment of an active fault is 1 or 2 years including all preparatory investigations.

- With respect to possible terror attacks, the following questions should be addressed:
  - Have any studies been or will be carried out on the threat posed by newer technologies, in particular potential attacks using civilian or military drones?
  - How is the result of the Nuclear Security Index 2023 (NTI 2025) for France assessed? Are improvements planned with regard to “security culture”, “cybersecurity” and “protection against insider threats”?

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

1. Enhance Transparency and Provide Clarity on Key Quantitative Data
   - Quantitative Data: The reports should provide the initial and final mass flow rates for the GCT-a valve flow capacity upgrade (PNPE1141), along with a comparison to the nominal operational flow. This is necessary to quantify the safety benefit.
   - Adverse Effects Analysis: The analysis of the uprated GCT-a capacity should be expanded to quantify the risk of increased radioactive release during a Containment Bypass scenario like a Steam Generator Tube Rupture (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
   - 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
    - Resolve Discrepancies and improve transparency: The reports present a mix of implemented and pending measures, with some items completed (e.g., PNPE1191 cable-axis reinforcement on unit 3) and others scheduled via specific programs with stated target dates (e.g., PNPE1323 BAN-chimney reinforcement to be deployed no later than end-2025), alongside ongoing monitoring/modelling enhancements to support future updates. To strengthen transparency and closure tracking, it would be useful to systematically provide unit-specific completion dates for each outstanding item marked “to be deployed” or “specific program,” and to concisely document the interim protections and verification in place until final resolution. Additionally, the report sometimes states facts without further explanation why a specific measure was not deployed (e.g. hard-core water source booster PNPE 1289 for unit 4).

10.4 Safety aspects of core melt accidents

- The EIA documents should include an overview of which of the planned measures are to be used to meet the ASNR requirements published at the end of the generic phase of the 4th PSR and when they are to be implemented.
- Studies that prove the sufficient thickness of the containment basements and the dimension of the spreading areas for Tricastin 3&4 should be provided.
- It should be explained which options were examined to limit the spread of radioactive substances via soil and groundwater after a core melt accident in accordance with Regulation [AG-D-III]. How is it justified that there is no need for additional measures with regard to safety risks?
- A systematic comparison between the safety level of the Tricastin 3&4 and modern safety standards of the EPR Flamanville 3 should be included in order to identify the gaps.
- Information about the core damage frequency (CDF) and the large (early) release frequency (L(E)RF) before the 4th PSR, after the outage of the 4th PSR and after implementation of all modification of

4th PSR (including Phase B, Supplementary Phase and the specific program) should be provided.

- The WENRA Safety Objectives for new NPP should be used to identify reasonably practicable safety improvements for Tricastin 3&4. The concept of practical elimination should be used in this approach. Especially since the goal of the 4th PSR is to move closer to the safety level of the EPR Flamanville 3.
- The status of the suggestions of the OSART mission (2022) concerning the issue of the severe accident management program for severe accidents taking place simultaneously in several units of the Tricastin NPP should be provided.
- The authorization for continued operation of Tricastin 3&4 should be issued only after the planned measures to mitigate the release in the event of a core melt accident have been fully implemented.

10.5 Radiological impact of accidents / Transboundary Effects

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

10.6 Assessment of the time frame

- Maintaining agreed schedule, or when possible, accelerating the safety improvements and LTO measures to be completed, where possible, even before 6 years deadline is strongly recommended.
- EDF should put the priority on the funding for the safety upgrade measures required in the 4th PSR and those related with the LTO, rather than on construction of a series of new EPR-2.
- Additional clarity of how the post Fukushima measures are being integrated with the measures that were decided on the basis of 4th PSR would be appreciated.

10.7 Radioactive Waste Management

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

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12 LIST OF FIGURES AND TABLES

List of Figures

List of Tables

13 GLOSSARY

AM	Ageing Management
ASG	Steam Generator Auxiliary Feedwater System
ASN	French Authority for Nuclear Safety
ASNR	French Authority for Nuclear Safety and Radiation Protection
BAN	Buildings for Nuclear Auxiliary Facilities
BK	Spent Fuel Building
BL	Electrical Equipment Building
Bq	Becquerel
CDF	Core Damage Frequency
CEPP	Primary Pump Sealing Circuit
CHF	Critical Heat Flux
CIRES	Repository for VLLW
Cs-137	Caesium-137
CSA	Repository for LILW-SL
DAPE	Dossier of Suitability for Continued Operation
DBA	Design Basis Accidents
DBE	Design Basis Earthquake
DEC	Design Extension Conditions
DID	Defence-in-Depth
EAS	Sprinkler System
EAS-ND	Hardened Safety Core Sprinkler System
EDF	Électricité de France
EDG	Emergency Diesel Generator
EIA	Environmental Impact Assessment
ENSREG	European Nuclear Safety Regulators Group
EPR	European Pressurized Reactors

EU	European Union
FARN	Force d'Action Rapide Nucléaire = Nuclear *Rapid* Action Force
FK	Release Category
FLA3	Flamanville Unit 3
GCT-a	Main turbine bypass system with direct venting of steam produced by the steam generators to the atmosphere
GMPP	Reactor coolant pump (primary pump) motor-pump group
GPR	Permanent Group of Experts on Reactors
HCTINS	High Committee for Transparency and Information on Nuclear Safety
HSC	Hardened Safety Core
HLW	High Level Radioactive Waste
ILW	Intermediate Level Radioactive Waste
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
LILW	Low and Intermediate Level radioactive Waste
LL	Long-Lived
LLW	Low Level radioactive Waste
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
OSART	Operational Safety Review Team
PGA	Peak Ground Acceleration
PIC	Program for Complementary Investigations
PM	Preventive Maintenance
PNGMDR	French national plan for the management of radioactive material and waste
PSHA	Probabilistic Safety Hazard Assessment
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PTR	Tank of Water Treatment and Cooling System of Pools
PTR bis	Mobile auxiliary cooling system for spent fuel pools
PWR	Pressurized Water Reactor
REA	Boron and Water Storage Tank
RIA	Reactivity Initiating Accidents
RIC	Reactor Core Instrumentation
RCP	Reactor Coolant Pump
RCS	Reactor Cooling System
RCV	Chemical and Volume Control system

RCV-RIS	Interface between the RCV and RIS systems
RFS	Règle Fondamentale de Sûreté
RGE	General Operating Rules
RIS	Safety Injection system
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
SGZ	Gas storage
SIS	Safety Injection Systems
SL	Short-lived
SMHV	Maximal plausible historical earthquake (Séisme Majoré Historiquement Vraisemblable)
SMS	Safe Shutdown Earthquake, Maximum safety earthquake, equivalent to design basis earthquake (Séisme Majoré de Sécurité)
SND	Séisme Noyau Dur – Seismic level for the hardened safety core
SOTA	State of the Art
SRL	Safety Refence Level
SSCs	Structures, Systems and Components
SSD	Reference Spektrum
TPR	Topical Peer Review

TOR	discrete (tout ou rien – TOR) as opposed to analogue used for measurement values and levels
TOR fuel pool	discrete indicators or alarm conditions for the fuel pool TOR
TTS	Target Technical Specifications
UHS	Ultimate Heat Sink
VD4	Quatrième Visite Décennale (Fourth ten-year inspection/outage)
VD5	Fifth ten-year inspection/outage
VLLW	Very Low-Level radioactive Waste
Vs	s-wave velocity of the top soil
WENRA	Western European Nuclear Regulators' Association

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