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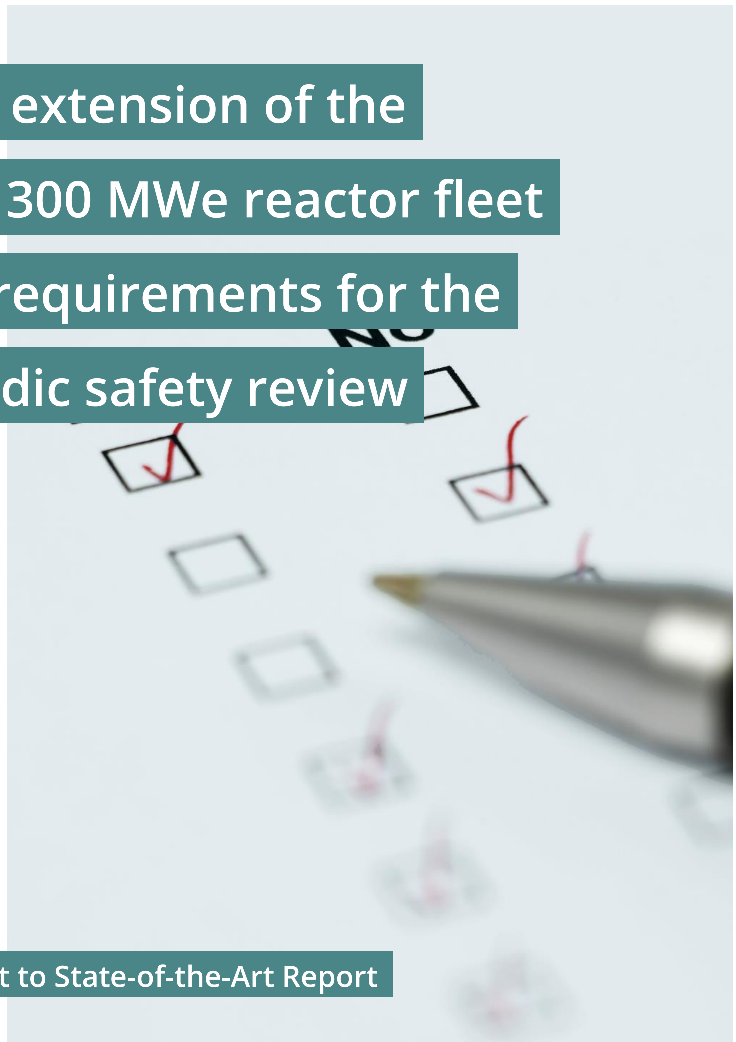

Task 4: Retrofit to State-of-the-Art Report

Federal Ministry
Republic of Austria
Climate Action, Environment,
Energy, Mobility,
Innovation and Technology

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Task 4: Retrofit to State-of-the-Art Report

REPORT
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SUMMARY

Twenty nuclear reactors of 1300 MWe installed capacity in France are now approaching forty years of operation, the end of their design life. The operator EDF intends to extend the lifetime of those plants. In France, once the design life-time of 40 years is reached, and the utility plans extending operation of a nuclear power plant (NPP) beyond its design lifetime, a comprehensive reassessment of the status of the plant is needed within the fourth periodical safety review (PSR4).

The French High Committee for Transparency and Information on Nuclear Safety (HCTISN) is organizing a public consultation process with the possibility to provide opinions on the generic phase of the PSR4, which covers topics relevant to all the 1300 MWe reactors. In case of a severe accident in a French NPP, significant impacts on Austria cannot be excluded. Therefore, Austria is participating in this consultation. For this participation, four task reports and a synthesis report have been prepared. The report at hand is task report no.4 focusing on retrofitting to the state of the art.

One safety objective of the 4th Periodic Safety Review (PSR4) of the French 1300 MWe reactor fleet (P4 and P'4 reactors) is to bring those reactors towards the state of the art as close as possible. The reference hereof is the EPR reactor in Flamanville (Flamanville-3), for which fuel loading has recently begun. Especially after the Fukushima accident, significant modifications have been made to increase the resilience of the reactors against natural hazards and sever accidents.

The P4 and P'4 reactors are pressurized water reactors that follow an n+1 safety concept, meaning that if one safety system fails, the other can still fulfil the safety function. The EPR, in contrast, has four trains of each safety system (n+3). Even if three trains fail out of various reasons, the last train can fulfil the full safety functionality. An aircraft crash–resistant shell covers the reactor building, the fuel building and two buildings - each housing two engineered safety trains. Provision for the prevention of severe accident conditions were already integrated from the design stage.

When comparing different aspects that affect the safety level of the P4/P'4 and the EPR reactors, the following points stand out:

- The EPR has a system for the stabilization of the core melt (corium) to prevent failure of the containment in case of severe accidents. Retrofitting of a so-called core catcher is possible due to space limitations below the nuclear reactor core. EDF claims that the provisions implemented in the 1300 MWe reactor series are in principle similar, but they rely on cooling the corium from above by flooding. R&D efforts aiming to show that this is possible (Licht 2023) are still on-going and have not shown convincing results yet.
- The total number of independent emergency electricity generating sources present at the reactor is about the same for the 1300 MWe reactors as for

the EPR after retrofitting measures. However, for the 1300 MWe reactors only two systems fulfil the more rigorous requirements for safety systems while this holds true for the four systems of the EPR.

- Nuclear power plants are designed to have redundant and diverse safety systems to ensure that essential safety functions are performed. The safety system of the 1300 MWe reactors consists of two trains. If one of those trains fails, the other train can fulfil the necessary safety functions (single failure criterion). The EPR safety systems consist of four trains. Backfitting the 1300 MWe reactors does not seem to address the redundancy and diversity of level 3 safety systems that handle design basis accidents.
- The Nuclear rapid intervention force (FARN) is located at different nuclear power plants sites all over France. It is supposed to be on-site in case of emergency within 24 hours and operational within 36 hours. FARN equipment, as mobile equipment in general, must not be credited in the safety assessment for design basis accidents and therefore is not part of the safety systems for design basis accidents on safety level 3. This makes it immediately clear that the FARN is acting on safety level 4, which is put in operation during design extension conditions to prevent and mitigate severe accidents. Systems relevant for safety should not be used to compensate for existing deficits at level 3. Further, FARN would also support in case of need at EPR sites and is thus not a measure decreasing the difference in safety levels between the 1300 MWe reactor fleet and the EPR.
- For the safety analysis of a nuclear power plant, not only natural but also events that are induced by human activities are considered. This includes the crash of an aircraft. The probabilistic analysis of the French 1300 MWe reactor fleet has been updated recently to also include large commercial aircraft. For the EPR, a deterministic approach was used and explicitly the case of an intended attack of an aircraft on a nuclear power plant was considered.
- The hardened safety core increases the resilience of the 1300 MWe reactor fleet against external hazards such as earthquakes and flooding. The design of the EPR against earthquakes and flooding is based on much more stringent requirements.

Based on the information available it can be stated that it is not possible to increase the safety level of the 1300 MWe reactor fleet to state-of-the-art. It should be emphasized, though, that significant improvements have been made by retrofitting compared to the original design.

1 INTRODUCTION

EDF intends to prolong the operational lifespan of its 1300 MWe fleet, consisting of 20 reactors across eight sites, beyond 40 years. This initiative necessitates proof of safety surpassing standard periodic safety review (PSR) requirements. In this context, EDF has proposed enhancing the safety standards of the 1300 MWe fleet to match those of third-generation reactors through retrofitting, with the EPR Flamanville 3 serving as the benchmark reactor. This objective, which is in accordance with the Nuclear Safety Directive (EURATOM 2014), has received provisional approval from the French regulatory authority ASN. Various retrofitting measures have been suggested, some of which have already been put into effect. This report assesses the safety systems and principles of both the 1300 MWe reactors and the EPR. Subsequently, it evaluates the suitability of different retrofitting approaches in elevating the safety standards of the 1300 MWe reactors to those of the EPR.

2 GENERAL PLANT DESIGN

In the following chapter, we describe the general plant design of the 1300 MWe P4 and P’4 reactors. Hereby, we set a focus on normal operation, safety systems and systems related to safety. Safety systems are designed to deal with design base accidents and must fulfill vigorous requirements regarding, e.g., redundancy and diversity. System related to safety are all possibly systems that might help during design extension conditions.

2.1 1300 MWe P4 and P’4

The P4 series PWRs originated from a Westinghouse design, building upon the 900 MWe "three-loop" CP(X) series. Eight P4 reactors are located at Paluel, Flamanville, and Saint-Alban. The P’4 series, consisting of 12 reactors, is situated across Belleville-sur-Loire, Cattenom, Golfech, Nogent-sur-Seine, and Penly (EUROPEAN NUCLEAR SAFETY REGULATORS GROUP 2012). While both series share a 1300 MWe electrical output and 3817 MWth thermal output, the P’4 design incorporates several modifications (Pistner et al. 2018).

Both P4 and P’4 utilize a double wall containment structure composed of prestressed concrete for the inner wall and reinforced concrete for the outer wall. Negative pressure is maintained within the inter-wall cavity for activity confinement. The P’4 design aims at a more compact layout for example by reducing the containment volume from 98,000 m³ to 83,700 m³. Further, the P'4 series incorporates distinct geometries for the turbogenerators and changes to the connections of the safety injection systems compared to the P4 design (Pistner et al. 2018).

Several P4/ P’4 reactors at one site operate independently as single units. Each reactor possesses dedicated safety and auxiliary systems housed in separate structures (Pistner et al. 2018).

2.1.1 Normal operation

The P4 and P’4 reactor designs utilize a four-loop PWR configuration. Each loop comprises a primary coolant pump and a steam generator (SG). The primary circuit water is heated to an inlet temperature of 285 °C and core outlet temperature 320°C and maintained at a pressure of 155 bar by the pressurizer (ASN 2009). Heat from the primary circuit is transferred to the secondary system via the four steam generators. Beside the main grid connection, there are several reserve grid connections (Pistner et al. 2018).

2.1.2 Safety systems

The safety injection system (SIS) is responsible for injecting coolant into the primary circuit during leak or pipe break scenarios. The SIS comprises two high-pressure injection pumps (ISMP) connected to the coolant storage tank (PTR) via dedicated trains. They use the same pipe for suction and can deliver coolant at high pressure when needed (Mertins 2021 p. 90). Additionally, motor-driven pumps can be utilized in case of ISMP failure, injecting borated water from the PTR into the primary circuit. The system also has four hydro accumulators that can feed cold water into the primary system. For low-pressure coolant injection during primary circuit depressurization events, two low-pressure feed pumps (ISBP), also part of the SIS, are employed (Pistner et al. 2018).

Following an n+1 safety concept, P4 and P'4 reactors have two emergency diesel generators each. If one fails, the other can deliver the full functionality.

In the event of pressure and heat buildup within the containment structure, a double-train containment spray system (EAS) is activated. This system utilizes coolant from the PTR or the containment sump for heat removal. The extracted heat is subsequently dissipated via an intermediate cooling water system (RRI) and a secondary cooling system (SEC) (Pistner et al. 2018).

The Steam Generator Emergency Feedwater System (ASG/EFW) serves a dual purpose: providing startup and shutdown support and facilitating secondary side heat removal during accident conditions. This dual-train system incorporates a motor-driven feedwater pump and a steam-driven turbo feed pump within each train. Both trains obtain water from one dedicated feedwater tank (ASG tank), which can be replenished by a passive inflow from the demineralized water storage tanks (part of the water distribution system - SER) (Pistner et al. 2018). For the PTR cooling there does not exist a diverse cooling option.

2.1.3 Systems relevant for safety

A mobile feed pump ("motopompe thermique mobile") is available to deliver coolant from the PTR to the primary circuit when the reactor pressure vessel is open during refueling operations.

After 2011, two ultimate backup diesel generators (DUS) were retrofitted per reactor ensuring redundancy for powering safety-related equipment during an incident. The system is further equipped with an emergency turbo generator (LLS), which is seismic qualified. Additionally, there is one emergency combustion turbine (TAC) per site. The TAC is not hardened against earthquakes. It is important to note that the TAC is not intended to be used for powering the safety injection system (SIS) (Pistner et al. 2018, IRSN 2015b).

EDF presented modifications for alternate heat sinks (ASN 2020). Modifications to allow for mobile equipment delivered by the rapid intervention force FARN were implemented until 2014 (ASN 2015, 2017). Those modifications were, such as many others, introduced under the label Hardened Safety Core (HSC) which is explained in more detail in chapter 4.2.7.

2.2 EPR Flamanville

The EPR is a third-generation pressurized water reactor (PWR) designed by a consortium of Areva, EDF, and Siemens. It represents the most recent operational reactor design deployed in Europe. Currently, beside the Flamanville EPR reactor for which fuel loading started in 2024, three EPR units are connected to the electrical grid and generating power, while two additional units are under construction.

- Operational: Olkiluoto-3 (Finland), Taishan-1 and Taishan-2 (China)
- Under Construction: Hinkley Point C-1 and C-2 (United Kingdom)

The EPR has a thermal power rating of 4,500 MWth. The system is engineered for a 60-year operational lifespan and is designed to achieve a net electrical output of approximately 1,600 MWe (AREVA 2013, EDF 2006).

2.2.1 Normal operation

The primary coolant system consists of four independent loops, each pressurized to 155 bar and containing a primary pump, a steam generator, and associated piping. Heat from the reactor core is transferred to the secondary system via the steam generators, where steam is produced at a pressure of 78 bar.

The turbine generator set incorporates a unique design featuring a single-flow, high-pressure and intermediate-pressure element housed within a common casing, followed by three tandem, double-flow low-pressure elements (AREVA 2013, EDF 2006).

The condensate and feedwater system (CFS) manages the closed-loop circulation of water within the secondary system. This system retrieves condensate from the condenser and delivers it back to the SGs for steam generation. Key components of the CFS include low-pressure (LP) feedwater heaters, a deaerator/feedwater tank, main feedwater pumps, high-pressure (HP) feedwater heaters, main feedwater isolation and control valves, culminating at the SG main feedwater inlet nozzles (AREVA 2013).

Four independent mechanical draft cooling towers are employed as the ultimate heat sink (UHS) for the power plant.

2.2.2 Safety systems

Safety systems are implemented in the four Safeguard Buildings. The buildings are strictly separated into 4 trains. Trains 2 and 3 are protected by hardened concrete shell and trains 1 and 4 by spatial separation. All Safeguard Buildings are also protected against the safe shutdown earthquake and explosion pressure wave (AREVA 2013, EDF 2006).

The combined Safety Injection/Residual Heat Removal System (SIS/RHRS) fulfils both normal shutdown cooling and emergency coolant injection / recirculation duties. Four independent trains guarantee system redundancy (n+3 configuration). Each train injects borated water into the primary circuit using either a medium-head safety injection (MHSI) pump or a low-head safety injection (LHSI) pump. The systems take suctions from the In-containment Refueling Water Storage Tank (IRWST) (AREVA 2013). The Emergency Feedwater System (EFWS) provides emergency water supply to the steam generators (SGs) to maintain water level and remove heat after a loss of normal feedwater in case of anticipated events (design-basis events) (AREVA 2013).

The Component Cooling Water System (CCWS) transfers heat from safety and safety-related systems (SIS-RHRS, CCHS to the UHS via the Essential Service Water System (ESWS). The UHS consists of four independent mechanical draft cooling towers, each linked to their corresponding ESWS pumps (AREVA 2013).

Each of the two Emergency Power Generating Buildings (EPGB) houses two emergency diesel generators (EDGs). Further, electrical I&C systems, and heating, ventilation, and air conditioning (HVAC) equipment for supporting safety systems during power outages are located in the EPGBs (AREVA 2013).

2.2.3 Systems relevant for safety

The Instrumentation and Control system (I&C) includes various systems which can be sorted in systems for normal operation, systems which correct deviations from normal operation, systems which stabilize or lessen the impact of an accident and Post-Accident Management Systems (AREVA 2013).

The plant receives offsite power via five transformers connected to the switchyard. Two dedicated transformers provide safety-related power for the Emergency Power Supply System (EPSS). Each transformer can independently power two of the plant's four safety divisions under normal conditions. However, in case of failure, they can collectively power all four divisions. The remaining three transformers supply the non-safety buses of the Normal Power Supply System (NPSS), with two transformers having sufficient capacity to handle the entire load. Notably, both EPSS and NPSS employ a four-train configuration for redundancy (AREVA 2013).

The Spent Fuel Pool Cooling (FPC(P)S/PTR[1]) aims at dissipating residual heat from spent fuel assemblies stored within the pool. It consists of two separate and independent trains that are backed by separate emergency diesel generators. Additionally, the inventory and capacity of the spent fuel pool is sufficient to compensate for normal evaporation losses for up to seven days (AREVA 2013, EDF 2006).

In addition to the two emergency diesel generators (EDGs) per EPGB each EPGB houses one ultimate emergency diesel generator (AREVA 2013).

The Core Melt Stabilization System (CMSS) manages and The Core Melt Stabilization System (CMSS) manages and cools a molten reactor core in a severe accident scenario. Molten core material is directed to a designated core melt spreading area within the containment. Water from the IRWST is released to cool and stabilize the molten materials, mitigating the consequences of a core meltdown event (AREVA 2013).

This is complemented by the Containment Heat Removal System (CHRS). This dedicated two-train system (one passive and one active flooding line per train) controls conditions within the containment structure and the IRWST following a severe accident. CHRS shares similarities with the US EPR's SAHRS, which has a single-train configuration with double redundancy in both passive and active flooding lines (NEA 2023).

[1] Whenever there is a distinct English and French abbreviation, we name both with the English first.

3 METHODOLOGY

The goal of this task was to assess in how far the safety level of the French 1300 MWe reactor fleet could be brought to state of the art by retrofitting. EDF committed itself to bring the safety level of the 1300 MWe fleet as close as possible to that of the reference reactor EPR Flamanville 3 as (ASN 2023).

We started by listing the different safety systems and give as many details as possible on

- the design goals: what is the safety function of the system? What are limiting parameters such as pressure or capacity?
- the redundancy of the systems: what is the general safety concept? How many trains exist? How many trains are required for the system to fulfil its safety function? Was the system designed according to single failure criterion, single failure and repair case, or something different? Do we know something about limits and conditions of operation of the system?
- the logical and physical independence, and protection of the system: Are the trains of the system independent from each other or are there components of the system which are used by more than one train (e.g. there might be three HPIS trains, but all of them draw suction from the same tank). Are the trains physically separated from each other, or are all trains located in the same room/building? Against which (seismic) hazards are the trains protected? Are all trains designed against the DBE?

Based on our research, we decided to structure and discuss the following aspects regarding differences between the 1300 MWe reactors and the EPR in more depth:

1. Provisions against accidents involving core melt
2. Provisions against extended station blackout
3. Redundancy of the safety systems
4. Rapid Nuclear Intervention Force (FARN)
5. Availability of heat sink
6. Impacts of Aircraft
7. Hardened Safety Core (HSC) with explicit discussion of earthquake
8. Spent fuel pool

However, these are not exactly the points that were requested by the Umweltbundesamt. We shortly discuss the input by Umweltbundesamt in the following and give some indication why we decided to not cover some of them or under which of the above listed aspects we summarized the other ones:

1. Implementation of a Core Catcher is discussed under point 1, “provisions against accidents involving core melt”, because EDF is only talking about retrofitting of systems like a core catcher.
2. Modified inter-containment annulus ventilation system, with severe accident qualification is a modification completed during the third periodic

safety review and not relevant for assessing the retrofitting of the 1300 MWe reactor fleet to state-of-the-art safety levels.

3. Provisions for deployment of the mobile containment water treatment unit is discussed along other mobile measures provided by FARN (point 4).
4. Modified power distribution and instrumentation and control systems for HSC conditions is covered by point 2, “provisions against extended station blackout” and point 7, “Hardened Safety Core”.
5. Primary system makeup and residual heat removal in HSC conditions is covered by point 4, mobile measures by FARN, and point 5, availability of heat sink.
6. Risk prevention measures for explosions, fire, internal flooding and extreme heat: Measures taken to tackle those aspects includes several non-technical measures such as risk management and training of personnel which are not part of this report. Measures regarding earthquakes are discussed in point 7 (HSC).
7. Strengthened auxiliary feedwater system, makeup to AFW tank, and replenishment of reactor cavity and spent fuel pool is covered by point 5, availability of heat sink.

We derived our recommendations and questions from the discussion of the above-mentioned points.

4 COMPARISON OF SAFETY FEATURES OF 1300 MWE REACTORS AND EUROPEAN PRESSURIZED WATER REACTOR

The safety of a nuclear power plant must be demonstrated for both operational states and accident conditions. Accident conditions are further differentiated into design basis accidents (DBA) and Design Extension Conditions (DEC). According to IAEA definition, a DBA is a "postulated accident leading to accident conditions for which a facility is designed in accordance with established designs criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits." (IAEA 2016a p. 6). Typical DBAs are steam line breaks, feedwater line breaks and loss of coolant accidents.

DEC are conditions that go beyond the DBAs and are defined as "accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions." (IAEA 2016b). Commonly postulated DEC are, among others, the failure of the reactor trip system to shut down the reactor in case of need (Anticipated Transient Without Scram - ATWS) and the complete loss of all AC power from off-site sources and main generator and standby AC power sources on-site. Design extension conditions are further differentiated based on the expected fuel degradation (without / with core melt).

Systems that are used to cope with DEC must not fulfil the same acceptance criteria as safety systems, e.g., regarding redundancy and diversity. Since they are nevertheless important for the overall safety of the nuclear power plant, they are often dubbed safety-related systems or systems relevant for safety to mirror the difference between them and safety systems (IAEA 2016a p. 35f). Additionally, for safety-related systems it is sufficient to use realistic instead of conservative assumptions.

It is important to emphasize that DECs are considered a subset of beyond-design-basis accident (BDBA) conditions. The rationale for this is that BDBA conditions extend to include accidents that, due to their extremely low probability of occurrence, are considered to be "practically eliminated". It is important to note that DECs would not include conditions that are considered to be "practically eliminated". Practical elimination of a scenario can be shown either based on being physically impossible or extremely unlikely with a high degree of confidence. This probabilistic assessment should be backed by design improvement, deterministic assessment and engineering judgment.

Systems designed for controlling severe accident conditions such as core melt accidents are level 4 systems of the defence in depth concept and need to be independent of safety systems of lower levels, especially level 3. Provisions under level 4 should not be used to compensate for shortcomings in safety systems of level 3 (IAEA 2008, IRSN 2016).

4.1 Overview

Before discussing the individual points, the following section provides an overview of the characteristics of the various safety systems and systems relevant to safety for the two reactor concepts.

Engineered Safety Systems for the 1300 MWe reactor and the EPR are listed in Table 1. In Table 2, different measurements and systems for different groups of accidents are listed for the 1300 MWe reactor in France and the EPR.

Table 1: Engineered Safety Systems for the 1300 MWe reactor and the EPR, italics refers to backfit (source: ASN 2011, AREVA 2013, EDF 2006)).

Engineered Safety Systems		
	1300er backfit	EPR
	2x 100%	4x 100%, Safeguard buildings 1 and 4 physically separated Main steam and feedwater valve stations physically separated into two pairs EG, ESWS, UHS physically separated
Reactor Protection System		
Instrumentation and Control	Mainly analog Refitted 2012	Almost exclusively digital; exception: manually trip reactor via hardwired actuation path; diverse digital platforms; three levels;
(Emergency) Core Cooling System		
Residual Heat Removal	n+1	n+3
Accumulators	4	4
Medium pressure injection	ISMP – medium pressure injection system	4 Medium head safety injection pumps
Low pressure injection	Low pressure injection system	4 Low head safety injection pumps
Protection against loss of AC	2 EDG	4 EDG (4x100%)
Heat Sink	one Alternate heat sink	Four division related + independent mechanical draft cooling towers connected to ESWS, water storage basins associated with each cooling tower
Time before exposure of fuel	Heat sink can be restored before fuel exposure	9 days – heat sink can be restored before (only one unit at site!)

External Hazards		
Earthquake Nuclear Island	NRC spectrum normalized to 0.15g at zero period SSE + 50% + probabilistic definition with a return period of 20,000 years (HSC)	EUR spectrum normalized to 0.25 g at zero period
Earthquake site structure	0.1/.15 g (site specific)	EUR normalized to 0.2 g zero period
Aircraft	Probabilistic, crash frequency of small aircraft	Hardened concrete shell, incl. fuel building; Attack load case considered

Table 2: Measurements and systems for different groups of accidents are listed for the 1300 MWe reactor in France and the EPR (source: AREVA 2013, ASN 2011, ENSREG 2012, IRSN 2023c). Backfitting measures are listed in italics.

Severe Accident Mitigation Features		
Corium Stabilization	Dry-spreading followed by gravity cooling from above Long term cooling via EAS-ND	CMSS (Core Melt Stabilization System)
Emergency Cooling Measure		
	Mobile motor driven pump – one per site Injection arrangement at the joints of the primary pumps (IJPP-ND)	Containment Heat Removal System (CHRS)
Essential Service Water Systems (ESWS)	emergency water supply to the steam generators (ASG-ND)	(4x100%)+one non-safety related dedicated train
Protection against loss of AC		
	One combustion turbine (TAC) per site, not qualified against earthquake 2 ultimate emergency diesel generators	2 non-safety related DG in 2 separate buildings
Batteries for guarantee of continuous supply	2 hours	2 hours; + two “12 hour” batteries
	Mobile backup means to open pressurizer safety valves	

Heat Sink		
	Alternate ultimate water sources installed (wells, ponds) (SEU) Pumping system for SEU has no final decision PTRbis (mobile system for the PTR circuit) to be installed by FARN after at least 24 h	alternate heat sink: reversed sea discharge channel
Loss of off-site electrical power supply		
Fuel autonomy	3.5 days 14 days	Not clear
Oil autonomy	3 days 14 days	10 days
Initial cooling water reserves	2 weeks	Min 10 days high temperature, min 22 days low temperature
Autonomy of generators	3 days 2 weeks	4 days
Loss of off-site electrical power supply + conventional backup supply		
Residual power	Removed by natural circulation (prim.sys. Closed); backup turbine generator LLS + ultimate backup dg GUS PTRbis if reactor connected to spent fuel pool	Ultimate backup diesel, 2hr +12hr batteries
Spent fuel pool	No electricity supply for cooling system	Cooling system can be supplied by backup DG, fuel exposure after 2 days when reactor cavity is full
Fuel exposure w/o external intervention	After a few days	9 days

4.2 Discussion of selected safety features

In the following, we will discuss selected aspects of the P4/P'4 and EPR designs that are relevant for assessing the level of safety.

4.2.1 Provisions against accidents involving core melt

A core catcher is a safety feature designed for nuclear power plants to mitigate the consequences of a severe accident. In case core cooling is lost, the reactor core might lose its structural integrity, form a molten pool of nuclear fuel and core supporting materials called “corium“. In case the accident progression cannot be halted, the corium may relocate in the lower plenum of the reactor pressure vessel. Eventually the reactor pressure vessel will fail, and the corium may accumulate in the core catcher. The catcher is designed to contain and cool the molten material, preventing it from attacking the base mat of the containment structure (molten core concrete interaction MCCI). Without the core catcher, the corium could breach the containment and would be released into the environment.

For the EPR, the core catcher is called Core Melt Stabilization Systems (CMSS). The molten core is cooled by the Containment Heat Removal System (CHRS) which also controls containment pressure and ensures cooling of the in-containment water supply (IRWST).

There is no equivalent system in the French 1300 MWe reactor fleet. To prevent the risk of loss of confinement in the event of an accident with core melt, failure of the reactor pressure vessel and subsequent ejection of corium into the reactor cavity, a measure is taken that is supposed to stabilize the corium.

After spreading the corium in a dedicated spreading zone, the molten core is flooded from above. Figure 2 shows the configuration of the 1300MWe fleet corium retention system. Containment sump, reactor cavity and the RIC compartment are located at lowest elevation of the containment. In case of RPV failure and melt ejection into the reactor cavity, a channel between reactor cavity and RIC compartment promotes spreading of the corium. If the containment sump level is sufficiently high, connections from the sump to the reactor cavity and RIC compartment can be opened and facilitate passive flooding of the reactor cavity and cooling of corium from above. Reconfiguration of the SIS allows active injection of water into the sump from the PTR tank, in case the containment sump is empty.

EDF states that the implemented system for corium stabilization is, “in principle, [is] similar to that implemented on EPR (“core-catcher”) type reactors” (EDF 2023a p. 160). In its 2020 statement, IRSN does not address the overall feasibility of the systems but focuses on molten core concrete interaction IRSN 2020d. They assume that, based on the findings, the thickness of the structural concrete slab will be evaluated individually for each system and reinforced accordingly.

The effectiveness of the system, however, depends, among others, on the coolability of the corium by flooding it from above. In the event of RPV failure and melt ejection into the cavity, a pool of corium accumulates at the bottom of the reactor cavity. Cooling water is then provided from above and will cool the corium by evaporation. The steam is condensed by the containment spray system and accumulated again in the containment sump. The key question, however, is how effective the pool of corium can be cooled by pouring water onto it from above. The current state of knowledge is that a crust would form on top of the pool of corium, preventing further heat removal. Below the crust, MCCI would proceed and sooner or later lead to containment failure (ASN 2024a). Regarding MCCI, the effectiveness of corium stabilisation management has been only provided for certain types of concrete. Demonstration is still needed for the mainly very siliceous concrete used in the 1300 MWe reactors (ASN 2023).

One important aspect herein is whether a molten crust on top of the corium when in contact with the coolant is formed or not. To understand the implications of this phenomenon, R&D efforts are underway. Most prominent is the NEA ROSAU project which is supposed to end in June 2024 (NEA, 2024) with participants from B, CAN, CZ, F, JPN, KOR, SW and USA at Argonne National Laboratory. However, until 2023 only four experiments have been conducted while seven were still missing (Licht, 2023).

4.2.2 Provisions against extended station blackout

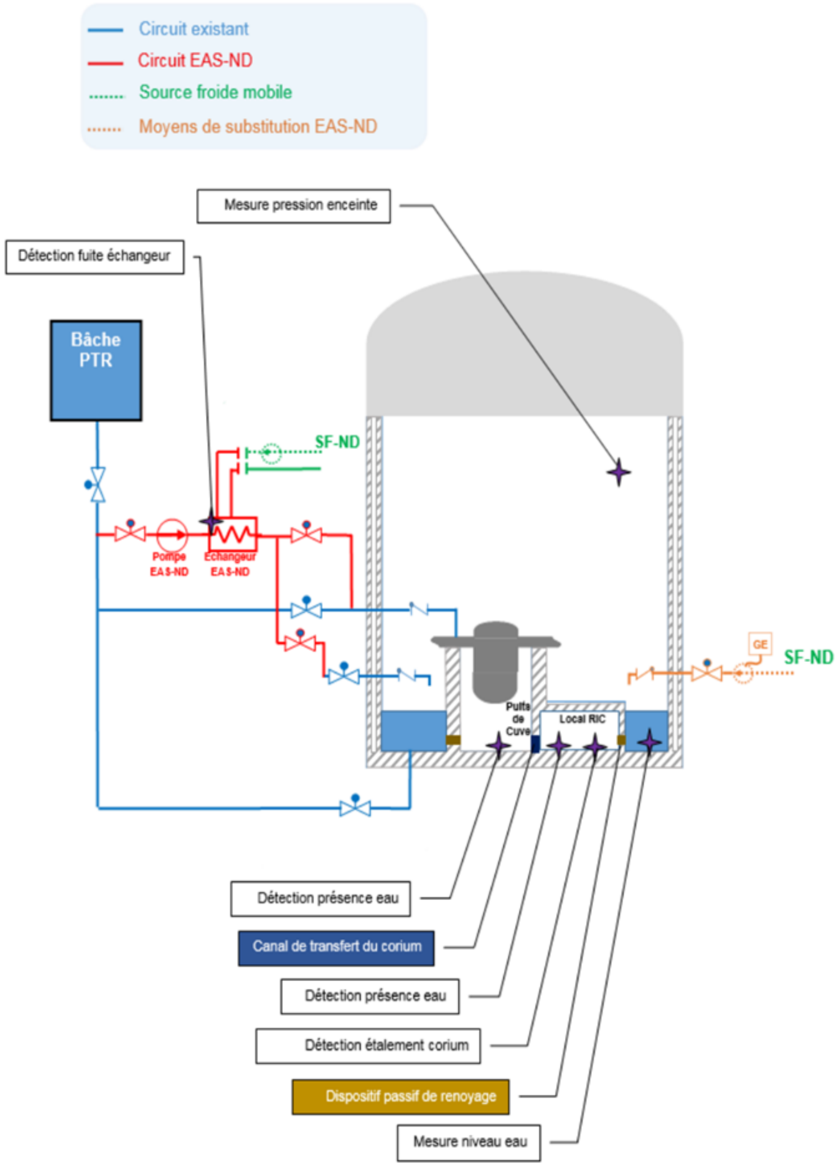
The safety systems of the French 1300 MWe reactor fleet follow a n+1 safety concept. Consequently, there are two emergency diesel generators per block. There exists, as systems relevant to safety, one steam-driven emergency turbo generator (LLS) per block and one additional emergency generator (TAC) per site. The LLS is not able to operate immediately after the failure of the emergency diesel generators. The TAC is not seismic qualified and not able to withstand a design basis earthquake. The high-pressure injection system cannot be used when electric power is supplied by TAC. After the Stress Tests following the Fukushima accident in 2011, the provision of ultimate backup diesel generators (DUS) which meet the hardened safety core requirements was requested by ASN and put in operation at all sites except Paluel until 2020 (ASN 2020 p. 17).

The EPR follows a n+3 safety concept. In addition to four emergency diesel generators, the EPR has two ultimate backup generators on site. Each division of the ESWS is driven by its own emergency diesel generator. If the emergency diesel generators fail, two trains (2x100%) can be rescued by the ultimate backup diesels (EDF 2006).

Regarding the autonomy of the on-site electrical power supplies, ASN considers the sites of the 1300 MWe reactor fleet to guarantee 3 days of autonomy for the generators sets. The statement of EDF that procurement is covered by contracts that guarantee deliveries within 24 hours is not sufficient to fulfil the requirement of the sites to be autonomous for two weeks under all circumstances. The EPR does not fulfil that goal either but can ensure autonomy for 4 days after loss of off-site electrical power supplies. Detailed numbers can be found in Table 2.

Figure 1: Measures to prevent containment breach through core melt (EDF 2023a).

Measures to prevent containment breach through core melt



Circuit existant - Existing circuit
Circuit EAS –ND - Emergency Auxiliary System circuit
Source froide mobile - Mobile cold source
Mesure pression enceinte - Containment pressure measurement
Detection fuite echangeur - Exchanger leak detection
Bache PTR - PTR tank
Detection presence eau - Water presence detection
Canal de transfer du corium - Corium transfer channel
Detection etalement corium - Corium spread detection
Dispositif passif de renoyage - Passive reflooding device
Mesure niveau eau - Water level measurement.

Source: EDF 2023

4.2.3 Redundancy

To provide multiple barriers against nuclear accidents safety systems are designed in a redundant and diverse manner. Redundancy means having multiple systems that perform the same task, e.g. cooling the reactor. If one system fails, another can take over and ensure the safety function is performed. To avoid common-cause failures, different types of systems should be used to achieve the same safety function. The P4/P'4 design pursues an n+1 concept, meaning that if one system fails, the other systems can fulfil the task (single failure criterion). The safety system of the EPR consists of four trains. With one train out of service due to maintenance, one failing to operate, and one affected by the initiating event that remaining train can still fulfil the safety functions (n+3).

Safety systems are the systems necessary to handle defence-in-depth level 3 accidents in nuclear power plants. According to the defence-in-depth-level concepts, the effectiveness of systems of the different levels should be independent. Several retrofitting measures for the 1300 MWe reactor fleet, e.g., emergency power supply or mobile measures, seem to fill deficits of level 3 systems while being level 4 systems.

4.2.4 The rapid response nuclear taskforce (FARN)

FARN is a team in charge of transporting equipment and people to assist plant personnel responding to an emergency with potential releases to the environment. Those specialized teams are "capable of relieving the shift crews and deploying emergency response resources in less than 24 hours, with operations starting on the site within 12 hours following their mobilisation" (ASN 2020 p. 38). Possible mobile means provided by the FARN are, e.g., diesel generators. FARN should also be able to provide the functions of the HSC in case of maintenance or failure.

The use of mobile ("non-permanent") equipment was introduced after the Fukushima accident to provide additional resilience against design extension events. That equipment, however, should not be regular means for coping with the short-term phase of DBA or DEC (IAEA 2024 p. 38).

FARN, by definition, is not a reactor-specific measure. FARN would – presumably - be deployed at an EPR reactor in case needed as well. It does not impact the difference between the 1300 MWe reactors and the EPR but increases overall safety of nuclear power plants that might be served in case of emergency.

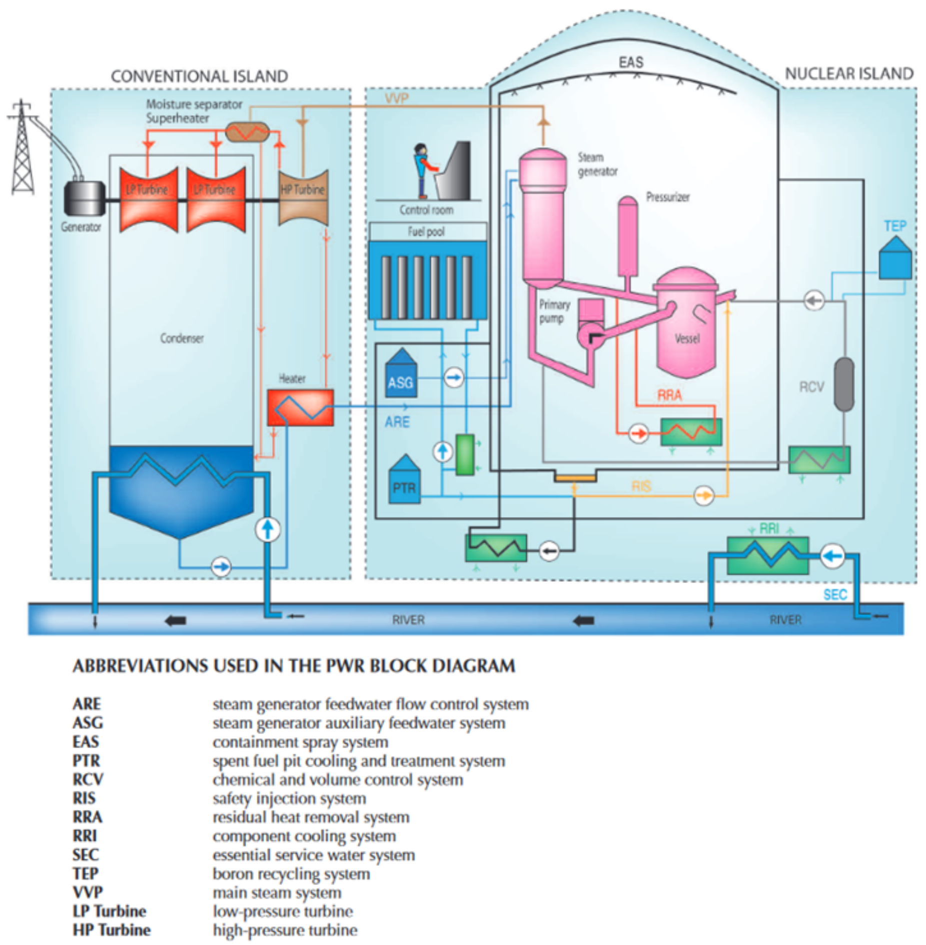
4.2.5 Availability of heat sink

The 1300 MWe reactor fleet lacks alternate heat sinks as per design. However, alternative measures such as wells and ponds have been implemented. At certain sites, the definitive design of the Ultimate Heat Sink (SEG) remains undetermined, depending on the chosen solution for ultimate water sources (SEU). For instance, EDF has yet to determine the inclusion of a filtration system at the gate station for sites utilizing basin water. Additionally, the design of the SEG pump required for specific sites remains unresolved. Currently, glandless motor pumps are kept on premises, but lack justification of design and operational monitoring methods, particularly concerning their robustness against seismic activity and particle loading in the SEU water (IRSN 2023c p. 5). The line-out of the cooling circuits of the nuclear power plant is shown in Figure 2.

The EPR has an alternate heat sink which comprises two independent systems, the ultimate enclosure cooling circuit (EVU) and the ultimate cooling circuit (SRU) which are made up of two redundant channels in the pumping station and can draw water from the main pumping station or from the outfall structure in the sea via a reversed sea discharged channel (ASN 2011 p. 87). The ultimate heat sink consists of four mechanical draft cooling towers.

Figure 2: Schematic illustration of the conventional and nuclear island (ASN 2011 p. 20).

Schematic illustration of the mobile cooling system PTZ bis



Source: ASN 2011 p. 20

The autonomy in case of loss of the primary (and at the time of design only) heat sink for the 1300 MWe fleet is at least 100 hours (ASN 2011 p. 87). For the EPR, the autonomy in case of the loss of the primary heat sink is 9 days with a potential cliff-edge effect after 2 days when switching from the EFWS to the safety-classified fire-fighting water production system (ASN 2011 p. 84).

4.2.6 External hazards (Aircraft)

French nuclear power plants are protected on a site-specific basis against the effects of small civil aircraft based on probabilistic analyses of the crash frequency of aircraft. Hereby, analysis was originally restricted to so-called "general aircrafts", meaning aircrafts with a mass up to 5.7 tons (FANC 2015).

For the 1300 MWe reactor fleet, probabilistic analyses were updated in 2023, including induced risks (direct and indirect effects) and risks associated with helicopters (EDF 2023a). EDF states that "the fall of an aircraft representative of general aviation on the most exposed wall of the combustible building does not entail a risk of mechanical degradation of the fuel assemblies, nor of loss of the water inventory of the pool which could lead to the melting of these assemblies" (EDF 2023a p. 127). This only addresses the small civil aircraft (aviation générale, mass less than 5.7 t). ASN states (2001) that French nuclear power plants are generally not designed to withstand other classes of aircrafts. This is considered current international practice. However, the possibility of terroristic acts and therefore impact of larger aircraft is acknowledged. Existing measures have been reinforced as part of the VIGIPIRATE plan, which is not disclosed to the public. In summary, no structural measures are taken to directly improve robustness against aircraft crashes.

Probabilistic studies were or are to be updated with current data for each site. EDF states: "The results of the studies carried out on the TTS sites at Paluel (P4) and Cattenom (P'4) show that the probability of an unacceptable release of radioactive substances at the boundary of the Paluel and Cattenom nuclear power plants due to air traffic is:

- less than 10-6 per year and per reactor for each of the 3 functions,
- at most of the order of 10-7 per year and per reactor for each of the 3 functions and per aviation family (general aviation, commercial aviation and military aviation)."

The three safety functions mentioned are: reactor shutdown and residual power removal, spent fuel storage and treatment of radioactive discharges.

For the EPR, adapted time-load functions were used in a deterministic approach, supported by risk considerations. According to the Safety Report for the EPR Flamanville, additional load cases introduced following the events of September 11, 2001, were considered and the initial design was adapted. It was confirmed that the EPR nuclear island and its general architecture could resist such an attack (EDF 2023a p. 326).

4.2.7 Hardened Safety Core (HSC)

The HSC encompasses a range of measures that are supposed to make the reactor more resistant against natural phenomena of an exceptional scale. Those phenomena might be combined and exceed events used for the design of the facilities. Examples are very long durations of loss of electrical power supply or cooling sources. The structures, systems, components, and equipment of the HSC should be qualified to be operable during design extension conditions. HSC measures were introduced after the stress tests following the Fukushima accident. They include, among others, new emergency control centres, an alternate water supply to spent fuel pool, emergency feed water tank and emergency core cooling system tank and an additional ultimate back-up diesel generator set for each reactor (ENSREG 2012). HSC components are not considered safety systems and must thus not fulfil diversity and redundancy requirements. However, the HSC is based on diversified systems, structures, and components (ASN 2017b p. 36).

Especially named external hazards covered by the HSC are earthquake, flooding, and “other natural hazards”. Special strategies for reactor driving under hardened core conditions are being developed (IRSN 2023c).

In the EPR design, a hardened concrete shell protects Safeguards building 2 and 3 (containing amongst others the main control room and remote shutdown station), the reactor building and the fuel building (housing fresh and spent fuel). The inner structure of the reactor building, safeguards building 2 and 3 and the fuel building are not connected (EDF 2006).

External hazards (Earthquake): To determine seismic risk, France complies with the methodology and criteria prescribed by the IAEA. In accordance with the IAEA recommendations, it notably sets a minimum overall site response spectrum of 0.1 g with infinite frequency (ASN 2017). The nuclear island of the 1300 MWe reactors is designed to withhold an earthquake of the NRC spectrum normalized to 0.15 g peak ground acceleration at zero period, while the site structure sometimes is only qualified to 0.1 g. For the hardened safety core, among others, the following requirements were set in accordance with technical prescription PT ECS-ND-7 (ASN 2017): “Encompassing the safe shutdown earthquake for the site, plus 50 %, and the probabilistic site spectra with a return period of 20,000 years'”. The respective earthquake is referred to as “noyau dur” (ND) earthquake or HSC earthquake.

In their evaluation of the “RP4 1300 - Anticipated instruction on the post-Fukushima hard core”, IRSN (2023c) finds several points which need further study. One of them is the leakage flow at the return from the primary pump seals. The values used in studies does not correspond to the maximum value that could be authorised in normal operation, and this value could increase following an ND earthquake. To date, EDF has not provided any information to characterise the potential effect of such an earthquake.

The EPR is designed for an EUR spectrum normalized earthquake at 0.25 g at zero period for the nuclear island and to 0.2 g for the site structure (ASN 2011).

For the EPR, the hardened concrete shell comprises not only the reactor building, but two full complete trains of the safety system, the main control room and the fuel building.

All four emergency diesel generators for the EPR are housed in two separate reinforced concrete buildings. The ultimate back-up reactors in the Switchgear building are protected by physical separation.

External hazard (flood): The baseline safety requirements for (external) flooding have been updated based on real events that happened in Blayais in 1999. In the first phase of the backfitting after Fukushima, flood resistance of a maximum 1000-year flood was reinforced (ASN 2017 p. 10). In the second phase additional protection such as a raised protected volume was installed (ASN 2020). Those measures were carried out on all sites concerned. Enhancing flood resistance is part of the hardened safety core.

For the EPR, external flooding considers, among others, the following parameters (EDF 2006 p. 328):

- River flood: Level reached by the millennium flood +15 % and level resulting from the largest known historical flood plus removal of a relevant structure.
- Seaside flooding / Tsunami: combination of maximum tide (coefficient 120) and millennial surge
- Dam breakage is considered as instantaneous removal of the dam.

The external hazard posed by flood is to a high degree site dependent.

1.1.1 Spent Fuel Pool

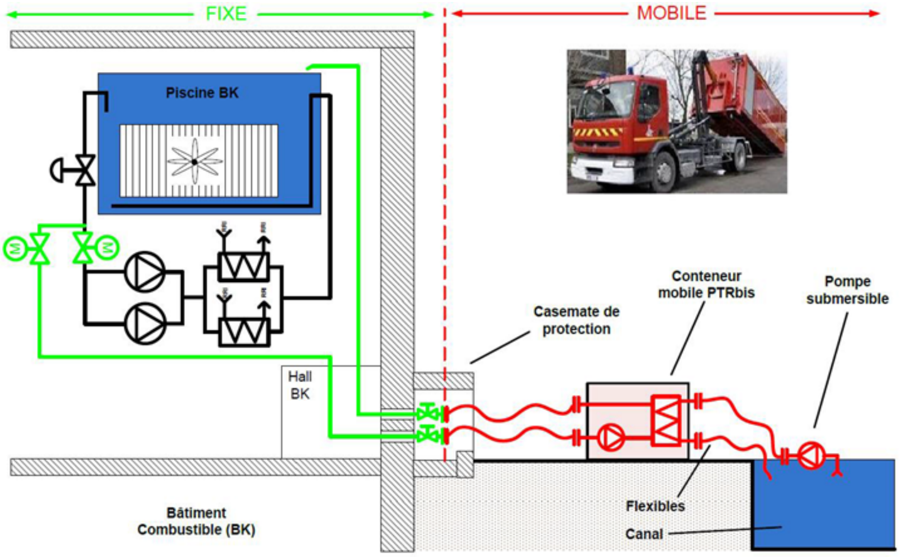
The spent fuel pool of the EPR is located in the fuel building which is enclosed by a hardened concrete protection shield. It has three trains of cooling (EDF 2006 p. 888). The third train has a diversified heat sink and can be resupplied with electricity by the ultimate backup generators (ASN 2011 p. 19).

For the 1300 MWe reactor fleet, there are two cooling trains. However, those rely on the same water supply and are not completely separate (Mertins 2021 p. 47). Electricity can be supplied by the emergency diesel generators. Spent fuel pool condition instrumentation has been reinforced and is part of the HSC (ASN 2020). At least for the 900 MWe reactor fleet, ASN stated that “the initial design and the current state of the spent fuel pools falls significantly short of the safety principles that would apply to a new facility” (ASN 2016a).

In case of failure of the two trains and the consequent loss of spent fuel cooling, a third train is to be provided by FARN using mobile equipment (PTRbis). For this case, the necessary piping has been permanently installed in the facade of the fuel building. However, this system is also not completely separate from the existing cooling trains (EDF 2023a).

Figure 3: Schematic illustration of the mobile cooling system PTZ bis (EDF 2023a).

Schematic illustration of the mobile cooling system PTZ bis



Piscine BK: spent fuel pool
Bâtiment combustible: fuel building
Casemate de protection: protective casemate
Conteneur mobile PTRbis: mobile emergency pump container
Pompe submersible: submersible pump
Flexibles: flexible piping
Canal: alternative heat sink

Source: EDF 2023

2 CONCLUSION AND RECOMMENDATIONS

The references are chapters in (EDF 2023a).

2.1 Planned provisions to terminate accidents with core melt at P4/P'4 reactors

Refers to I.2.4.2.2 "Measures implemented to deal with situations involving the risk of core melt".

2.1.1 Motivation/Observation:

The EPR has a system for the stabilization of the core melt (corium) to prevent failure of the containment in case of severe accidents. Retrofitting of an EPR type core catcher to the P4/P'4 reactors is not possible due to space limitations below the reactor pressure vessel. EDF is planning to retrofit other measures to stop accident progression in case of core melt accidents instead and claims that those provisions are similarly effective. However, they rely on a number of assumptions, including that the corium would spread on a large area and that the corium, once spread on the containment floor, could be effectively cooled by flooding with water from above. The OECD/NEA Project "Reduction of Severe Accident Uncertainties" ROSAU plays a crucial role in the demonstration of the effectiveness of the corium retention system for the 1300 MWe reactor fleet, but those R&D efforts (LICHT 2023) are still on-going and have not shown convincing results yet.

2.1.2 Recommendation:

It is recommended to require full experimental proof and the demonstration of applicability before approving LTE.

2.2 Qualification of emergency diesel generators

Refers to I.2.7.3.1 "Ultimate backup Diesel generators",

2.2.1 Motivation/Observation:

There are two emergency diesel generators per 1300 MWe reactor associated with level 3 safety systems (level of defence 3, design basis systems) and three sources of electricity associated with level 4 safety systems (DEC systems). TAC is shared among several reactor blocks and cannot supply the safety injection system (SIS). Additional AC power could be provided and be operational by FARN after a maximum of 36 hours. The EPR concept against loss of power consists of four emergency diesel generators (for design basis events) of which two are in buildings which are protected against external hazards. In addition, there are two additional so-called ultimate backup diesel generators for safety level 4, DEC, available.

2.2.2 Recommendation:

For the 1300 MWe reactors only two EDGs fulfil the more rigorous requirements for level 3 safety systems, there is only a single redundancy, while this holds true for four EDGs of the EPR. While the EPR emergency diesel generators follow a n+3 redundancy concept, the P4/P'4 reactors follow a n+1 approach. It is recommended to try to elevate the safety level of the 1300 MWe fleet to the EPR also at the level of design basis safety systems, or, in case this is not possible, clearly state the deltas and evaluate the resulting additional risk.

2.3 Design basis accidents – redundancy of safety systems

2.3.1 Motivation/Observation:

There is a considerable gap regarding design basis safety systems redundancy between the 1300 MWe fleet and the EPR. The general safety philosophy for the 1300 MWe reactors design basis consists of two trains (2x100% safety concept). If one of those trains fails, the other train can fulfil the necessary safety functions (single failure criterion). The EPR safety systems, which are the target for retrofitting of the 1300 MWe fleet, consist of four trains (4x100% safety concept). The planned backfitting measures of the 1300 MWe reactors do not to address the redundancy and diversity of level 3 safety systems that handle design basis accidents.

2.3.2 Recommendation:

It is recommended to retrofit additional safety systems and qualify them as design basis safety systems to elevate the P4/P'4 design closer to the EPR.

2.4 Consideration of FARN in the safety evaluation of P4/P'4 reactors

Refers to I.2.7.4. "FARN"

2.4.1 Motivation/Observation:

The Nuclear rapid intervention force (FARN) is located at different nuclear power plants all over France and is supposed provide support on-site in case of emergency within 24 hours. This makes it immediately clear that the FARN is operating on safety level 4 which is put in operation during design extension conditions to prevent and mitigate severe accidents. Those systemas relevant for safety should not be used to compensate for existing deficiencies at safety level 3. Further, FARN would also support in case of need at EPR sites and is thus not a measure decreasing the difference in safety levels between the 1300 MWe reactor fleet and the EPR.

2.4.2 Recommendation:

Non-permanent measures such as the implementation of FARN are covered in IAEA Safety Standard SSG-88. One aspect is that mobile equipment should not be relevant in the short-term phase of design basis and design extension conditions. It is therefore recommended to not credit FARN equipment when comparing the safety level of 1300MWe reactors against state of the art (EPR).

2.5 State of the art consideration of aircraft crash

Refers to I.2.2.2.1.1.16 "air risk management"

2.5.1 Motivation/Observation:

For the safety analysis of a nuclear power plants not only natural but also events that are induced by human activities are considered. This includes the crash of an aircraft. The probabilistic safety analysis of the French 1300 MWe reactor fleet has been updated recently to also include large commercial aircraft. For the EPR, a deterministic approach was used and explicitly the case of an intended attack of an aircraft on a nuclear power plant was considered.

2.5.2 Recommendation:

It is recommended to elevate the P4/P’4 level of protection against aircraft crash to the EPR level, which would include the same assumption on load cases and require the same method of analysis. In case this is considered to be not feasible, it is recommended to point out this gap in the safety level to the EPR and evaluate the resulting risks in a risk report.

2.6 Design basis protection and giving credit for HSC/FARN

Refers to I.2.7 “Contribution of the Hardened Safety Core to the Objectives of the Re-examination”

2.6.1 Motivation/Observation:

The hardened safety core increases the resilience of the 1300 MWe reactor fleet against external hazards such as earthquakes and flooding. The design of the EPR against earthquakes and flooding is based on much more stringent requirements.

2.6.2 Recommendation:

It is recommended to address deficits in dealing with the design-based earthquake for the 1300 MWe reactor fleet by retrofitting and to also seismically harden the LLS and the TAC. It is recommended to enhance the safety level of the P4/P’4 fleet beyond providing HSC systems and FARN.

2.7 Spent fuel pool cooling/emergency cooling systems

Refers to I.2.3 “Spent Fuel Pool"

2.7.1 Motivation/Observation:

The spent fuel pool of the EPR is part of the hardened fuel building. It can be cooled by three different separate trains. The spent fuel pool of the 1300 MWe reactor fleet has been reinforced and improved as part of the Hardened Safety Core. An additional ultimate cooling water supply and installations for the use of mobile equipment increase the reliability of cooling, while all three trains are intermeshed. In addition, the spent fuel pool is located inside the containment in the EPR, while housed in a separate building at the 1300MWe reactors which level of protection against hazards like aircraft crash is significantly below the level of the EPR containment.

2.7.2 Recommendation:

It is recommended to retrofit spent fuel pool cooling systems/emergency cooling systems to match the degree of redundancy and diversity of the EPR spent fuel pool cooling systems and strengthen the structures which are housing the spent fuel pool to the level of the EPR containment.

LIST OF FIGURES

LIST OF TABLES

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GLOSSARY

AC .... Alternating Current

ASN..................................... Autorité de Sûreté Nucléaire

SGEFW / ASG .... Steam Generator Emergency Feedwater System

ATWS .... Anticipated Transient Without Scram

CCWS.................................. Component Cooling Water System

CFS...................................... Condensate and Feed Water System

CHRS .... Containment Heat Removal System

CMSS .... Core Melt Stabilization System

CP(X)................................... Chicago Pile

DBA .... Design Basis Accidents

DEC..................................... Design Extension Condition

DUS .... Ultimate Backup Diesel Generators

EAS .... Containment Spray System

ECC / CCL........................... Emergency Control Centre

ECCS .... Emergency Core Cooling System

ECR .... Emergency Control Room

EDF .... EDF

EDG .... Emergency Diesel Generators

EFWS .... Emergency Feedwater System

EPR .... Evolutionary Power Reactor// European Pressurized Reactor

EPGB .... Emergency Power Generating Buildings

EPSS .... Emergency Power Supply System

ESWS .... Essential Service Water System

EVU..................................... Ultimate Enclosure Cooling Circuit

FARN .... Rapid Response Nuclear Taskforce

FPCPS .... Fuel Pool Cooling and Purification System

HP .... High Pressure

HSC .... Hardened Safety Core

HVAC .... Heating, Ventilation, and Air Conditioning Systems

IRWST .... In-containment Refueling Water Storage Tank

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

ISBP .... Low-pressure Feed Pumps

ISMP .... High-pressure Injection Pumps/Medium Head Safety Injection Pumps

I&C .... Instrumentation and Control System

JP .... Event of Fire

LHSI .... Low-head safety injection pump

LLS .... Emergency Turbo Generator

LP .... Low Pressure

MCCI .... Molten Core Concrete Interaction

MCR .... Main Control Room

MHSI .... Medium-head Safety Injection Pump

MSS .... Main Steam System

MW .... Megawatts

MWe .... Megawatts electric

MWth .... Megawatts thermal

NEA .... Nuclear Energy Agency

NPSS .... Normal Power Supply System

NRC .... Nuclear Reactor Containment

PSR .... Safety Surpassing Standard Periodic Safety Review

PTR .... Pool Treatment and Cooling System

PWR .... Pressurized Water Reactor

ROSAU .... Reduction of Severe Accident Uncertainties

RHRS .... Residual Heat Removal System

RIC .... Reactor In-core Cooling

RPV .... Reactor Pressure Vessel

RRI .... Intermediate Cooling Water System

RSS .... Remote Shutdown Station

R&D .... Research and Development

SAHRS .... Severe Accident Heat Removal System

SEC .... Secondary Cooling System

SED .... Demineralized Water Distribution System

SER .... Water Distribution System

SEU .... Ultimate Water Sources

SG .... Steam Generator

SIS .... Safety Injection System

SMS .... Seisme Majore de Securite

SRU .... Ultimate Cooling Circuit

SSE .... Safe Shutdown Earthquake

TAC .... Emergency Combustion Turbine

UHS / SEG .... Ultimate Heat Sink

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